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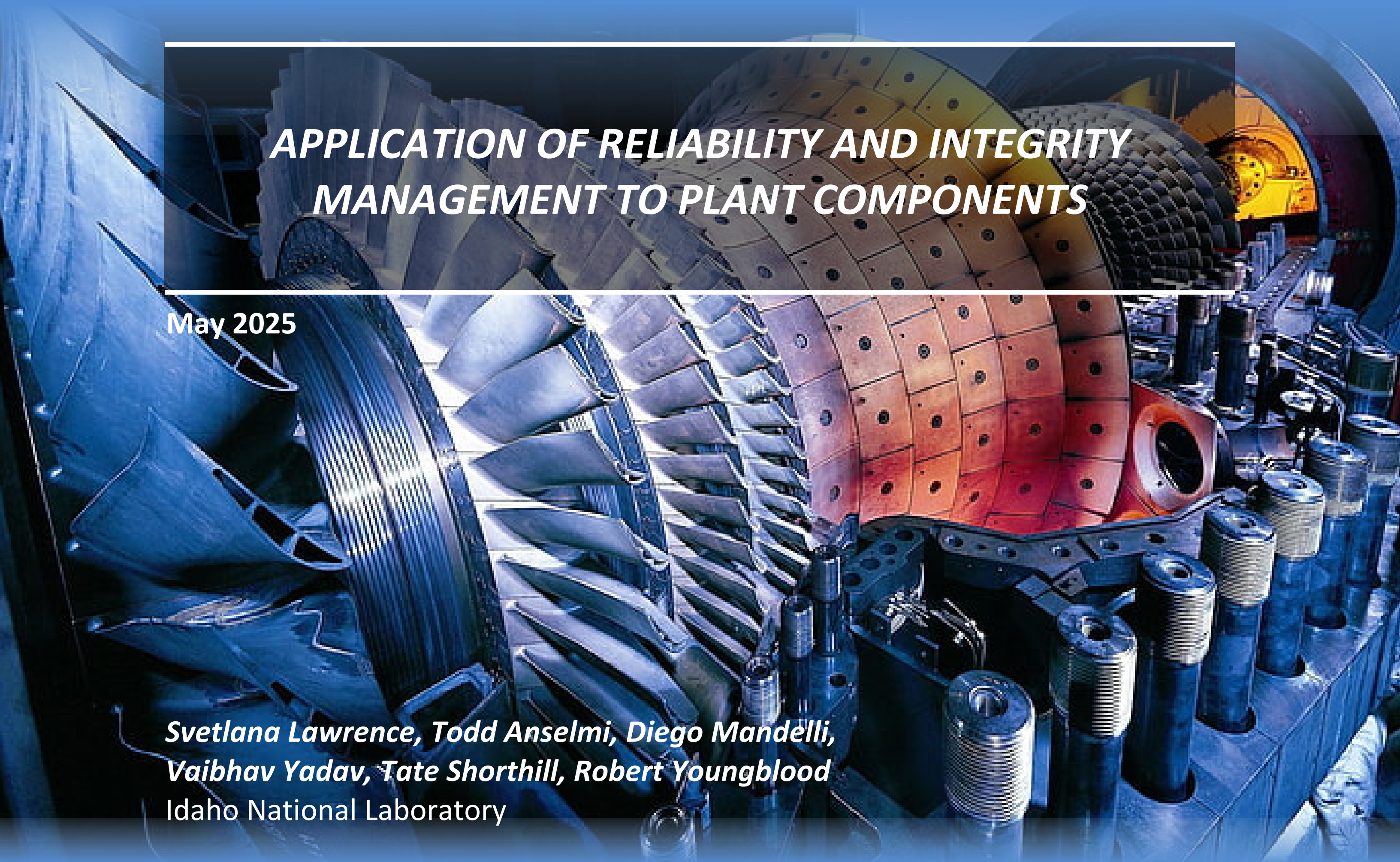

APPLICATION OF RELIABILITY AND INTEGRITY MANAGEMENT TO PLANT COMPONENTS

May 2025

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EXECUTIVE SUMMARY

U.S. Nuclear Regulatory Commission (NRC) regulations in Title 10 of the Code of Federal Regulations (CFR) 50.34(b)(6)(iv) and 52.79(a)(29)(i) require all applicants for operating and combined licenses to include plans for conducting normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components (SSCs). However, the regulations prescribe specific preservice inspection (PSI) and inservice inspection (ISI) program requirements for boiling- and pressurized-water-cooled nuclear power reactors. The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Division 2 (BPVC XI-2), provides a process for developing a Reliability Integrity Management (RIM) program similar to a traditional PSI and ISI program under ASME Boiler and Pressure Vessel Code, Section XI, Division 1, for all types of nuclear power plants.

A RIM program has not yet been developed and presented for use by a nuclear power plant licensee in the United States; however, it is anticipated that applicants which operate new non-light-water reactors may include a RIM program as part of plans for SSC maintenance, surveillance, and periodic testing. In preparation for using BPVC XI-2 in RIM program development, evaluation of applying a RIM program to plant SSCs was performed and described in this report. The report provides an assessment of how various RIM strategies could be used to achieve the established performance targets considering the factors that affect reliability, including but not limited to design strategies, material selection, operating practices, PSIs and ISIs, and repair and replacement practices.

There are connections between RIM and the Licensing Modernization Project approach described in Nuclear Energy Institute 18-04, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” and endorsed in NRC Regulatory Guide 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors.” Both processes utilize reliability target establishment and allocation as an integral part of the process.

Continuous Monitoring: Traditional ISI programs most often utilize direct indications of degradation, such as visual examinations and volumetric examinations. A RIM strategy may rely on continuous monitoring through indirect means like sodium detectors or humidity detectors that indicate a subsequent action is to be taken that then implements a direct means of identifying degradation. In the case of pressure boundary leakage, continuous monitoring often identifies the leak after it occurs. This approach for monitoring for leakage instead of degradation is different from the current ISI practice where monitoring strategies are developed and applied to ensure the absence of unacceptable levels of damage with the specific goal of avoiding any leaks. However, fundamentally the two approaches are similar as both have the goal to detect degradation prior to the occurrence of an unsafe condition. Acceptable leakage monitoring would enable effective performance monitoring strategy.

Although BPVC XI-2 is a code for ensuring the integrity of passive pressure boundaries, the contribution of active components (e.g., electrical systems) influences the reliability targets and determination of RIM strategies. A potential consideration with excluding the active equipment is that the results may be skewed due to the application of an overconservative treatment of passive components (suggested by

BPVC XI-2) since no credit is given to active components for precluding or mitigating accidents. Consideration of both passive and active SSCs as part of an overall RIM strategy is a holistic approach that would result in a better understanding of the plant performance and allow for efficiencies in performance monitoring of important system functions in addition to meeting specific RIM program requirements.

The key steps in developing a RIM strategy may have more than one approach, and the example cases in Section 6 and Section 7 of this report provide an effective understanding of what an Owner may present during the development of its RIM program and associated RIM strategies. The example case in Section 6 is a light-water reactor which was selected based on an ability to utilize well-known and readily available information. Specifically, the reactor coolant system (RCS) for a pressurized-water reactor (PWR) design is utilized in this example. The sodium fast reactor design used as an example in Section 7 was selected because of the ability to use publicly available information and is not intended to be representative of any particular reactor design currently under development. In this example, a RIM strategy is developed for the intermediate heat transfer system. Each of these example cases points out aspects to consider when developing a RIM program and presented the key steps in establishing the RIM strategy.

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CONTENTS

FIGURES

TABLES

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ACRONYMS

ACS	air cooling system
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
BDBE	beyond design basis event
BOL	beginning of life
BPVC	boiler and pressure code
CDF	core damage frequency
CFR	Code of Federal Regulations
CL	cold leg
CLR	component-level requirement
CSCC	caustic stress corrosion cracking
DBE	design basis event
DID	defense in depth
DM	degradation mechanism
DMA	degradation mechanism assessment
ECT	eddy-current testing
F-C	frequency-consequence
FOM	figure of merit
HL	hot leg
HX	heat exchanger
IHTS	intermediate heat transfer system
IHX	intermediate heat exchanger
INL	Idaho National Laboratory
ISI	inservice inspection
LMP	licensing modernization project
LOCA	loss-of-cooling accident
LWR	light-water reactor
MANDE	monitoring and nondestructive examination
MANDEEP	MANDE expert panel
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NPP	nuclear power plant
NRC	Nuclear Regulatory Commission
NUREG	U.S. Nuclear Regulatory Commission technical report designation
O&M	operation and maintenance
pdf	probability density function
PFAD	plugging filter aerosol detector
PHTS	primary heat transfer system

POD	probability of detection
PRA	probabilistic risk assessment
PSI	preservice inspection
PWR	pressurized-water reactor
PWSCC	primary water stress corrosion cracking
RAMI	reliability, availability, maintainability, and inspectability
RCS	reactor coolant system
RG	regulatory guide
RHR	residual heat removal
RIM	reliability integrity management
RIMEP	reliability integrity management expert panel
RVACS	reactor vessel auxiliary cooling system
SAR	safety analysis report
SFR	sodium fast rector
SG	steam generator
SID	sodium ionization detector
SS	stainless steel
SSC	structure, system, and component
SWRPRS	sodium-water reaction protection relief system
TASCS	thermal stratification, cycling, and striping
TT	thermal transient
U.S.	United States
Var	variable
VT	visual examination

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APPLICATION OF RELIABILITY AND INTEGRITY MANAGEMENT TO PLANT COMPONENTS

1 BACKGROUND

U.S. Nuclear Regulatory Commission (NRC) regulations in 10 Code of Federal Regulations (CFR) 50.34(b)(6)(iv) and 52.79(a)(29)(i) [1] require all applicants for operating and combined licenses to include plans for conducting normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components (SSCs). However, the regulations prescribe specific preservice inspection (PSI) and inservice inspection (ISI) program[a] requirements for boiling- and pressurized-water-cooled nuclear power reactors. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section XI, Division 2 [2], provides a process for developing a Reliability Integrity Management (RIM) program similar to a traditional PSI and ISI program under ASME BPVC, Section XI, Division 1 [3], for all types of nuclear power plants (NPPs).

The development of an alternative to the current PSI and ISI requirements results from the different design features included in advanced non-light water reactors (ANLWRs). These design features often include a different coolant (e.g., sodium rather than water) and higher operating temperatures. These different design features introduce various degradation mechanisms (e.g., creep and thermal stresses) that may necessitate nondestructive examination (NDE) techniques not required for today's light-water reactors (LWRs). In addition, the longer timeframes between shutdowns may necessitate an increased reliance on continuous monitoring. The approach to monitoring the structural integrity of the pressure boundaries of these new reactors will differ from the current ISI-required examinations prescribed for current LWRs.

Through the issuance of U.S. NRC Regulatory Guide (RG) 1.246, "Acceptability of ASME Code, Section XI, Division 2, 'Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants,' for Non-Light-Water Reactors," [4] the NRC endorses the use of ASME Section XI, Division 2, with conditions, for developing a RIM-type program. It further states that RIM provides a process for identifying degradation mechanisms and reliability targets for SSCs in the RIM program and developing a RIM strategy for performance monitoring, including evaluating uncertainties. ASME Section XI, Division 2, provides general guidance on developing the RIM strategy but leaves the user the flexibility to determine the most effective means to establish the monitoring and NDE techniques that should be performed to assure adequate plant SSC performance.

The current state of knowledge for the use of RIM has been assessed and the results are presented in "Reliability and Integrity Management Scoping Study" [5],which detailed three cases of RIM program development, including approaches for establishing reliability targets.

The intentional flexibility in the ASME Section XI, Division 2, makes it necessary for users of ASME Section XI, Division 2, to establish the strategies for RIM programs and show how the strategies link condition monitoring with reliability targets that represent adequate performance. In preparation for reviewing future RIM strategies and RIM program submittals, the NRC contracted Idaho National Laboratory to evaluate methods of developing reliability targets and assessing how RIM strategies may be used to achieve the reliability targets while considering how different factors can affect reliability.

[a] Reference to an ISI program in this report is inclusive of required PSI requirements when explicitly citing PSI in connection with ISI.

2 INTRODUCTION OF RIM

ASME Section XI, Division 2, "Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants," is a technology-neutral standard of the ASME BPVC. It was determined to be an acceptable process by the NRC for developing a RIM program like a traditional PSI and ISI program under ASME Code, Section XI, Division 1, for non-light-water reactors. The RIM program contains provisions beyond a traditional ISI program, such as using a probabilistic risk assessment (PRA) to develop reliability targets for SSCs within the scope of the program. It also relies on establishing the monitoring, NDE, and repair and replacement practices to maintain component reliability based on the determination of what degradation mechanisms may exist throughout the life of the plant [2].

Article RIM-2 of ASME Section XI, Division 2, provides an overview of the RIM program, and a description of the full process is not the subject of this report. This report focuses on how SSCs may be monitored in accordance with a RIM strategy and how that monitoring correlates to established reliability targets.

2.1 RIM Operational Context

Generally speaking, an SSC can be defined as two connected elements (see Figure 1): its form (i.e., the physical entity and its constituent components) and the set of functions it is supporting. For example, when considering a centrifugal pump, the form element consists of all the components and subcomponents that make the pump (e.g., motor, stator, shaft, impeller), and the function element indicates the function of the pump (i.e., increase fluid pressure). Similarly, for passive components and subcomponents in a piping system, the form element consists of all pipes, junctions, and welds that are typically designed to contain and transport some substance (e.g., fluid, air).

When thinking in terms of degradation mechanisms (DMs) (e.g., flow-accelerated corrosion, chemical erosion), it is relevant to note that they directly affect the form element (see Figure 1), which can in turn potentially impact the element's function (e.g., reduction of pump flow rate, fluid leak out of pipe weld). The rate of degradation caused by DMs is influenced by external environmental conditions noted as variables (Var) in Figure 1 (e.g., fluid chemical composition, temperature, pressure). Note that the reverse of this causal representation is also allowed: form or function degradation might affect external environmental conditions.

The data collected by monitoring and from NDE may correspond to different elements of the diagram shown in Figure 1. This data may be acquired both directly and indirectly for the variables, form, or function elements being monitored. SSC degradation can be assessed using the collected data from monitoring as well as data trends. Actions informed by the collected and analyzed data can be performed to improve the condition of the "form" (i.e., the physical asset), such as:

- Maintenance activities (e.g., pump shaft replacement, pipe weld repair) to restore the form of an SSC, which consequently restores its function.
- Update monitoring strategy if the initial SSC degradation hypotheses are no longer valid (e.g., improve monitoring system performance, increase surveillance frequency). While this action does not affect the form of an SSC, it is intended to better inform decisions about maintenance activities.
- Update SSC lifecycle plan (e.g., periodic SSC replacement or maintenance).

Note that the reliability properties associated with an SSC typically refer to the function element(s) of such an SSC. Basic events in a plant PRA correspond to various ways an SSC could fail to perform an assigned function (i.e., failure modes, such as an injection pump that fails to run). A system fault tree represents the functional relations between SSCs within the system and relationships with SSCs in supporting systems. An important argument about Figure 1 is that it explicitly shows how SSC reliability (i.e., function node in Figure 1) balances actual SSC degradation phenomena (i.e., changes of form), available monitoring data (including its quality and its coverage), and the decisions made based on available data.

As part of the RIM decision process for each SSC, it is important to identify form-function relationship(s) (i.e., at what point a degraded form can no longer support an intended function), DMs that might affect SSC performance, variables affecting degradation mechanisms, and relations between available monitoring data and the elements shown in Figure 1. This process is imperative to understanding how a chosen SSC RIM strategy satisfies the postulated performance requirements for an SSC where performance requirements are expressed as reliability targets.

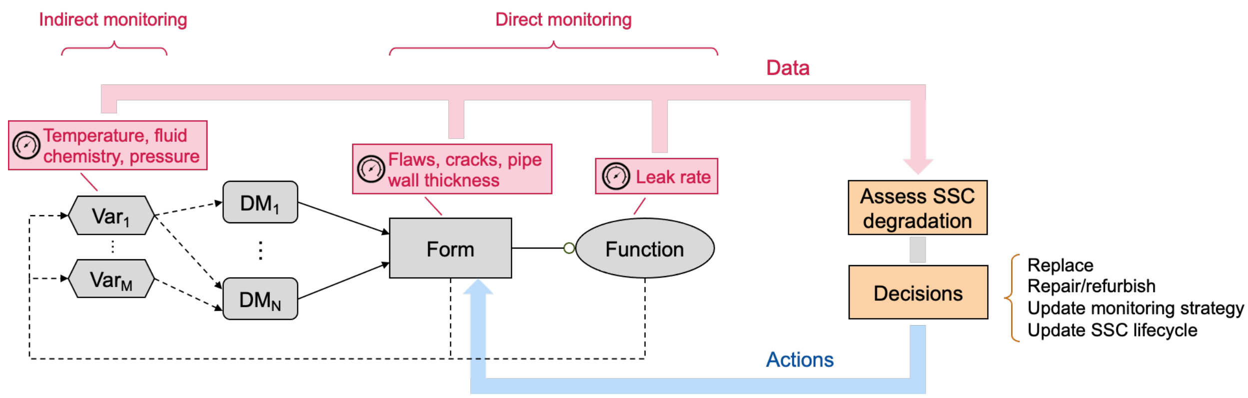

Figure 1. Representation of the relations between an SSC (shown as a coupled form-function), applicable degradation mechanisms (indicated as DM), and the external environmental conditions that affect degradation mechanisms (indicated as Var).

2.2 Initial Considerations About SSC Reliability Modeling

Using physics-based considerations, an undesired SSC event (e.g., SSC failure) can be considered an outcome resulting from the SSC microscopic structure (i.e., the *capacity*) not being able to withstand stress (i.e., the *load*) imposed by operating conditions, either normal or abnormal. While this statement has been phrased in very generic terms, it can be expanded with more details depending on the SSC operational context using mechanical, chemical, nuclear, or electrical arguments. This supports the main argument that SSC failure is the product of deterministic (i.e., physics-based) phenomena. This load vs. capacity argumentation is graphically represented in Figure 2 where load and capacity are represented using probability density functions (pdfs). When the load exceeds capacity, there is a chance for failure. The area under the intersection of the two probability curves represents the probability of failure. Note that load and capacity could be represented as point values (rather than pdfs) and similar reasoning applies—when the load exceeds capacity, SSC failure can be expected. In the case of point values, there is no probabilistic value describing a failure, rather the notion is the failure is essentially guaranteed when the load exceeds capacity.

Following the postulation of failure phenomenon, the opposite exists; when load and capacity curves do not overlap, SSC failure cannot occur (top figure of Figure 2). SSC degradation over time causes the capacity curve to decay (i.e., move to the left), and when load and capacity curves overlap, SSC failure may be expected (bottom figure of Figure 2).

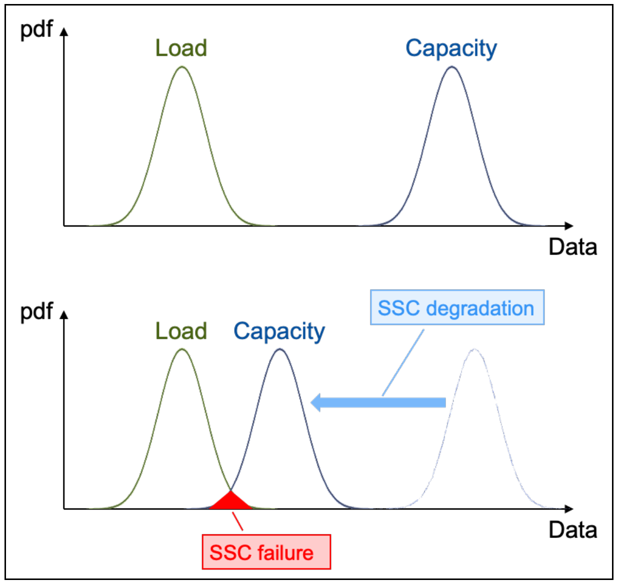

Figure 2. SSC failure represented in terms of load and capacity.

The transition from deterministic (i.e., physics-based) to probabilistic arguments is warranted in the presence of multiple variables and associated uncertainties making a single-value representation of load and capacity infeasible. In order to ensure a single-value representation is valid for any covered scenario, a bounding case must be used where limiting assumptions and boundary conditions are applied to estimate both load and capacity values. This has been the approach historically applied in the nuclear industry since the onset of commercial nuclear power. However, this bounding approach typically results in overdesigned plant systems, safety features, and operating processes, which may cause an altered risk profile since some risks are overestimated compared to others. The nuclear industry has been increasing its use of risk-informed, performance-based approaches. This increased use necessitates the application of best-estimate plus uncertainty or validation methods to the set of model parameters, assumptions, and boundary conditions. Therefore, probabilistic approaches to measure safety and risk are warranted to replace or supplement traditional deterministic approaches.

When probabilistic arguments are used, the failure rate represents the probability of an SSC failure within a time interval. Assuming an undesired SSC event is quantified in terms of a failure rate λ, then using classical reliability modeling notions, SSC reliability R is quantified as function of λ (i.e., $R = R(\lambda)$). The SSC failure rate λ accounts for the entire path—from the environmental condition(s) to damage mechanism(s) to the physical degradation of the SSC form to the SSC function failure shown in Figure 1. Conceptually, SSC reliability (often conveyed in terms of a failure rate or a failure probability) is a probabilistic measure associated with the occurrence of an event (i.e., a loss of function for the generic SSC representation shown in Figure 1). The reliability of a given type of SSC is updated based on cumulative performance history for similar SSCs. Reliability values for nuclear systems and components are calculated based on observed failures during periodic testing of on-demand equipment (e.g., safety injection pumps), observed failures of normally operating equipment (e.g., service water pumps), and observed pressure boundary failures (e.g., pipe breaks).

As a starting point, consider an SSC that is not subject to any form of testing/surveillance/condition monitoring or maintenance operation. In this operational context, the occurrence of the SSC is an event which is completely aleatory in nature. After a failure of such SSC, a decision is made to either replace or repair a failed SSC. Figure 3 graphically shows the relationship between data (i.e., the occurrence of an event—SSC failure) and a decision (i.e., SSC repair), which is sequential in nature: event occurrence "causes" SSC repair action. Thus, based on rate of occurrence of SSC failure events, SSC reliability is updated (e.g., through a Bayesian updating process).

This report extends the concept of SSC reliability from failure rates based on observed failures into a RIM context, as indicated in Figure 1. RIM is defined here as the set of available monitoring data (i.e., form, function, or external environmental conditions) and decisions made based on the collected and analyzed data. Unlike a reactive decision shown in Figure 3, available monitoring data can provide information about SSC performance (for either form or function), which can be used to trigger proactive actions designed to prevent SSC failure (e.g., increase frequency of inspections, restore SSC when approaching limiting conditions). Consequently, the relationship between data and decisions becomes integrated and iterative in nature.

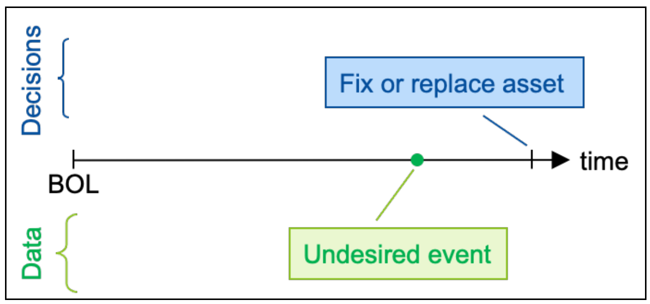

Figure 3. Graphical representation of the relationship between data (occurrence of an undesired event) and decisions (asset repair) where BOL is the beginning of life.

In a RIM context, SSC reliability is still a probabilistic measure (inferred) associated with the occurrence of an event (i.e., SSC failure) where such an event can occur if any of these conditions are satisfied:

- Missed or misinterpreted identification of SSC degradation from direct or indirect monitoring data (e.g., degradation is not detected, degradation is not appropriately categorized).
- Inability to take action before event occurrence after the SSC degradation has been detected.

The first condition points to the elements of the SSC RIM plan that are *unknown* for one reason or another. Some examples which would fall under this point include: 1) monitoring data might not be available continuously; 2) monitoring data might not provide a precise assessment of SSC degradation; 3) monitoring data might not exhaustively cover all physical elements of the SSC; 4) monitoring system detecting capabilities might be imprecise (measured values might be affected by uncertainties or monitoring system might not be able to reliably detect degradation phenomena).

The second condition points to the elements of the SSC RIM plan that are *known* (i.e., degradation phenomena have been detected, and their evolution can be tracked). The presence of such degradation poses a critical element that might lead to SSC failure, which can be quantified using load vs. capacity considerations described above (i.e., deterministic in nature). In addition, the prescribed corrective actions defined in the SSC RIM plan cannot be immediate and, hence, the time to perform such actions needs to be compared with the progression rate of the degradation phenomena.

Thus, the goal of SSC reliability modeling within the RIM context is to capture the probabilistic nature of the conditions listed above; such reliability considerations are presented in detail in Section 5.3.

3 CONDITION AND PERFORMANCE METRICS

There are two categories of metrics to describe SSC performance, probabilistic and deterministic. Reliability is a probabilistic measure of an SSC's condition. In the RIM program and the Licensing Modernization Project (LMP) approach described in Nuclear Energy Institute (NEI) 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light-Water Reactor Licensing Basis Development," [6] the minimum acceptable reliability values are defined as reliability targets.

3.1 Reliability and Performance

The LMP and NEI 21-07 [7] defines two terms, reliability and capability. The term "reliability" refers to the reliability performance metrics involved in the estimation of safety function failure probabilities in the PRA. Reliability is not observable but rather is calculated based on observed performance measures and available generic evidence. Capability is a performance measure used to establish the success criterion required to quantify the failure probability in the PRA. Based on these definitions, reliability in the LMP framework is a probabilistic measure while capability is a deterministic measure of SSC performance where the two terms are correlated. The deterministic metrics of an SSC's condition and performance are typically expressed in physical terms, such as pipe wall thickness or flaw size. When determining if the condition is acceptable for continued use, the measured value may be compared against a predetermined condition that can initially conclude that an SSC's condition is acceptable for continued use and the value representing the condition is recorded as part of ongoing monitoring and trending. If the condition exceeds the predetermined acceptance criteria, the condition may be more formally evaluated against a limiting condition value, such as the minimum pipe wall thickness or a maximum flaw size.

The probabilistic metric of an SSC's condition is a reliability value of an SSC typically expressed as a compliment of the probability of failure (i.e., reliability = 1 – probability of failure). Similar to the above paragraph, the acceptability of an SSC for continued use is determined based on the comparison of the SSC's current reliability and the predetermined acceptance criterion also expressed in probabilistic terms (i.e., accepted reliability). If the calculated current reliability value is below the predetermined acceptance criterion, the condition may be more formally evaluated against a limiting reliability value, formally defined as a reliability target.

There could be a situation where *reliability targets* are established only for the plant or a function-level allowing flexibility with the SSC-level reliability targets (i.e., as long as the plant or function actual reliability is within the predetermined *reliability targets*, deviation of reliability values on a component-level is acceptable). In this case, the SSC limiting condition can be expressed in deterministic terms (i.e., SSC capability and SSC conditions). The actual SSC reliability will be calculated based on the SSC performance and observed conditions, and this SSC actual (i.e., as-built, as-operated) reliability will be used in the plant or function reliability estimates in the as-built, as-operated PRA. The as-built, as-operated plant or function reliability will be compared to the plant or function *reliability target* predetermined in the plant license commitment(s). As long as the plant or function actual reliability is within the predetermined *reliability targets*, SSC performance expressed in deterministic terms is adequate and acceptable.

There are multiple difficulties associated with setting a *reliability target* for each SSC, a probabilistic metric for an SSC performance monitoring. First, a predetermined *reliability target* for each SSC will dramatically limit plant operating flexibility that otherwise will be available if the *reliability targets* are set only on a plant or function-level. Second, as demonstrated in Reference [5], there are millions of combinations of SSC-level reliability values that would satisfy the *reliability target* on a plant function level, so the questions become how to select *reliability targets* on a component-level and why the *reliability target* associated with a component is needed. Third, probabilities of failure are often inversely proportional to metrics representing physical conditions (e.g., the probability of a 2 in^2 pipe break is smaller than the probability of a 0.5 in^2 pipe break, which makes it impossible to use corresponding probabilities of failure as reliability targets).

Let's illustrate reliability vs. performance using an example where the capability of a service water system to provide cooling to plant equipment (i.e., safety function) is defined as a minimum flow rate of 2,000 gallons per minute (gpm). The function is considered failed if the flow rate is lower than 2,000 gpm. If this system is capable of delivering up to 2,400 gpm during normal operation, it means that up to 400 gpm could be lost (e.g., through a pipe break) before the system's function to provide cooling is considered failed. The maximum leak rate of 400 gpm can be easily correlated to a corresponding size of pipe break given the system pressure, say 2 in^2. Therefore, in this scenario, the maximum allowed pipe break size is 2 in^2.

A service water system pipe break can be used as an example to demonstrate the third point. Given the service water system material, water chemistry, and operating pressure, the probability of a pipe break with a given size can be estimated using either historical data, probabilistic fracture mechanics techniques, or a combination of methods. For the purpose of this argument, the probability of a 2 in^2 pipe break is 7.5E-8 and of a 0.5 in^2 break is 5.5E-7. The limiting condition when the system is losing its function is a pipe break size of 2 in^2, with the corresponding probability of such a break being lower (i.e., reliability is higher) than for a smaller break size of 0.5 in^2. So, what value in this case should be used as the reliability target for the pipe?

To resolve the topic presented as the third point above, the performance target (i.e., the limiting condition of an SSC to be used in RIM) should be set based on the expected capability of the SSC to perform its assigned function(s). The probabilistic reliability metrics are useful to indicate the current SSC condition and to forecast the degradation of its reliability into the future to inform decisions about corrective actions.

As an alternative, SSC reliability targets in probabilistic terms could be established following the process described in Section 3.1.1, where limiting the probability of failure of a given SSC is decomposed from the plant reliability targets. In this case, the actual capability of an SSC is considered only implicitly. Instead, a mathematical computation is performed to calculate limiting not to exceed failure probabilities (i.e., *reliability targets*) for all the SSCs contributing to the plant-level failure rates by decomposing the plant- and function-level reliability targets into SSC-level reliability targets.

This is a set of definitions based on what has been described above and in Sections 2.1 and 2.2:

- *SSC failure:* Observed instance of the SSC not being able to perform its intended function(s)
- *SSC reliability*: Measure expressed in probabilistic terms of the ability of the SSC to perform its intended function(s)
    - *SSC performance*: Measure (framed in either deterministic or probabilistic terms) of the ability of the SSC to perform its intended function(s).
    - *SSC performance target*: Measure (framed in either deterministic or probabilistic terms) of the minimum acceptable ability of the SSC to perform its intended function(s). The performance target is the limiting not to exceed condition for the SSC performance that may be expressed in probabilistic terms (e.g., probability of failure or reliability) or deterministic terms (e.g., minimum pipe wall thickness or maximum flaw size).
    - *SSC reliability target*: Measure (framed in probabilistic terms) of the minimum acceptable ability of the SSC to perform its intended function(s). Similarly to the performance target, the reliability target is the limiting not to exceed condition for the SSC performance expressed in probabilistic terms.

- *SSC actual reliability* (see Figure 4): Measure (framed in probabilistic terms) of the as-is, i.e., as-built, as-operated condition, of the ability of the SSC to perform its intended function(s) based on observed performance (derived from past SSC failure occurrences) and actual as-is SSC condition (based on present monitoring data).
- *SSC condition:* Measure (framed in deterministic terms) of the SSC physical conditions that influence the ability of the SSC to perform its intended functions defined in observable measurable physical terms (e.g., wall thickness, flaw size).

- *SSC RIM strategy:* Set of operations that are being (or planned to be) performed on the SSC designed to assess and quantify SSC physical architecture (in search for SSC degradation) and SSC behavior in terms of its intended functions and restore SSC physical conditions (e.g., through SSC replacement or maintenance activities) to ensure satisfactory behavior.

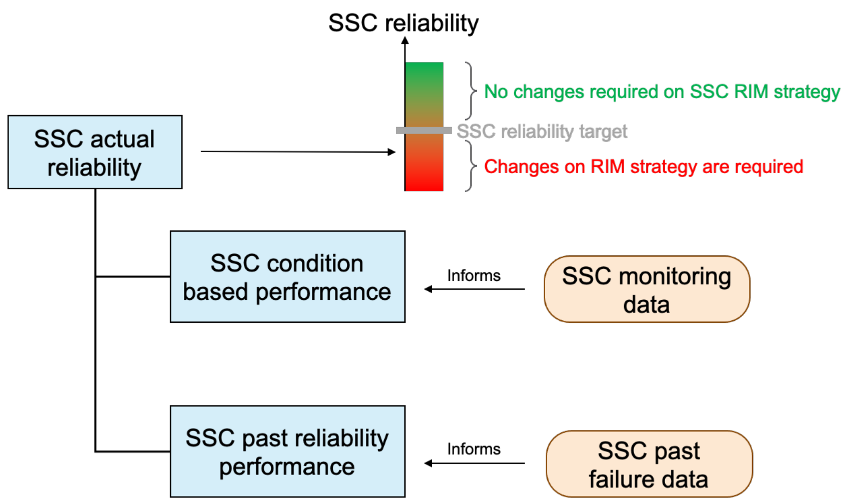

Figure 4. Decomposition of SSC actual reliability into condition- and reliability-based components.

Thus, based on the arguments presented above (including the ones indicated in Section 2) an SSC reliability is a probabilistic measure (typically in terms of failure rates or probabilities) regarding the failure occurrence of an SSC (or a population of SSCs operating in similar operating conditions). The set of SSC monitoring activities and associated operations designed to avoid SSC failure (see Figure 1) are means to improve SSC reliability to keep it under the limiting condition expressed as a reliability target.

Once the plant is operating, the SSC reliability target will be compared with the as-is SSC reliability values estimated from observed data (e.g., number of failure occurrences in a time window). Based on such a comparison, the SSC RIM strategy can be changed accordingly.

Lastly, another difficulty with the use of probabilistic metrics as SSC reliability targets is that some SSCs may not be modeled in the PRA when a higher-level PRA satisfies the plant needs. In this case, the PRA model could have details only for functional or system levels with the reliability of SSCs bundled into the system-level reliability estimates.

3.1.1 SSC Reliability Target Allocation

The process of allocating reliability targets involves considering multiple requirements, which can be grouped into two categories:

- Regulatory limits on the risks, frequencies, and radiological consequences of licensing basis events determined based on multiple considerations, including deterministic analyses and evaluations, insights obtained from the PRA models, and defense in depth (DID) aspects.

- Requirements for plant availability and investment protection are defined by the limits on the risks related to the loss of production and loss of assets determined by the plant reliability, availability, and investment protection considerations.

The objective of selecting SSC reliability targets is to establish a benchmark that will be used for evaluating the SSC reliability performance while the plant is operating. An important element is that an SSC RIM strategy (e.g., SSC preventive maintenance, inspection, or periodic replacement) can directly affect SSC reliability. Thus, reliability target allocation can inform system engineers in the design phase on SSC design elements (e.g., materials), SSC inspectability, and SSC replacement or refurbishment schedules. As such, reliability targets ideally should be developed during the plant design phase where the system and SSC design, operational context, and planned RIM strategies are used as inputs to the SSC reliability target allocation process along with the regulatory limits and requirements, such as those represented in Figure 5.

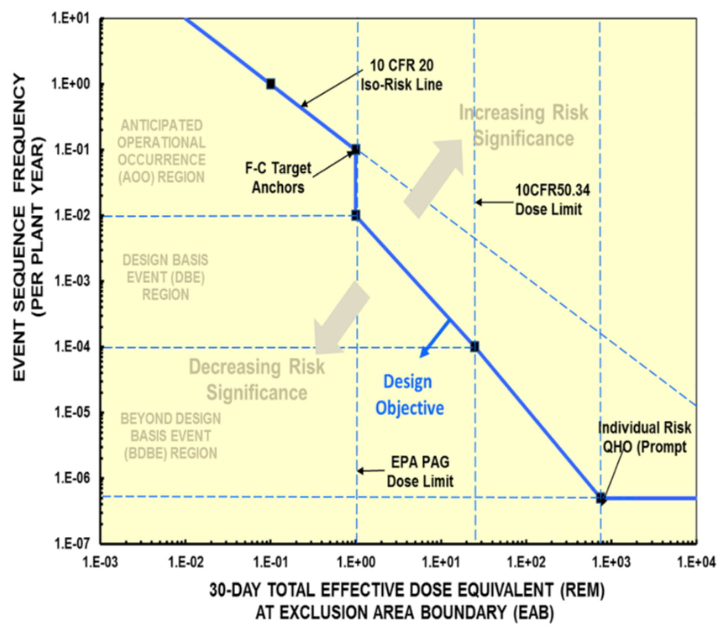

Figure 5. NEI 18-04 frequency vs. consequences (F-C) curve.

Figure 6 provides an overview of the SSC reliability target allocation process in a graphical form where input, output, models, and constraints are explicitly indicated. Note that this figure also highlights how such a process can be iterative when the plant is in the design phase.

In the reliability target allocation process shown in Figure 6, several models can be used to determine specific figures of merit (FOMs) that are compared against regulatory, availability, or investment constraints. These models can be:

- Plant PRA models (e.g., to determine accident event sequences and their associated probabilistic frequencies)
- Plant deterministic models and evaluations designed to assess plant behaviors under accident scenarios (e.g., core radionuclide inventory, safety limits, success criteria)
- Plant architecture models designed to identify plant operational (e.g., environmental conditions for SSC inspections) and safety characteristics (e.g., DID, system redundancy and diversity).

The degree of freedoms in this iterative process (i.e., the input elements) can be:

- SSC design properties (e.g., materials)
- SSC operational context (e.g., type, temperature, and pressure of fluid moving through a piping system)
- Estimated SSC reliability performance based on SSC design and operational context
- Candidate SSC RIM plan (e.g., type and performance of monitoring system, planned SSC lifecycle plan, such as periodic replacement). Here, codes and standards documents provide guidance on possible choices for monitoring and surveillance depending on the operational conditions.

This information is fed to the models to assess specific plant regulatory (e.g., NEI 18-04 frequency vs. consequences curve shown in Figure 5), availability, and investment FOMs that need to be compared to the actual predefined constraints. If such constraints are satisfied, the following information is generated for the considered SSC: SSC performance target(s) that could be expressed as a probabilistic metric (i.e., reliability target) or as a deterministic condition metric and a planned SSC RIM strategy. If the constraints are not satisfied, the degrees of freedom listed above are changed until the constraints are satisfied.

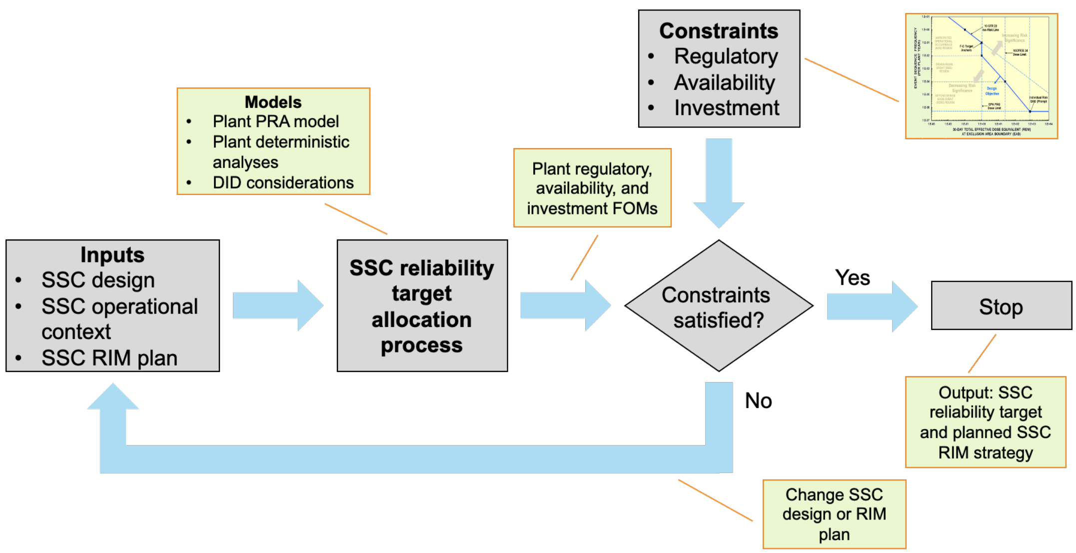

Figure 6. Overview of the SSC reliability target allocation decision process in the plant design phase.

An overview of the steps in the reliability target allocation process, the probabilistic approach, is:

- *Step 1: Plant-level reliability targets, radiation dose limits*. The starting point is the radiation dose limits to the public. The radiation dose limits are the same regardless of reactor design, but compared to LWRs, new designs may have additional requirements for dose limits due to different release sources.
- *Step 2: Plant-level reliability targets, accident scenarios*. The accident scenarios that could lead to a release associated with source terms identified in Step 1 are defined. The accident parameters include SSC failure modes and associated probabilities of failure that may lead to an accident. As the result of this step, frequency of each possible accident scenario is determined.
- *Step 3: System-level reliability targets*. The reactor design must meet the radiation dose regulatory limits with various options to meet the requirement, meaning one plant could accomplish it through a high redundancy of mechanical systems (e.g., three-train system) while another design may accomplish it through a very high reliability of the primary system (e.g.,

passive cooling system) supported by a backup system for DID. The system-level reliability targets are assigned in a way that the plant-level reliability targets remain in the designated licensing basis event category— anticipated operational occurrence (AOO), design basis event (DBE), and beyond DBE (BDBE)—including uncertainty considerations.

- *Step 4: Component-level reliability targets*. This is the step where failure modes and probabilities are defined for each SSC to inform the SSC-level reliability targets. The SSC reliability targets are then input into the system-level reliability targets, and the evaluation is performed to check if the initially set system-level reliability targets are met. If not, SSC-level reliability targets are adjusted (i.e., reliability values are increased or decreased) until the system-level reliability targets are satisfactory.

The above discussion assumes the considered SSC plays a role in any of the model elements indicated in Figure 6; i.e. if an input element associated with the SSC (e.g., SSC design or RIM plan) is changed, at least one output FOM generated by the employed models also changes (e.g., frequency of a given dose equivalent). As an example, the SSC is represented by a set of elements in the plant PRA model (i.e., basic or initiating events); by changing the SSC design or RIM strategy, the values associated with these PRA elements also change, which consequently changes the location of the set of accident sequences in the NEI 18-04 frequency-consequence plot (see Figure 7).

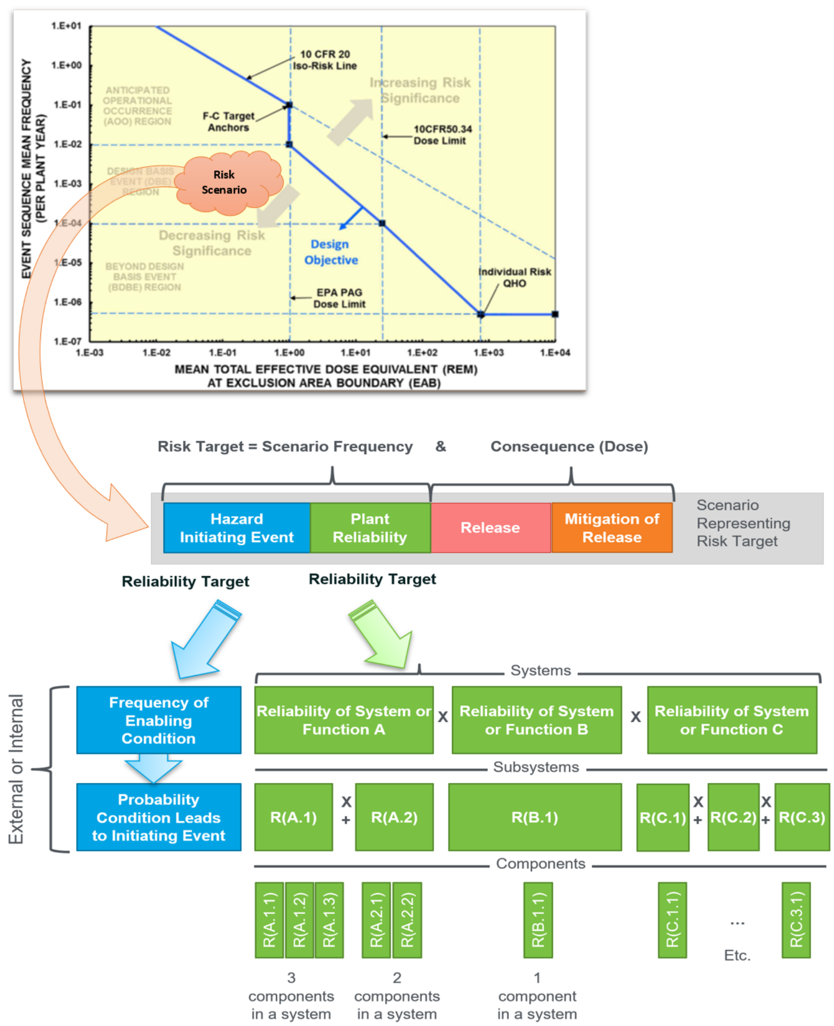

Figure 7. Schematic of reliability target allocation process.

Once the plant is operating, the actual plant performance is observed and recorded (e.g., SSC degradation trends, SSC failure occurrences). Based on this data, an actual SSC reliability value is quantified; if such a value is above the SSC reliability target and such a deviation has an impact toward the plant constraints (e.g., an unacceptable change of the location of the set of accident sequences in the NEI 18-04 frequency-consequence plot—see Figure 5), the SSC RIM strategy needs to be changed accordingly to retain a reliability within the predetermined reliability targets. Such a process is graphically shown in Figure 8. The integration of observed SSC reliability performances into the quantification of the actual SSC reliability value are described in Section 3.1.

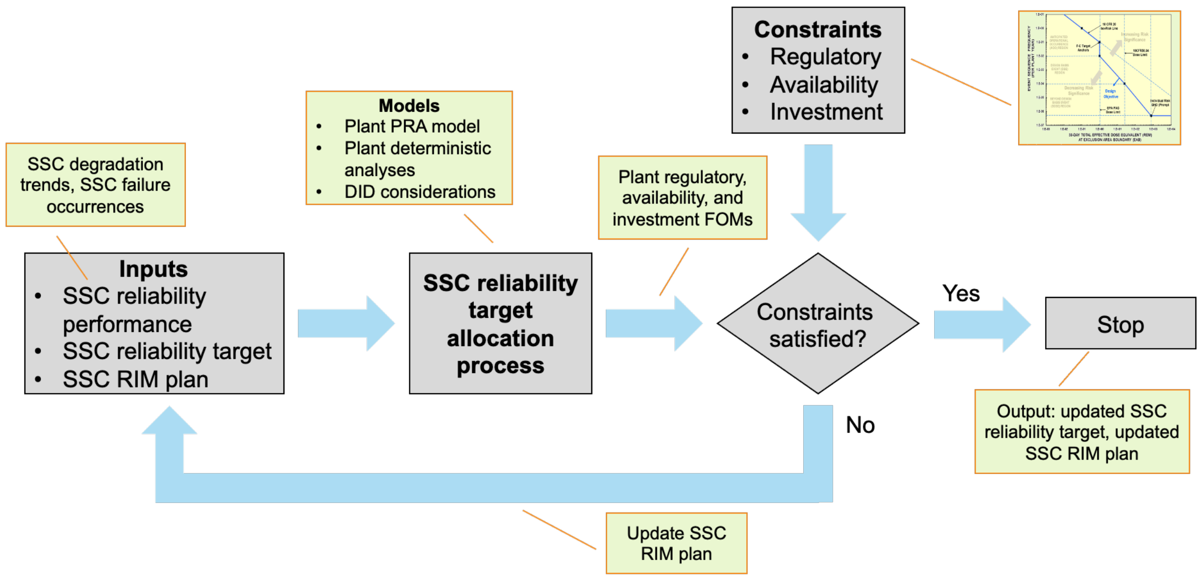

Figure 8. Overview of the SSC reliability target allocation decision process in the plant operational phase.

Note that the considerations about SSC reliability presented so far give the analyst some degree of freedom on the specific elements of the plant SSCs that require an associated reliability target value. More precisely, the analyst is required to define a precise decomposition of the plant into its constituent elements. Figure 9 provides a generic plant architecture decomposition into its constituent SSCs. In particular, note that each system can be decomposed further into its subset of constituent components and structures. At this point, the analyst is required to identify the subset of SSCs that require a reliability target based on safety and risk considerations. For each identified system, the analyst can decide to decompose the system reliability target into reliability targets associated with the components and structures that belong to such a system.

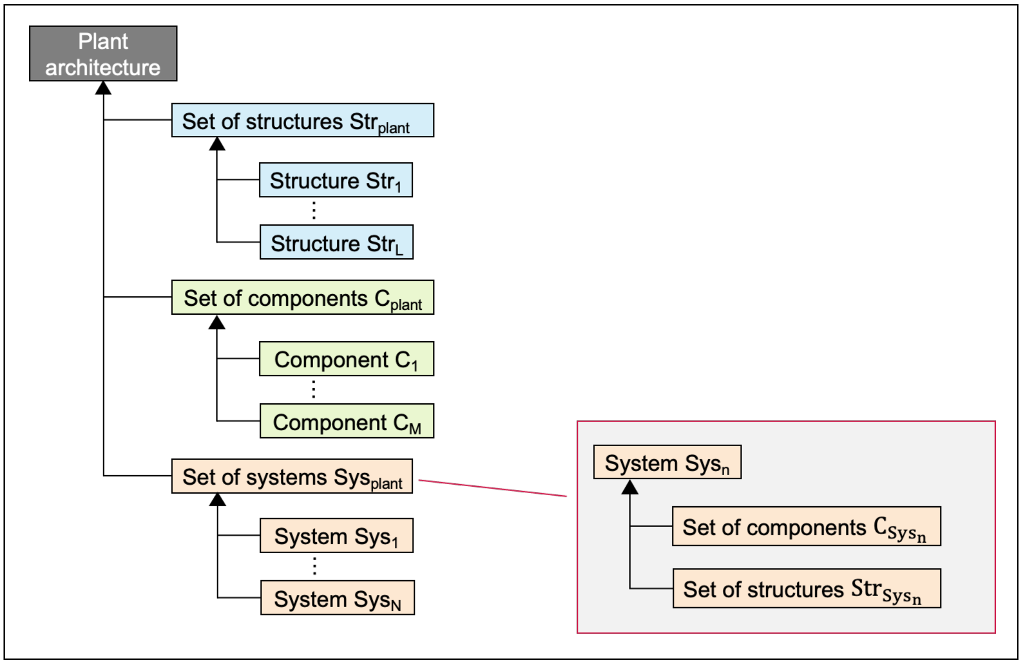

Figure 9. Decomposition of plant architecture into its constituent SSCs with decomposition into a subset of constituent components and structures.

3.2 Component Condition

The condition of an SSC can represent a deterministic performance metric while still demonstrating that the component is meeting its reliability target. An example is when a flaw in a pipe becomes a through-wall flaw and fluid starts leaking from the pipe. For an ISI program developed in accordance with ASME Section XI, Division 1, a limiting condition value for the identified flaw is established as part of the examination as the ISI program requires a periodic examination to preclude through-wall leakage. For a RIM program, an allowable leak rate may be more desirable rather than establishing only a limiting flaw size. The duplicity of monitoring a leak rate is that the assessment of that leak rate correlates it to the crack size that results in the leak rate and identifies that the crack size is still below a limiting crack size value (based on its leak rate) and that the reliability of the SSC is still being maintained (see Section 3.1 for a service water system example for a pipe break size being used as a limiting condition). This example demonstrates that a RIM strategy may deploy monitoring of a leak rate (or absence of a leak) to represent that the reliability of the SSC is still sufficient.

The reliability of SSCs should be considered early in the design stage of the facility. For established designs, historical data informs the designer of the failure rates that have been established for the SSCs. However, for newer designs, reliability data may have more limited availability because similar SSCs have not been in operation long enough to reduce the uncertainty in that data.

Even when limited historical data is available, the use of this data can be improved when considering the type and frequency of maintenance performed because maintenance can affect the reliability and thereby affect the SSC's performance. When the maintainability of an SSC is considered in the design, it can assist in lowering the uncertainty when applying limited historical data with a sufficient understanding of the degradation and failure mechanisms expected during SSC operation.

When the reliability and maintainability of an SSC is correlated to the desired availability of the function that the system and supporting components provide, the effectiveness of that SSC's performance can be determined. This type of analysis is traditionally known as a reliability, availability, and maintainability analysis. When inspectability is singularly identified, it is known as a reliability, availability, maintainability, and inspectability (RAMI) analysis.

The use of a RAMI analysis supports the idea that establishing a performance target associated with the desired availability of a system and its functions can be an effective means of monitoring an SSC's performance and comparing it to a performance goal or target. The RAMI analysis not only assesses unavailability resulting from the failure of a specific system function (i.e., the reliability) it also factors in the planned maintenance, tests, and inspections to be performed on the equipment, which results in additional periods of time that the system is unavailable. In many respects, this may provide a more complete "picture" of the overall availability and, subsequently, the reliability of the system. There is sufficient information and tools available for a further assessment of the conduct of a RAMI and how the individual aspects contribute to the overall health and performance of a system and its functions.

3.3 Performance Monitoring

The objective of performance monitoring is to ensure the SSCs can perform their function(s) by identifying degradation prior to the failure of the function an SSC performs. The activities implemented for identifying and detecting degradation are part of a RIM strategy. Performance monitoring can constitute monitoring for conditions that lead to degradation and the potential of failure or directly identify that degradation is present. For a RIM strategy developed in accordance with ASME Section XI, Division 2, monitoring activities must also be shown, through analysis and/or performance demonstration, to reliably accomplish what the monitoring is intended to identify.

3.3.1 Indirect and Direct Monitoring

Detecting degradation in a system, or component, can be performed either directly or indirectly. Direct detection could often be accomplished through a direct visual examination of a potential degradation condition, such as a through-wall leak, or through ultrasound to detect a flaw within the component's pressure boundary material. These direct means of examination are performed on some periodicity that may be based on a likelihood that the periodicity can detect the onset of a degraded condition prior to that condition resulting in system or component failure.

Monitoring does not require that a component only be directly examined or inspected for a specific degraded condition. Indirect monitoring can provide continuous monitoring that results in an ongoing awareness of the system or component condition that, when a condition is detected, prompts a subsequent action for further evaluation and likely results in the conduct of a direct means of detecting degradation. An example of indirect monitoring is temperature measurement at a location in a system; while temperature measurement does not directly detect any degradation, a specific temperature threshold could indicate the onset of degradation. Continuous monitoring essentially provides an indicator of component health on a continuous basis that can notify the operator of a condition that may indicate an impending failure or undesirable condition that may be missed if only periodic and direct condition monitoring are employed.

3.3.2 Means of Monitoring

LWRs in operation today apply several means of monitoring system and component performance by collecting data for parameters such as pressure, temperature, flow, level, chemistry, and vibration. The sensors' operating environments are well known, and the performance of sensors in these environments has been satisfactorily demonstrated. Advanced reactors are expected to monitor these same parameters. However, in many cases, the operating environments are expected to be harsher with higher operating temperatures and radiation levels and potentially less frequent access to the sensors and instrumentation to perform maintenance that ensures their reliable operation.

Monitoring the performance and conditions of an SSC is dependent on several factors, including understanding the underlying physics of each degradation mechanism so that suitability can be confirmed and ensuring that the monitored parameters are appropriate for identifying the presence and characteristics of degradation being monitored. Many monitoring attributes are addressed during the design stage, especially for SSCs that support reactor control and operation and safety-related systems. For advanced reactors, these same parameters may require higher reliability because the monitoring function it provides serves as a means of monitoring under the RIM program (provided the monitoring meets the requirements for developing a RIM strategy).

3.3.3 Monitoring Means Performance Demonstration

The probability, or likelihood, of detecting the degradation mechanisms, characterizing the degradation mechanisms (e.g., flaw size), and taking actions to correct identified conditions are key attributes in determining the type of performance monitoring to be implemented and have an influence on the ability to demonstrate that the reliability target for an SSC is being met.

An ISI program for current LWRs recognizes that there are different influences on the ability of an examination to effectively detect the presence of a flaw. The ISI programs require a performance demonstration for the equipment, personnel, and procedures used for the examinations being performed. The performance demonstration of identification and characterization techniques is a key attribute for a RIM strategy developed in accordance with ASME Section XI, Division 2, as it is called out as part of the general examination and monitoring requirements and applies the requirement for leakage monitoring.

ASME Section XI, Division 2, Mandatory Appendix IV, "Monitoring and NDE Qualification," provides requirements for qualifying monitoring and nondestructive examination (MANDE) methods. However, it does not provide prescriptive requirements on how the performance demonstration is to be performed. It relies on the use of the MANDE Expert Panel (MANDEEP) to consider the inputs shown in ASME Section XI, Division 2, Figure I-1.1-6, "Qualification Process for MANDE." and establish performance demonstration requirements to ensure that the RIM strategy will support the achievement of an SSC's performance target(s). ASME Section XI, Division 2, includes the responsibilities and qualification requirements for MANDEEP to ensure it has subject matter experts with sufficient knowledge that a prescribed method is not necessary.

Not all monitoring strategies will require a detailed demonstration and justification of their performance, but a basis for why the equipment is an appropriate means for monitoring will need to objectively justify that the use of the monitoring equipment is appropriate and sufficient in achieving the performance target(s) for an SSC. An example is the use of a sodium ionization detector in a molten-salt-cooled reactor to detect sodium in the vicinity, which may indicate a leak of the primary cooling system fluid. When monitoring for leakage as part of a RIM strategy, calibrations and other means of periodic confirmation of an instrument's accuracy supports the basis and demonstration of the reliability for detecting and locating a leak.

4 FACTORS AFFECTING RELIABILITY

Ensuring SSC reliability is not limited to how it is maintained. Decisions made in the design, manufacture and construction, operation, and maintenance of an SSC can affect its ability to provide reliable operation.

To understand how to monitor the condition and performance of an SSC, it is important to understand how these decisions can influence reliability. ASME Section XI, Division 2, recognizes the potential impact and requires (as part of RIM-2.5.1) that a RIM strategy accounts for factors that contribute to reliability.

4.1 SSC Design and Material Selection

As early as the conceptual design, considering the type of degradation mechanisms that an SSC may experience influences decisions on the type of material to be used and ensures access to perform appropriate condition monitoring. A decision on what design and construction code to use can also influence the reliability of an SSC. These factors should be accounted for, and if needed, become design requirements for an SSC to ensure reliable operation. A RIM program developed in accordance with ASME Section XI, Division 2, identifies "design requirements" as part of developing a RIM strategy.

There may be multiple options for the design and material selection of an SSC, and it should not be assumed that the option that results in the highest reliability must be selected. An evaluation of the design and material(s) may show that the more expensive option does not provide a sufficient improvement in reliability to justify the necessary resources or design and construction costs. The important aspect when determining a design or material selection option is to ensure that there is a sufficient basis for the selection and that, if applicable, a design requirement is included in the RIM strategy to ensure that that the option is considered in the future design change related activities.

4.2 Fabrication Procedures

A material's susceptibility to degradation of its material properties, and subsequently to its reliability, can be affected through fabrication practices. A change in susceptibility to degradation may be achieved through specific actions such as post-weld heat treatment or cold working of materials, impact to grain structure when applying manual or automated welding processes, and surface finishing for NDE. The fabrication procedure for an SSC should be evaluated for the resultant impact to its reliability and management of degradation and, if it is a key attribute for managing degradation, becomes part of a RIM strategy.

One additional consideration for including fabrication practices and procedures in a RIM strategy is that this will also need to apply to the fabrication of any test specimens, especially a specimen that will be used for a performance demonstration of MANDE that may be applied to ensure an SSC achieves its performance target(s).

4.3 Operating Practices

Aging and degradation mechanisms are affected by the environment in which an SSC operates. These environments are not only influenced by the design of the plant but its operation (both periodic testing and normal operation) and can degrade an SSC internally as well as externally.

One common means of degradation can come from the fluid being contained by the pressure boundary. The chemistry of the fluid can influence the degradation that the pressure boundary material may encounter; this may simply be an increased or decreased corrosion rate of the internal surface or another relevant condition, such as a fabrication flaw, to initiate or propagate in a different manner compared to the observed processes in LWRs.

If, during the design or as a result of operating experience, a specific operating practice is identified for managing or reducing the potential for degradation, it becomes a part of a RIM strategy and demonstrates how it affects the ability of an SSC to meet its performance target(s). To illustrate this, the RIM strategy can require that an operating procedure includes a requirement to ensure that a flow rate is not exceeded because it was identified that a degradation mechanism can be present due to an undesirable change in temperature that results in exceeding the flow rate condition. This example is not intended to require that all flow restrictions like this become part of the RIM strategy but if there is a sufficient impact on reliability, it is a good practice to identify it in the RIM strategy.

4.4 Preservice and Inservice Inspections

PSI and ISI are standard activities common to maintenance programs in nuclear and non-nuclear industries and is very common to NPPs implementing ISI programs developed in accordance with ASME Section XI, Division 1. PSI activities are conducted during the preoperational phases of the plant to ensure a proper baseline of an SSC's material condition is accurately identified and can be used for the comparison and disposition of conditions identified during operation resulting from ISI-related activities (or other maintenance activities performed independent of the formal ISI program).

Implementing a RIM program developed in accordance with ASME Section XI, Division 2, requires conducting a PSI for examining SSCs that have been selected as part of a RIM strategy. It further requires that all SSCs of a similar category be examined even when only a percentage of SSCs are selected for that examination. This provides for a baseline of all SSCs for use if an expanded sample of SSCs needs to be performed.

ASME Section XI, Division 2 (RIM-2.7.4), requires that the preservice baseline examinations be performed in accordance with the same processes and procedures that are to demonstrate achievement of the SSC performance target(s). It is important that the PSI activity is performed to the same level (i.e., using similar personnel and equipment qualifications) as the ISI because the PSI may record a condition that is acceptable and used as a baseline for future inspections or examinations. If the PSI is not performed at the same qualification level as the ISI, the condition may not be detected and evaluated and subsequently be of little to no benefit when evaluating conditions identified during service.

Although RIM-2.5.1 identifies PSI and ISI as factors that need to be considered in a RIM strategy, the activities contained in the strategy may not be referred to as PSI or ISI. An ISI activity is more likely to be identified as the specific activity being conducted, such as a visual inspection or NDE.

4.5 MANDE and Testing

The ability of an SSC to reliably achieve its function and its performance target(s) is dependent on the ability to identify degradation prior to it resulting in failure of the SSC to perform its function. The ability to detect failure is contingent on the type of inspection, test, or examination being performed and on its frequency. A unique aspect of developing a RIM program in accordance with ASME Section XI, Division 2, is that the program requires demonstrating that the prescribed MANDE and testing[b] activities, including continuous monitoring, supports meeting the SSC performance target(s).

Several attributes of examination and testing can influence the ability to detect degradation (or flaws, being used interchangeably) and, subsequently, to ensure adequate performance. This includes attributes such as the SSC geometry and ability to locate the degradation, accuracy of sizing and probability of detection, the size or extent of the degradation prior to it becoming critical, and selection of an appropriate sample size and periodicity of the test, inspection, or examination.

Each of the cited attributes may impact reliability, and quantifying the impact is technique and situation specific. As is the case for the design and material selection, a strategy that has the highest probability of detection may not always be technically feasible or may result in a significant financial burden for little to no improvement in reliability (e.g., requires unique and challenging system lineups or a plant shutdown outside of normally scheduled shutdown activities).

4.6 Maintenance, Repair, and Replacement Practices

The impact of maintenance practices on SSC reliability is typically well understood by operators of an NPP. Many preventive maintenance programs contain a basis for the type and periodicity of performance of a preventive maintenance activity. Some programs may implement software that determines an expected failure rate for the type and periodicity of preventive maintenance activities being performed. Often, these programs are applicable to the active function of SSCs which are not part of a RIM program.

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[b] The term testing when discussed with the monitoring and NDE activity is related to leak or leakage testing.

Other maintenance practices, such as the control of foreign objects or work area cleanliness, are intended to reduce the chance of introducing a degradation mechanism that would not normally be present if the activity had not occurred. If there is a known degradation mechanism that can be introduced through a maintenance practice or repair and replacement activity, such as chlorides on the external surface of a pressure boundary SSC, and it has been demonstrated to reduce SSC reliability, the RIM strategy can include the exclusion of the material type or highlight the need for a specific maintenance practice.

Similarly, if a corrective maintenance activity is performed, the corrective action, including replacement, are performed in accordance with maintenance programs that institute appropriate verifications and confirmatory actions to reduce the likelihood of increasing the potential for failure. Errors introduced during maintenance can impact reliability of an SSC, however, a specific review during SSC design for each likely situation may not be practical, especially given the lack of operating experience with many of the new equipment types. During operation, a maintenance decision or practice could be identified to specifically reduce occurrence of a credible failure mode and become a part of a RIM strategy.

Repair and replacement activities are considered maintenance activities and if performed on an SSC that is part of a RIM program developed in accordance with ASME Section XI, Division 2, it may be required to follow specific repair and replacement protocols prescribed under ASME Section XI, Division 1. Considerations for repair and replacement may already be covered under another factor and simply re-enforced when developing the repair and replacement activity (and not specifically called out within the RIM strategy as a repair and replacement activity).

5 RIM STRATEGIES TO ASSURE PERFORMANCE

RIM-2.1 provides an overview of the RIM program, which includes in RIM-2.1.1(c) a single summary statement that appropriately describes the objective of a RIM program:

> The RIM program shall select the combination of inspection, monitoring, operation, examinations, tests and maintenance requirements that enable the SSCs to meet its reliability target in an efficient and cost-effective manner.

This “combination” comprises what is referred to as a RIM strategy.

5.1 Identification and Evaluation of RIM Strategies

Figure I-1.1-2, “RIM Program Development and Integration,” from ASME Section XI, Division 2, provides a flowchart representing the key activities in the development of a RIM program. The flowchart identifies the development of RIM strategies following the determination of reliability targets and completion of the degradation mechanism assessment. In addition to these key activities, other factors and inputs influence the development of RIM strategies, including the factors that affect SSC reliability previously described in Section 4.

5.1.1 Scope

Developing a RIM program requires the Owner[c] to specify the scope of SSCs to be included in the RIM program. ASME Section XI, Division 2, does not prescribe what SSCs are within the scope of the RIM program other than to require the inclusion of SSCs whose failure could adversely affect plant safety and reliability.

The Owner is expected to categorize SSCs in accordance with an appropriate categorization process and will likely identify SSCs as either safety-related, non-safety-related with special treatment, or non-safety-related [6]. Owners will provide a basis for the selection of SSCs within the scope of the RIM

[c] The licensee for operation of the nuclear plant is referred to as the ‘Owner’ when applying ASME Section XI, Division 2. Owner will be used throughout the report and is synonymous with the use of the term licensee.

program. Although there is no requirement to include non-safety-related SSCs requiring special treatment, an Owner is required to provide a basis for the exclusion of any SSC from the scope of the RIM program so there will at least be a consideration of these components for inclusion in the RIM program.

5.1.2 Degradation Mechanism Assessment

An obvious step in determining an appropriate approach to monitoring the performance of a component or system is to identify what to monitor. The reliability and availability of an SSC is dependent on the failure modes and degradation mechanisms that challenge the health and performance of the SSC. Therefore, an assessment of the SSC's operating conditions and degradation mechanisms is an integral part of developing a plan for performance monitoring.

The conduct of the degradation mechanism assessment (DMA) can be approached in multiple ways. For commercial NPPs, the most notable approaches are those conducted as part of an aging management assessment to support a license extension in accordance with 10 CFR 50.54 and a similar assessment performed when developing a risk-informed ISI program to meet the requirements of 10 CFR 50.55a.

Details on performing a DMA are provided in Section RIM-2.3 of Article RIM-2. RIM-2.3 requires an application of the appropriate table of the degradation mechanism attributes and attribute criteria contained in Mandatory Appendix VII, "Supplements for Types of Nuclear Plants." These supplements are applied based on a reactor type, and they provide information similar to that applied in a risk-informed ISI program for LWRs.

In addition to the applicable degradation mechanisms contained in the Mandatory Appendix VII (for the applicable plant type), RIM-2.3 also requires the consideration of conditions that affect reliability discussed in Section 4 of this report. These conditions are included to ensure that the Owner considers unique aspects of the plant design and does not only apply the degradation mechanism information contained in the appendix. As an example, when considering the factors of RIM-2.9(3), there may be operating and transient conditions or a service environment that may present a condition not fully encompassed by the attribute criteria in Table VII-X.2-1.[d]

Not all plant types are included within Mandatory Appendix VII, which will prompt owners to develop the DMA using other information and processes. One approach was included in the analyses supporting material qualification for the Kairos Fluoride Salt-Cooled High-Temperature Reactor [8]. In this example, Kairos was applying Alloy 316H as an approved material for the use in high-temperature reactors. This specific qualification activity is not related to the development of a RIM program, but it demonstrates an approach to determine the presence of potential degradation mechanisms. Kairos utilized a Phenomena Identification and Ranking Table methodology to identify environmental degradation issues applicable to the design. In the absence of information, such as that to be included in Appendix VII of ASME Section XI, Division 2, or to support the utilization of information currently present in the same appendix, an approach such as a Phenomena Identification and Ranking Table methodology may provide sufficient information to satisfy the needs and requirements of a DMA.

5.1.3 Design Requirements

Section 4.1 discusses how the design and material selection can contribute to reliability and reflects the requirement from RIM-2.5.1(a)(1) of ASME Section XI, Division 2, which states that the RIM strategy should account for design strategies, including material selection.

ASME Section XI, Division 2, is not a design or construction code, and it does not directly state that the Owner applies this code as part of its design basis. However, in acknowledgement of the importance of the design to the SSC's reliability, development of a RIM program should be started during early

[d] The X is used in this table number to indicate that there may be multiple tables where the X represents multiple subsections.

phases of the design to ensure that cost-effective solutions to ensuring reliable SSC performance are developed.

This is reinforced by the wording of subsection RIM-2.7.4, specifically on design requirements for RIM. The same requirement for the inclusion of design requirements in a RIM strategy is repeated there, but it goes further by bounding the requirement as one that "…may be required to prevent or reduce the susceptibility to degradation mechanisms determined…" during performance of the DMA or "otherwise needed to support a selected RIM strategy." Therefore, to comply with the requirements of RIM-2.5.1 and RIM-2.7.4, it is important that the Owner considers the development of a RIM program early in the design of the facility's SSCs.

The fact that these design attributes become design requirements is important because, as the plant is operated, a design attribute that is important to SSC reliability needs to be managed in a manner similar to other design requirements and ensure that modifications to the SSC are assessed for its impact on reliability.

5.1.4 MANDE and Testing

ASME Section XI, Division 2, builds on the progress the industry has made regarding the understanding of degradation mechanisms and the ability to detect them. Implementing ISI in the current LWR fleet requires a performance demonstration for many of the examination techniques being applied. Although many of the advanced reactors will be using materials operating in temperatures higher than originally qualified for use in LWR nuclear systems, the experience gained in demonstrating these examination techniques can be applied to the conducting of examinations under a RIM program.

This is important because the selection of MANDE and testing performed for a RIM program is influenced by its contribution to meeting the reliability target for the SSC being monitored. This means that the monitoring technique, such as an ultrasonic examination, will have a likelihood of detection based on experience, qualification of the examiner and equipment, and periodicity of the examination. The influence is continued through the effectiveness of characterizing the degradation (e.g., flaw size, orientation, and growth) to accurately predict its behavior and determine the correct actions to take to address the condition.

The demonstration of the ability to detect degradation is not limited to only the examination techniques traditionally applied by an ISI program developed in accordance with ASME Section XI, Division 1. Article IV of ASME Section XI, Division 2, provides requirements for the performance qualification of both monitoring and MANDE methods and techniques. One requirement that stands out is that the qualification of monitoring methods and techniques requires approval from the MANDEEP for procedures, techniques, and equipment that are used for the monitoring techniques. Therefore, the Owner must have documentation that shows the evaluation and analysis for the use of the monitoring equipment (including equipment installed in the plant) that shows how it was considered in the overall reliability of the SSC that it monitors.

Take for example, a RIM strategy that requires continuous monitoring for system leakage. Several factors must be analyzed and possibly supported by performance demonstration, which could include test documentation on the accuracy and ability of the monitoring system to perform in the environment the system operates. This may require a set of analyses designed to show that when an instrument or sensor is out of service, the accuracy and location of remaining instruments or sensors ensure that leakage can be detected and continue to meet the reliability target.

With the expected longer operating time frame before refueling, or other planned maintenance, can be performed, there may be a challenge with ensuring the instruments and sensors are operating within their expected range and require less frequent calibration. The LWR industry has already identified the benefit of reducing the frequency of instrument calibration and is demonstrating proper operation through techniques such as online monitoring, which can reduce the need for periodic and intrusive maintenance activities [9] for instruments being used to control processes. The same approach can be applied in a RIM strategy.

5.1.5 Strategy Selection

ASME Section XI, Division 2, clearly identifies many attributes that are considered when developing a RIM strategy. Understanding the strengths and weaknesses for each attribute is important because the value, or contribution, of each attribute may be more effectively addressed by another attribute, and it may be more cost-effective to limit focus to only one, or some, of the attributes, especially when the change in risk is similar. In other words, focusing resources on more important contributors is important to reliability.

Developing a RIM program, either in accordance with ASME Section XI, Division 2, or by other means, focuses on SSCs that contribute to the safe and reliable operation of the plant. Attention may be placed on safe operation, but cost-effectiveness plays a role because it is necessary to focus resources in the most effective manner. This is why strengths and weaknesses for each method of monitoring and NDE are evaluated rather than attempting to fully implement all possible means of monitoring and NDE.

Consider accessibility as an example. For today's LWRs, many SSCs are only accessible when the system or the plant is shut down. Advanced reactors are anticipated to have longer periods between plant shutdowns, which drives some potential MANDE and testing activities to be performed less frequently. The evaluation of the MANDE or test to be performed should include a determination of the potential for a flaw to occur and grow during the time between MANDE or test performance. Coupled with the probability of detecting a flawed condition, the evaluation may conclude that the periodicity of tests that aligns with plant shutdowns is appropriate.

The evaluation of MANDE and testing may indicate a need to incorporate monitoring capabilities that provide a more frequent check for the conditions that could lead to degradation rather than a direct examination. This could result in a need to incorporate into the design a means of indirect monitoring for conditions (e.g., temperature or vibration) that may indicate an increased potential for degradation (or absence of the condition precludes a need for increased attention). Additionally, it could be determined that a material change could nearly eliminate a degradation mechanism and that, upon further investigation, the RIM strategy could be more cost-effective over the life of the plant and achieve or exceed a performance target(s) by maintaining the selected material in the design and ensure appropriate access is provided for certain direct monitoring activities.

An SSC could have many approaches as part of a RIM strategy and multiple approaches to achieve the desired performance. Therefore, it is not appropriate to bound a RIM strategy by developing a typical approach.

5.2 Correlation of RIM Strategies to Component Performance

The objective of a RIM strategy is to conduct appropriate monitoring of an SSC's condition to support achieving its performance target(s). When developing a RIM program in accordance with ASME Section XI, Division 2, an initial RIM strategy undergoes several prescribed processes, most notably, activities that ensure the prescribed MANDE and testing will satisfy adequate performance.

An initial challenge with correlating a component's performance with a reliability target expressed in probabilistic terms is that the performance monitoring of the component entails an observation of a physical condition, which may not directly confirm that the reliability target is being achieved. The term "reliability target" is used throughout the main text body of ASME Section XI, Division 2, but it isn't until Mandatory Appendix II on the derivation of component reliability targets where the term "component-level requirement (CLR)" is introduced. Although ASME Section XI, Division 2, requires that the CLR be derived in the form of a reliability target, the CLR expressed in deterministic terms, like SSC condition, may be more readily aligned with an observable condition and may be a more useful term in establishing the RIM strategy. The RIM strategy will also include the basis for monitoring for an observable condition, and meeting the CLR by default achieves the reliability target and is a value that can be reviewed and approved by the RIM Expert Panel.

To compound the challenge, observing an unexpected component condition may not necessarily indicate improper performance, meaning that interim criteria should be established for the Owner to act on without necessitating a heightened level that prompts a regulatory response (sometimes referred to as an alert level).

To illustrate this issue, a RIM strategy may establish an observation or examination for a material condition (which is represented in Figure 10 in terms of crack size). The expected crack size (an observable condition) can be expressed as a function of time. The bridge between observed data and SSC reliability is provided in terms of load vs. capacity (see Section 2.2) where deterministic and probabilistic considerations are used to assess how likely it is that the load will exceed capacity. Note that such considerations can also be framed in terms of margins, which also highlights that safety margins (e.g., imposed by codes or standards) might provide further SSC reliability assurance.

From a RIM-related point of view, decision makers might be interested in assessing how degradation phenomena evolve with time and comparing it with past operational data. When an observed condition does not follow trends based on past operational data, such a condition is simply unexpected, and unless it is at a level of unacceptable degradation, represented by the RIM target in the observable space, the Owner must first act to further understand the basis for being outside of the expected range and plan for recovery strategies (e.g., a reduced SSC lifetime).

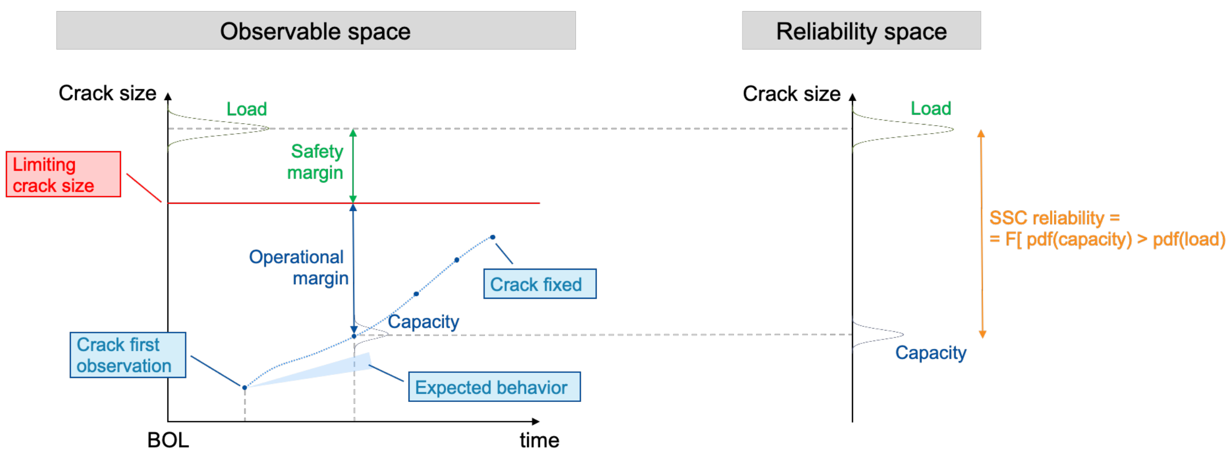

Figure 10. SSC reliability assessment based on observable data (i.e., crack size).

Figure 10 supports the fact that an observable condition may not be a safety consideration if it is outside of the expected condition. This is important because a new reactor design may have limited operational experience and could have early operating data where the condition is outside of the expected region, including conditions below the expected degradation. This potential may be recognized by the RIM program by conducting the monitoring activity on a more frequent basis early in plant operation and then less frequently as the observed condition is confirmed to be as, or better than, expected.

Note that the data provided by a monitoring system is often affected by uncertainties, which may have an impact on the actual reliability value that is being compared to the target reliability. In this respect, data uncertainties in the measurement system (represented as a pdf) can be factored into the decision process using a load vs. capacity argument.

The scenario is different when the monitoring system can detect the presence of a flaw but is not able to characterize it (e.g., its three-dimensional size). Such data cannot be represented as a data point and would not provide enough knowledge to precisely pinpoint an informed RIM strategy. In such cases, the best engineering practices should be followed.

The risk-informed ISI program for current LWRs incorporates many of the same monitoring attributes that will be applied in the RIM program for an advanced reactor. As discussed previously, advanced reactor designs may have limited accessibility to many locations where direct examinations (e.g., ultrasonic testing or visual examinations) are necessary. Therefore, advanced reactors are expected to rely more on indirect monitoring for system parameters that lead to degradation and, to the extent practical, may include more advanced continuous monitoring techniques.

As introduced in Section 2, two monitoring approaches are considered, monitoring for conditions that may indicate the presence of a degradation mechanism (indirect monitoring) and monitoring to detect the degradation mechanism (direct monitoring). Both approaches can be applied for the same component and often complement each other. As an example, an analysis may identify that the presence of a degradation mechanism is highly unlikely and support a RIM strategy based solely on continuous monitoring for conditions that indicate the potential for a degradation mechanism (i.e., differential temperatures or pressure surges). Achieving the reliability target may also require periodic NDE. In all cases, the RIM strategy must include a basis demonstrating that continuous monitoring and/or NDE achieve the performance target(s).

The basis for how a monitoring approach ensures adequate performance must address the ability to detect degradation. The ability to detect degradation is dependent on the attributes for monitoring. Instrumentation and sensors used for continuous monitoring require demonstrating equipment accuracy as well as demonstrating that monitoring location(s) are appropriate, and the sample size is sufficient to detect degradation and would allow the Owner to determine the next action based on the identified condition.

As mentioned earlier, new reactors could operate in novel environments potentially using novel materials, which can be a challenge for the technologies available for sensors and monitoring capabilities. Current technologies are believed to be adequate; however, the industry has been performing research to further develop and qualify sensors and instruments for use with advanced reactor technologies. One research program of note is through the U.S. Department of Energy's Advanced Sensors & Instrumentation Program [10]. One goal of this program is the provision of new capabilities for the measurement, control, and operation of current and future nuclear reactors through the development of sensors designed for the anticipated harsh environments.

5.3 Correlation of SSC Performance to SSC Reliability

As indicated in Figure 1, both direct and indirect monitoring elements can provide quantitative indications of SSC degradation. Such indications can then be used to update the SSC RIM strategy (e.g., increase surveillance frequency) or inform specific corrective actions (e.g., fix or replace SSC). The challenge now is to quantify SSC reliability under a RIM program where monitoring data is available. The data provides indications about SSC form and functional performances and is used to perform specific actions informed by collected and analyzed data.

Again, SSC reliability is a probabilistic measure of the occurrence of an undesired event: the SSC loss of function. To understand the meaning of SSC reliability in a RIM context, let's consider two extreme situations:

- No knowledge about SSC performance (i.e., no monitoring data). In this scenario, SSC reliability is a probabilistic measure calculated based on observed historical tested performance. The reliability metric is associated with the physical phenomena that cause the occurrence of an undesired event.

- Complete knowledge about SSC performance. This hypothetical scenario assumes that available monitoring data provides a continuous and complete understanding of the degradation phenomena that leads to the occurrence of an undesired event. In addition, it is also assumed that the SSC can be rapidly restored (compared to degradation phenomena progression rate) once degradation is observed. Under these circumstances, an undesired SSC event cannot occur (i.e., SSC reliability is 1.0).

Because these are extremes, it is recognized that an SSC subject to a RIM strategy will have some monitoring data (not precise, nor complete) that shows that SSC performance lies between the two extreme situations listed above and will have a direct impact on how to model SSC reliability. As indicated in Section 2.2, the combination of monitoring data, decisions, and actions implies that SSC degradation has a deterministic (also referred as performance-based) aspect while still having a probabilistic aspect to its performance. In fact, when degradation phenomena are observed, characterized, and tracked, physics-based considerations (through experimental and past operational data) can be used to assess the evolution of such degradation phenomena and evaluate the likelihood of SSC failure based on the previously introduced load vs. capacity arguments. Based on this knowledge, appropriate proactive actions are taken to avoid SSC failure.

Thus, from these considerations, causes of SSC failure can be decomposed into two categories: not being able to detect SSC degradation and not being able to timely fix and restore SSC performances[e]. The first category is based solely on aleatory elements and combines the fact that there are physical limitations of the available monitoring system (e.g., probability of detection), data are collected periodically, and uncertainties associated with data are present. The second combines physics-based considerations (i.e., performance) that are informed by available monitoring data with the actual SSC RIM strategy (e.g., time to take corrective action), which might be aleatory in nature (see Figure 11).

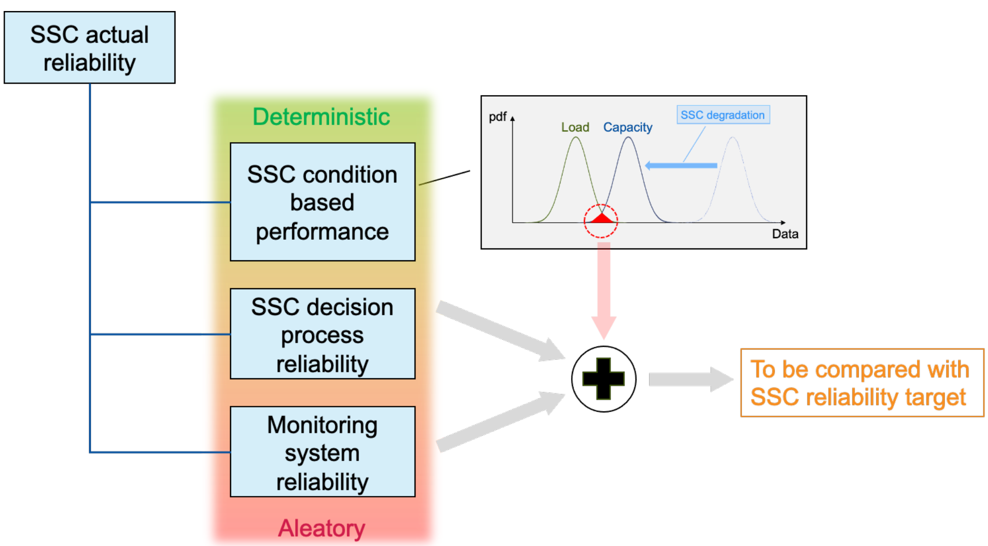

Figure 11. Decomposition of SSC reliability: SSC performance informed by available monitoring data (deterministic in nature), reliability associated with monitoring system (aleatory in nature), and reliability associated with the decision process (deterministic and aleatory in nature).

The following presents three common SSC operational scenarios that describe the elements that affect actual SSC reliability. The common theme throughout is that SSC reliability is a measure that balances SSC performance (i.e., data) and decisions. Consider the scenarios where SSC performance is not

[e] The assumption here is that once a flaw (or a leak) is detected, plant personnel will not deliberately lead the SSC to the undesired event.

monitored (Section 5.3.1), SSC is monitored through periodic surveillance inspections (Section 5.3.2), and SSC is subject to continuous monitoring (Section 5.3.3).

5.3.1 SSC Without Monitoring or Inspection Programs

This first scenario considers an SSC where neither monitoring system nor surveillance programs are used. Thus, once the undesired event occurs (e.g., SSC failure), corrective actions are performed (see Figure 12). SSC reliability is a measure of the occurrence of an undesired event, which is considered completely aleatory in nature. The rate of occurrence of the undesired event is estimated using historical or experimental failure data. In addition, Bayesian updating methods can be performed while the SSC is operating and operating data (e.g., number of undesired event occurrences through the SSC life) is available.

Rate of occurrence of the undesired event is then compared to the actual SSC reliability target; if actual SSC reliability is below the predetermined reliability target (e.g., failures occur too frequently), the SSC RIM strategy needs to be updated (e.g., add monitoring strategy, see Section 5.3.2 and 5.3.3)

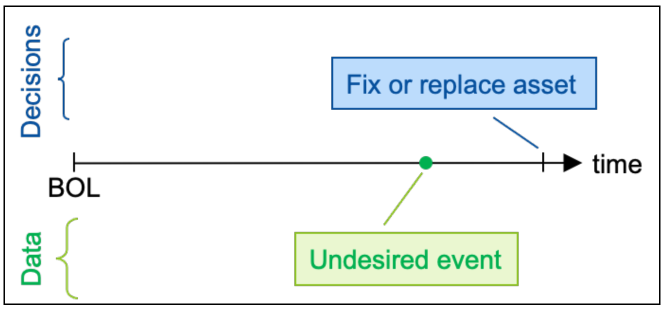

Figure 12. Temporal relations between data and decisions for an SSC where monitoring and inspection programs are not in place.

5.3.2 SSC Under Periodic Inspections

Consider a scenario where an SSC is periodically examined to identify flaws at a fixed interval set as ΔT, (see Figure 13). Once a flaw is detected, its progression is monitored at either the same frequency (i.e., with ΔT as the time interval between inspections) or at an increased frequency. Once degradation progression reaches predefined limits, corrective actions (i.e., the SSC is either fixed or replaced) are taken before the failure. As indicated in Section 2.2, the occurrence of such undesired events can occur if either the flaw is not detected, or corrective actions are not taken in a timely manner prior to event occurrence. Table 1 presents the information used for modeling from a reliability perspective. Note that the set of listed variables are associated with flaw emergence, physical phenomena behind degradation progression, and actual SSC RIM program. Regarding the phenomena behind degradation progression, such knowledge is mostly physics-based using experimental and past operational data to quantify the temporal evolution of a flaw-to-failure process. Advanced modeling techniques based on fracture mechanics are now available as well for a more accurate modeling of degradation progression over time. As an example, an article in Sensors, "A Crack Propagation Method for Pipelines with Interacting Corrosion and Crack Defects," [11] provides indications about crack propagation in piping systems in corroding environments.

Given the information provided in Table 1, the SSC reliability under periodic inspections can be presented in terms of $Prob(event)$, which indicates the probability[f] associated with the occurrence of the SSC undesired event; this term can be computed as:

[f] Note that the occurrence of the SSC undesired event can be framed in terms of either probability or rate.

$$Prob(event) = Prob(no\_detection) + P(no\_timely\_reaction|flaw\_detected) = \\ = f(r_{flaw}, T_{flaw}, POD, \Delta T) + g(\Delta T, T_{corr}, T_{flaw}) \quad (1)$$

where the $f(.)$ and $g(.)$ functions indicate the probability that the flaw is not detected (i.e., $Prob(no\_detection)$) and the probability that corrective actions are not performed prior to failure of the SSC given that the flaw is detected (i.e., $P(no\_timely\_reaction|flaw\_detected)$).

At this point, $Prob(event)$ is compared to the reliability target; if $Prob(event)$ is above the target, the SSC RIM strategy needs to be updated. Two possible courses of action can be followed: increase the frequency of inspection (i.e., reduce ΔT) or the improve the monitoring system probability of detection (POD) to allow for a more precise calculation of the probability of failure. If these changes in the RIM strategy result in a probability of failure that is still above the target, actions may be needed for the SSC form (i.e., its physical condition), such as repair or replacement.

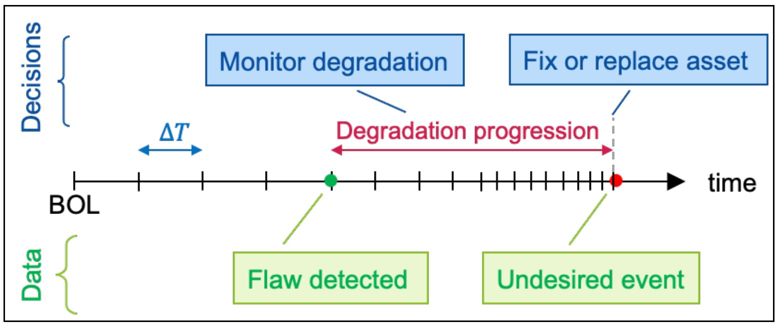

Figure 13. Temporal relations between data and decisions for an SSC under periodic inspection.

Table 1. Reliability modeling parameters associated with an SSC under periodic inspection.

Symbol	Name	Reliability Modeling Notes
r_{flaw}	Probabilistic rate of flaw occurrence	The flaw occurrence in an SSC is considered aleatory in nature
T_{flaw}	Flaw temporal progression	Estimated time for a flaw to cause the undesired event
ΔT	Frequency of periodic inspections	Inspection frequency originally set in the SSC RIM program
POD	Probability of flaw detection of monitoring system	The probability of flaw detection is considered aleatory in nature and the estimated performance of detection system based on past operational experience and testing
T_{corr}	Time required to take appropriate corrective actions	Time specified in current SSC RIM program and estimated based on past operational experience

5.3.3 SSC Under Continuous Monitoring

Figure 14 presents a scenario where SSC performance is monitored continuously. Once a flaw is detected, its progression continues to be monitored until the degradation progression reaches predefined limits; at this point, corrective actions to repair or replace the SSC are taken before the failure occurs. As indicated in Section 2.2, such undesired events can occur if either the result of a flaw is not detected or if timely corrective actions are not taken before the event occurrence.

Given the information provided in Table 2, the SSC reliability under continuous monitoring is here indicated as $Prob(event)$, which indicates the probability associated with the occurrence of the SSC undesired event; this term can be computed as:

$$Prob(event) = Prob(no\_detection) + P(no\_timely\_reaction|flaw\_detected) = \quad (2)$$
$$= h(r_{flaw}, T_{flaw}, POD, U_{monit}) + k(U_{monit}, T_{corr}, T_{flaw})$$

where the $h(.)$ and $k(.)$ functions indicate the probability that the result of a flaw is not detected (i.e., $Prob(no\_detection)$) and the probability that corrective actions are not be performed before SSC failure given that the flaw is detected (i.e., $P(no\_timely\_reaction|flaw\_detected)$).

At this point, $Prob(event)$ is compared to the target; if $Prob(event)$ is above the target, the SSC RIM strategy needs to be updated. Two possible courses of action can be followed: improve monitoring performance (i.e., POD) or improve monitoring system unavailability to allow for a more precise calculation of the probability of failure. If these changes in the RIM strategy result in a probability of failure that is still above the target, actions may be needed for the SSC form (i.e., its physical condition), such as repair or replacement.

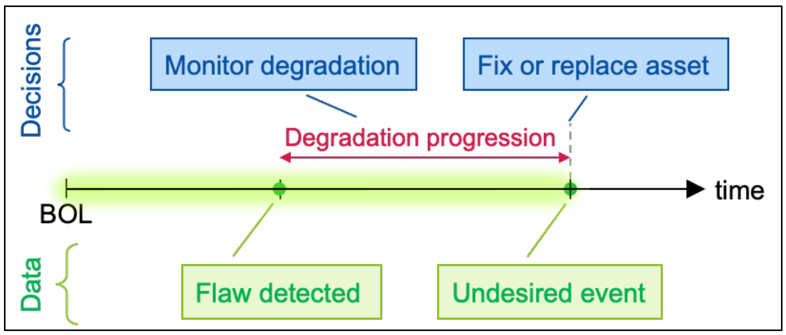

Figure 14. Temporal relations between data and decisions for an SSC under continuous monitoring.

Table 2. Reliability modeling parameters associated with an SSC under continuous monitoring.

Symbol	Name	Reliability Modeling Notes
r_{flaw}	Probabilistic rate of flaw occurrence	The occurrence of flaw in an SSC is considered aleatory in nature
T_{flaw}	Flaw temporal progression	Estimated time for a flaw to cause the undesired event
U_{monit}	Unavailability of monitoring system	This term includes unavailability caused by sensors and instrumentation and control system failure
POD	Probability of flaw detection of monitoring system	The probability of flaw detection is considered aleatory in nature and the estimated performance of detection system based on past operational experience and testing
T_{corr}	Time required to take appropriate corrective actions	Time specified in current SSC RIM program and estimated based on past operational experience

5.4 Adequacy of RIM Strategies

Assume that the plant is operating and utilizing continuous monitoring and periodic surveillances are providing performance information about several SSCs. The goal now is to validate the SSC RIM strategy by integrating the monitoring data into SSC reliability models and comparing the actual SSC reliability value with the SSC reliability target specified in the SSC RIM strategy.

RIM-2.5.2, “Evaluation of RIM Strategy Impacts on SSC Reliability,” in part specifically requires the Owner to assess the impact of each strategy on the reliability of each SSC in the scope of the program. These are further explored in the following subsections.

5.4.1 Assessment of SSC Reliability Methods

Data analytical methods are employed to process and integrate observed failure data (e.g., failure data of SSC operating in similar operating conditions), data generated by physics-based models (either deterministic or probabilistic), and data collected from expert opinion. Assumptions behind data sources and analytical methods need to be assessed and validated based on actual SSC operating conditions.

5.4.2 Assessment of SSC Reliability Values

The process of integrating monitoring data into SSC reliability models can be performed using classical or Bayesian statistical analysis techniques. As an example, from the outcomes (i.e., number of flaws detected) observed from the NDE inspections performed throughout the SSC lifetime, the rate of occurrence of flaws (per weld or per meter of piping) can be estimated. Similarly, the POD of the NDE inspection can be estimated by counting the number of detected leaks (a leak can be considered an undetected flaw) over the number of performed NDE inspections.

Once the data has been used to “update” the SSC reliability model, the newly calculated SSC reliability value is compared with the actual SSC reliability target. If the SSC reliability value is below its corresponding reliability target, the SSC RIM strategy elements need to be changed such that the SSC reliability value is kept above the reliability target. The degrees of freedom here are the deterministic parameters related to the RIM strategy, such as frequency of NDE inspections or POD.

5.4.3 Assessment of SSC RIM Strategy Reliability

A RIM strategy is based on several factors (e.g., failure data, expert opinion, SSC operating conditions) that might be affected by uncertainties or other elements that might vary during the SSC lifecycle. Hence, the impact of such uncertainties and variability on the SSC RIM strategy needs to be assessed.

5.4.4 Assessment of SSC RIM Strategy Performance

Here the performance of the employed RIM strategy is evaluated and compared to the performance initially estimated in the design phase. Examples of RIM strategy performance include:

- POD of flaw monitoring system
- Time to detect a leak in the containment
- Fraction of SSC inspected per surveillance instance
- Flaw characterization capabilities of flaw monitoring system
- Ability to access SSC for inspection
- Ability to identify abnormal behaviors of SSC based on current operational data.

6 EXAMPLE APPLICATION – LWR CASE STUDY

There may be multiple approaches in the development of a RIM strategy and the provision of an example case can provide an effective understanding of what an Owner may present during development of its RIM program and associated RIM strategies. To demonstrate the process in the development of a RIM strategy two example cases are provided in this report. The first is an LWR example case that utilizes well-known and readily available information for an LWR[g]. Specifically, the reactor coolant system (RCS) for a pressurized-water reactor (PWR) design is used.

It is recognized that ASME Section XI, Division 2, has not been endorsed for use by LWRs. This example case is meant for demonstrating the process of developing a RIM strategy as part of a RIM program that is developed in accordance with ASME Section XI, Division 2. Use of this example does not imply that this is an NRC approved RIM strategy for application to an LWR system and does not suggest that the operating reactors should change the current ISI practices.

6.1 Degradation Mechanism Assessment

When looking at the types of degradation mechanisms experienced at a PWR, loop 4 of a four-loop PWR provides a good example because of the different geometries and degradation mechanisms that result from the connection to the pressurizer. Information from a risk-informed ISI program for a PWR is used in the example case as it provides credible degradation mechanism information and precludes the need to formally conduct this part of the example.

The RCS piping included in this example case is ASME Class 1 and SS 300 stainless steel (300 series). This includes the main loop piping and lines connected to the pressurizer. Through the connection to the RCS hot leg, RCS pressure is controlled using the pressurizer where steam is formed through use of heaters (expansion) or condensed through the use of spray (contraction). In addition to the lines used for adding the sprays, there are lines connecting relief and safety valves to the pressurizer.

When performing the DMA, piping and welds are evaluated for both normal and transient conditions, including conditions experienced during plant heat up, cool down, and shutdown cooling. Emergency or upset transient conditions are not part of the evaluation.

For this example, results are similar to the conduct of a DMA performed for a risk-informed ISI program. As stated in Section 2.4 of this report, the user is expected to consider the attributes that contribute to reliability in addition to applying the attributes in Table VII-1.2-1, “Degradation Mechanism Attributes and Attribute Criteria (LWR),” of ASME Section XI, Division 2. Details supporting the application of each degradation mechanism are discussed below.

6.1.1 Thermal Stratification, Cycling, and Striping (TASCS)

During plant startup or shutdown, a temperature difference is often experienced between the water in the pressurizer and the RCS hot leg temperature. With certain flow rates, this temperature difference is sufficient to result in low flow thermal stratification TASCS in the pressurizer surge line due to insurges and outsurges during plant heatup and cooldown. For the example case, TASCS is identified for the pressurizer surge line.

For the pressurizer spray line, a bypass flow is usually present to prevent thermal shock in the spray piping during spray operations. The mixing of this bypass flow with the pressurizer steam in horizontal lines near the spray nozzle can result in TASCS (low flow and two-phase flow). For this example, the nozzle and horizontal regions of the spray line are identified as being susceptible to TASCS.

For the pressurizer surge line, the potential for TASCS extends to the RCS connection in the loop 4 hot leg. The residual heat removal (RHR) system takes suction from the RCS Loop 4 hot leg, so it is also evaluated for susceptibility to TASCS. For this example, the RHR system is placed into service controlling the system parameters that precludes the susceptibility to TASCS in this line.

[g] Although ASME Section XI, Division 2, applies to LWRs, its use for LWRs has not been endorsed by the NRC in the development of a RIM program in LWRs.

6.1.2 Thermal Transient (TT)

Similar to the potential for TASCS in the pressurizer surge line, the temperature differences between the pressurizer and RCS hot leg, in combination with higher potential flow rates experienced during plant cooldown (much higher than that required for the previously described TASCS), the pressurizer surge line has the potential for thermal-transient-driven degradation.

For this example, other lines such as drains and pressurizer spray do not have sufficient flow rates (much higher than that required for the previously described TASCS) and resultant mixing to be susceptible to TT. As previously discussed for TASCS in the RHR suction line, the line is also not susceptible to TT for this example.

6.1.3 Stress Corrosion Cracking

Primary water chemistry control is usually sufficient to limit the presence of oxygen, halides, and corrosion-initiating contaminants and exclude the need to monitor for intergranular stress corrosion cracking and transgranular stress corrosion cracking in the RCS. Additionally, controls are assumed to be sufficient to preclude the presence of external chloride corrosion cracking.

Many of the operating PWRs have evaluated the RCS for the potential susceptibility to primary water stress corrosion cracking (PWSCC). Often, the nozzle-to-safe end welds are found to be susceptible to PWSCC. Therefore, for this example, the nozzle-to-safe end welds for nozzles in the pressurizer surge line, spray lines, and lines to the pressurizer safety and relief valves will include the susceptibility to PWSCC.

6.1.4 Localized Corrosion and Flow Sensitive DMs

In some instances where flow is stagnant and subject temperatures may result in a potential susceptibility to micro-biologically induced corrosion; however, this is not a common degradation mechanism in the RCS that an ISI program monitors for. Other corrosion mechanisms, such as pitting and crevice corrosion, are a potential degradation mechanism in the RCS when oxidizing species and other contaminants (e.g., fluorides and chlorides) are present especially when regions of low flow and geometries such as thermal sleeves are present. Localized corrosion may be addressed through chemistry control and monitoring. Therefore, for this example, localized corrosion is an expected degradation mechanism for monitoring by a RIM program.

A PWR RCS does not usually have the potential for cavitation sources that makes it susceptible for erosion-cavitation degradation nor are there regions of two-phase flow that require monitoring for flow-accelerated corrosion. Therefore, for this example, the RCS is not susceptible to these degradation mechanisms.

The locations selected for this example case and the resultant degradation mechanisms can be summarized in Table 3.

Table 3. DMA information.

Description	NPS	Degradation Mechanism	Possible Monitoring
Pressurizer nozzles to safe ends—Surge line, relief and safety valves	4, 6, 14	PWSCC at nozzle to safe ends. Crack initiation from inner surface (expected) at welds and heat-affected zones and areas of stress concentration. Crack growth can be relatively fast and may progress to through the wall within a 40-month inspection period.	Periodic ultrasonic examination—crack detection and characterization. Continuous monitoring—monitoring RCS temperature.

Description	NPS	Degradation Mechanism	Possible Monitoring
			Continuous monitoring—indications of leakage.
Pressurizer spray lines	4, 6	TASCS in the spray lines near the spray nozzles (potential for low flow when bypass spray is secured). Crack initiation from inner surface (expected) at welds and heat-affected zones and areas of stress concentration. Crack growth is slow and not expected to progress to through the wall within a 40-month inspection period. Locations include nozzles but can occur over extensive portions of the pipe.	Periodic ultrasonic examination—crack detection and characterization, including a specified distance down the length of piping. Continuous monitoring—indications of leakage.
Pressurizer surge line	14	TASCS due to potential for low flow in this line, including welds from line to RCS, and connection to the RCS during heat up and cooldown surges. This line is also susceptible to TT as cold fluid might be injected into hot piping or hot fluid into colder piping during surges. Crack initiation from inner surface (expected) at welds and heat-affected zones and areas of stress concentration. Crack growth is slow and not expected to progress to through the wall within a 40-month inspection period. Locations include nozzles but can occur over extensive portions of the pipe.	Periodic ultrasonic examination—crack detection and characterization, including a specified distance down the length of piping. Continuous monitoring—monitor temperature along line. Continuous monitoring—indications of leakage.
RCS	NA	There is potential for pitting and crevice corrosion throughout the RCS when oxidizing species or other contaminants exist.	Primary Water Chemistry Control Program will monitor the RCS for presence of oxidizing species and contaminants that contribute to corrosion as applicable.
RHR Suction line	10	None identified	Continuous monitoring—indications of leakage.

6.2 Reliability Target Allocation

As indicated in Section 3.1.1, the allocation of the reliability target is influenced by several factors (e.g., safety, risk, economics). For this specific case, a leak in the pressurizer surge line is a contributor to the initiating event "small-break loss-of-cooling accident (LOCA)." Such an initiating event is typically modeled within the plant PRA with its own event tree, which generates its own set of accident sequences that would be mapped as points in the frequency and consequence plot (see Figure 5). The probabilistic rate of occurrence associated with this initiating event (along with the reliability value of the safety systems queried to place the plant in a safe state or mitigate the radiological consequences) would contribute to the plant overall risk expressed as the frequency value (i.e., along the y-axis in the frequency and consequence plot, see Figure 5). Based on the plant acceptable risk, the analyst can determine the target locations of the accidence sequences in the frequency and consequence plot and, consequently, set the target rate of occurrence of the considered set of initiating events, such as the "small-break LOCA."

6.3 RIM Strategy Development

Although the designs of advanced reactors are still in progress, it is anticipated that there will be fewer reactor shutdowns which provide opportunities to physically access SSCs that require examination. Instead, advanced reactor designs will likely rely on the use of indirect monitoring of system conditions (e.g., leakage monitoring).

Section 2.1 provides the rationale behind the argument that SSC monitoring and decision-making have an impact on SSC reliability. Based on the predicted contributions of leaks from the pressurizer surge line to the "small-break LOCA" event, the analyst can estimate the probabilistic rate of occurrence associated with "small-break LOCA" from the pressurizer surge. Leakage monitors could be installed to measure either flow rate in the system or humidity on the area as an alternative means of detecting a leak.

6.3.1 Identification of What to Monitor

With the determination of the degradation mechanisms for the subject locations, it is necessary to determine what should be monitored, including the area of examination or monitoring and the acceptance standards, to effectively identify the anticipated degradation mechanisms. Once this is understood, a determination of what monitoring is available can be made and applied.

Assigning an examination category is not a requirement of ASME Section XI, Division 2; however, for LWRs, ASME Section XI, Division 2, includes, as part of Mandatory Appendix V, a catalog of NDE requirements and areas of interest that are aligned with the assignment of examination categories for the locations to be examined and the type of examination to be performed. Assigning an examination category in Mandatory Appendix V aligns with typical examination geometries and examination volumes for the subject SSC (note that applicability must be confirmed by the MANDEEP).

Applying the examination categories from Mandatory Appendix V to the SSCs of this example case provides a direct means of assigning the type and location for examination. Focusing the example case to two locations of the pressurizer surge line and a general (not specified) nozzle at the top of the pressurizer that is greater than 4 inches in diameter results in prescribing the NDE shown in Table 4.

Table 4. Required nondestructive examinations.

Part(s) Examined	Examination Method	Examination Boundary	Acceptance Standard(s)
Pressurizer Spray Line Nozzle & Surge Line Nozzle: NPS 4 (DN100) or larger nozzle to safe end butt welds (Category K)	Volumetric and Surface	Division 1 Fig. IWB-2500-8	RIM-3 & Appendix VII-1.4.3 & VII-3.4.3
Pressurizer Surge Line: NPS 4 or larger (DN 100) circumferential welds (Category J)	Volumetric and Surface	Division 1 Fig. IWB-2500-8	RIM-3 & Appendix VII-1.4.3 & VII-3.4.3

Not all of the reactor types are addressed by the Division 2 appendices and as of the 2019 Edition of ASME Section XI, Division 2, only data for LWRs exists in Mandatory Appendix V. It will be necessary for each developer of a RIM strategy to identify the areas to perform applicable examinations and demonstrate that the methods of examination have been demonstrated to effectively identify and characterize flaws and cracks. Selection of the examinations to perform in this example case assumes that the methods of detection have been demonstrated to effectively identify and characterize flaws.

6.3.2 Monitoring and NDE

After identifying the degradation mechanisms and relevant SSC parameters that need to be examined or monitored, the next step is to determine the means of monitoring, this includes direct and indirect monitoring.

For this example, the method of examination for the locations identified in Table 4 is already established based on the utilization of the requirements provided by ASME Section XI, Division 2, Appendix V, which subsequently applies the volumetric and surface examinations prescribed by ASME Section XI, Division 1. The detection of a flaw in the subject examination areas will be performed once per 10-year inspection interval as is prescribed for today's LWRs, including associated performance demonstrations.

The RIM strategy for an advanced reactor design may not have enough historical information to be able to limit the monitoring to the traditional NDE performed for LWRs using ASME Section XI, Division 1. The RIM strategy could require a system or facility shutdown to allow for NDE, such as an ultrasonic examination. As a result of this limitation on the conduct of NDE, the strategy may need to include the monitoring of system operating conditions that lead to degradation, such as changes in temperature, to maintain awareness of the potential for the onset of degradation. The pressurizer surge line can be used in this example to show how the RIM strategy may incorporate continuous monitoring associated with indirect monitoring for degradation.

Because of the potential for insurges and outsurges of fluid in the surge line, the DMA identified the potential for stress corrosion cracking introduced by thermal transients resulting from temperature differences in the fluids participating in the flow surges. Although it is not a requirement of ASME Section XI, Division 2, for a RIM strategy to specifically address the cause of the thermal transients, a form of monitoring for the resulting degradation must be performed. Rather than looking for the degradation, the Owner may choose to monitor for the conditions that lead to the degradation by installing temperature sensors for a defined length of piping up to and including the pressurizer surge nozzle. This will allow the Owner to monitor for the potential of stress corrosion cracking. It can also be used as part of an evaluation of the system when something is identified during the ultrasonic examinations. The data acquired from the instrumentation can then be used for trending and possibly integrated into a fatigue monitoring program that many of the operating PWRs have in place today.

Now the focus is placed on continuous monitoring for leakage. An advanced reactor design may have design conditions that justify the use of continuous leakage monitoring without the need to conduct nondestructive examinations as are used for this example application. Some of the considerations in developing a monitoring system are the effectiveness of direct monitoring sensors, proximity to likely DM locations, number of sensors required for an effective overall monitoring, reliability of each set of instrumentation, and overall reliability of the monitoring system.

There is experience and precedence with monitoring of RCS leakage in LWRs as evident through the guidance that the NRC staff considers acceptable for the monitoring and detecting of leakage contained in NRC RG 1.45, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage" [12]. Components or locations listed as "critical" for leakage monitoring by RG 1.45 include the reactor vessel head, control rod penetration nozzles, pressurizer nozzles, and dissimilar metal weld regions. The regulatory guide provides further information on the usefulness of monitoring parameters, such as the level or flow rate to tanks and sumps, airborne particulate radioactivity, airborne gaseous radioactivity, containment atmosphere humidity, containment atmosphere pressure and temperature, and condensate flow rate from air coolers.

As discussed in RG 1.45, leakage can be detected by radiation monitoring systems as reactor coolant normally contains sources of radiation that could be released to the containment. Large coolant leakage in the containment can also be inferred by large or abnormal fluctuations in indirect monitoring, such as air temperature and pressure levels. Dewpoint temperature measurements can be used for continuous monitoring of humidity in the containment atmosphere, where a rise in the humidity level may indicate a release of water vapor from a leak.

Once a leak is detected, it is critical that the leakage monitoring system can determine the source or location of the leakage as soon as practical. Humidity sensors, acoustic monitoring sensors, and remote visual surveillance capabilities (e.g., radiation resistant video) may be considered in the initial design for installed capabilities for directly identifying the source of the leakage.

The effectiveness of the monitoring system to detect and locate a leak will depend on the effectiveness and location of each instrument. The quantity and locations of these instruments could be determined by accounting for the range and effectiveness of each instrument, the coverage required throughout the containment, and conservatism accounted for in the analysis that provides the basis for the overall reliability of the system to perform its function. Accuracy, operational sensitivity, and detection response time of an instrument are important considerations for the effectiveness of leakage monitoring. An instrument should have adequate sensitivity to reliably detect small leakages and must have adequate sensitivity even for a large leakage in order to provide reliable leakage rates [12].

Estimating the POD of the overall leakage detection system will be a function of the performance measures of each instrument, such as accuracy, sensitivity, and detection response time. For certain instruments, especially those that have undergone a rigorous performance qualification process for nuclear reactor applications, the estimates of accuracy, sensitivity, and detection response time, with their respective uncertainties, could be provided by the manufacturer or supplier. Performance testing and reactor application could provide leakage detection data for an instrumentation that can be used for estimating $POD = n_d/n$, where n_d is the number of leaks successfully detected by the instrumentation out of the total n leakage occurred. The better the accuracy, sensitivity, and detection response time of the instrumentation, the higher the number of leaks that are detectable (n_d), and therefore the higher value of the POD. The number and location of instrumentation in a leakage monitoring system would affect the estimate of the POD of an entire leakage monitoring system with several instruments.

The monitoring and trending capabilities adopted for leakage location and quantification could be extended to determining the severity of the underlying degradation, such as the crack length. Analytical capabilities could be developed using a combination of physics-based calculations, historical data, and expert judgement to receive input of measured quantities, such as humidity or radioactivity levels, and provide output in the form of crack length associated with the given input quantities. Such analytics could provide estimates of crack length, rate of crack propagation, and whether the crack growth is stable or unstable. Additionally, research in continuous monitoring techniques that direct detection and characterization capabilities, including continuous monitoring for flaws in piping materials, may provide a more reliable means of monitoring in the future.

Although the example case is associated with a system in a PWR, this RIM strategy assumes that a basis has been established that demonstrates the capability of the leakage monitoring system to reliably detect and locate the source of a leakage in an appropriate time that allows action to be taken prior to piping failure. The basis will include appropriate justification of the location and quantity of system instrumentation and, if necessary, provide analytical or performance demonstration data as part of its qualification to meet the requirements of ASME Section XI, Division 2, Mandatory Appendix IV.

6.3.3 RIM Reliability Modeling

As indicated in Section 2, the starting point to perform reliability modeling in a RIM context is to understand the SSC under consideration (its form and function) and its functional relations with considered DMs and available monitoring data. As an example, for the pressurizer surge line, the formalisms described in Figure 1 are used to identify the relations shown in Figure 15 for this specific case.

The undesired event associated with this SSC is a leakage from any of the welds or piping greater than a specified value. It is here assumed that the RIM strategy associated with the pressurizer surge line includes the elements described in Table 5. This table presents the available monitoring data and associated set of O&M decisions for each element (also indicated in Figure 15).

Table 5. RIM strategy details for the pressurizer surge line regarding available monitoring data, related decisions, and impact on surge line reliability.

RIM Strategy	Data	Decisions	Impact on Surge Line Reliability
Periodic NDE inspection of piping and welds in search for flaws.	Number of flaws identified throughout surge line life.	Track damage progression. Assess criticality of flaw (e.g., load vs capacity argument). Plan for pipe and weld repair.	Update rate occurrence of flaws and POD of NDE monitoring system.
Periodic inspection of piping and welds in search for leakages.	Number of leaks identified throughout surge line life.	Track damage progression. Assess criticality of leak (e.g., load vs. capacity argument). Plan for pipe and weld repair.	Update rate occurrence of leaks and POD of leak monitoring system.
Monitoring of temperature of the water in the hot leg that is shared with the surge line.	Continuous data stream.	Track and identify water temperature excursions that might aggravate DMs.	N/A.
Monitoring of water leaked from the primary system through humidity sensors located in the containment.	Continuous data stream.	Assume observed leak is from surge line. If leak is within limits, fix pipe and weld at first outage instance, otherwise shutdown plant.	If observed leak is from surge line, update rate occurrence of flaws and leaks, and POD of monitoring system.

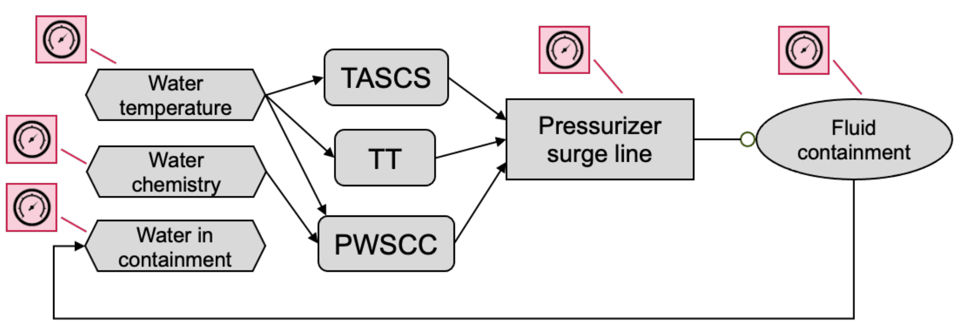

Figure 15. Schematic representation of the RIM strategy of the pressurizer surge line (its form and function): DMs (i.e., TASCS and TT), operating variables, and available monitoring data (highlighted in red) based on the considerations indicated in Table 5.

The reliability model that can be built based on the considerations indicated in Sections 2 and 3 are context dependent. Such models can be developed from first-principle reliability considerations or by relying on models developed in the literature (an example is provided in [11]). The information contained in Table 5 is structured in such a way that, for each element of the SSC RIM strategy, a clear connection between data, decisions, and reliability is set.

The actual probabilistic values for the parameters of such models can be estimated using past operational experience (for components operating in similar conditions) or by relying on assumptions based on initial testing. These probabilistic values can also be validated while the plant is operating. Once the probabilistic parameters have been established, the set of deterministic values (e.g., frequency of periodic inspections) can be set in such a way that the computed SSC reliability is below the SSC reliability target.

For the specific pressurizer surge line test case shown in Figure 15, the SSC reliability model is designed to capture the probabilistic occurrence of the undesired event based on the stochastic emergence of flaws and on the monitoring strategies specified in Table 5. In this respect, Figure 16, shows the temporal progression of the degradation (i.e., from flaw emergence to leak) that is considered for the surge line. It is here assumed that a flaw in the surge line can emerge according to its own probabilistic rate of occurrence r_{flaws}^{surge} . Consequently, the flaw becomes a crack until a leak is initiated and the crack grows until the amount of leakage meets a limiting condition (defined in the plant tech specs) (i.e., the undesired event). Figure 16 also indicates the temporal location of planned outages (set every 2 years) where NDE of the surge line is performed, with an estimated POD value.

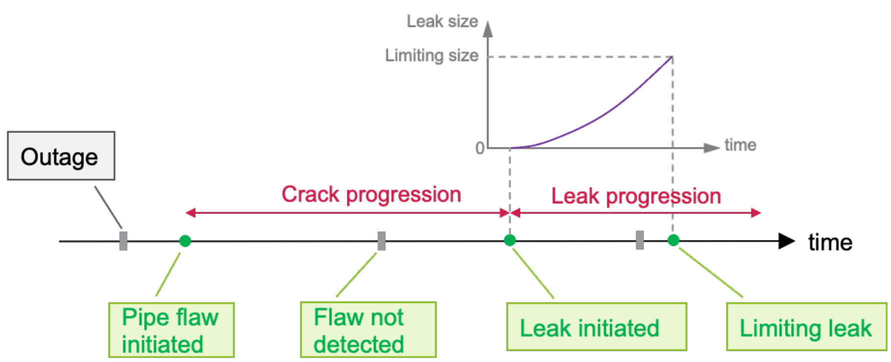

Figure 16. Timeline of flaw progression into leak for an element of the surge line.

6.3.4 RIM Strategy as Assurance of Adequate Performance

In this section, we assume that the plant is operating, and monitoring data is providing the appropriate information regarding surge line conditions. The goal of the following discussion is to demonstrate how the selected RIM strategy supports the demonstration of adequate performance of the surge line. The actual (i.e., as-is) reliability value of the surge line is calculated based on obtained data. Then, the actual reliability value is compared with the reliability target specified in the RIM program.

As indicated in Figure 11, the assessment of the impact of identified flaws and leaks on SSC reliability can be performed in a load vs. capacity mindset using modeling and simulation tools, statical analysis, or past operational experience. Actual SSC reliability combines the reliability of the employed monitoring system and the reliability of the decision process.

Regarding the reliability of the employed monitoring system, Table 6 provides an example of data that can be gathered for two RIM strategies shown in Table 5. From the number of flaws observed in the surge line piping (indicated as N_{flaws}^{pipe}), the probabilistic rate of occurrence of such flaws (indicated as

r_{flaws}^{pipe}) can be updated through Bayesian statistical methods [13]. As an example, assuming that the plant has been operating for 20 years and only one flaw has been detected, then using a constrained noninformative prior, the pdf of r_{flaws}^{pipe} can be analytically determined. In this respect, Figure 17 shows the pdf of the employed prior for r_{flaws}^{pipe} and the obtained posterior given the observed data. From the posterior pdf, it is possible to get the new value of r_{flaws}^{pipe} as the mean value of the posterior distribution; for this specific case, the obtained value is 6.0E-02.

The probability of failure for the surge line is recalculated using Equation 1 based on the updated value of flaw initiation r_{flaw}. The calculated probability of failure of the surge line at the beginning of plant operation was 7.5E-04, and the recalculated probability of failure at the 20-year mark is 6.4E-04. The limiting probability of failure (i.e., reliability target) specified for the surge line in the original RIM program is 2.0E-03. Both of the actual probability of failure values, at the beginning of plant operation (i.e., 7.5E-04) and the currently observed (i.e., 6.4E-04), are below the limiting probability of failure. Therefore, the condition and performance of the surge line are satisfactory.

Regarding the reliability of the decision process, for the identified flaws and leaks, an assessment of their progression rate is used to evaluate how the planned repair strategy is effective to avoid pipe failure. This can come in either deterministic or probabilistic terms using a load vs. capacity mindset by factoring in when the flaw and leak will be fixed.

Table 6. Subset of observed data under a specific RIM strategy and the corresponding reliability parameters that need to be updated.

RIM Strategy	Observed Data	Reliability Parameter
Periodic NDE inspection of piping and welds that in search flaws.	N_{flaws}^{pipe}: Number of flaws observed in surge line piping system N_{flaws}^{welds}: Number of flaws observed in surge line welds	r_{flaws}^{pipe}: Probabilistic rate of occurrence of flaws in piping systems r_{flaws}^{welds}: Probabilistic rate of occurrence of flaws in welds
Periodic inspection of piping and welds in search of leakages.	N_{leaks}^{pipe}: Number of leaks observed in surge line piping system N_{leaks}^{welds}: Number of leaks observed in surge line welds	POD_{flaw}^{pipe}: POD of flaws in piping systems POD_{flaw}^{welds}: POD of flaws in welds

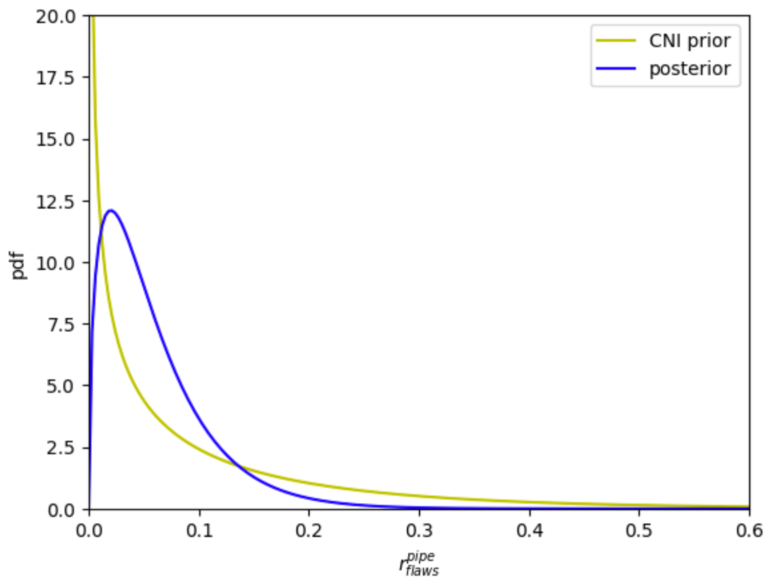

Figure 17. Graphical representation of the pdf of r_{flaws}^{pipe} : chosen prior and obtained posterior given the employed data.

As indicated in Figure 16, a leak that reaches the specified limiting conditions (in terms of maximum allowed gpm) can occur when of all the following conditions are satisfied:

- A flaw is initiated and becomes a crack
- The crack is not detected during the first examination for flaws (e.g., the first outage instance)
- The crack creates a leak that remains undetected by the leak monitoring system
- The leak grows until it reaches the specified limiting conditions.

Regarding the leak detection system, we consider a scenario where a set of five redundant sensors are located within the containment building for detecting leaking water from the RCS system. The information generated by the sensor is fed into a controller located outside the containment, which provides an indication of the possible presence of a leak and informs the reactor operators.

The locations of each sensor have been chosen in such a way that they are able to detect leaks from all elements of the RCS. This was performed by employing simulation tools where temporal evolution of humidity inside the containment were calculated for leaks of different sizes and RCS locations. In particular, simulation analyses were performed to assess the detection performance of the considered leak detection system for different values of leak sizes located in the pressurizer surge line. Of interest is the time required by the detection system to detect a leak after it is initiated based on different crack sizes.

In addition, the temporal evolution of crack growth was estimated from past operational experience and fracture mechanics analyses. This information is used to determine the time distribution for a crack in the pressurizer surge line to generate a leak greater than the specified limiting threshold.

Table 7. Suge line leak: phenomena considered and reliability modeling considerations.

Phenomena	Reliability Considerations
Flaw initiation	Flaw in the surge line is modeled through a probabilistic rate of occurrence r_{flaws}^{surge}

Crack not detected during outage	NDE inspection of surge line is characterized by a POD_{NDE}^{surge}
Leak develops from cracks	Physics-based consideration based on fracture mechanics simulations
Leaks not detected between two outage instances	POD between two outages POD_{leak}^{surge} of leak detection system

From the available reliability database, the reliability values of the employed sensors and controller can be estimated with the goal of estimating $Prob(no\_detection)$ of Equation (2). For this specific case, a leak can reach the maximum allowed gpm if all the conditions listed in Table 7 are met. Regarding the first element of Table 7, r_{flaws}^{surge} is estimated from past operational data. The probability of an NDE inspection of the surge line to not detect a flaw (i.e., POD_{NDE}^{surge}) is estimated by evaluating the detection performance of similar NDE inspection systems. The temporal transition from cracks to leaks (third element of Table 7) is estimated to occur in about 1–1.5 years. Regarding the second element of Table 7, assuming the sensors can be replaced only during an outage (e.g., every 2 years) and assuming a failure rate for each sensor set to $\lambda_{sens} = 1E - 4\ yr^{-1}$, POD_{NDE}^{surge} can be calculated, given five sensors, as $Prob(no\_detection) = (1E - 4\ yr^{-1} \cdot 2 \text{ years})^5$.

Based on these considerations, the probabilistic rate of occurrence of a leak to reach the maximum allowed gpm can be quantified as

$$r_{leak}^{surge} = r_{flaws}^{surge} \cdot POD_{NDE}^{surge} \cdot POD_{leak}^{surge} \tag{3}$$

From a RIM point of view, r_{leak}^{surge} is compared to the target rate $r_{leak}^{surge,target}$, which is specified in the surge line RIM strategy; if $r_{leak}^{surge} > r_{leak}^{surge,target}$, the RIM strategy needs to be modified. Possible course of actions are:

- Improve POD_{NDE}^{surge} for the NDE inspection of surge line which can be accomplished by improve the frequency of NDE inspection and/or improve the quality of the NDE system.
- Improve POD_{leak}^{surge} for the leak monitoring system which can be accomplished by increasing the number or the reliability of humidity sensors.

6.4 LWR Case Study Summary

This example case study demonstrated one possible RIM strategy for the pressure boundary of a system in a LWR. Presentation of this example RIM strategy does not constitute NRC acceptance of this example RIM strategy nor for a system with similar operating conditions and degradation mechanisms.

The study presented the key steps in establishing the RIM strategy: determination of reliability targets for system parts/components, DMA, and development of the RIM strategy in light of the available monitoring methods and tools. The case study also pointed out aspects to consider when designing the RIM strategy.

7 EXAMPLE APPLICATION - SFR CASE STUDY

The second example case provided in this report is associated with a sodium fast reactor (SFR) advanced reactor design, a RIM strategy applied to an SFR intermediate heat transfer system (IHTS). The case study compliments the earlier LWR work and increases the scope of this report. Similar to the LWR example case study, this example case study is meant for demonstrating the process of developing a RIM strategy and use of this example does not imply that this is an NRC approved RIM strategy for application to an SFR IHTS nor for other similar systems.

7.1 Introduction of SFR Case Study

For this specific scope of work, the IHTS of a SFR design was selected as an example case for developing a RIM strategy. The example case does not constitute an NRC evaluation or determination of acceptability of a RIM strategy for the purposes of licensing an SFR. The selection of a system from an SFR for this example case is based on the availability of publicly available information such as that for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor. Selection of the IHTS does not constitute NRC acceptability of the results of the evaluation and development of a RIM strategy for the IHTS or any system design with similar attributes as that presented in the example case.

In this RIM strategy demonstration, the basic steps are to define the scope, identify degradation mechanisms, establish reliability targets, select strategies for meeting those targets, and evaluate the proposed strategies.

7.2 System Description

7.2.1 Description of Sodium Fast Reactor Example Case Boundaries

Publicly available information was used to define the system boundaries and operating parameters for this example case; however, the availability of SFR information readily applicable to this example case was limited. In establishing this example case, most of the information was pulled from two sources: NUREG-1368, "Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor" [14]; and the PRISM Preliminary Safety Information Document [15]. The example design was augmented by additional information pertaining to other similar designs and by assumptions that provided the necessary parameters for enabling the application of a RIM strategy.

7.2.2 Description of Intermediate Heat Transfer System

The IHTS is comprised of two IHTS heat exchangers (IHXs) within the reactor vessel, piping from the IHXs to the steam generators (SGs) (the shell side of the SG contains the IHTS sodium, also referred to as secondary sodium), the IHTS sodium pumps, and IHTS isolation valves (located just outside the reactor enclosure). Rupture disks, designed to fail at 325 psi (2240 kPa), are used to provide system overpressure protection that may result from SG tube rupture, with the relief flowpath to the Sodium-Water Reaction Protection Relief System (SWRPRS). Figure 18 provides a general overview of the PRISM IHTS.

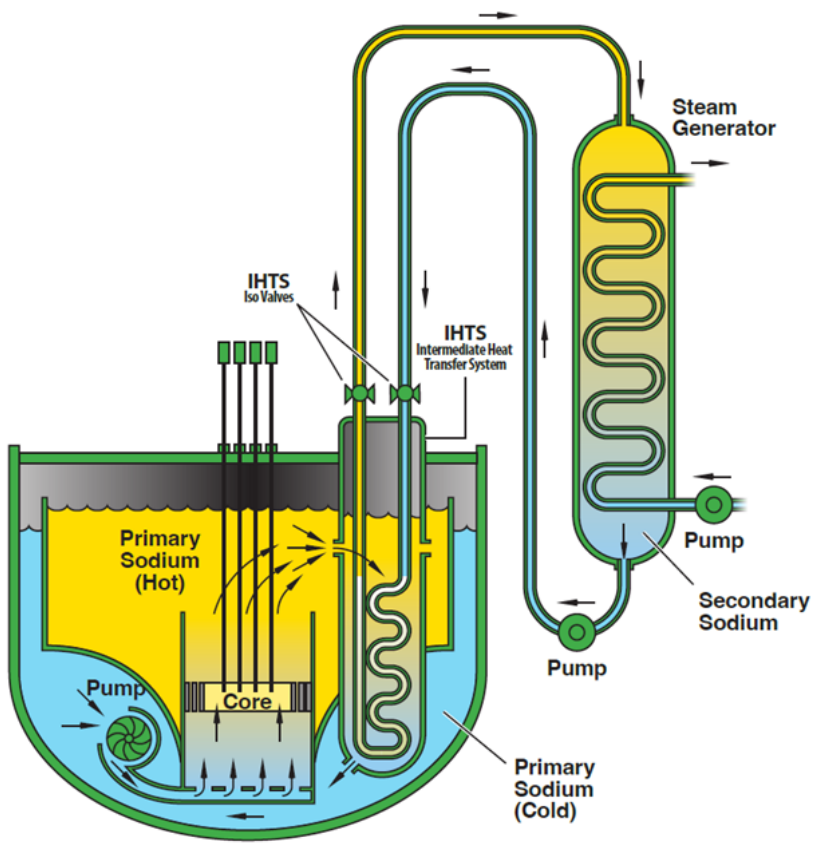

Figure 18: Simplified overview of the IHTS.

Subsection 5.5.1 of NUREG-1368 states that the IHTS, with the exception of the IHX, is non-safety-grade. However, the referenced documentation on PRISM uses both the terms "safety-grade" and "safety-related" throughout and identifies more than just the IHX as being safety-grade (e.g., NUREG Figure 5.1 shows portions of the IHTS—from the reactor vessel to the isolation valves—as being safety-grade [with the remainder being non-safety-grade] or applying industrial codes and standards). Furthermore, it states that although it is non-safety-grade, it will be inspected in accordance with ASME Section XI, Division 3[h].

ASME Section XI, Division 2 [2], does not provide a basis for what is in the RIM program scope, stating only that the Owner is responsible for identifying the scope. The importance of the SSC functions thus becomes a driver for determining said scope, implying that safety-related SSCs will likely fall within the scope of an RIM program. The IHTS does feature important functions such as ensuring that the IHX can transfer heat from the primary sodium in the reactor vessel without compromising the integrity of the Primary Heat Transfer System (PHTS)/IHTS pressure boundary, but these important functions are not necessarily significant in terms of nuclear safety. The importance of different system functions is determined as part of a licensing application. A risk-informed performance-based approach to licensing could be applied via the LMP [6], which is complimentary to the RIM risk-informed performance-based method.

[h] Although this reference is no longer applicable and ASME Section XI, Division 2, is an endorsed code for assuring the structural integrity of a pressure boundary, it can be used as a basis for including in this example case without insinuating that this piping in similar liquid metal reactor designs should be within scope of a RIM program.

7.2.3 Intermediate Heat Transfer System Heat Exchangers

The PRISM preapplication safety evaluation report (NUREG-1368) states that one of the IHXs' safety objectives is to isolate radioactive primary sodium from the intermediate (secondary) sodium and provide the mechanical boundary between the two outside the containment boundary [14]. Because the primary sodium pressures are near atmospheric and the IHTS operates at approximately 100 psi, a basis could be developed to classify the IHXs as non-safety-related, since secondary sodium would be leaking into the primary sodium and not vice versa. For this example, the IHXs are considered to fall within the scope of the RIM program, as they have the important function of providing the boundary between the radioactive sodium and the intermediate sodium outside the reactor enclosure.

The IHX design is a basic shell-and-tube HX consisting of upper and lower tubesheets separated by straight tubes. The IHX is kidney shaped (see Figure 19), with a central downcomer and riser for the incoming and outgoing secondary sodium, respectively. The primary and secondary sodium flow paths are also illustrated in Figure 19.

The two IHXs are located partially within the reactor vessel annular space and penetrate the vessel baffle plates from the hot pool to the cold pool. Primary sodium enters the shell side of the IHX and flows down from below the upper tube sheet to the lower tube sheet, transferring heat through the IHX tubes containing the secondary sodium loop. The secondary sodium flows down the central downcomer (entering from the reactor vessel closure head) and back up through the tubes that exit exiting the heat exchanger through the riser.

The IHX is vertically supported by the reactor vessel through two concentric pipes (i.e., the riser and the downcomer) that transfer their load to the reactor top plate. Minimal information can be found on how the IHX is attached to the reactor vessel—other than that it is welded to the IHX support flange, which is in turn a flange-bolted connection to the reactor top plate.

The IHX is constructed from Type 304 austenitic stainless steel (SS), with a tube bundle containing over 2,000 tubes. Although the tube bundles and heat exchanger are expected to be designed for long-term operation (e.g., 40 years or more), the complete IHX is assumed to be replaceable. The IHX support flange and associated bolting are fabricated from grade 304 SS material, while the IHX tubes and internal components may be fabricated from grade 316 or 316N SS material.

One IHX performance consideration noted in NUREG-1368 is the potential damage to the IHX as a result of flow instabilities and pressure loss. Performance testing was to be performed to ensure that the IHX and its parts, including the IHX tubing, would not be damaged or cause a malfunction by either flow-induced or seismic vibrations. It further identifies that the IHX's dominant failure mode is creep fatigue/damage in its upper (hotter) portions, and that creep damage results from residual stresses created during temperature excursions in the reactor vessel.

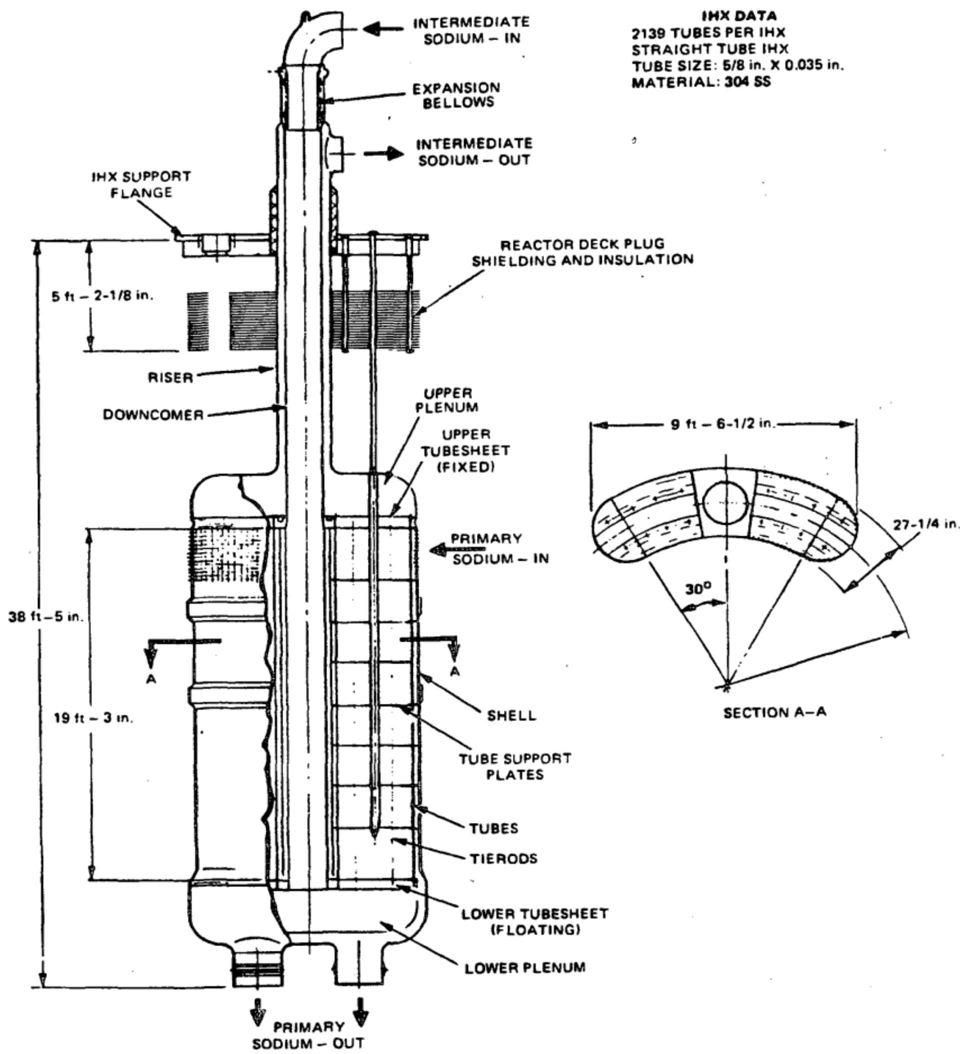

Figure 19: PRISM IHX diagram (taken from [14]).

7.2.4 Expansion Bellows

The expansion bellows at the upper end of the IHX and at the IHTS penetration through the reactor enclosure allows for differential thermal growth between the tube bundle and the IHX downcomer.. The upper end of the IHX features a shield plug that interfaces with the top reactor plate and is considered an integral part of the IHX. Limited information exists on the design of the bellows, and for the sake of simplicity in this example case, it will not be considered when assessing the degradation mechanisms and associated condition monitoring or examination. However, exclusion of the bellows does not imply that it is not an SSC in similar plant RIM programs.

7.2.5 Sodium-Water Reaction Pressure-Relief System

In the event of SG tube rupture, the SG rapidly depressurizes via a steam- and water-side blowdown system initiated in conjunction with the sodium dump of the IHTS by the SWRPRS.

The IHTS overpressure protection afforded by the SWRPRS includes the use of two 28-in. safety-grade rupture disks, in series, that are designed to rupture at 325 psi (2241 kPa) and provide a relief path out of the IHTS. The SWRPRS design accommodates all sodium-water reaction products—even should a complete break of all the SG tubes occur—maintaining an IHTS backpressure of below 700 psi (4826 kPa).

Overpressure protection of the IHTS resulting from a SG tube rupture is afforded by isolation valves (see the description below) in addition to the rupture disks. The IHTS piping and components are designed for faulted conditions equal to full steam pressures of 1000 psig (6996 kPa).

The SWRPRS is separate from the IHTS and is not included within the scope of the example case.

7.2.6 Intermediate Heat Transfer System Isolation Valves

Safety-grade[i] IHTS isolation valves, located immediately outside the reactor enclosure protects the IHXs from the high pressures that can occur due to a SG tube rupture. These valves can be closed on a high-pressure signal, protecting the IHX tubes from increased corrosion potential as a result of sodium-water reaction products. The valve pressure boundary is within the scope of the example RIM strategy. The ability to isolate on a high-pressure signal is not included in the example RIM strategy[j] but highlights the importance of precluding corrosive materials from entering the IHXs, as well as how the PRISM design reduces or precludes that from occurring.

7.2.7 Piping

The IHTS piping is designed in accordance with ANSI/ASME B31.1. Guard piping surrounds the intermediate loop piping inside the reactor enclosure, limiting the possibility of sodium fires and spill hazards in this area, which is more difficult to access when necessary to locate the source of sodium leaks and the resultant generation of smoke[k]. NUREG-1368 states that both the safety- and non-safety-grade portions of the IHTS will be continuously monitored by a leak detection program.

Parameters necessary for assessing the IHTS degradation mechanisms are summarized in Table 8. Many of these values were taken from the already referenced PRISM safety analysis and safety evaluation documentation [14], [15], with adjustments having been made per other reactor designs and engineering judgement in order to fill in gaps or make the data relevant to the example case.

Table 8. IHTS design and operating parameters.

IHTS Component	Design Parameters
IHTS sodium temperature SG HL/CL (PRISM SAR Table G.2.2-2)	830°F (443°C) / 540°F (282°C)
IHTS flowrate	41,250 gpm
IHX shell and support material	304 SS
IHX tubes and internal components	316/316N SS
Reactor vessel material	316 SS
IHTS piping material	304 SS
Guard vessel material	2 ¼ Cr-1Mo
Guard vessel wall thickness	1.0 in. (2.54 cm)

[i] The term safety-grade is used as a descriptor similar to safety-related.

[j] The IHTS isolation valve may be included in a RIM strategy as part of complying with a licensee's inservice testing program but the function to isolate flow is not the pressure boundary function that ASME Section XI, Division 2 applies to.

[k] The SER specifically states that the guard piping is present to provide an investment protection function.

7.3 Determination of a Performance (Reliability) Target

In the context of a RIM program developed in accordance with ASME Section XI, Division 2 [2], allocation is a decision-making process that satisfies a set of high-level performance targets (e.g., plant- or function-level risk/reliability targets) via investments that ensure that a particular set of lower-level (e.g., component-level) risk/reliability targets are achieved. Satisfaction of a particular set of lower-level risk/reliability targets implies satisfaction of the higher-level targets. Article RIM-2.4.3 of ASME Section XI, Division 2, states that the detail of a PRA be sufficient to develop reliability targets for each SSC within the RIM program and if the PRA is not sufficient to associate the SSCs with an element of the PRA, then to modify the PRA. This implies that mapping of the component-level reliability targets to the higher-level targets will be accomplished by conducting a PRA of all components within the scope of the RIM program.

For a given set of component-level targets, MANDE may need to be conducted to show that the component-level targets—and thus the higher-level ones—are satisfied. Establishing the targets at various levels is a responsibility of the RIM Expert Panel (RIMEP). Because the targets cannot be set without regard to the feasibility of demonstrating their satisfaction through MANDE, targets must be established iteratively with the MANDE Expert Panel.

In order for a component-level allocation to make sense, it must imply satisfaction of the higher-level target(s), with sufficient assurance that achievement is feasible. That is, MANDE must be able to provide the needed assurance of performance: the observations (kind and frequency) must demonstrate component states in enough detail to support competent RIM.

7.3.1 Satisfaction of Higher-Level Targets

Higher-level targets may address performance at several levels: the facility-level, the functional-level, the system-level, and all the way down to the component-level. The RIM process leaves it to the RIMEP to develop these targets. The properties of a good allocation are at least partly those of the entire collection of targets. If limited credit is taken for a given system, then the situation may require more credit for a different system (via larger investments and a greater focus on quality assurance, testing, and maintenance of that system). This is suggested notionally in Figure 20.

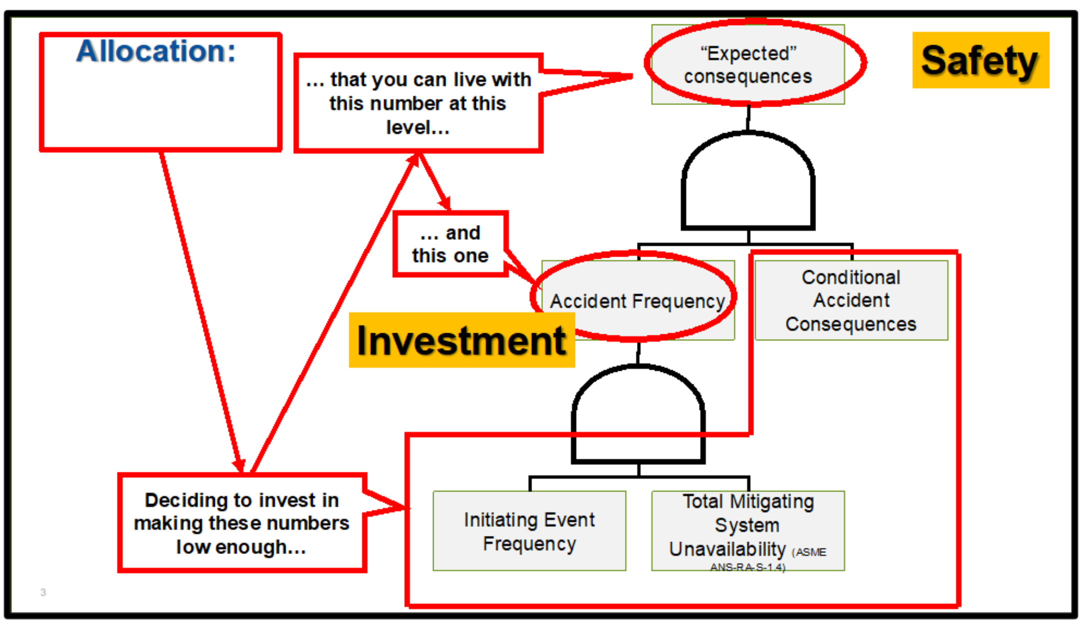

Figure 20: Notional representation for the allocation of higher-level performance targets.

For example, in the case of today's operating pressurized-water reactors, many chose to take credit for both the auxiliary feedwater and the primary feed and bleed, rather than attempt to argue that the auxiliary feedwater system operates perfectly. A logic model for the failure of the auxiliary feedwater and feed and bleed would be an input to the "Total Mitigating System Unavailability" box in Figure 20 above. The term "allocation" is not usually applied to that example, but the core damage frequency (CDF) subsidiary objective[1] should be satisfied. For presently operating light-water reactors, the CDF subsidiary objective in effect becomes a plant-level target within a RIM-like framework, and it is desired that individual contributors to the CDF metric be significantly smaller than this plant-level target. The transient initiating event frequency enters the calculation, as does auxiliary feedwater reliability and the reliability of the feed and bleed function. If thermal-hydraulic uncertainty exists in the success probability of the feed and bleed, it also enters the calculation and may influence how much credit is taken for auxiliary feedwater and/or having a low initiating event frequency. Maintenance practices regarding these systems' components, as well as operator performance, also influence the calculation.

In short, a given high-level target has multiple influences, in turn placing multiple constraints on the admissibility of a given component-level allocation.

7.3.2 Satisfaction of the Component-Level Targets

Performance targets must be practical to achieve (e.g. the implied costs must be affordable and justifiable, the inspections must be effectively performable, the monitoring must be effective, etc.). Assigning overly small failure probability targets to the various components may make it difficult for any practical RIM strategy to show that those targets are being satisfied. In general, the principle of DID is not to overly rely on any one thing in formulating a safety argument. This means that reasonable targets will be derived, to the extent practical, for a broad complement of components featuring collective redundancy, rather than relying on only a few specific components. It is possible the RIMEP may decide to assign intermediate (function- or system-level) targets in such a way that their satisfaction deliberately precludes overreliance on a particular subset of components. For example, when several redundant systems are each required to perform at a given level, this precludes total reliance on any single one of them.

Though much of the above discussion has addressed areas in which the facility has redundancy (e.g., multi-train systems in current operating reactors), it also applies to individual events that directly contribute to a facility-level risk metric, since a plant-level target on that metric is the sum of all contributors to that metric. Suppose that the RIM program is informed with a target on "frequency of small loss-of-coolant accident," and also that this was chosen by postulating extremely high reliabilities for all boundary components except one, whose assumed unreliability is close in value to the target. Any uncertainty in that one number becomes a critical uncertainty in the overall evaluation.

If a plant design offers many success paths, each comprising numerous components, a wide array of possibilities may exist for satisfying a top-level objective, potentially making it difficult to select the best option. Computational techniques (e.g., Top Event Prevention Analysis) can assist in this regard, but are computationally challenging to apply to large problems.

The System Based Code [16] concept was applied in Japan to allocate performance to passive components to prevent reactor core damage. CDF was used as a plant-level reliability target metric. This technique simplified the problem by taking little or no credit for mitigating systems and instead focused on achieving an extremely low initiating event frequency. At present, it is unclear whether such an

[1] The "subsidiary objectives" are derived from the Commission's Safety Goals. One of them is on core damage frequency.

approach will be attractive to all vendors of new designs. However, a consensus on an approach will be valuable to both RIM and the LMP phases/aspects of licensing applications.

Idaho National Laboratory (INL) conducted a study by using an approach similar to the System Based Code technique [17], demonstrating reliability target allocation from the top-level to the component-level by decomposing the top-level target into systems and components that contribute to the potential failure of that top-level function. That approach utilized optimization techniques in which the lowest possible reliability target (i.e., highest unreliability) on the component-level that would satisfy the plant-level reliability target was preferred, as higher unreliability usually corresponds to lower costs. The lowest cost was the objective function, and satisfaction of the high-level reliability target was the constraint.

7.3.3 Intermediate Heat Transfer System Reliability Target

As previously mentioned, and as represented by Figure 18, the IHTS supports electricity production by transferring heat from the reactor core to the SGs. This heat is transferred by the flow of secondary sodium through the IHX tubes (absorbing primary sodium heat) to the shell side of the SGs (with the tube side of the SGs containing water). This is the normal operating means of removing heat from the reactor core. Per NUREG-1368, the PRISM design is equipped with three methods of decay heat removal [14]:

1. Cooling the condenser in conjunction with the intermediate sodium and SG systems
2. The auxiliary cooling system, which removes heat from the SG via natural convection of air after transporting heat from the core via natural convection in the primary and intermediate systems
3. The reactor vessel auxiliary cooling system (RVACS), which removes heat passively from the reactor containment vessel via natural convection of air.

As per the earlier discussion on allocation, Figure 21 illustrates options for decay heat removal.

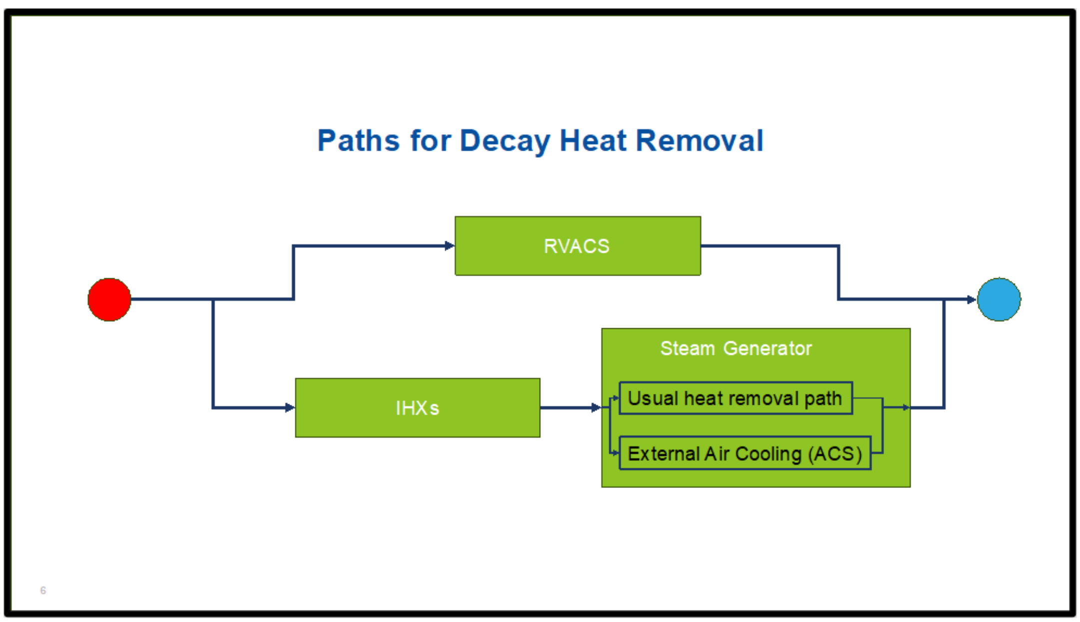

Figure 21: Options for decay heat removal.

RVACS is a simple, passive system identified as the safety-grade means of removing heat [14]. If the RVACS alone is sufficient to remove the decay heat, the IHTS need not be credited in post-trip heat removal unless the high-level target on loss of decay heat removal is extremely stringent. Thus, the IHTS's main function is not a significant part of the safety case.

The next action is to determine whether an IHTS malfunction could compromise any other higher-level functions or cause an initiating event. One such event is the potential for an SG tube leak substantial enough to result in a sodium-water reaction that increases the pressure in the IHTS and challenges the IHX in the reactor vessel.

The IHTS contains isolation valves that are installed just outside the reactor vessel enclosure and are designed to close on a high-pressure signal to specifically protect the IHX tubes from potential corrosion stemming from the sodium-water reaction products. As previously stated, analysis of the PRISM design revealed that the pressures in the IHTS would be limited to 700 psi (4826 kPa) even if all tubes in the SG were to rupture. The IHTS piping and components are designed for faulted condition pressures of 1000 psi (6996 kPa), with IHTS overpressure protection afforded by rupture disks designed to kick into action at 325 psi (2240 kPa).

NUREG-1368 discusses whether the analysis of the proposed worst-case SG tube rupture event adequately considered the combined effects of safety-grade system design and failure (including rupture disk malfunctions), along with the proposed failure of non-safety-grade systems. A containment bypass path may appear in the event an isolation valve was to fail. For this example, this issue was assumed to have been satisfactorily addressed.

Because the SGs are located outside the reactor enclosure, the IHX tubes provide the main barrier between the primary and secondary sodium contained in the IHTS sections outside the reactor vessel enclosure. Were an IHX tube failure to occur without the failure of any other component or portion of the IHTS, secondary sodium would leak into the primary sodium because the primary sodium is more or less under atmospheric pressure, which is a low value relative to the pressure in the intermediate loop (~100 psi).

A large sodium leak outside the reactor enclosure could theoretically cause a fire, an initiating event that may compromise safety functions and have associated investment protection considerations.

7.3.4 Considerations in Setting the Reliability Targets

Since the IHTS is not relied on to perform any required safety functions, reliability targets are unnecessary from a nuclear safety perspective. Instead, they are desired from an investment protection standpoint. Given that a loss of integrity occurred at the Monju plant [18], it is appropriate to mention that incident briefly so as to inform the present example. A detailed review is beyond the scope of the present illustration, but certain features are relevant.

Though the Monju design does not directly correspond to the PRISM design, the leak location, as indicated in Figure 22, does correspond notionally to a particular portion of the PRISM design: the piping outside the reactor enclosure that transports sodium from the IHXs to and from the SG. Perhaps noteworthy is the fact that the isolation valves in the PRISM IHTS are much closer to the containment boundary than those shown for the Monju plant.

Failures in piping are often unexpected. However, the Monju event was not a piping failure, but rather a failure of a thermocouple well projecting through the pipe wall into the sodium flow stream. At least one source attributed it to flow-induced vibration of the thermocouple well. It is uncertain whether comparable devices were considered in the PRISM design, but if so, the operating experience obtained from Monju could lead to the inclusion in the review of similar designs.

Though no significant radiological consequences resulted from the event, other types of consequences were indeed significant, such as shutdown of the reactor followed by expensive repairs and cleanup activities. Given the significance of these consequences, deliberate efforts should be made to preclude such events. Thus, for the purposes of demonstration and discussion, this example case selects two IHTS targets focused on investment protection—namely, targets for leak occurrence and leak response.

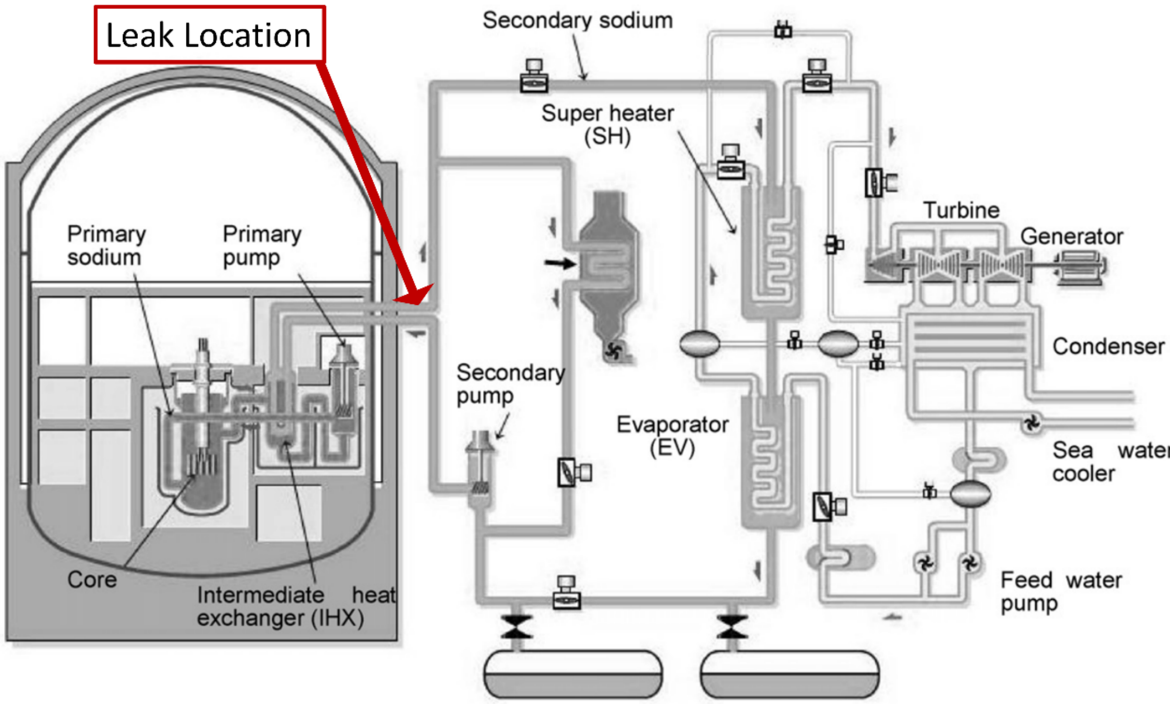

Figure 22: Representation of the Monju plant from [19], leak location added from [18].

When setting a performance target, the event sequence diagram and subsequent event tree given in Figure 23 and Figure 24, respectively, usefully illustrate how one might arrive at reasonable target values for leak occurrence and response.

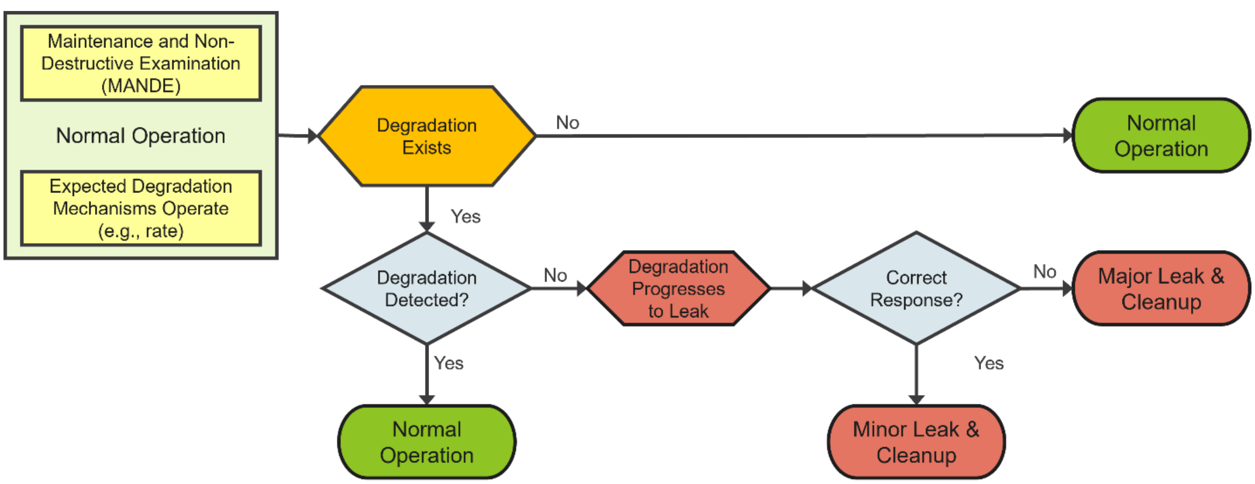

Figure 23: Simplified event sequence diagram.

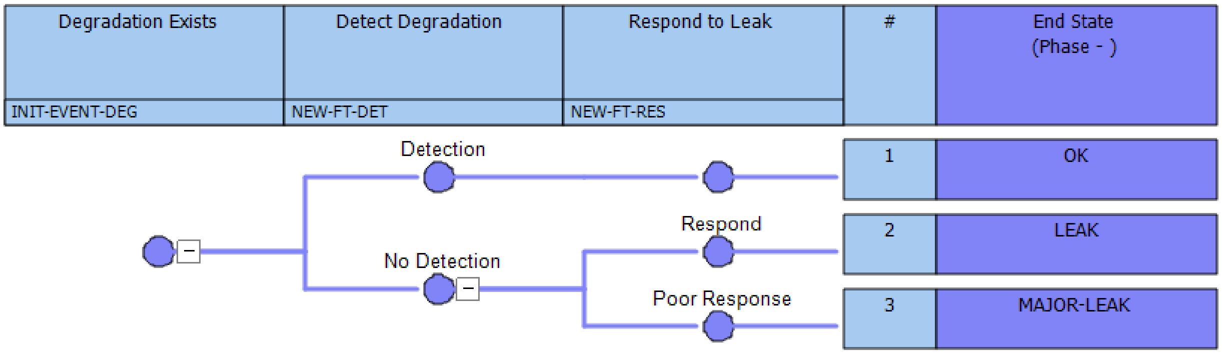

Figure 24: Event tree perspective for arriving at target values.

The event sequence diagram in Figure 23 provides a high-level overview of how events might progress to a leak. In this case, during operation, degradation mechanisms may lead to leakage, for which there are varying degrees of response. The same sequences can be shown as an event tree (see Figure 24). The following offers clarity on the figures.

- Degradation mechanisms occur during operation and may lead to degradation. Certain levels of degradation is expected and normal operation is not questioned.
- Degradation may be detected (e.g., through MANDE) and actions taken to correct, or accept the condition, which again puts the system in a condition for normal operations.
- Degradation may proceed to leak prior to detection with MANDE.
- Occurrence of a leak can be followed by a predetermined set of actions (responses) that keep the leak small, with minimal or no consequences; or by a failed response that results in a large leak with large consequences.

MANDE should be formulated based on a pipe rupture frequency that is acceptable to the stakeholders. Using probabilistic fracture mechanics techniques, the probability of pipe rupture may be established based on the phenomenology of degradation mechanisms that cause leaks which may in turn lead to pipe ruptures. If MANDE reveals degradation that is more rapid than anticipated and desired, corrective actions may be taken. Should the observations, sensors, or MANDE efforts prove insufficient (i.e., fail to detect the presence and progression of degradation), a leak may occur, perhaps ultimately resulting in a pipe rupture.

Figure 24 presents outcomes for a leak, based on whether the leak detection and response were effective. If ineffective, this is not necessarily a safety issue but may lead to significant consequences such as reactor shutdown, expensive cleanup, and significant adverse publicity. In principle, target determination for detection and response should be informed by the leak frequency target and its uncertainty. The costs associated with an ineffectively mitigated leak, including costs not explicitly financial in nature, should also be considered.

Two targets are discussed in this example: one to preclude a leak (the “LEAK” end state in the event tree in Figure 24), and one to preclude a large break or pipe rupture that would necessitate a large cleanup (the “MAJOR-LEAK” end state in the event tree in Figure 24).

The target allocation starts from the top (i.e., from the undesirable end state). In this case, the undesirable end states are LEAK and MAJOR-LEAK. From an investment protection standpoint, preclusion of major leaks is a must, and avoidance of small leaks is desired.

Thus, the reliability target for leakage is postulated as “less than once in a lifetime” along with a desired safety margin. The plant lifetime is assumed to be 100 years, so the small-leak frequency should be less than once every 100 years, or 1E-02. When considering a safety margin of 2, the unreliability target for leakage becomes 5E-03 (i.e., a reliability target of 0.995). Given that major leaks must be avoided, a much larger safety margin equal to 100 is applied, making the unreliability target for major leakage equal to 1E-04 (i.e., a reliability target of 0.9999).

As a means of discussing target allocation, consider the following:

- **Degradation exceeds limits**: One possible assumption is that IHTS components are designed for the plant's entire lifetime. Therefore, degradation in excess of the limiting conditions is not expected to occur or may be reached at a single point during the plant lifetime. With an assumed plant lifetime of 100 years, the probability of this occurring is 1/100, or 1E-02. To satisfy this reliability target, the materials and manufacturing processes (e.g., welds) for the IHTS components must be demonstrated capable of withstanding the anticipated degradation mechanisms throughout the plant lifetime.
- **Detect degradation**: Given the desire to preclude leakage, the plant designer/owner/operator is willing to establish measures for continuously monitoring the system conditions. The probability of successfully detecting degradation should be estimated based on monitoring techniques that consider equipment reliability, procedure quality, and the qualifications of the personnel performing the monitoring (if manual). For this example, it was assumed that MANDE and other monitoring techniques had to hit a reliability target (or probability of degradation detection) of 0.99, with a corresponding unreliability of 1E-02.
    - **Probability of leakage**: Given the two unreliability targets established above, the probability of a leak occurrence is 1E-02 x 1E-02 = 1E-04, which is smaller than the set unreliability target of 5E-03 for the leak. Thus, the low-level reliability targets were satisfactory for meeting the higher-level reliability target.
- **Respond to a leak**: An effective response to a leak includes actions to limit damage (i.e., preclude a large pipe break or rupture). These actions could be to shut down the reactor and repair the small leak before it propagates into a large break or rupture. The credibility of the response should be confirmed by demonstrating the adequacy and reliability of the leak monitoring equipment, the procedural guidance for the operator response to the leak, or the adequacy of the automated response (e.g., by demonstrating the automated system's performance in response to the leak). For this example, the required reliability of the response actions was set to 0.99, with a corresponding unreliability of 1E-02.
    - **Probability of major leakage**: Given the three unreliability targets above, the probability of a major leak is 1E-02 x 1E-02 x 1E-02 = 1E-06, which is smaller than the set unreliability target of 1E-04 for the major leak. Thus, the low-level reliability targets are satisfactory for meeting the higher-level reliability target.

The above example also demonstrates that the low-level reliability targets could be even lower yet still satisfy the high-level reliability target. For example, the probability of degradation exceeding the limit could be as high as 0.5. With a degradation detection reliability of 0.99, the probability of leakage would still be 0.005 (5E-03), which is the set reliability target. This relaxation would allow plant designers to select lower-grade materials; however, the plant operator would have to heavily rely on the degradation monitoring. On the other hand, less rigorous degradation monitoring approaches could be selected with detection probabilities as low as 0.5, with the quality of materials and construction driving an initiating event frequency that supports the 1E-02 estimate. This simple example demonstrates the flexibility that exists in allocating component-level reliability targets. With dozens of higher-level targets and hundreds of component-level reliability targets, the solution space becomes very large, with potentially millions of combinations that would all lead to a satisfactory result. Various modeling and simulation techniques could support optimization of reliability target allocation.

The numbers above are not presented as being “recommended” or even “representative.” They simply demonstrate typical occurrence frequency calculations for given event sequences. Neither the LEAK nor MAJOR-LEAK end states in Figure 24 are a threat to the public~~.~~

To support a quantitative approach similar to that demonstrated above, the potential for degradation mechanisms and how quickly degradation can progress should be understood. Whenever the degradation mechanism phenomena or the rate of degradation are insufficiently understood (e.g., due to use of a novel material or to a known material being used in a novel operating environment), measures could be put in place to monitor the progression of degradation in order to collect operating experience data that in turn support an in depth understanding of how the degradation progresses. In this scenario, margins between the current (i.e., as-is) state and the undesirable state(s) are monitored and assessed. INL research [20] proposed such an approach, which (1) supports performance monitoring as part of a RIM program and (2) could serve as the basis for a regulatory approach to accelerated deployment of new material or applications in advanced reactors.

As already discussed, the present example will not address SSC-level reliability targets derived from severe-accident metrics but does establish reliability targets for investment protection purposes. For investment protection, two classes of events suggest themselves: events that call for shutdown in order to repair (replace) a component (e.g., IHX tube leakage), and events that require cleanup in addition to component replacement or repair (e.g., IHTS piping outside of containment).

Eide et al.’s research [21], as presented in Table 9, suggests that IHX leakage events may occur on the order of 1E-2 per year. According to NUREG-1368, the IHTS design is meant to allow replacement of the IHX. Though investors would prefer to avoid IHX failures, there does not appear to be a compelling safety reason for a stringent target on failure probability. Moreover, inspecting the IHX may be infeasible or very costly. Thus, a goal of 1E-2 per year for “IHX replacement” was selected. This value corresponds to a replacement rate of less than once in a plant’s lifetime, which is assumed to be 100 years or less.

The IHTS includes pumps, valves, pipes, heat exchangers, and steam generators. Table 9 suggests that the dominant contributors to external sodium leakage pertain to the steam generators, isolation valves, and IHTS pumps. These events have tabulated rates on the order of 1E-6 per hour, yielding annual rates almost on the order of 1E-2. In [21], the SG failure values were qualified as having been taken from a system that used water coolant (not sodium), and the valve failure rates were estimated based on zero observed failures. The latter are, in the words of Eide et al., “most likely conservative.”

Table 9. Failure rates for components in a sodium working fluid [21].

Component	Failure Rate (per hour)	Failure Rate (per year)
IHX		
Shell leakage (external)	1.00E-6	8.76E-03
Tube leakage (per unit)	1.00E-6	8.76E-03
Isolation		
Motor-operated valves (external leakage)	5.00E-7	4.38E-03
Pump		
Electromagnetic pump (external leakage)	3.00E-6	2.63E-02
Piping		
Piping leakage per foot	3.00E-9	2.63E-05
Steam Generator		
SG shell (external leakage)	1.00E-06	8.76E-03
SG tube (per unit)	5.00E-06	4.38E-02

Note: Certain details on the calculations and data underlying the above numbers are provided in [21].

As mentioned, the desired probabilities for IHTS failure events may be less than 5E-3 per year. For a given component failure rate, periodic examination can, in principle, drive the probability of an actual leak to below a desired target value by detecting conditions that signal an incipient leak; it is just a question of how often such inspections would be needed. One option is to derive an inspection interval $T_{inspection}$ based on the tabulated failure rate λ, such that $\lambda T_{inspection} << 1$. A sufficiently frequent inspection interval provides confidence that failure is not about to occur. For example, if the rate of developing leakage (i.e., λ) is on the order of 1E-2 per year, then monthly inspection interval will push the external leakage event occurrence below 1E-3 within the inspection period.

Another option is to analyze historical data that reported no observed failure (such as those that produced the artificial failure rates cited in Table 9) and evaluate whether an engineering basis exists for such numbers. A third option is to select material and welding techniques (for the IHTS piping) that have a demonstrated leak-development probability, including uncertainties, lower than the desired probability. This would require demonstration of adequacy via testing or operating experience.

In summary, the discussion of target allocation has yielded targets for leak occurrence and response, though the actual best-estimate rates have yet to be assigned. A quantitative discussion indicated how the event sequences would influence the frequency of these targets and was supplemented by an investigation into leak rates for various IHTS components. Condition monitoring and inspection can reduce the probability of event sequence occurrence. The event diagrams assumed a detection likelihood of 99%; however, if detection is not nearly as effective, the likelihood of event occurrence will increase. These discussions indicate a necessary balance between monitoring strategies and anticipated rates of occurrence. Subsequent sections afford greater insight with respect to inspection and monitoring.

7.4 Degradation Mechanism Assessment

As described in Section 5.1.2 of this report in regard to completing a DMA, the developer of an RIM program for a LWR can refer to the Mandatory Appendix VII, Supplements for Types of Nuclear Reactor Facilities, found in ASME Section XI, Division 2. This section is not currently populated for an SFR, nor is it endorsed for use in RIM program development. However, an addition to ASME Section XI, Division 2 has been approved for incorporation into Article VII-2 which provides SFR degradation mechanism attributes and attribute criteria [22]. Although this has not yet been published by ASME nor endorsed for use by the U.S. NRC, it represents a source of information readily applicable to the SFR example case.

Degradation mechanism information from the draft ASME Section XI, Division 2 code change action [22] is presented for each of the identified degradation mechanisms evaluated for this example case.

7.4.1 Thermal Fatigue

Cracking can initiate in the base metal, heat-affected zones, and welds whenever there is potential mixing of hot and cold fluids, or inflow of a cold fluid into a hot fluid location (possibly during a reactor scram). This may also be the case when a thermal transient occurs due to a relatively rapid temperature change.

The possibility of a sudden mixing of hot and cold fluids increases with the occurrence of operational transients, especially a reactor scram. The PRISM preapplication safety evaluation report (NUREG-1368) includes discussions on thermal shock, thermal fatigue, thermal stratification, and other issues regarding mixing. However, these are focused on the PHTS or the reactor vessel and internal structures.

The IHTS design does not introduce colder sodium flows (or other fluids) from other systems in response to operational transients. This eliminates one degradation mechanism attribute stemming from thermal-fatigue-type degradation mechanisms. Normal shutdown operations require that the power be reduced in a manner that limits thermal transients. Operation of the IHTS should allow for maintaining a relatively constant flowrate for normal operational transients, thus limiting the temperature changes and reducing the potential for thermal transients. For reactor scrams, it is assumed that the IHTS flows will be proportionally reduced, with PHTS flows performed to reduce stress on the reactor vessel and internal components, including the IHX.

With no introduction of a colder sodium flow to the IHTS, along with the ability to control IHTS flows during normal and transient shutdowns, thermal fatigue will not be a degradation mechanism that is monitored in this example RIM strategy.

7.4.2 Vibration Fatigue

Cracks can initiate in base metals, heat-affected zones, and weld materials whenever there are high flow velocities and structural natural frequencies in the range of flow-induced vibration frequencies. Mechanical fatigue can occur when a component (e.g., a pump) induces the vibration source.

The PRISM preapplication safety evaluation report [14] identified vibration caused by the electromagnetic pump that circulates the IHTS sodium as a potential degradation mechanism requiring further consideration. Section 5.5.5 of the PRISM report [14] specifically stated that a subsequent review will need to include evaluation of the natural vibration frequencies of the IHTS in order to preclude the intermediate pump from causing harmful resonance vibrations in the IHTS, as little information existed on the subject at that time. A review of newer information on designs similar to PRISM identified no additional publicly available information.

Although the final design of the IHTS will likely address the consideration, vibration fatigue will be considered a degradation mechanism that must be considered in the example RIM strategy. It is also recognized that no analysis has yet been performed to provide information on specific monitoring locations in either the IHX or the IHTS piping.

7.4.3 Corrosion

High flow velocities can cause wall thinning (surface treatments can reduce these effects). Dissolution of surface alloy metals has been shown to occur when the operating temperatures of liquid-metal systems are >400°C and the sodium purity is insufficient, especially as it pertains to dissolved oxygen control.

Discussions on the sodium flow velocity for the PRISM reactor are predominantly based around internal reactor vessel flows of the primary sodium. The same mechanism in the piping can result in wall thinning when there is sufficient fluid velocity at the boundary layer of the sodium with the internal pipe wall surfaces. It is assumed that sufficient studies will be performed to ensure that the IHTS design precludes the occurrence of wall thinning due to high flow velocities. For systems in which wall thinning may be expected, the design is expected to account for wall thinning in thickness calculations for the pressure boundary pipe.

IHTS operating temperatures in the hot leg to the SG are expected to exceed 400°C, increasing the potential for dissolution of surface alloy metals. The potential for corrosion is enhanced when the levels of dissolved oxygen are high, so greater importance is placed on maintaining oxygen impurity levels at the desired levels.

The PRISM reactor uses a cold trap to remove oxygen impurities in the primary sodium. The cold trap can also be used to monitor the impurity levels in a sodium loop. Use of a cold trap in the IHTS is not specifically identified for the PRISM. However, for the purposes of this example RIM strategy, a cold trap is assumed to be used for controlling and monitoring impurities in the IHTS.

Although the PRISM reactor employs a means of managing oxygen impurity levels, corrosion may potentially exist in the hot leg portion of the IHTS piping from the IHX to the SG. Thus, this is a degradation mechanism that must be accounted for in the example RIM strategy.

7.4.4 High-Temperature Degradation

The ASME Section XI, Division 2, code action that presents SFR degradation mechanism information [22] identifies several degradation subtypes that are associated with the high temperatures (>375°C [for ferritic materials] or >425°C [for austenitic SSs]) that can occur at surfaces in contact with the liquid metals and are enhanced when the sodium purity (especially dissolved oxygen) is insufficient and stress is induced. The resultant degradation can take the form of either cracking or ductility reduction in the base metal, heat-affected zone, and welds.

Additionally, stress relaxation cracking can initiate cracking in heat-affected zones of austenitic alloys with large grain sizes, high residual stresses from manufacturing, and certain geometries (e.g., notches) that cause stress concentrations.

The PRISM IHTS design was not yet mature enough for determining whether locations exist where manufacturing will lead to stress relaxation cracking, or whether a geometry will be present that increases the potential for such cracking. The lack of design maturation is not intended to lessen the importance of the presence of stress relaxation cracking, as this is an important attribute to address during the manufacture of piping and components for materials that operate under susceptible temperatures. A RIM strategy developed in accordance with ASME Section XI, Division 2, does offer a mechanism for addressing an attribute that can be managed during manufacturing; this is accomplished by introducing a design requirement governing how a material is used in SSC fabrication. In the example RIM strategy, stress relaxation cracking will not be monitored for.

Other high-temperature degradation mechanism subtypes to be assessed are affiliated with creep. The PRISM Preliminary Safety Information Document [15] presents several discussions on the effects of creep, but these discussions are limited to the reactor vessel internal structures and fuel. The regulatory review documented in the Preliminary Safety Evaluation Report [14] stated that additional analysis is required to address considerations associated with creep, creep rupture, and creep fatigue when it comes to longer service lifetimes at higher operating temperatures.

Although at the time of submitting the Preliminary Safety Information Document [15] and the Preapplication Safety Evaluation Report [14], ASME Section III did not address the use of materials at the higher operating temperatures expected for PRISM, the IHTS was to be designed to meet all ASME Section III requirements. In the time since, ASME Section III, Division 5 has been published, providing requirements on the design of components that will be exposed to higher operating temperatures.

7.4.5 Degradation Enhancement Phenomena

The ASME Section XI, Division 2 code action [22] also identifies several degradation mechanism subtypes within the degradation mechanism phenomena category. Creep strength and ductility in the base metal and welds were shown to be lessened under specific combinations of conditions. Among the primary influences were the neutron irradiation environment, the impact of sodium chemistry on ferritic steels, and the presence of notches, stress concentrators, and temperature gradients.

The IHX includes secondary sodium fluid, and although it is inside the reactor vessel, the IHTS will not be evaluated for the degradation mechanism subtype of neutron embrittlement. Also shown in the degradation mechanisms table being added to the ASME Section XI, Division 2 code action for SFRs is the subtype called liquid metal embrittlement (LME), which can reduce ductility in the base metal (especially in ferritic steels without appropriate stress relief treatment) [22]. For this example case, the IHTS design was not reviewed in enough detail to identify locations with LME attributes such as notches, stress concentrators, or areas with potential exposure to slow strain loading. Thus, LME will not be considered in the example RIM strategy.

The degradation mechanism attributes table identifies caustic stress corrosion cracking (CSCC) as a degradation mechanism phenomenon subtype potentially seen in base metals, heat-affected zones, and welds in austenitic SSs and low alloy steels. The PRISM preapplication safety evaluation report [14] does identify the IHTS as a closed-loop system with an expansion volume integral to the SG and argon cover-gas. A system design will entail a stress analysis to identify stress locations in piping and components and to determine whether degradation mechanisms such as CSCC may be present. Such analysis was unavailable to review for this example case, and CSCC will be assumed to be absent. Thus, monitoring of CSCC will not be required in this example RIM strategy.

7.4.6 Deformation

Stress relief of the pressure boundary metals is a deformation subtype that can occur in systems where the operating temperature exceeds 375°C (for ferritic materials) or 425°C (for austenitic SSs), and where secondary stress is present. Ratcheting effects may may become an important consideration when cyclic stresses result in an incremental inelastic deformation and shakedown does not occur.

A NUREG review to evaluate deformation found that, as part of addressing General Design Criteria 31 (fracture prevention of reactor coolant pressure boundary), the preapplication for the PRISM reactor stated that the reactor vessel, IHX, and reactor closure head are to be fabricated from materials capable of meeting the deformation and fatigue failure modes. In the absence of additional information on the design, deformation of the IHTS and associated components will be assumed inapplicable to this example case.

7.4.7 Looseness

Looseness of bolting, resulting in a reduction of the bolting tension can occur, especially in the absence of a rotation lock.

As discussed above, the IHTS piping is assumed to be welded to the reactor top plate, which is then flange bolted to the reactor top head. The bolting is associated with the reactor vessel and not the IHX. The PRISM safety evaluation and safety analysis documentation provides insufficient information to conclude that other potential locations such as the rupture disks and isolation valves will feature bolting associated with the pressure boundary of the IHTS.

With the absence of a confirmed bolted connection in the scope of the IHTS pressure boundary, looseness of bolting is not a degradation mechanism to be monitored in the example RIM strategy.

7.4.8 Spatial Phenomena

Spatial phenomena are identified in the degradation mechanism attributes table, along with a note stating that loose parts or areas with restricted access are not degradation mechanisms but rather conditions that should be monitored because they can lead to degradation.

The PRISM design is to include the use of a Loose Parts Monitoring System that meets the intent of Regulatory Guide 1.333, “Loose Part Detection Program for the Primary System of Light-Water Reactors.” No information on the design and function of the Loose Parts Monitoring System was found, indicating a consideration that is unique to the PRISM design and requires special attention in being included in the RIM program. Nor is the physical plant configuration fully established to enable identification of any additional considerations when it comes to obtaining access for inspecting for loose parts within the IHTS. Therefore, a degradation mechanism specifically associated with spatial phenomena is not considered in this example case.

Table 10 summarizes the assessment of IHTS degradation mechanisms. Among the primary degradation subtypes to monitor for are corrosion, vibration, and cracking due to high-temperature degradation mechanisms.

Table 10. Summary of the degradation IHTS mechanism assessment.

Mechanisms	IHX	Piping	Pump
Thermal Fatigue	N	N	N
Vibration Fatigue	Y	Y	Y
Corrosion	N	Y	N
High-Temperature Degradation (creep)	Y	Y	N
Degradation Enhancement Phenomenon	N	Y	N
Deformation	N	N	N
Looseness / Spatial Phenomena	N	N	N

7.5 Development of Reliability Integrity Management Strategy

Figure I-1.1-2, “RIM Program Development and Integration,” from ASME Section XI, Division 2 [2], shows a flowchart representing the key activities for developing a RIM program. The flowchart identifies the development of RIM strategies following the determination of reliability targets and the assessment of degradation mechanisms. In addition to these key activities, other factors and inputs influence the development of RIM strategies, including those factors that affect SSC reliability (see Section 4 of the main report).

7.5.1 Considerations for Condition Monitoring

After assessing the degradation mechanisms to which the IHTS is subject, an appropriate MANDE approach is assigned. For the example RIM strategy, it is only intended to assess those MANDE types that would be considered for the IHTS, not to present a quantitative basis for the technique’s effectiveness. The following is a discussion on the various monitoring techniques applicable to the degradation mechanisms presented in Table 10. When relevant, multiple approaches are discussed, though these may not actually show up in the final strategy.

7.5.1.1 Vibration Fatigue Monitoring

In this example RIM strategy, the need for vibration monitoring stems from having insufficient information on the natural frequencies of vibration that are encompassed in the IHTS design and may result from resonance vibrations created by the electromagnetic IHTS sodium pumps. Though this consideration may be sufficiently evaluated in current designs similar to PRISM, it will be included in our example RIM strategy in order to help demonstrate the expectations of a given RIM strategy.

The resulting degradation is expected to take the form of cracking in the base metal, heat-affected zones, and/or weld. The applicable NDE method depends on where the crack is expected to initiate. Identification of cracking on the external surface may necessitate techniques (e.g., surface examinations) in addition to a volumetric technique, while internal crack initiation may entail a volumetric examination as well as potential digital imaging when internal surfaces can be exposed for visual examination.

The locations susceptible to vibration and those where vibration should be monitored were not analyzed in this example RIM strategy. As the pumps are on the cold-leg side of the IHTS from the SGs to the reactor vessel, selection of the monitoring locations centered around the cold leg, with vibration monitoring instrumentation being installed at multiple cold-leg locations, along with the IHTS pumps (for pump performance monitoring and for comparing the modeled vibration against the actual observed vibration).

Depending on the analyzed susceptibility to vibration, monitoring of the cold leg may include NDE in tandem with direct vibration monitoring. In this example RIM strategy, cracking resulting from vibration will be bound by monitoring for cracking induced by other degradation mechanisms, and monitoring for vibration fatigue will focus on vibration monitoring.

7.5.1.2 Corrosion Control

Online corrosion monitoring is not currently performed, as all known measurement techniques are technically challenging and time consuming to implement, due to the harsh process conditions involved (e.g., high temperatures and gamma radiation levels). One approach may be to employ corrosion probes or manually extract a test sample and then apply metallographic analysis or in-situ visual observation. The sodium can also be sampled for testing; however, the concentrations of dissolved metals may be too low to effectively measure [23].

Given the limitations of directly monitoring for corrosion, monitoring the coolant for oxygen levels (a primary contributor to the onset of corrosion) becomes a viable option, as it helps prevent corrosion. Chemistry control of sodium can include multiple chemical parameters in addition to oxygen levels and may differentiate between the primary and the secondary sodium. Monitoring of the secondary sodium is assumed to be performed as part of a separate piping loop that directs a portion of the IHTS sodium through a cold trap. This is not meant to imply that it is the only approach to monitoring sodium chemistry, as multiple locations and methods may in fact exist.

Corrosion control could include monitoring for leakage from the feedwater within the SG, as introducing sodium/water reaction products enhances the chances of corrosion. Hydrogen gas is produced during the exothermic reaction when sodium and water are exposed to each other. The strategy of detecting hydrogen has also been tested in terms of placement in the SG cover-gas space or directly in the sodium [23].

7.5.1.3 Volumetric Examination and Monitoring for Cracking and Creep

Creep that develops under a combination of temperature and stress over time is characterized by deformation, is especially prevalent in higher temperatures, and can lead to cracking. The condition of creep can be monitored via destructive testing of surveillance specimens, potentially identifying degradation prior to the onset of cracking. Such monitoring requires that the specimen be placed in the same heat transfer fluid and under the same operating conditions as the material being monitored. This could entail utilization of a separate flow loop off the IHTS containing the surveillance specimens, which are then removed under prescribed service times. The key to the use of surveillance specimens is ensuring that the locations of the surveillance specimens apply similar stresses and environments.

NDE for sodium-containing systems, as well as for advanced reactors in general, will focus on detecting cracks in hard-to-access components and in regions readily accessible to probes deployed in the cooling medium. A nondestructive means of identifying cracking may require multiple means of monitoring and detection, such as ultrasonic testing (UT), visual examination (VT), and eddy-current testing (ECT). Cracking can occur in the base metal but is most likely to be seen around welds, which can contain fabrication defects, porosity, and a lack of fusion whose presence may initiate damage. Additional influences stem from material property differences between the weld deposits and heat-affected zones, increasing the potential for cracking and also accelerating crack propagation.

Creep and other high-temperature degradation mechanisms that lead to cracking can be mitigated during design by selecting the proper materials and ensuring the identified degradation mechanisms are understood and accounted for during manufacturing. As an example, operating experience from the Fast Breeder Test Reactor, which identified that manufacturing defects in valves led to sodium leaks [23], can be factored into today's advanced reactors so as to reduce the potential for degradation. The design can also be affected by ensuring access to the area of interest in conducting the necessary NDE. Both these aspects, when specifically considered to achieve the reliability target, become part of the RIM strategy.

As a result of higher-temperature operating environments, the current technologies employed for MANDE in advanced reactor designs are being further developed. VT of the IHTS piping can be performed on the external surfaces, including via remote visual equipment, but becomes more difficult for cracks that initiate on internal surfaces, as draining may be required due to the sodium being opaque. Remote visual technologies may represent a solution for examining the IHTS piping inside the reactor enclosure when specific access into the space between the pressure-retaining piping and the guard pipe is required.

Technological advancements for UT and ECT are being made on higher-temperature pressure boundary systems. However, more remains to be done, as the techniques have not yet been fully demonstrated for deployment in advanced reactors. In addition to enhancements made to these techniques, new parameters of monitoring have also shown promise at affording greater sensitivity to the degradation mechanisms of interest. Once such parameter is the measurement of acoustic harmonics that have proven sensitive to the material changes caused by mechanisms such as cracking and creep [23]. Among the challenges in using acoustic harmonics are sensor material survivability in the operating environment (radiation, temperature, and coolant compatibility) and improving the interpretability of the measurements.

Access to locations that require monitoring must also account for whether the NDE requires the reactor to be shut down, or whether remote capabilities can be used. Assuming successful deployment of the technologies, locations identified for crack monitoring will be directly examined using ECT, UT, or a combination of the two, as driven by the location and configuration. Acoustic harmonics may be used to supplement direct examinations, either to reduce uncertainty or address other considerations such as the reliability of the technology for the specific location, or the time between examinations.

7.5.1.4 Continuous Leakage Monitoring

Sodium leak detectors afford an indirect means of identifying degradation by detecting sodium leaks, which indicate that degradation has occurred or remains underway. Due to limited access and the ability to perform some of the examination activities on the IHTS, the ability to continuously monitor for leakage from the IHTS has increased in importance.

The PRISM design includes sodium-to-gas leak detection capabilities for the IHTS piping and components. Sodium aerosol or contact-type detectors monitor the pressure boundary and scan for valve stem leakage. In addition to detection near the leakage source, cable- or spark-type plug detectors monitor the collection of pooled sodium beneath major components.

To prevent leaks from growing into large pipe breaks or ruptures, the location and the number of sodium leak detectors should be optimized to ensure that any leaks are detected soon after they form[m]. The detection must afford sufficient time for taking appropriate action to locate the leak, isolate it, and potentially drain the location of sodium to directly stop the leak from growing. When an indication is received, the plant operators must confirm and locate the leak. In some cases, direct observation is possible, though it may be necessary to use remote means such as cameras or smoke detectors (when the leak is big enough to produce a detectable amount of smoke).

The PRISM design cites the use of aerosol and direct contact detection. At present, multiple types of sodium leak detection can be used. The benefits and challenges of each must be weighed when estimating the capability to detect leaks and take action so as to reduce their impact. Successful detection/response is factored into evaluating potential achievement of the IHTS performance target.

Several methods and emerging technologies exist for monitoring sodium leakage, a few of which have already been selected for potential application to the example RIM strategy.

Plugging Filter Aerosol Detector
A plugging filter aerosol detector (PFAD) utilizes the differential pressure drop across a filter that is collecting deposition of aerosol sodium released by the leak. PFADs can be used in larger areas containing the equipment being monitored, or can be placed in close proximity in order to sample gases between the pipe/component and the thermal insulation.

Sodium Ionization Detector
As an alternative to the PFAD approach, a sodium ionization detector (SID) detects aerosols that are thermally ionized when deposited onto a heated filament, alerting the user upon exceeding a predetermined concentration level.

Cable Detector
A cable detector directly detects sodium leakage without requiring the accumulation of aerosols. It consists of an SS-sheathed mineral-oxide-insulated cable with holes perforating the outer grounded sheath, where the presence of sodium lowers the resistance of the electrical current.

Electrical Contact Detector
For valves with a bellows-seal, an electrical contact detector, or spark plug detector, is placed between the bellows and the backup steam seal, where a sodium leak will short circuit the electrical contact of a detector probe and the exposed end of a central wire. The presence of sodium in the open end provides a short circuit between the insulated electrode and the grounded sheath.

When a means of continuous monitoring (e.g., use of leak detectors) is credited for achieving a reliability target, the use and capability (e.g., the expected reliability of detection) of the monitoring equipment is identified in the system design basis as a design requirement so as to ensure that the basis for use is incorporated into the plant's configuration management system and not incidentally removed by future design changes. As an example, the RIM strategy may establish a requirement on the capability to detect small leaks, such as "should detect a leak with a rate of 2 grams per minute within 16 hours of leak initiation." The basis for the leak size and the timeliness of the detection supports the response times for locating and correcting the leak.

[m] A formal analysis of leak detectors is not a part of the example RIM strategy as this strategy is intended to highlight the process and key aspects of developing a RIM strategy and not to provide a basis for why some MANDE activities are acceptable or unacceptable.

Depending on the IHTS design, leakage monitoring can be supported through sodium level monitoring. The Fast Flux Test Facility included a means of monitoring the sodium level in both the reactor vessel and IHTS. The Molten Salt Reactor Experiment employed multiple means of monitoring sodium levels, including the use of a float level system, conductivity probes, and ultrasonic level probes [23]. As these were test/experimental facilities, instrumentation throughout the reactor and supporting systems may have been more prevalent than in an operating reactor. However, experience in the use of these level monitoring means, coupled with potential improvements learned from these facilities, may lead to a reliable means of sodium level monitoring that can support continuous leakage monitoring for the IHTS. Although level monitoring could be part of a RIM strategy to monitor leakage, the example RIM strategy is instead focused on the use of leak detectors.

7.5.1.5 Condition Monitoring of the Intermediate Heat Transfer System Heat Exchangers

The IHX presents a unique configuration, as it is supported by the riser and downcomer piping at the IHX top flange and sits inside the reactor vessel, where materials are exposed to the hot and cold primary sodium pools within. The PRISM preapplication safety evaluation report [14] states that the IHX will be analyzed and designed such that it will not become damaged or cause a malfunction as a result of flow-induced vibration. It further states that the dominant failure mode is creep fatigue/damage in the upper, hotter portions of the IHX, and that it is expected to be analyzed and designed to reduce the creep effects.

Although the IHX design utilizes an expansion bellows to absorb the thermal growth between the tube bundle and the IHX downcomer, the available information on the IHX design is insufficient to conclude that stresses at the top of the IHX and the reactor flange will be eliminated; thus, one condition to monitor will be the absence of cracking at the IHX top flange and its attachment to the downcomer/riser piping. Vibration monitoring at this location is part of the example RIM strategy, which provides indirect monitoring for degradation at this location. However, a more direct means should be included.

As technology is improving in regard to performing UT on higher-temperature surfaces, greater opportunities are being afforded for conducting UT. The periodicity of UT is dependent on the deployment of high-temperature capabilities and operating status of the reactor. For the purpose of this example RIM strategy, it is assumed that a reactor shutdown will be planned that places the reactor in a condition that allows for UT every 6 years.

The flow-induced vibrations and creep fatigue are a topic to address for the inaccessible portions of the IHX located within the reactor vessel, or the IHX internals. The PRISM IHX design states that the entire IHX can be replaced if needed, though this may not be the case for similar SFRs currently being designed. No design information on the reactor vessel top flange that provides access to the internals was found. An inspection port can provide access for certain limited examinations, including VT; however, this should not be a planned periodic condition monitoring activity, and should be conducted only as needed.

Because the primary degradation mechanisms affect inaccessible portions of the IHX, an alternative may be to use a surveillance specimen that can be removed for destructive examination and to acquire information on the structural integrity of the materials inside the reactor vessel. This may not require access to the reactor vessel if a separate specimen loop can be designed that pulls hot pool sodium into a separate circuit that exposes the surveillance specimens to similar operating and environmental conditions. Operation in a similar operating environment includes ensuring that the specimens are subject to similar stresses. This presents an additional challenge to design.

7.5.2 The Reliability Integrity Management Strategy

Several methods of IHTS condition monitoring were presented as they apply to the degradation mechanisms expected during IHTS operation. Because identifying strategies for meeting reliability targets is not a prescriptive process, the RIM program can be flexible in response to case-specific needs provided it ensures that it accounts for the degradation mechanisms applicable to each SSC that falls under the RIM program scope. Additionally, not all of the presented methods need to be included in the example RIM strategy and further selection can be based on the effectiveness in being able to detect the specific degradation mechanism or condition that leads to degradation. Therefore, each method is presented along with a discussion on the level of confidence it warrants.

7.5.2.1 Corrosion Monitoring

As already discussed, dissolved oxygen is a primary contributor to corrosion. Smith [24] discusses the following primary methods for monitoring, either directly or indirectly, the oxygen concentrations within liquid-metal sodium:

1. Vacuum distillation technique
2. Mercury amalgamation method
3. Oxygen solubility at cold trap (Na-Na_2O equilibrium) temperature
4. Vanadium-equilibrium method
5. Electrochemical oxygen meters.

Of these, Smith states that the vacuum distillation and mercury amalgamation methods are tedious as well as susceptible to large errors in the low ppm range [24]. This leaves options 3, 4, and 5. Cold traps are common for controlling concentrations of dissolved oxygen but are not intended for either direct or real-time measurements. Ensuring that the cold traps maintain the sodium at the desired purity requires some calibration. The difficulties involved with such calibration were discussed by Smith [24], but appear to have been resolved in the subsequent decades. A 2010 report [25] from Argonne National Laboratory indicates that oxygen can be maintained at 1 ppm in operation conditions (600°C).

Electrochemical oxygen meters, discussed in the Sodium-NaK Handbook [26], can monitor oxygen levels, but entail certain temperature constraints that require them to operate in a cooler section of the liquid sodium. Smith [24], in agreement with Volume III of the Sodium-Nak Handbook [26], indicates that these sensors are generally poorly suited for long-term operation. Of course, this was in light of 1970s technology, which may have improved in the decades since.

The vanadium-equilibrium method, another technique for measuring oxygen concentrations within sodium, can be used for calibration purposes to support cold trap techniques [27]. Use of cold traps is likely the most robust approach for maintaining sodium purity, but it must be augmented by calibration or testing to ensure that the cold traps perform as desired. A discussion on cold traps is provided in [28] where Kozlov et al. highlight the observed reliable use of cold traps for the entire lifespan of the BN-350 reactor. Furthermore, some cold traps for the BN-600 reactor have been successfully operated for 35 years [28]. These experiences lend credence to the robustness of cold traps.

As an example of sensor performance, one source describes the study of electrolytic oxygen meters [29]. A collection of 15 meters was tested. Though not all were identical in composition, they were identical in purpose. The test results showed an average of 253 days without failure. For short-lived sensors, the best use may be to support periodic inspections that may verify cold trap performance. Ultimately, sensors should be rigorously tested with an aim toward commercial deployment.

7.5.2.2 Vibration

Vibration was not a confirmed degradation mechanism for the PRISM design. Flow-induced vibration was deemed a topic to address, since it had not been fully analyzed, though there may be several potential sources of vibration within the IHTS, such as the SG internals, possibly the IHX, or flow-induced vibration in the piping system.

In the current LWR-type plants, vibration is predominantly performed through the use of contact-type sensors, using accelerometers attached to the monitored surface. Accelerometers were used to detect leaks in the Experimental Breeder Reactor II SGs [23], and standoffs were utilized to account for the higher temperatures at the monitored locations for the Experimental Breeder Reactor II.

Another means of vibration monitoring is to utilize a displacement transducer, which is an electromechanical device for converting mechanical motion or vibrations into a variable electrical current, voltage, or electric signals. Displacement transducers are not expected to be applicable to the IHTS—especially in locations close to pumps—due to possible electrical interference. Therefore, vibration monitoring for the example RIM strategy will rely on externally placed accelerometers. Locations should be based on engineering vibrational analysis, which is not the focus of the present work.

Accelerometer failure data for use in radiation areas are not readily available; however, given that certain manufacturers offer rad-hardened accelerometers, operational experience certainly exists. One source [30] that documented sensors and equipment reliability as part of a risk assessment gave an accelerometer failure rate of 9.4E-5 per hour), though there was no indication whether this value was for rad-hardened accelerometers.

The locations at which to install the accelerometers will be based on the vibration analysis. For the purpose of the example RIM strategy, the accelerometers will be assumed to be located close to the reactor top head, on the hot leg outside the reactor enclosure near the isolation valve, and downstream of the IHTS pump. As this is for monitoring purposes and identifying the presence of increased vibration over time, setpoints for increased attention will be provided such as for a digital twin but need not trigger an alarm in the control room unless another need is identified.

7.5.2.3 Monitoring for Cracking

Cracking can stem from several mechanisms, including vibration, stress corrosion, and creep, all of which entail different approaches to detection. The DMA identified that potential cracking resulting from vibration and creep should be monitored for.

The assessment was not detailed enough, nor is sufficient design information available, to formally identify those locations most susceptible to creep and vibration as part of developing this example RIM strategy. Therefore, the locations for performing UT are simply assumed to be where the cracking may result from vibration and are the locations on the hot leg that may be susceptible to creep.

One challenge already discussed is the state of the technology for UT in high-temperature locations. Additionally, access to the locations that require monitoring must account for whether the NDE requires that the reactor be shut down, or if remote capabilities can be used.

Typical NDEs for commercial NPPs have included multiple means, such as UT, VT, surface examinations using dye-penetrant testing, magnetic particle testing, and ECT. Advancements in these methods are being pursued, along with the development of new methods that utilize strain gauges and harmonics for in-situ monitoring of creep.

Assuming that new technologies are successfully deployed, the locations identified for the monitoring of cracking will be direct examined using either ECT, UT, or a combination of both, as driven by location and configuration. Acoustic harmonics may be used to supplement direct examinations, either to reduce uncertainty or for other considerations such as the reliability of the technology for the specific location, or the time between examinations.

To maximize reliable examination, ASME Section XI, Division 1 requires a performance demonstration under Mandatory Appendix VII. This is similar to what is expected when qualifying personnel and monitoring under ASME Section XI, Division 2. When coupled with the years of experience and history pertaining to inservice inspection of current LWRs, a basis for and a level of confidence in a RIM strategy can be developed.

A simplified view of crack monitoring can be employed, with confidence in the crack monitoring being focused on weld areas and based on the POD. POD is used when discussing the efficacy of weld inspections, as mentioned in [31] and [32] . As a complete detailed assessment of weld inspections is outside the scope of this example case, put simply, the probability of defect detection varies in accordance with the skill, the methods employed, and accessibility.

7.5.2.4 Leakage Monitoring

Various leak monitoring methods exist, each corresponding to a different metric. Among the various metrics employed are sodium fluid level, water level, acoustic frequency, atmospheric concentration, sodium chemistry, and pressure. Changes to each of these metrics present diverse indications for leakage and contribute differently to the given monitoring strategy.

Monitoring for leakage is based on two types of assessments. The first is from the perspective of leakage between two systems, when said leakage does not breach the system pressure boundary to the atmosphere. Here, leakage may be indicated by pressure or level changes in both systems. From an IHTS leakage perspective, indications of an SG tube leak have already been discussed and does not represent a leak of secondary sodium from the IHTS. Similarly, the IHTS is designed to operate at a higher pressure, which will leak into the primary sodium should a leak in the IHX tubing occur. This can be detected by changes in the expansion tank or by other means of level monitoring in one or both systems.

Development of a reliability target for the example case is more centered around a leak to the atmosphere, and leakage monitoring will rely on the use of a PFAD, SID, and cable detector. The level of confidence in the use of these detectors is assessed as follows:

> **PFAD** and **SID**: These have proven able to detect leaks in both air and nitrogen, on the order of grams/hour [33]. The sensitivity is increased when the liquid sodium leak is above 500°F. Aerosol detector performance is improved with atmospheric circulation rates. Long-term endurance testing of SIDs and PFADs indicates they perform well in typical reactor environments [33]. Technology similar to PFADs can be used to monitor impurities in liquid sodium. A secondary source for sodium aerosol detectors indicates a sensitivity of up to 1 nanogram per cubic centimeter of carrier gas. Typical installations are capable of detecting within 10 minutes [34].
>
> **Cable detector and electrical contact detector**: Both are capable technologies but may be better suited to large leaks or leaks in a confined space [33]. The Monju accident [18] can serve as a basis for performance confidence in these detector methods. The accident sequence was indicated first by a fire alarm (smoke detector), then by a sodium leak alarm that indicated the leak. To stop the leak, sodium was drained, starting at around 22:40 and finishing at around 00:15. In the end, a little more than 1 cubic meter of sodium residue remained, implying a leak rate of 1 cubic meter in 4.5 hours, or 200 g/hour (for the given density of 997 grams/cubic meter). For this calculation of leak rate, detection of sodium aerosols occurred almost immediately, supporting confidence in being able to detect leaks within a single 8-hour shift.

Key aspects that will aid in detection include sensor proximity, atmospheric circulation, and the density of the sensors employed. Note that secondary or diverse means of indicating external leaks can be as simple as smoke detectors and visual inspections. Small leaks are more likely to first be detected by smoke detectors and visual inspections over volumetric measures of sodium inventory.

7.5.2.5 RIM Strategy Summary

ASME Section XI, Division 2 does not prescribe how the RIM strategy is formatted nor how it is applied to the SSCs within the scope of a RIM program. A RIM strategy can apply to a single SSC, a group of SSCs, or the system as a whole. The example RIM strategy developed in this case study applies to the IHTS as a whole. As demonstrated with the evaluation of options to monitor for potential degradation mechanisms in the IHTS example case, some SSCs may have more than one means of monitoring or NDE and the selection of means is dependent on the confidence in a method to reliably detect the degradation.

The IHTS example RIM strategy focuses on three primary MANDE methods that look for the presence of degradation, or conditions that can lead to degradation, prior to the degradation leading to a loss of the IHTS pressure boundary. These include:

1. Monitoring IHTS chemistry to detect unacceptable dissolved oxygen levels that can lead to corrosion
2. Monitoring vibration at specified locations to ensure that the potential for flow-induced vibration doesn't lead to cracking
3. Performing volumetric examinations (UTs) at locations where the onset of creep is a consideration and can lead to cracking.

Leakage monitoring is also a part of the IHTS example RIM strategy. The occurrence of a leak in IHTS does not represent a challenge to the safe operation of the plant since detection and mitigative actions can be taken in a timely manner with a significant available margin to preclude any severe consequences. A basis can be developed for the approach to the detection of sodium that demonstrates the timeliness of detection is such that appropriate steps can be taken to locate the leak and take steps to stop the leak.

The other MANDE methods in the example RIM strategy have acceptance criteria that can be associated with the parameters being monitored. For leakage monitoring, we suggest that the acceptance criteria should not be simply established as 'presence of a leak'. The strategy should include how the location is determined and what means are relied on for alerting plant personnel to the presence of the leak as well as what activities such as calibrations or other tasks (e.g., periodic walkdowns) are necessary to ensure that the equipment for leak detection is operating reliably.

The example RIM strategy has its foundation on the identified degradation mechanisms of the IHTS and the methods that can monitor for them. Information was provided to offer confidence in these methods and to provide a basis for evaluating the targets of the RIM strategy. A demonstration evaluation of the strategy is provided in the next section where metrics are based on assumed reliability values for demonstration.

7.6 Fault Tree Discussion

For this example case, quantification of leak occurrence is based on best-estimate failure rates. First, a useful perspective on the RIM is given by the fault tree in Figure 25, which highlights how the target event can be influenced by a failure to detect ongoing degradation. Here, degradation detection requires monitoring equipment (e.g., sensors) and associated processes that are performed manually or automatically. The processes are supported by programmatic controls (e.g., training, procedures). The "failure to detect" can occur when either the hardware malfunctions or processes fail (e.g., a technician performs NDE incorrectly, operators fail to recognize indications of degradation provided by the sensors).

The following provides an example calculation using assumed reliability values:

- Accelerometers have a manufacturer-specified failure probability of 1E-3.
- The probability of failure to detect a leak from manual processes is estimated at 1E-3, using human reliability analysis techniques based on clear indications, procedure quality, and periodic training. Thus, the *Process & Personal Failures* event is assigned a failure probability of 1E-3.
- *Failure to Detect* can be caused by a hardware failure or a process failure, and is calculated as the sum of the two, giving 2E-3.
- The *Existence of Degradation Beyond Limits* is estimated: the vibration limits sufficient to cause pipe failure may occur 5 times during a plant's lifetime. For a lifetime of 100 years, this gives 5E-2 events per year.
- The probability of the top event, *Failure of IHTS Boundary*, is then calculated by multiplying the two events under the AND gate: 5E-2 x 2E-3 = 1E-4.

The calculated probability of failure of 1E-4 is smaller than the unreliability target of 5E-03, demonstrating the adequate plant performance and the appropriateness of the selected RIM strategy.

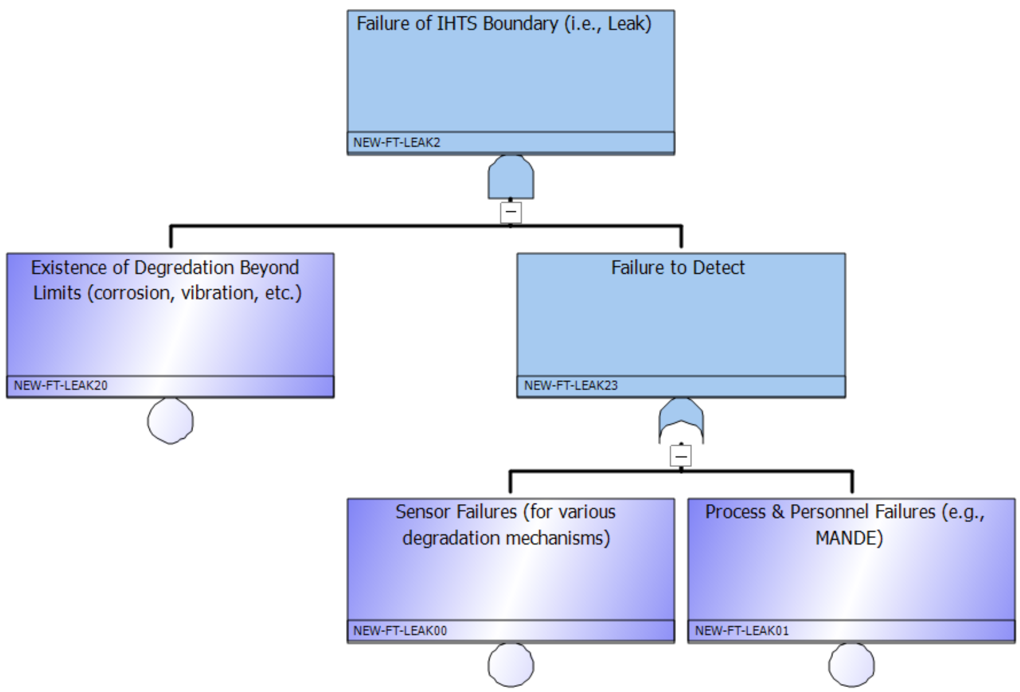

Figure 25: Simple fault tree for leak occurrence.

7.7 SFR Case Study Summary

This example case study demonstrated one possible RIM strategy for the pressure boundary of a system in a sodium-cooled nuclear reactor design. Presentation of this example RIM strategy does not constitute NRC acceptance of this IHTS example RIM strategy nor for a system with similar operating conditions and degradation mechanisms.

The study presented the key steps in establishing the RIM strategy: determination of reliability targets for system parts/components, DMA, and development of the RIM strategy in light of the available monitoring methods and tools. The case study also pointed out aspects to consider when designing the RIM strategy.

8 CONCLUSION

The development and implementation of a Reliability Integrity Management (RIM) program, as outlined in ASME Section XI, Division 2, offers a robust and flexible approach for ensuring the safety and reliability of advanced non-light water reactors. By leveraging probabilistic risk assessment and integrating continuous monitoring, NDE techniques, and proactive maintenance strategies, RIM programs can address the unique challenges posed by the different design features and degradation mechanisms of nuclear reactors.

The U.S. NRC's endorsement of ASME Section XI, Division 2, through Regulatory Guide 1.246, underscores the regulatory acceptance of this approach. The intentional flexibility within the RIM framework allows for the development of customized strategies that cater to the specific needs of various reactor designs, thereby promoting reliability and safety while accommodating operational and environmental differences.

Key elements of a successful RIM program include the thorough assessment of degradation mechanisms, the establishment of reliability targets, and the application of advanced monitoring and inspection techniques. The integration of both deterministic and probabilistic approaches ensures a comprehensive understanding of SSC performance and reliability, enabling timely and informed decision-making to prevent failures. The Idaho National Laboratory's evaluation of methods for developing reliability targets and assessing RIM strategies further supports the NRC's review process and aids in the continuous improvement of these programs. By addressing factors that affect reliability, such as SSC design, material selection, fabrication procedures, operating practices, and maintenance activities, the RIM program provides a holistic approach to managing the lifecycle of critical nuclear power plant components.

The implementation of a RIM program, as detailed in this report, provides a comprehensive approach to ensure the reliability and safety of plant components through a structured process of identifying degradation mechanisms, establishing reliability targets, and developing appropriate monitoring and maintenance strategies. The report illustrates the application of the RIM methodology using two example case studies: a portion of the Reactor Coolant System for a LWR and the Intermediate Heat Transfer System for a sodium fast reactor.

In the LWR case study, the RIM strategy involved the identification of various degradation mechanisms, such as thermal stratification, thermal transients, and primary water stress corrosion cracking. Continuous monitoring, periodic NDE, and advanced leak detection systems were integrated into the strategy to ensure the reliability targets were met.

The SFR case study focused on the intermediate heat transfer system, addressing degradation mechanisms specific to high-temperature environments, such as creep and vibration fatigue. The RIM strategy incorporated monitoring for dissolved oxygen levels to control corrosion, vibration sensors to detect flow-induced vibrations, and volumetric examinations for high-temperature degradation. Continuous monitoring for the presence of sodium leakage was also included to identify leakage in a timely manner that allows for prompt mitigation.

Both case studies underscore the importance of a holistic approach to RIM, combining direct and indirect monitoring techniques, performance demonstrations, and the integration of reliability targets into the overall plant safety and operational strategy. The flexibility of the RIM methodology allows for tailored strategies that address the unique conditions and challenges of different reactor types and components.

In conclusion, the RIM program exemplifies a forward-thinking, performance-based approach that ensures the safety and reliability of commercial nuclear reactors. By combining rigorous technical assessments with regulatory frameworks and industry standards, the RIM program ensures that nuclear reactors can operate safely and efficiently, meeting both regulatory requirements and the evolving needs of the nuclear industry.

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