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Reliability and Integrity Management Scoping Study

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EXECUTIVE SUMMARY

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) 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 pre- service inspection (PSI) and in-service inspection (ISI) program requirements for boiling and pressurized-water-cooled nuclear power reactors. ASME Boiler and Pressure Vessel Code (BPVC), Section XI, Division 2 (BPV XI-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 (BPV XI-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 U.S.; however, it is anticipated that applicants to operate new non-light water reactors may include a RIM program as part of plans for maintenance, surveillance, and periodic testing of SSCs. In preparation for using BPV XI-2 in RIM program development, an assessment of the current state of knowledge of RIM use was conducted.

Experience with the use of RIM stems from two primary activities, the Pebble Bed Modular Reactor (PBMR) Demonstration Power Plant (DPP) intended for operation in the Republic of South Africa, and application of a system-based code (SBC) concept derived to support the development of Japanese fast breeder reactors. Although these RIM related activities applied to reactors that are to operate outside of the U.S., close coordination with ASME BPV personnel resulted in the simultaneous development of the content for BPV XI-2.

The purpose for development of the PBMR RIM program was to establish strategies for the reliability and integrity management of passive components for the PBMR DPP. The technical approach to developing RIM strategies for the PBMR DPP was coordinated with the code development activities of the ASME Section XI Special Working Group for high temperature gas reactors. The pilot study conclusions were submitted to ASME to improve the technical approach considered by the working groups responsible for developing BPV XI-2. When accounting for differences in the safety philosophy and materials/operating conditions, the risk-informed ISI approach helped establish the PBMR RIM approach.

Similarly, application of the SBC concept in Japan, through the development of a fitness-for-duty code by the Japan Society of Mechanical Engineers, was coordinated with the BPV XI-2, culminating in the approval of ASME Code Case N-875, “Alternative ISI Requirements for Liquid-Metal Reactor Passive Components Section XI, Division 3”. Although Code Case N-875 was not approved by the NRC for application in the U.S., aspects of its application are included as Non-Mandatory Appendix A to BPV XI-2.

The RIM approach prescribed by BPV XI-2 establishes individualized strategies (i.e., RIM strategies) that apply monitoring and examination activities—referred to as monitoring and non-destructive examination (MANDE)—for an SSC or group of SSCs within the scope of the RIM program. This approach utilizes certain principles used in forming a risk-informed ISI program under a BPV-XI-1-based ISI program. An early step requires an assessment of the potential degradation mechanisms to which a material is subjected (i.e., a degradation mechanism assessment), and a determination of the monitoring and examinations necessary to detect the resultant degradation. RIM is a living program, as it is continually updated throughout operation and considers operating experience such as degradation mechanisms and expected material conditions.

Development of the SBC approach and the details associated with developing the PBMR RIM program have been beneficial in developing the RIM requirements under Division 2 of ASME Section XI, and a significant number of attributes remain in the 2019 Edition. Through the issuance of NRC Regulatory Guide 1.246, “Acceptability of ASME Code, Section XI, Division 2, Requirements for Reliability and Integrity Management (RIM) Programs for NPPs, for non-LWRs", the NRC endorsed,

with conditions, the use of the 2019 Edition of ASME Section XI, Division 2, for development and implementation of a PSI and ISI program for non-LWR reactors.

Because a RIM program has not been fully implemented in an operating nuclear power plant, the technical basis for the two RIM projects, in combination with information from a case study performed on application of RIM (funded by the US Department of Energy), was used as experience in implementing RIM and forming insights into future implementation. The assessment also considered current actions being performed by the ASME Section XI group responsible for BPV XI-2 because these actions can provide an understanding of the development of a RIM program in accordance with BPV XI-2.

The assessment concluded that there is sufficient insight from the current state of knowledge on RIM to provide recommendations for the future development and implementation of a RIM program developed in accordance with BPV XI-2. Although the recommendations relate to BPV XI-2, addressing the recommendations is not the responsibility of only the ASME but apply to the industry (comprised of reactor designers, ASME volunteers, operators, research organizations, and regulating authorities). Therefore, the following recommendations do not identify a suggested organization for addressing each recommendation.

RIM program scope: BPV XI-2 relies on the owner to determine what SSCs are included in the scope of the RIM program. Although acceptable, there is a lack of guidance on what SSCs should be in the program.

Use of the PRA: RIM-2.4.2 of BPV XI-2 requires the user to employ the PRA model to allocate SSC reliability. Because implementation of RIM entails consideration of the plant design, the PRA must be developed to the extent that its data can be used to determine the ability to achieve the reliability targets, and that it is sufficient for understanding the uncertainties of the RIM strategy. However, in the early design stages, a PRA may be insufficiently mature to effectively allocate reliability targets (CLRs) to a component. Assumptions may need to be made to compensate for the lack of design maturity in systems, including active systems that interface with the passive pressure boundary components within the scope of the RIM program.

Limitation to passive components: Although BPV 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 the determination of RIM strategies. A potential concern with exclusion of the active equipment is that the results may be skewed due to the application of an over- conservative treatment of active components (suggested by BPV XI-2). Including both passive and active SSCs as part of an overall RIM strategy may also result in efficiencies for performance monitoring of important system functions in addition to meeting specific RIM program requirements.

Connection with the NRC Regulatory Guide 1.233: There are connections of RIM with the Licensing Modernization Project (LMP) 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. To ensure consistency across these processes, there should be consistency in the definition, or meaning, of a reliability target, and how it is derived.

Continuous Monitoring: Traditional ISI programs most often utilizes 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 that subsequent action is to be taken that then implements a direct means of identifying degradation. The PBMR DPP RIM program and the presentation of SBC emphasize the appropriateness for an increased reliance on continuous monitoring without a need for periodic examinations such as a volumetric examination, including the primary heat transfer systems that comprise the Class 1 SSCs. In the case of pressure boundary leakage, continuous monitoring often identifies the leak after it occurs. This approach is fundamentally different from the current ISI practice where monitoring strategies are developed and applied to ensure absence of any damage, i.e., the specific goal is to avoid any leaks. It is important to develop sound rationale that leakage monitoring is an acceptable strategy of performance monitoring.

Degradation Mechanism Assessment: BPV XI-2 contains a mandatory appendix that contains degradation mechanism information that is to be considered. However, the appendix currently addresses only a couple of reactor types. Increased priority to populate the information for other reactor types should be taken to ensure that consensus information can be applied consistently across all RIM programs of each reactor type.

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CONTENTS

FIGURES

ACRONYMS

ASME	American Society of Mechanical Engineers
BPVC	Boiler and Pressure Vessel Code
CDF	core damage frequency
CFF	containment failure frequency
CFR	Code of Federal Regulations
CLR	component-level requirement
DPD	differential pressure detector
DMA	degradation mechanism assessment
HTGR	high-temperature gas reactor
INL	Idaho National Laboratory
ISI	in-service inspection
JSME	Japan Society of Mechanical Engineers
LBB	leak-before-break
LBE	licensing basis event
LMP	license modernization project
LMR	liquid-metal reactors
LWR	light-water reactor
MANDE	monitoring and non-destructive examination
MANDEEP	MANDE Expert Panel
NDE	non-destructive examination
NRC	Nuclear Regulatory Commission
PBMR	pebble-bed modular reactor
RG	Regulatory Guide
RIMEP	RIM Expert Panel
PHTS	primary heat transfer system
PSI	pre-service inspection
PRA	probabilistic risk assessment
RIM	reliability and integrity management
SBC	Systems Based Code
SID	sodium ionization detector
SSC	structures, systems, and components
SWG	special working group

Reliability and Integrity Management Scoping Study

1 INTRODUCTION

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) [2] 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 pre-service inspection (PSI) and in-service inspection (ISI) program requirements for boiling and pressurized-water-cooled nuclear power reactors. ASME Boiler and Pressure Vessel Code (BPVC), Section XI, Division 2 [3] (BPV XI-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 [4] (BPV XI-1), for all types of nuclear power plants.

Development of an alternative to the current PSI/ISI requirements result is due to the different design features included in new advanced reactors. These design features often include a different coolant (e.g., sodium rather than water) and higher reactor coolant system operating temperatures. These different design features introduce degradation mechanisms (e.g., creep and thermal stresses) that may necessitate non-destructive examination (NDE) techniques that differ from the current ISI-required examinations prescribed in BPV XI-1.

Documentation on the approach to developing a RIM program can be traced back to the activities of an ASME Section XI Special Working Group (SWG) on high-temperature gas reactors (HTGRs) [5] in the mid-2000s, and to the technical documents used as input to the SWG. The SWG built the initial RIM program code based on lessons learned through developing and implementing the NRC-endorsed risk-informed in-service inspection (RI-ISI) programs of BPV XI-1. This information was used as part of the RIM program development and documentation for the Pebble-Bed Modular Reactor (PBMR) Demonstration Pilot Plant (DPP) [6].

In addition to the development activities for a RIM program, Systems Based Code (SBC) concepts were being introduced in Japan. The introduction of SBC concepts can be credited to the work of Asada et al. [7], [8] conducted as part of Japanese R&D on fast breeder reactors. Several technical papers were written on the fundamentals of an SBC approach and how it could result in the development of an ISI program for a liquid-metal reactor (LMR).

The progress from these efforts supported the basis for the development and approval of BPV XI-2. NRC staff reviewed the 2019 edition of BPV XI-2 and found it to be an acceptable approach for enabling non-light-water reactors (non-LWRs) to develop a PSI and ISI program. Acceptance was provided through the issuance of NRC Regulatory Guide (RG) 1.246, Acceptability of ASME Code, Section XI, Division 2, "Requirements for RIM Programs for Nuclear Power Plants," for non-LWRs [1].

In preparation for using BPV XI-2 in ISI program[1] development for non-LWRs, NRC contracted Idaho National Laboratory (INL) to conduct an assessment of the current state of knowledge of RIM use, focusing on three primary scopes: a pilot study of the application of RIM for the PBMR Demonstration Pilot Plant (DPP) [6], a recent INL case study on the target reliability allocation within the RIM process [9], and SBC application for sodium-fast reactors in Japan [10], [11], [12].

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

2 RIM PROCESSES—GENERAL DESCRIPTION

This report is structured so as to communicate the assessment results for each of the three previously introduced scopes, or processes. The results are divided up into each of the discussion areas. A general description is given here, followed by a more detailed description in subsequent sections.

2.1 Pebble-Bed Modular Reactor Demonstration Pilot Plant

The purpose of the PBMR DPP RIM Pilot Study [6], herein referred to as the PBMR RIM Report, was to establish strategies for the RIM of passive components for the PBMR DPP and to guide development of ASME RIM requirements for modular HTGRs. The study provided recommendations on what combination of strategies were both necessary and sufficient to achieve target reliability goals for passive components.

The pilot study objectives are as follows:

- Provide the technical basis for the PBMR DPP RIM program
- Provide technical input to ASME code development in order to establish RIM requirements for HTGRs
- Demonstrate the feasibility of a technical approach for formulating RIM requirements in an ASME HTGR RIM white paper
- Provide RIM strategies that are consistent with the risk-informed, performance-based licensing approach for PBMR design certification by NRC.

The technical approach to developing RIM strategies in the PBMR RIM Report is based on the ASME Section XI SWG for HTGRs [5]. The pilot study conclusions were submitted to ASME to improve the technical approach considered by the working groups responsible for developing BPV XI-2. When accounting for differences in the safety philosophy and materials/operating conditions, the RI-ISI approach helps establish the PBMR RIM approach.

The PBMR RIM approach is similar to RI-ISI and generally includes the following actions:

- Determines degradation mechanisms
- Applies an inspection requirement that is appropriate for the given degradation mechanism, and calculates the probability of pipe failure
- Removes non-risk-significant exams and adds inspections to some locations included under a traditional ISI program
- Originates from core damage frequency (CDF) and large early release frequency values, and includes a minimum of 10% for ASME Class 1 inspection locations.

2.2 Systems Based Code

Use of conventional codes/standards to design SSCs often results in an excessive design margin, as the margins accumulate and essentially add to each other. The SBC concept, or simply "SBC," enables the optimization of margins for integrity of SSCs and the evaluation of integrity throughout the entire life of the SSCs, from conceptual design to decommissioning [10].

The most active development of the SBC concept in the nuclear power industry has occurred in Japan, where several papers and presentations were communicated by personnel from the Japan Atomic Energy Agency and other companies working to develop a fitness-for-service-type code with the Japan Society of Mechanical Engineers (JSME). These JSME activities were coordinated with the commercial nuclear power industry in the United States, culminating in the approval of ASME Code Case N-875, Alternative ISI Requirements for Liquid-Metal Reactor Passive Components Section XI, Division 3 [13].

ASME Code Case N-875 applies to ASME Section XI, Division 3 and was intended to address reactor systems that contain liquid metals (sodium or sodium alloys) as coolant. Division 3 was never completed, so to meet the international need for development of inservice inspection requirements for LMRs, ASME Code Case N-875 was issued[2]. In the 2021 Edition of ASME Section XI, Division 3 is shown to have been deleted, and its requirements now fall within the scope of Division 2. ASME Code Case N-875 is being used as the basis for development of a Liquid-Metal Reactor supplement in Division 2, planned for possible inclusion in the 2025 Edition. Once the applicable edition of ASME Section XI, Division 2 is endorsed by NRC, Code Case N-875 will be annulled.

Apart from Non-Mandatory Appendix A to ASME Section XI, Division 2, no specific document in the United States describes the usage of SBC in developing an ISI program. In a recent GEN IV International Forum presentation on use of SBC [16], it was communicated that the process steps and actions contained in ASME Code Case N-875 (as well as in BPV XI-2) are integral to the ISI standard for sodium-fast reactors being developed in Japan.

This report's SBC-related content is based on reference presentations and papers, as well as on certain details also contained in ASME Code Case N-875. Thus, the content of ASME Code Case N-875 is still referenced in this report to provide information beneficial to understanding applications of SBC, when such details may not be found in other referenced materials.

SBC entails two primary stages that each provide an evaluation path, in series, to determine whether the SBC process can be implemented. If the two evaluations are determined to meet the code case criteria, the alternative examination requirements of the code case may be applied.

Stage I – Structural Reliability Evaluation

The first-stage evaluation assesses the component's structural integrity under design basis conditions, as well as the potential for failure in light of the identified degradation mechanisms. This evaluation requires the development of component-level requirements (CLRs) based on the allowable degradation limits for the various components. CLRs are described in accordance with the plant safety evaluation, using quantities (e.g., break size) postulated in an accident scenario. If the probability of component failure is determined to meet the CLR, the Stage I evaluation is considered satisfied and users can proceed to Stage II.

Stage II – Evaluation of Ability to Detect Degradation

The first-stage evaluation did not consider the ability to detect a flaw (i.e., ISIs). The second-stage evaluation, on the other hand, does consider the ability to detect a flaw before it exceeds its maximum allowable size. If the Stage II evaluation results are acceptable, the user may implement the alternative examination requirements.

2.3 ASME Section XI, Division 2

The RIM approach establishes individualized strategies (RIM strategies) that apply monitoring and examination activities—referred to as monitoring and non-destructive examination (MANDE)—for an SSC or group of SSCs within the scope of the RIM program. SSCs within the RIM program are those that can directly affect plant reliability, including some non-safety-related SSCs.

Developing a RIM program utilizes certain principles used in forming a RI-ISI program [4] under a BPV-XI-1-based ISI program. An early step requires an assessment of the potential degradation mechanisms to which a material is subjected (i.e., a degradation mechanism assessment [DMA]), and a determination of the monitoring and examinations necessary to detect the resultant degradation.

RIM recognizes the unique qualities that the new reactor designs introduce to the development of an ISI program. RIM provides requirements for demonstrating examination techniques and qualification for personnel who perform inspections and examinations—particularly, requirements that differ from BPV XI-1 techniques that have been implemented for years.

[2] ASME Code Case N-875 has not been endorsed for use by NRC.

Similar to the BPV XI-1 ISI program, the RIM program is considered a living program, as it is continually updated throughout operation and revised. It also considers operating experience such as degradation mechanisms and expected material conditions. The importance of the SSCs and the expected performance results will drive periodic checks and revisions of the RIM strategy.

The PBMR DPP RIM Pilot Study and activities associated with develop the SBC were integral to the development of BPV XI-2. In the case of SBC, it is included as a non-mandatory appendix to BPV XI-2 in order to provide an alternative to the prescribed NDE requirements of RIM.

3 RIM PROCESSES - KEY ATTRIBUTES

3.1 Pebble-Bed Modular Reactor Demonstration Pilot Plant

The technical approach to developing the PBMR RIM program is described by the steps and framework represented in Figure 1 [6].

While this framework identifies seven primary steps, the PBMR RIM Report aligns with six major elements for developing the RIM program. This review of the PBMR RIM Report was performed using a combination of the framework and the major steps, and is supported by a technical presentation on the development of the PBMR RIM program [14].

The six major elements are as follows:

1. Determine plant-level requirements.
2. Apply a risk-informed licensing approach.
3. Determine SSC-level requirements for RIM.
4. Determine the scope of SSCs within the scope of the RIM program.
5. Define the nature and extent of inspections for each SSC in the RIM program.
6. Monitor performance and changes to RIM inputs.

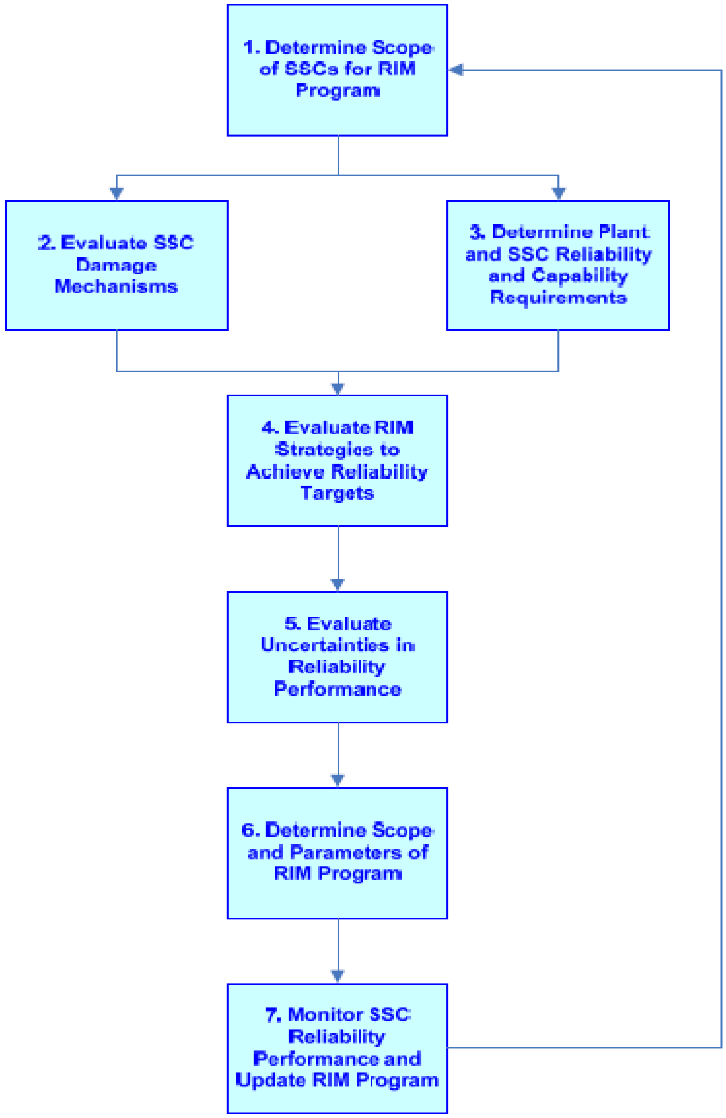

Figure 1. Framework for developing the PBMR RIM program.

At the time of the PBMR RIM program development, experience related to the application of risk-informed concepts pertained to LWRs. Applying that experience to an advanced reactor design required an understanding of the differences between the safety design philosophies of currently licensed LWRs and the PBMR. The PBMR RIM Report states that the safety philosophy of currently licensed LWRs focuses on mitigation and prevention of loss-of-coolant accidents, in which the primary heat transfer system (PHTS) is breached and the integrity of the fuel is subsequently challenged. In contrast, the PBMR focuses on fuel integrity without relying on the performance of active engineered systems.

3.1.1 Plant-level Requirements

Plant-level requirements are the initial point in any plant design. Design requirements (e.g., licensing requirements, selection of codes and standards, and stakeholder requirements) collectively represent the plant-level requirements. These often include special requirements for the plant's reliability and availability performance, transient and load following performance, investment risk, and safety/licensing. Understanding the plant-level requirements is not only essential to commencing design, but also to developing component-level reliability requirements and ensuring that appropriate strategies are employed to meet these reliability targets.

3.1.2 Risk-informed Licensing Approach

The PBMR RIM program recognized that the RIM-type programs of the past were deterministically based, as evidenced through the implementation of 10 CFR 50 general design criteria [2] and the development of ISI programs in accordance with ASME Section XI (prior to the use of risk-informed insights). Development of the PBMR RIM program was intended to apply a risk-informed licensing approach utilizing a plant-specific probabilistic risk assessment (PRA) to determine licensing basis events (LBEs) and information, which will be shared with the designer to assist in the selection of safety-related SSCs.

At the time of developing the PBMR RIM program, U.S. HTGRs were expected to be licensed via a risk-informed approach that uses a plant-specific PRA to determine LBEs and information necessary to identify those SSCs that will be classified as either safety related, non-safety with special treatment, or non-safety-related [15]. For the PBMR DPP, a more deterministic approach was implemented by classifying equipment as safety related, given that SSCs are relied upon to ensure that event consequences remain within the prescribed limits of the design basis and prevent high-consequence, lower frequency events from being considered as part of the plant design basis.

3.1.3 Plant- and SSC-level Reliability and Capability Requirements

For the PBMR, the reliability and capability requirements for the plant and SSCs were derived from regulatory requirements that limit the frequency and consequences of LBEs. In addition to derivation of requirements based on regulatory limits, the PBMR RIM program established Reactor Cavity Cooling System requirements for meeting non-mandatory plant reliability and investment risk management goals.

The following is an example of a reliability goal established in the PBMR RIM program: the frequency of event sequences involving main power system depressurization due to failures of helium pressure boundary SSCs with an equivalent break size exceeding 10 mm shall be less than 10^{-2} per reactor year.

SSC-level reliability goals were established using pipe reliability modeling, with the mean frequency of each component being determined as a function of break size (including uncertainties).

3.1.4 Scope of SSCs in the RIM Program

When considering all the requirements for plant reliability, safety, and investment protection, the scope of SSCs can be large. The scope of SSCs for the PBMR RIM program was limited to SSCs that perform passive safety functions and have a safety classification that requires a level of special treatment in order to meet regulatory requirements.

The PBMR RIM Report includes a full section on the scope selection, with emphasis placed on functions associated with radionuclide control, heat generation and removal, and chemical attack, as well as with maintaining reactor vessel and core geometry.

3.1.5 RIM Strategies to Achieve Reliability Targets

RIM strategies are based on their ability to achieve the degree of reliability and capability necessary to meet the regulatory special treatment requirements. Strategies that address the evaluated degradation mechanisms, including design approaches, are identified for implementation. Additional defense-in-depth may also be implemented, depending on the uncertainties in the development of the reliability and capability requirements. This may result in the addition of different examinations or the increased periodicity of such examinations.

The degradation mechanism assessment performed for the PBMR RIM project was very similar to the assessment performed for RI-ISI programs that meet BPV XI-1. Additionally, uncertainty in predictions of passive component reliability was accounted for.

The impact on reliability that resulted from the type of monitoring and the prescribed RIM strategy NDE was factored into the final reliability targets.

3.1.6 Monitoring SSC Reliability and Updating of RIM Program

The PBMR RIM Report identifies that monitoring SSC performance, including the inputs to decisions that form the basis for the RIM strategies, is critical for providing an effective RIM program and serves as a key element in meeting risk-informed and performance-based regulations.

3.2 System Based Code

As previously stated, SBC consists of a two-stage evaluation process, the satisfaction of which resulting in the application of the prescribed alternative ISI requirements. Figure 2 provides an overview of the logic used to determine the applicability of the alternative examination methods.

The first-stage evaluation is a structural reliability evaluation. Component-level requirements (CLRs), defined as the allowable limits for flaws in components from a safety point of view, are used in the initial determination of a component's potential for failure, prior to crediting the ability to detect a flaw during ISI activities (which is considered in the second-stage evaluation).

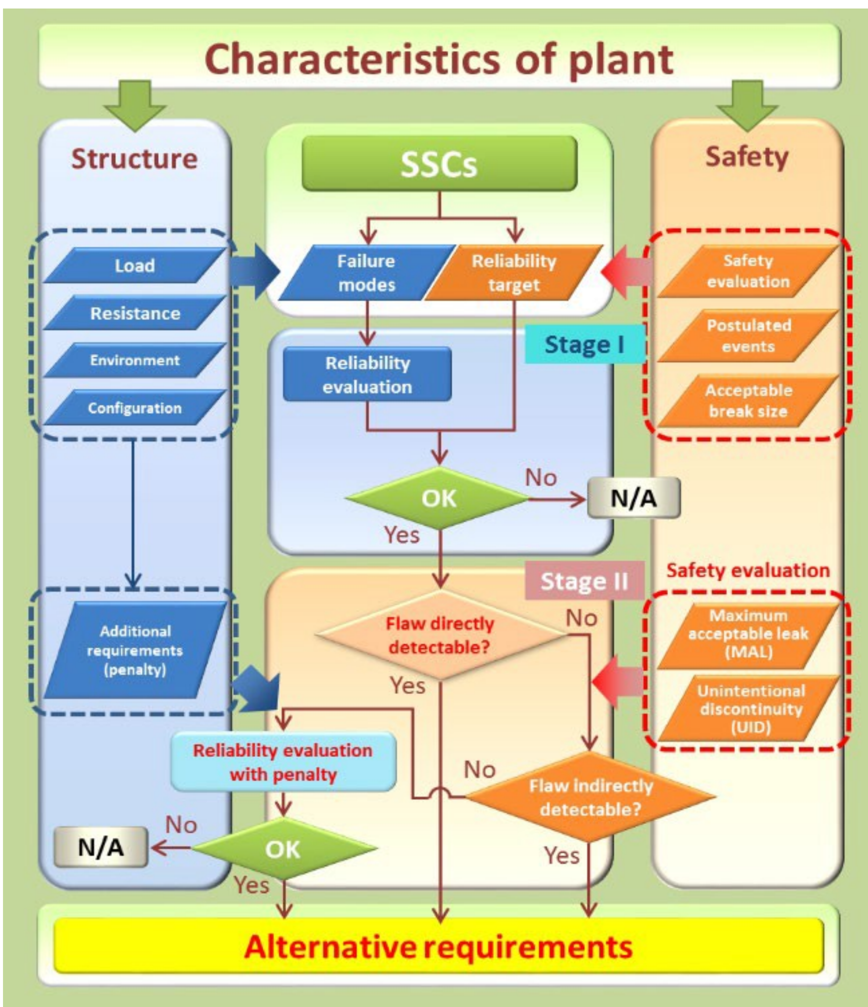

Figure 2. SBC logic flow diagram [16].

Any determination of the postulated degradation mechanisms and potential failure modes to which a component is subjected must quantify the component probability of failure. This probability is compared to the previously determined CLR.

3.2.1 Derivation of Component-level Requirements

Inputs used in determining CLRs consist of deterministic and probabilistic information. Break size and other accident scenario information are understood to be deterministic in nature. From a quantitative perspective, application of ASME Code Case N-875, Mandatory Appendix I [13] is required in deriving component target reliabilities by using CDF and containment failure frequency (CFF) or large early release frequency quantities.

Mandatory Appendix I identifies the key steps necessary for deriving CLRs from plant-level safety requirements. The key steps are as follows:

- Allocate target reliabilities.
- Identify groups of SSCs for which loss of function would have the same effect on the PRA model as was identified during the reliability target allocation.
- Conduct an initial assignment of target reliabilities and impact on plant-level risk.
- Finalize target reliabilities (i.e., CLRs).

Mandatory Appendix I also requires the user to confirm that the PRA for deriving the CLRs is sufficient to perform the derivation. It states that ASME/ANS RA-S-1.4-2021, "Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants" [17], can be used to demonstrate the adequacy of the PRA model; however, it does not require the standard to be used.

3.2.2 Structural Reliability Evaluation of Passive Components

Design parameters that impact the structural integrity of a component are used to determine a failure probability for that component. This information is assumed to be statistical (i.e., quantitative) in nature; if not, the user is directed to establish a data provision approach that can be compared against the CLRs. A procedure for performing this structural reliability evaluation for LMRs is provided in ASME Code Case N-875 Mandatory Appendix II [13], "Procedure for Structural Reliability Evaluation for Passive Components of Liquid-Metal Reactors (LMR)." Although this procedure is specific to an LMR, the approach is generic enough to effectively apply to other reactor types.

The requirements in Mandatory Appendix II are written succinctly, affording little detail. Its sections and subsections are often single requirements, with most of the content being focused on failure scenario modeling.

The structural reliability evaluation determines the component failure modes that can be compared to CLRs through component target reliabilities. If the evaluated reliability is determined to meet or exceed the CLRs, the user may proceed to the next stage of SBC.

The second stage of SBC considers the ability to detect a flaw before it reaches its maximum acceptable size and challenges the ability to safely shut down the plant. The detectability assessment enables both direct and indirect flaw detection.

Direct detection includes conventional technologies (e.g., ultrasonic examination), for which sufficient information is typically available to apply a probability-of-detection perspective. It is less common to detect flaws via indirect means. For indirect examination, SBC and ASME Code Case N-875- utilize two approaches associated with the two terms of maximum acceptable leak and unintentional discontinuity [13, 16].

Maximum acceptable leak can be used to directly compare an indication of a leak in a system pressure boundary against a leak rate that is small enough to enable a controlled shutdown prior to reaching an unacceptable leak rate. Unintentional discontinuity accounts for means that indicate a potential leak without observing or confirming the presence of the leaked fluid, using plant parameters such as coolant temperature and velocity. The ability to apply direct and indirect methods of examination opens up a more effective solution for monitoring the integrity of the pressure boundary, given the various plant designs and features expected in the new advanced reactors.

Inability to detect a flaw does not preclude the user from implementing SBC or the alternative requirements of the code case. ASME Code Case N-875 describes the derivation of additional requirements that can be satisfied and that can exempt the user from having to demonstrate the ability to detect a leak or flaw. The derivation is accomplished by adding margin in the structural reliability evaluation; in SBC, this need to add margin is referred to as a penalty. The inclusion of penalties, such as increasing the load or reducing the resistance in the structural evaluation, ensures that each component under the service conditions maintains functionality, without the need to conduct monitoring or examinations. If it is concluded that a flaw or leak can be detected in sufficient time to safely shut down the reactor, the alternative requirements of the code case can be applied.

3.3 ASME Section XI, Division 2

The flowcharts contained in Appendix I of Division 2 represent the RIM process by BPV XI-2. Key attributes of the RIM process are represented by Figure 3, and include the following:

- Assess potential degradation mechanisms for SSCs within the scope of RIM.
- Identify a reliability target that a plant desires the SSCs to meet, through the allocation of plant-level reliability to the SSC level.
- Identify RIM strategies that employ specific MANDE techniques to maximize detection of both known and potential degradation mechanisms.
- Understand the uncertainties that exist in the ability to detect degradation, and apply additional MANDE—or revise frequencies of MANDE—to further maximize the ability to detect degradation prior to SSC failure.

RIM recognizes that the new reactor designs are likely to involve materials that operate in systems and environments different from those of currently operating LWRs. RIM has built into the process (rather than requiring subsequent technical or regulatory approvals) an evaluation of the adequacy of testing and NDE techniques, as well as the ability to detect degradation in materials operating in these different environments.

A RIM strategy is developed for SSCs within the RIM program that implements MANDE activities for identifying credible and postulated degradation mechanisms. MANDE is evaluated, including a possible performance demonstration to identify the postulated degradation and achieve the target reliability for the given SSC. The MANDE evaluation is supported by a MANDE expert panel (MANDEEP). The subject matter experts comprising the MANDEEP establish appropriate criteria for each SSC in order to ensure that the SSCs are adequately monitored and evaluated prior to the onset of credible damage mechanisms, as well as to ensure that the implemented MANDE can support the established reliability targets for the SSC lifecycle.

The MANDEEP is one of two expert panels that are integral to ensuring that the RIM program effectively achieves the desired reliability of system functions important to safe, reliable plant operation. BPV XI-2 integrates an expert panel as part of the overall RIM program development, documentation, and implementation, and this panel is referred to as the RIM Expert Panel (RIMEP). The responsibilities of the RIMEP are broad, and include concurrence on the program scope, the SSC reliability targets, and the RIM strategies to be implemented by the program.

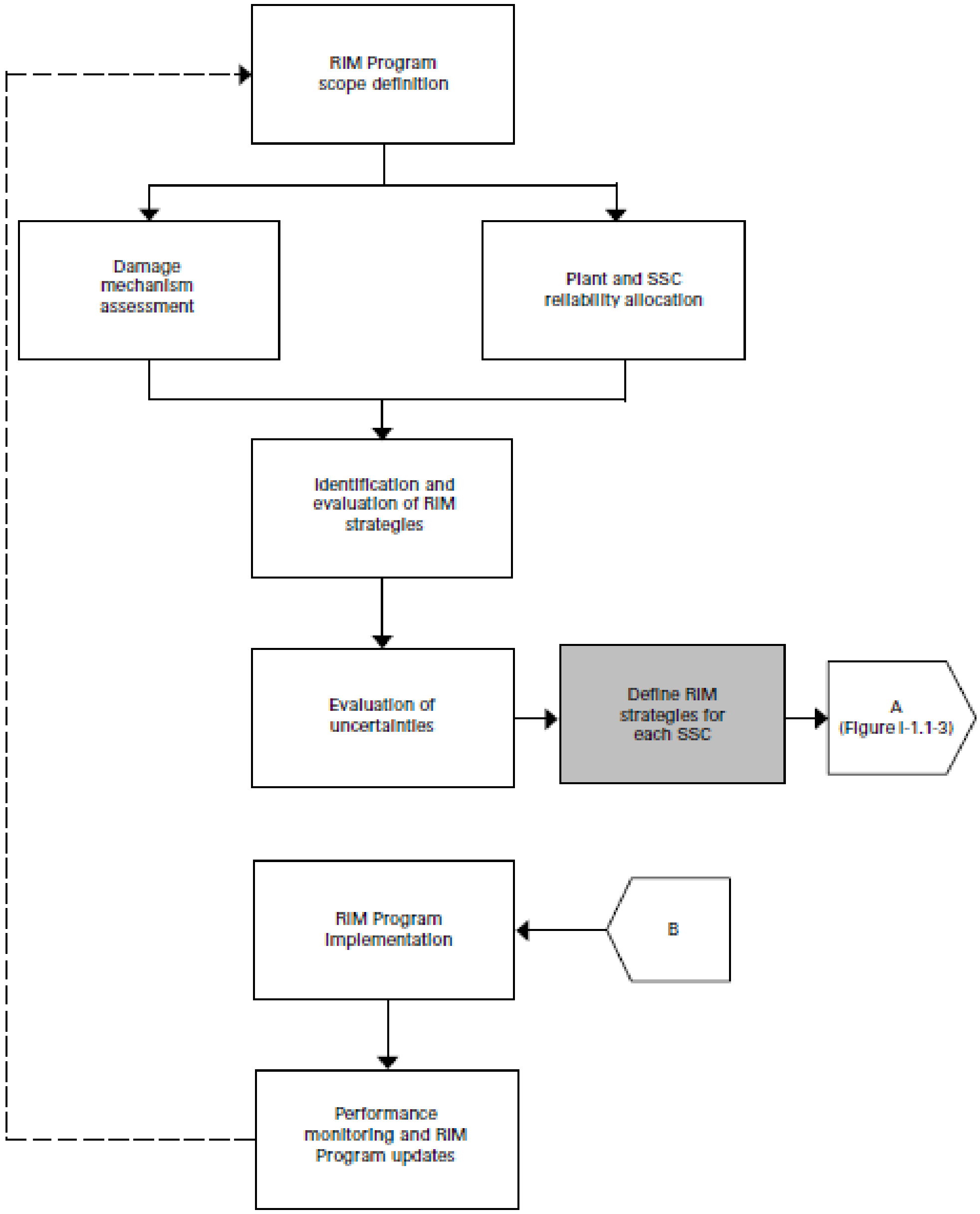

Figure 3. RIM process overview (taken from Figure I-1.1-2 of [5]).

The process of developing a RIM strategy is described in accordance with the first several steps depicted by Figure I-1.1-2 of BPV XI-2 [5].

3.3.1 Defining the RIM Program Scope

The program owner is responsible for defining the scope of the RIM program. BPV XI-2 simply states that the scope shall include SSCs whose failure could adversely affect plant safety and reliability, and that the basis for excluding an SSC from the scope must be documented. This implies that all SSCs will be assessed for inclusion or exclusion.

From a RIM program perspective, when RIM program development commences, the initial SSC scope may change throughout the plant design process, as the contribution of an SSC or group of SSCs can change throughout this process. The program scope may become more finalized as the process of SSC reliability allocation is carried out in subsequent steps.

3.3.2 Degradation Mechanism Assessment

Determination of the inspection and examination type to be applied in the RIM program is dependent on the type of degradation (known or credible) for the material and operating environment. As in the development of a RI-ISI program under BPV XI-1, a formal assessment of the degradation mechanisms is performed.

Details on performing a DMA are provided in Section RIM-2.3 of Article RIM-2. RIM-2.3 requires the listing of conditions and the subsequent application of the appropriate table of degradation mechanism attributes and attribute criteria contained in Mandatory Appendix VII, Supplements for Types of Nuclear Plants. These supplements are applied based on reactor type. Figure 4, which gives an excerpt from Table VII-3.2-1, provides an example of the degradation mechanism attributes and attribute criteria for HTGR-type plants.

Table VII-3.2-1 Degradation Mechanism Attributes and Attribute Criteria (Cont'd)

Degradation Mechanism		Attribute Criteria	Degradation Features and Susceptible Regions	Table VII-3.3.3-1 Examination Category
HTC	C	– high temperature helium environment, and – graphite is present, and – no monitoring or control of impurities in the helium, and – component is carbon steel or low allow steel with an operating temperature >350°C (662°F), or – component is austenitic stainless steel with an operating temperature >500°C (932°F)	cracks can initiate in welds, HAZ, and base metal at the component inner surface affected locations can include regions of the component exposed to the hot helium coolant flow stream degradation can occur over extensive portions of the pipe inner surface crack growth is relatively slow, and through-wall cracking is not expected within an inspection period	Volumetric methods as approved by MANDEEP
	CP	– operating temperature is higher than the allowable design temperature in ASME Section III	cracks can initiate in welds, HAZ, and base metal at the component inner surface affected locations can include areas of stress concentration crack growth is relatively slow, and through-wall cracking is not expected within an inspection period	

Figure 4. Table VII-3.2-1 of BPV XI-2, for HTGR-type plants.

3.3.3 Plant and SSC Reliability Allocation

As shown in Figure 4, the DMA is performed in parallel with determining reliability targets for each SSC. Both are performed prior to determining the strategy for applying MANDE.

Plant-level reliability criteria must be derived from regulatory limits on risk, frequencies, and radiological consequences from LBE defined in the PRA. To implement the RIM program, the plant-level reliability criteria are allocated to the components that fall within the scope of the RIM program as CLRs. Article RIM-2 and Mandatory Appendix II provide the requirements for deriving SSC reliability targets, the stated objective of setting a reliability target for a component being to establish a point of reference against which system performance can be measured in order to meet plant-level requirements.

The method of allocation, or CLR determination, is not detailed in either RIM-2 or Mandatory Appendix II. Considerations for the allocation are provided, but the user is given flexibility in how the reliability targets are derived. Because a plant PRA must be used in deriving the CLRs, RIM provides requirements for the scope, level of detail, and technical adequacy of the PRA.

3.3.4 RIM Strategies for Each SSC

After the DMA has been completed and the CLRs are developed, initial RIM strategies can be determined. RIM-2.5.1 requires the user to account for a minimum set of factors when developing the RIM strategies.

A review of these factors reinforces the application of RIM during the design of a plant. The factors are not limited to plant operation, and include the following:

1. Design strategies, including material selection
2. Fabrication procedures
3. Operation practices
4. Pre-service and in-service examinations
5. Testing
6. MANDE
7. Maintenance, repair, and replacement practices.

An important difference between Division 2 and Division 1 of ASME Section XI is that the development of the RIM program commences during the design of the plant, not during preparation for operation. This is because a RIM strategy may include a specific design requirement/feature that must be implemented to ensure the component reliability target can be achieved. This may be required to prevent or reduce susceptibility to the identified degradation mechanisms, improve the potential for detection by incorporating leak detection capabilities, or ensure the geometry is appropriate for a NDE technique application. These changes are often more economical to incorporate during the conceptual design phase.

Applied strategies are sufficient and necessary to achieve the SSC reliability in support of reliability targets. RIM-2.5.2 provides requirements for how to evaluate the RIM strategy impacts on SSC reliability.

3.3.5 Evaluation of Uncertainties

To ensure that the principles of risk-informed decision making are maintained, incorporating operating experience and accounting for the impacts of uncertainty are factored into the RIM process. In addition to considerations by the MANDEEP for uncertainty in detecting degradation, the RIMEP is responsible for ensuring that the initial RIM strategy accounts for uncertainties in predicting SSC reliability performance. This may result in identifying additional MANDE, or MANDE revisions, that are necessary to ensure that the reliability targets will be achieved and maintained throughout the service life of the components.

BPV XI-2 refers the user to BPV XI-1 requirements for SSC repair/replacement, and to the acceptance criteria for MANDE—similar to what is done for current LWRs. Because new advanced reactor technologies are expected to require different approaches to examinations and to use new materials and configurations, Division 2 also provides guidance and requirements on the qualification of personnel and techniques for the MANDE contained in the RIM strategy. Specific RIM sections (i.e., articles) and appendices provide guidance and requirements on these subjects, affording users the flexibility to develop MANDE activities without having to get regulatory approval for each application.

As with the issuance of ASME Code Case N-875, which was intended to apply SBC as an alternative to the ASME Section XI, Division 3 requirements, BPV XI- 2 provides a non-mandatory appendix that provides a similar alternative to the determined MANDE using an SBC approach. It is uncertain how much this non-mandatory appendix will be utilized, given the status of some of the reactor designs in the U.S. commercial nuclear power industry.

4 RESULTS OF IMPLEMENTATION

Although final implementation of RIM-type programs has not been achieved, the results of implementation, either through program development or through case study, can be used to gain insights for future implementation.

The summary of the implementation is formatted in like fashion to the key subject areas represented by the RIM process overview presented back in Figure 3. This allows for discussing the differences across the three different RIM applications, based on a key subject area. The key subject areas for discussion are the scope of SSCs within the programs, the development of reliability targets, and the development of RIM-type strategies.

4.1 SSCs Within Process Scopes

4.1.1 Pebble Bed Modular Reactor RIM Program Scope

A key outcome from developing the PBMR RIM program was identifying the opportunity to influence the reliability of passive metallic components during the reactor design stage by applying a broader set of options to ensure SSC reliability. This differs from traditional ISI programs, as the ability to ensure reliability is limited to deciding where and how often to apply NDE techniques during operations after the design is already complete [14].

The PBMR RIM program scope was established by considering the importance of the function that the SSC performs. Scope development was performed in conjunction with the ASME work to establish RIM requirements for HTGRs. Therefore, a predetermined requirement was not present beyond that presented under BPV XI-1.

The basis for the RIM program scope was presented as part of the PBMR RIM program development. The scope of the PBMR RIM program included all passive metallic SSCs that perform a required safety function. Within the scope of SSCs in the PBMR RIM program were aspects such as:

- The reactor pressure vessel, including all nozzles, penetrations, bolted connections, and structural support components
- Representative SSCs on the helium pressure boundary
- Representative SSCs in the reactor cavity cooling system.

4.1.2 SBC Scope

Although SBC was integrated into an ASME code case, experience on implementing SBC is limited to application to the PHTS at the previously operating Monju nuclear plant. Its use in the development of an ISI program is documented through various means; however, two recent documents provide the best information pertaining to the experience with using SBC. The first document was published in the Journal of Pressure Vessel Technology [12], and the second was a recent Japan Atomic Energy Agency presentation given as part of the GEN IV International Forum series of presentations [16].

The SBC activities in Japan specifically focused on the PHTS of the Monju nuclear plant. There was no general scoping activity for equipment subject to a RIM program based on SBC concepts. The PHTS can be correlated to the ASME Class 1 pressure boundary systems within a BPV XI-1-based ISI program.

4.1.3 Scope of the ASME Section XI RIM Program

In October 2022, NRC issued RG 1.246, Acceptability of ASME Code, Section XI, Division 2, “Requirements for RIM Programs for Nuclear Power Plants,” for non-LWRs [1]. This RG provides an approach that is acceptable to NRC staff for developing an ISI program in non-LWRs. Issuance of the RG was important to the industry, as the RIM program can provide input to the plant design. Since many reactor designs have either not started development or are very early in the planning stage, no submittals to NRC with use a RIM program and thus no experience is available to learn from.

Unlike BPV XI-1, which states that equipment classified as ASME Class 1, 2, or 3 fall within the scope of the ISI program, BPV XI-2 does not require that specific SSCs be included within the scope of the RIM program. Although a regulatory agency can prescribe the classification of equipment and how they are scoped into a RIM program, the RIM process of BPV XI-2 focuses on the importance of the function of the SSC and its contribution to meeting plant reliability goals.

4.1.4 Summary and Comparison of Scopes for Each Process

Little insight into defining the scope of a RIM program can be gained from assessing the PBMR and SBC processes. This is because these processes focused on a specific system (e.g., the SBC focused only on the PHTS) or systems, rather than equally applying to all systems that may be important to the operation of the nuclear power plant. The systems included in the application of these two processes are typical of sodium fast reactor and HTGR designs, and an application of a RIM program developed in accordance with BPV XI-2 can be expected to include these same systems and produce likely similar results.

4.2 Development of Reliability Targets

4.2.1 Pebble Bed Modular Reactor Reliability Targets

The PBMR DPP case study involves a simple reliability target derivation strategy: "reliability targets are set ranging from a factor of 10 to a factor of 3 below the level they would need to be to just meet the plant-level goals [6]." The selected reliability target values were tested and confirmed by evaluating the RIM strategies necessary to meet the component-level reliability targets. The reliability targets in the PBMR DPP case studies were established based on risk-informed and deterministic insights. The probabilistic information in the reliability target selection process was used to control LBE frequencies and consequences. The deterministic information was used to provide assurance that regulatory requirements will be met and to address uncertainties and defense-in-depth principles.

4.2.2 SBC Reliability Targets

The process of deriving reliability target values for SSCs that follow SBC is described by Kurisaka et al. [10]. It relies on the PRA model and the quantitative safety goal. In this case, the safety goal is expressed in terms of CDF and CFF. A reactor cooling system for the Japan sodium-cooled fast reactor was chosen to demonstrate the reliability target allocation process.

The process starts with the distribution of risk to contributing initiating events (i.e., internal events, seismic events, and other external hazards). The distribution then continues by allocating portions of the risk to plant operating states (i.e., full-power operation and shutdown). Next, the risk is further distributed according to accident sequence type (i.e., sequences directly caused by an initiating event and sequences that combine initiating events and associated mitigating systems).

The SBC scope is limited to the pressure-boundary-retaining components (i.e., application of an ISI program is specific to passive components). As such, the contribution of active and human-related failures are removed from consideration by applying a conservative approach, setting up their probability equal to one.

The values initially assigned as reliability targets for plant systems were processed through an aggregated analysis to determine whether the safety goal would be met. If not, the reliability target values were adjusted using Fussell-Vesely importance values derived from the PRA model. The process was repeated until the selected reliability targets resulted in a plant overall risk that was below the safety goal.

The reliability target allocation process based on the internal events was repeated for seismic initiating events.

4.2.3 ASME Section XI RIM Program Reliability Targets

While BPV XI-2 does not provide methodologies for establishing reliability targets, it does contain instructions for the general process. The RIM program developers are directed to begin with the plant-level reliability targets and goals derived from the regulatory limits placed on the risks, frequencies, and radiological consequences of LBEs. The PRA-defined plant-level reliability targets should be included, and other plant availability targets and goals may be added if desired. The plant-level reliability targets are to be used as the basis for the SSC-level reliability targets—a logical approach, as the overarching goal of the RIM program is to ensure that plant-level performance remains within the acceptable regulatory limits and desired availability goals.

The code provides further directions to focus on passive components when deriving SSC-level reliability targets from the plant-level targets. This requirement is based on the fact that BPV codes are developed for passive components. Active components are addressed elsewhere. Additional guidance is offered in Mandatory Appendix II on reliability target derivation, including a possible way to segregate passive portions of PRA from active contributors. However, the approach suggested in the appendix is not the only possible way to derive reliability targets while remaining compliant with the code.

One RIM-related activity is currently in progress through funding provided by the U.S. Department of Energy. INL initiated a project to develop guidance based on BPV XI-2. The report INL/RPT-22-68899, “Reliability and Integrity Management Program Implementation Approach” [9], was issued in September 2022 to communicate the initial results of the project. A broader implementation is currently underway, and additional results are expected to be communicated in 2023. This report provides the first publicly available insight into the use of RIM within the U.S. commercial nuclear power industry.

The demonstration project, which aims to provide guidance on allocating reliability targets to the component level, focuses on a single system in an advanced reactor design as the case study for developing the guidance.

One key lesson learned during the project relates to the availability of reliability data and the maturity of the PRA. Because RIM implementation requires consideration of the plant design, the PRA must be developed to the extent that its data can be used to determine the ability to achieve the reliability targets, and the data must be sufficient to understand the uncertainties of the RIM strategy without having to incorporate overly conservative compensatory measures.

The lack of data or PRA maturity can also result in the use of different materials or the quality level of the plant design components. Developing a RIM strategy during the plant design can inform the designer as to how using more expensive but more reliable components compares against ability to rely less (i.e., lower the target reliability) on the associated function of both the component and the system in which it operates, thereby decreasing the overall costs of manufacture and future testing/inspection. These cost-benefit evaluations enable developers to select optimal design strategies that satisfy regulatory requirements via the most cost-effective system architectures, components, and monitoring strategies during facility operation.

4.2.4 Summary and Comparison of Reliability Target Derivation for Each Process

The reliability target derivation processes described in the above guidance documents are all similar to each other—they all correlate the plant-level reliability target with the SSC-level reliability targets in order to ensure that the plant-level reliability target is met and that the SSC-level reliability targets are achievable. The PBMR DPP case study does not describe in detail the actual approach via which the plant-level reliability target is decomposed to the SSC-level reliability targets. Similarly, BPV XI-2 does not include any specific approaches or methodologies to establish SSC-level reliability targets. On the other hand, SBC offers a specific step-by-step approach for deriving SSC-level reliability targets from the plant-level safety limit, which in this case serves as the plant-level reliability target.

The industry would benefit greatly from guidance on methodologies for reliability target derivation.

Ideally, several methodologies should be developed and tested, allowing reactor developers to use the one best suited to their licensing approach and/or reactor technology.

4.3 RIM Strategies

4.3.1 Pebble Bed Modular Reactor RIM Strategies

To meet the reliability targets for an SSC, a RIM strategy is developed to address the identified and credible degradation mechanisms and assess the ability to detect degradation. The damage mechanisms included in the PBMR RIM program DMA were those identified for LWR piping systems, which apply to the PBMR and to the unique mechanisms specific to PBMR service conditions. Because of the uniqueness of the HTGR design, some RIM strategies that would typically rely on NDE may not be as effective in identifying the damage mechanisms; therefore, the RIM strategies relied on a combination of design, fabrication practices, NDE, and continuous leakage monitoring.

An online leakage detection system is part of the RIM strategies implemented for the PBMR RIM program. A specification was developed for the online leakage detection system, ensuring that it can identify a leak in the helium pressure boundary within 24 hours of occurrence, at a probability of detection of 90%.

Selected SSCs were volumetrically examined in a manner similar to the selection process for a LWR RI-ISI program. This included selecting at least 10% of the Class 1 piping butt welds. Examinations were to be performed in accordance with BPV XI-1 requirements and acceptance criteria, within an overall program inspection interval of 12 years. During this interval, examinations can be performed at times that correspond to scheduled maintenance activities planned for a 6-year cycle.

4.3.2 SBC RIM Strategies

In general, SBC follows the same approach as the other processes for determining a RIM strategy, in that it identifies the applicable degradation mechanisms and determines the appropriate monitoring and NDE techniques for identifying the degradation. As previously stated, the criteria of the two evaluation stages must first be met in order to provide a basis for what RIM strategy is applied.

Stage I - Structural Reliability Evaluation

For the Monju PHTS, two degradation mechanisms were identified: fatigue-creep and corrosion. Fatigue-creep was considered because the operating temperature range and cyclic stresses were determined to exist. Because the inner surfaces of the PHTS are in contact with sodium coolant, the sodium environment can affect the nickel and chromium alloys of the piping.

Fatigue-creep was addressed by the design code by restricting any resultant damage to below the design allowable level. In terms of corrosion effects, the impact on the material thickness is dependent on total operating time, temperature, and dissolved oxygen concentration. The limitations on dissolved oxygen concentrations in the Monju PHTS resulted in a determination that only a 0.1 mm loss would be experienced over 30 years.

The CLR derived for the Monju PHTS was based on the break size postulated in the plant safety evaluation. It was a through-wall crack with the opening area of 22 cm^2. Because it was determined that a crack would not initiate and propagate to the size selected as the CLR, the criterion for meeting the Stage I evaluation was achieved.

Stage II – Evaluation of Ability to Detect Degradation

The second stage evaluation considers the ability to detect a postulated flaw. Several types of leak detection systems were installed in Monju, including a sodium ionization detector (SID) and a differential pressure detector (DPD). A SID can detect leaks by monitoring changes in the ion current produced by ionizing sodium aerosol in an inert gas atmosphere, whereas a DPD detects leaks by using the differential pressure across the detector filter. These detectors can detect leaks as small as 100 g/h, and when compared to the postulated break size and a leak rate of 80 kg/s, the detectors were considered to appropriately identify the presence of a leak with sufficient time to safely shut down the reactor.

A key attribute of SBC is the conclusion that continuous monitoring for leakage, through use of the previously described SID and DPD, is an effective ISI activity for piping and vessel welds in LMRs. Use of these detectors is considered an indirect means of detection, rather than direct verification of a leak. To support the effectiveness of the ability to detect leaks, the leak-before-break (LBB) concept [18] was taken into consideration. Applying the LBB analysis demonstrated that leakage will be identified so that proper action can be taken before a flaw reaches a critical length and causes subsequent failure of piping or vessels that retain sodium coolant [12].

LBB assessment guidance is being developed in conjunction with a fitness-for-service code for sodium fast reactors in Japan, through JSME. The goal is to provide procedures via which the LBB concept can be applied to piping and vessels that retain sodium coolant. The guidance document has been issued, but it is in Japanese and has not yet been translated at the time of this review on the current state of RIM knowledge.

4.3.3 ASME Section XI, Division 2 RIM Strategies

Experience in determining RIM strategies in accordance with BPV XI-2 is limited. The referenced case study by INL is limited to a single system and provides some insight into the impact of applying different RIM strategies. Final conclusions have not yet been presented in regard to what RIM strategies should be applied to that system.

Some understanding of the implementation of ASME Section XI, Division 2 can be gained from the actions being taken by the ASME Boiler and Pressure Vessel Code committees and working groups responsible for BPV XI-2. Insights gained during NRC's review of BPV XI- 2—insights obtained in support of the issuance of RG 1.246 [1]—were initiated by the associated working groups and committees so as to either clarify certain requirements through the use of a white paper within the ASME Section XI revision process, or to make specific changes to the code. Some of these actions are also specifically intended to clarify regulatory positions identified in RG 1.246 on BPV XI-2's acceptance for use.

One area that needs clarification is the assignment of examination categories, which occurs when applying the BPV XI-1 requirements to an LWR. As part of referring the user back to BPV XI-1 for acceptance criteria, Mandatory Appendix V of BPV XI-2 provides examination requirements and areas of interest for examination. Each table in the appendix is associated with an examination category from BPV XI-1, and within each table is an associated item number that is also used in BPV XI-1. The general information at the beginning of the appendix states, "The Tables in this Appendix make reference to Division 1 examination categories and examination volumes. These references are not intended to imply that either Class 1 or Class 2 Division 1 requirements are to be utilized for a particular RIM SSC." Other references are made to the use of an examination category in the Owners Activity Report in Mandatory Appendix III and in the supplements for each reactor type in Mandatory Appendix VII. The clarification provided by the previously mentioned BPV XI-2 white papers indicates that BPV XI-2 does not require assignment of an examination category, but also recognizes the need for clarifying information on the use of an examination category.

Mandatory Appendix VII of BPV XI-2 contains the supplements for each reactor type and is incomplete. Not all reactor types are contained in the appendix, articles are shown as "in Course of Preparation" for the supplements for LMR-type plants, molten-salt-reactor-type plants, Generation-II LWR-type plants, and fusion-machine-type plants.

4.3.4 Summary and Comparison of RIM Strategies for Each Process

Developing a RIM strategy via both the SBC and the PBMR processes shares significant similarities with the current BPV XI-2 process. This is primarily because the two processes were in development either prior to or in conjunction with the development of BPV XI-2. In all three processes, it is necessary to determine what degradation mechanisms are expected for the associated materials and operating environments. Similarly, all three processes require the determination of a reliability target (regardless of how it is derived), as well as assurance that the RIM strategy will be developed (and demonstrated) to reliably detect the expected degradation.

Comparison with the SBC of the resultant strategy (i.e., continuous monitoring) is limited to just the strategy for continuous leakage monitoring, as that is the only strategy that has been communicated thus far. However, the RIM strategies developed in the PBMR RIM report demonstrate the importance of implementing aspects of RIM early on in the design phase, as the strategies consider PBMR design features, such as reducing or eliminating degradation mechanisms and ensuring that the design enables the mechanisms to be detected.

BPV XI-2 is more similar to the RIM strategy development approach presented in the PBMR report. However, BPV XI-2 recognizes that the strategy may need to be more oriented to a deterministically derived reliability target, and presents an alternate approach for establishing MANDE (i.e., a RIM strategy) through implementation of an SBC-type approach. This alternative is presented as Non-Mandatory Appendix A.

5 CONCLUSIONS

Development of the SBC approach and the details associated with developing the PBMR RIM program have been beneficial in developing the RIM requirements under Division 2 of ASME Section XI, and a significant number of attributes remain in the 2019 Edition. Although there is limited experience in implementing any of the three RIM-type processes, there are sufficient insights for developing recommendations on the future implementation and development of RIM programs. Some insights learned from analyzing the sources are offered below.

RIM Program Scope

BPV XI-2 is a code that can be used internationally and is subject to endorsement by multiple regulatory authorities. From a RIM program scope perspective, clarification on how a scope is determined would be beneficial. In the absence of a regulatory authority prescribing what SSCs should be assessed for inclusion, clarification may be a simple statement, within BPV XI-2, clearly reflecting that the owner determines what SSCs fall within the RIM program, and provides guidance on determining the scope—potentially as a non-mandatory appendix.

Use of the PRA

RIM-2.4.2 of BPV XI-2 requires the user to employ the PRA model to allocate SSC reliability. Because implementation of RIM entails consideration of the plant design, the PRA must be developed to the extent that its data can be used to determine the ability to achieve the reliability targets, and that it is sufficient for understanding the uncertainties of the RIM strategy without having to incorporate overly conservative compensatory measures. However, in the early design stages, a PRA may be insufficiently mature to effectively allocate reliability targets (CLRs) to a component. Assumptions may need to be made to compensate for the lack of design maturity in systems, including active systems that interface with the passive pressure boundary components within the scope of the RIM program.

Focus on Passive-only Components

Although BPV XI-2 is a code for ensuring the integrity of passive pressure boundaries, the contribution of active components (e.g., electrical systems) influences the determination of RIM strategies. This is based on the ability to increase or decrease the reliability of a specific SSC based on its contribution to the reliability of the overall system functions, where function is typically influenced by active and passive components. This reinforces the importance of PRA maturity as a necessary consideration when developing a RIM program and specific RIM strategies.

A potential concern about the intentional separation of active and passive components during reliability target allocation is that they skew the representation of risks, due to the application of an over-conservative treatment of active components (suggested by BVP XI-2). While it is understandable that the separation is needed—since BVP XI-2 focuses on passive-only components—the process could be set up such that all the SSCs are considered for reliability target allocation process initially, but with only passive components included in the scope of the RIM program at the end.

Connection with the NRC Regulatory Guide 1.233

RG 1.246 endorsed BPV XI-2 as an acceptable approach for developing and implementing an ISI program for non-LWRs, and lists RG 1.233 [20] as related guidance. RG 1.233 endorses the technology-inclusive, risk-informed, and performance-based methodology of informing the licensing basis and application content for non-LWRs as presented in Nuclear Energy Institute 18-04 [15], known as the Licensing Modernization Project (LMP).

There are a few clear connections between BPV XI-2 and LMP:

- The LMP dictates (emphases added) "**derivation of requirements** necessary to **support SSC performance** of safety functions in the prevention and mitigation of LBEs that are modeled in the PRA." This LMP requirement can be accomplished by implementing a RIM program, and BPV XI-2 is the NRC-endorsed approach for a RIM program.
- The LMP, just as with BPV XI-2, relies on reliability targets. In the LMP, reliability targets are used to

support the development of design and special treatment requirements for SSCs as well as evaluations of the adequacy of plant defense-in-depth.

- The purpose of the SSC performance requirements in the LMP is to "provide reasonable confidence in the SSC capabilities and reliabilities in performing functions identified in the LBEs consistent with the Frequency-Consequence (F-C) Target and the regulatory dose limits for design basis accidents." The F-C Target and regulatory dose limit are plant-level safety requirements in the LMP, which means that the LMP, as with BPV XI-2, makes specific connections between the plant-level and component-level performance requirements, expressed as reliability targets.
- The LMP guidance on deriving the reliability targets is similar to the BPV XI-2 guidance: "Information from the PRA is used as input to the selection of reliability targets and performance requirements for SSCs."

Given that reliability targets are required by both the LMP and the RIM program, there should be consistency in the meaning of the reliability target term. Also, the overlap suggests that consistency should exist in the reliability target derivation process in both the LMP and the RIM program.

Continuous Online Monitoring

The PBMR RIM program and the presentation of SBC emphasize the appropriateness for reliance on continuous leakage monitoring, including the PHTSs that comprise the Class 1 SSCs. In both cases, a technical basis is provided for the accuracy of online monitoring systems and to serve as the basis for properly applying them (i.e., detecting with sufficient accuracy and notification so that actions can be taken prior to catastrophic failure of the SSC).

Given that the safety philosophy of many of the new advanced reactor designs focuses on fuel integrity, and that from a safety perspective, there is less reliance on the integrity of PHTS, continuous leakage monitoring is expected to be a key attribute for many systems in future advanced reactors regardless of reactor technology.

Degradation Mechanism Assessment

BPV XI-2 contains a mandatory appendix that contains degradation mechanism information that is to be considered. However, the appendix currently addresses only a couple of reactor types. Increased priority to populate the information for other reactor types should be taken to ensure that consensus information can be applied consistently across all RIM programs of each reactor type.

6 REFERENCES

[1] NRC RG 1.246, "Acceptability of ASME Code, Section XI, Division 2, Requirements for Reliability and Integrity Management (RIM) Programs for NPPs, for non-LWRs," U.S. Nuclear Regulatory Commission, October 2022.

[2] NRC 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," U.S. Nuclear Regulatory Commission, Title 10, Code of Federal Regulations (CFR), Part 50.

[3] ASME Section XI, Div.2, "Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants," American Society of Mechanical Engineers, ASME Boiler & Pressure Vessel Code, Section XI, Division 2, BPVC.XI.2-2019, 2019.

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