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ADVANCED NONLIGHT-WATER REACTORS
Summary of Gap Identification and Recommendations on Consensus Codes and Computational Codes

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TABLE OF ACRONYMS

3-D	three dimensional
AFCEN	French Association for Nuclear Codes and Standards
AGR	advanced gas-cooled reactor
ALE	arbitrary-Lagrangian-Eulerian
AMR	advanced modular reactor
ANL	Argonne National Laboratory
ANLWR	advanced nonlight-water reactor
ANSTO	The Australian Nuclear Science and Technology Organization
API	American Petroleum Institute
ARMI	advanced reactor modeling interface
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BPVC	Boiler and Pressure Vessel Code
C	Celsius
CAE	computer-aided engineering
DOE	U.S. Department of Energy
EDF	Électricité de France
EMPA	Eidgenössische Materialprüfungs- und Forschungsanstalt (Swiss Federal Laboratories for Materials Science and Technology)
EPRI	Electric Power Research Institute
F	Fahrenheit
FEAM	finite-element alternating method
HPB	helium pressure boundary
HTGR	high-temperature gas-cooled reactor
INL	Idaho National Laboratory
ISI	inservice inspection
ISO	International Organization for Standardization

JSME	Japan Society of Mechanical Engineers
LBB	leak before break
LLNL	Lawrence Livermore National Laboratory
LWR	light-water reactor
MANDE	monitoring and nondestructive evaluation
mHTGR	modular high-temperature gas-cooled reactor
MOOSE	Multiphysics Object-Oriented Simulation Environment
MSR	molten salt reactor
NASA	National Aeronautics and Space Administration
NDE	nondestructive examination
NDM	nondestructive monitoring
NRC	U.S. Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PBMR	pebble bed modular reactor
PNNL	Pacific Northwest National Laboratory
PRA	probabilistic risk assessment
PVP	pressure vessel and piping
PWSCC	primary water stress-corrosion cracking
RG	regulatory guide
RIM	reliability and integrity management
RPV	reactor pressure vessel
SBC	System Based Code
SCC	stress-corrosion cracking
SFR	sodium-cooled fast reactor
SNL	Sandia National Laboratories
SSC	structure, system, and component
UK	United Kingdom
UT	ultrasonic testing
VFT	Virtual Fabrication Technology

WG-HTFE	Working Group on High Temperature Flaw Evaluation
WRS	weld residual stress
XFEM	Extended Finite Element Method
xLPR	Extremely Low Probability of Rupture (Code)

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EXECUTIVE SUMMARY

This report identifies potential gaps in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section XI, Division 2, based on the operating experience issues compiled and summarized in the previous technical letter report (Turk et al., 2019). This includes high-temperature damage mechanisms and potential licensing issues of future advanced nonlight-water reactors (ANLWRs). While it is likely that some prior experience with older reactors may not be applicable to future ANLWR designs, the operating experience report (Turk et al., 2019) summarizes potential service issues that could be considered by the various consensus code and standards organizations (e.g., ASME). While ASME is developing and enhancing rules for ANLWRs in ASME BPVC, Section III, Division 5, this report focuses on ASME BPVC, Section XI, Division 2.

This report is a high-level summary of the reliability and integrity management (RIM) program recently included in ASME BPVC, Section XI, Division 2. The RIM program was inspired by experience with the Japanese System Based Code concept, which was the product of inservice inspection requirements for Monju, a prototype fast breeder sodium-cooled reactor. This report provides two examples of RIM applied to ANLWR designs.

The report also summarizes recent enhancements to the United Kingdom (UK) R5 Fitness for Service procedure used in ANLWR structural component integrity analysis and contrasts these with other code methodologies. R5 is currently undergoing enhancements to include the consideration of carburization (which is unique to the carbon-dioxide-cooled UK high-temperature gas reactors) and improved nonsteady-state creep crack assessments. R5 incorporates high-temperature fracture mechanics assessment procedures not available in the current ASME BPVC.

In addition, the report discusses nondestructive examination and inservice inspection technologies for high-temperature reactors. It summarizes the concept of nondestructive monitoring, which targets online monitoring of active degradation mechanisms at susceptible locations.

The final sections of this report examine computer codes applicable to the design and assessment of ANLWRs. The report also reviews some of the available commercial, open source, government, and NRC computer codes. The discussion of each computer code includes perceived needs for improvements (or computer code gaps). The report examines the following computer codes:

- **Commercial finite-element computer codes** discussed include ABAQUS, ANSYS, and other commercial codes.

- **NRC-developed analytically based computer codes** covered include extremely low probability of rupture (xLPR), Version 2, as well as PROMETHEUS, FAVOR, ALT3D, NRCPIPE, NRCPIPES, SQUIRT, and related codes and modules from the xLPR

probabilistic code, developed initially to assess light-water reactors. The first three of these are probabilistic codes used to assess the uncertainties associated with inservice damage to reactor piping.

- **Open source codes** examined include WARP3D, which has many features applicable to ANLWR material degradation and fracture assessment. In addition, since it is open source with extensive documentation, additional features can be added. Other ABAQUS USER subroutines (e.g., creep USER subroutines) can be used with WARP3D with minor modifications.

- **Government codes** discussed include GRIZZLY, MOOSE, SIERRA, DIABLO, and ALE3D. These are multiphysics-based, finite-element codes developed by Idaho National Laboratory, Sandia National Laboratories, and Lawrence Livermore National Laboratory. The GRIZZLY code is currently being developed and used for nuclear applications.

- **Graphite computer codes** are covered through a limited summary of computational graphite modeling efforts and available codes. Many of the modeling efforts in the United Kingdom use either a commercial code or an open source code with special graphite damage mechanisms implemented either with the use of USER material subroutines or directly as subroutines into open source codes.

The recommendations and road map presented in this report identify computer code improvements recommended for structural integrity assessment of ANLWRs. The plan is aggressive and may stretch technical resources and staffing during development.

For example, the codes GRIZZLY and ABAQUS can be used to perform both deterministic and probabilistic assessments. The development of subroutines for GRIZZLY and ABAQUS will take time to develop but will aid in validating the development of xLPR for use by ANLWRs. GRIZZLY developments will be the most powerful for ANLWR structural component life assessments when complete but will take the longest to develop and train for use.

Performing probabilistic risk assessments based on GRIZZLY or ABAQUS solutions will be challenging, as xLPR experience assessing rupture in piping systems susceptible to primary water stress-corrosion cracking has demonstrated that tens and even hundreds of thousands of realizations may be necessary to ensure convergence to a realistic risk number for low-probability events. Hence, the use of surrogate models based on the solution space obtained with GRIZZLY or ABAQUS may be necessary.

1. INTRODUCTION AND BACKGROUND

Multiple domestic and international corporations have indicated their intent to conduct licensing or prelicensing activities for advanced nonlight-water reactors (ANLWRs) with the U.S. Nuclear Regulatory Commission (NRC) in the next 5 to 10 years. Currently, several ANLWR concepts are being considered, including sodium-cooled fast reactors (SFRs), high-temperature gas-cooled reactors (HTGRs), and molten salt reactors (MSRs).

These advanced designs require modifications to the design rules in American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, to handle temperatures higher than previously considered. These rule changes are being considered for ASME BPVC, Section III, Division 5, and the NRC is assessing them in a separate effort, outside the scope of this report. ASME has developed BPVC rules for the ANLWRs considered in this report. The 2019 Edition of ASME BPVC, Section XI, Division 2, provides requirements for the reliability and integrity management (RIM) program and contains supplemental information for applying the RIM program to ANLWRs. The RIM program encompasses the entire life cycle of the plant and is applied to each in-scope passive structure, system, and component (SSC). The RIM program includes a combination of monitoring, examination, tests, operation, and maintenance requirements ensuring SSCs meet reliability targets, which are defined as performance-based goals for the probability that an SSC will complete its specified function to achieve plant-level risk and reliability goals.

Advanced reactors generally operate at higher temperatures (500–900 degrees Celsius (932–1,652 degrees Fahrenheit (F)) compared to light-water reactors (LWRs) (274–310 degrees C (525–590 degrees F)). At these temperatures, materials behave inelastically, and the allowable stresses are explicit functions of both time and temperature. ASME BPVC, Section XI, Division 2, does not currently address rules for time-dependent crack growth and fracture, although they are currently under development.

Due to the extreme environments, and in lieu of experimental testing, computer codes are being developed to assist the design and licensing of ANLWRs. Computational codes used for the development and expected licensing of ANLWRs belong to one of five categories:

(1) Commercial finite-element computer codes are used for analyzing the complex issues associated with ANLWR design and life estimates.

(2) NRC-developed computer codes, including probabilistic codes used to assess uncertainties associated with reactor damage development, finite-element codes, and analytically based legacy codes, aid in fracture assessment, leak-rate predictions, and other aspects of reactor safety.

(3) Open source codes have no license fees associated with their use and have many features applicable to ANLWR damage and fracture assessment. WARP3D is an example of such an open source code.

(4) The U.S. Government developed multiphysics-based finite-element computer codes. The U.S. Department of Energy (DOE) national laboratories have developed three separate codes (GRIZZLY, DIABLO, and ALE3D) for structural assessments of ANLWRs. The NRC and contractors may be able to access these nonpublic codes. These codes operate within powerful multiscale modeling environments, permitting the use of computer modules to add additional capability.

(5) Proprietary codes include, for example, the advanced reactor modeling interface (ARMI) code used by TerraPower LLC. Due to their proprietary nature, only a very limited amount of information is publicly available on such codes, but the ARMI code is known to have open source components, such as Idaho National Laboratory's (INL's) MOOSE (Multiphysics Object-Oriented Simulation Environment) and the Massachusetts Institute of Technology's OpenMOC and OpenMC. This report does not discuss proprietary codes.

2. CONSENSUS DESIGN CODES AND FITNESS FOR SERVICE PROCEDURES CONSIDERED

This report identifies international consensus design codes for nuclear installations in other countries and provides limited comparisons to ASME BPVC, Section XI, rules where gaps are identified. However, resource limitations did not allow for a full comparison. International codes examined include the French RCC-MRx code and British R5 Fitness for Service procedure. Within the scope of this effort, it was not possible to obtain consensus codes from Germany, Japan, India, China, and Russia. ASME (2012a) summarized rule comparisons between ASME codes and those from France (French Association for Nuclear Codes and Standards (AFCEN)), Japan (Japan Society of Mechanical Engineers (JSME)), Korea (Korea Energy Agency), and Canada (Canadian Standards Association). However, that report focused on rules related to lower temperature LWRs. Because R5 was used to manage the advanced gas-cooled reactors (AGRs) in the United Kingdom (UK) for many years, it is the precursor to other ANLWR fracture assessment codes. Most international codes follow the procedures in R5, especially for the crack assessments. Additional ASME standards and technology programs highlighted technical gaps in the ASME codes needed for the design of ANLWRs (ASME, 2011; ASME, 2012b), including ASME BPVC, Section XI, Division 2. The most notable gaps in the Section XI, Division 2, RIM program are the supplemental appendices for individual reactor designs in mandatory Appendix VII.

This section presents a short overview of the new ASME BPVC, Section XI, Division 2, RIM program, which is a risk-informed code assessment process developed for all types of reactors, including ANLWRs. This section gives an overview of the purpose and the process or the procedure rather than a complete assessment of the new RIM program. Because this is a new code assessment process, there are no direct applications of RIM for U.S. reactor vendors. However, the report provides two examples of Japanese and South African reactor designs that apply RIM.

The Japanese proposed a process for inservice inspection (ISI) requirements for Monju, a prototype fast breeder SFR. The process to manage component reliability was called the System Based Code (SBC) concept (Asada, Tashimo, and Ueta, 2002a; Asada, Tashimo, and Ueta, 2002b; and Asada, 2006). At the time, the overall Japanese codes and standards system was considered inflexible and rigidly structured. The structural integrity standards consisted of independent codes for materials, design, construction, and operation, and they are self-inclusive and independent of each other, including margins. This inflexibility and rigid structure resulted in overlapping and excessive design margins. Takaya et al. (2015) demonstrated the application of an SBC, including an example for Monju. This inspired the development of the RIM program, which was incorporated into the 2019 Edition of AME BPVC, Section XI, Division 2.

The designers of the South African gas-cooled pebble bed modular reactor (PBMR) also developed a RIM program for assessing the passive metallic components, as discussed by Fleming, et al. (2008). This effort provided guidance for the development of ASME BPVC, Section XI, Division 2, as well. The South African effort was inspired by lessons learned from risk-informed ISI programs developed for LWR piping systems by the NRC (Regulatory Guide (RG) 1.178, "An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping," first issued May 1998).

This section also discusses other consensus codes.

2.1. ASME BPVC, Section XI, Division 2, Reliability and Integrity Management

Initial RIM developments of ASME BPVC, Section XI, Division 2 (summarized by Schaaf, 2014) were introduced into the JSME code in 2004 and called SBC. ASME developed RIM rules as part of ASME BPVC, Section XI, Division 2 (ASME, 2019). These rules are meant to apply to both LWRs and ANLWRs. The RIM program has appendices to deal with degradation mechanisms, flaw evaluations, and acceptance criteria specific to different types of ANLWRs. Some members of the Japanese SBC team and the South African RIM team were integral parts of the RIM program in Section XI, Division 2.

The RIM program addresses each passive SSC in the RIM program scope over the plant's lifetime and is informed by probabilistic risk assessment (PRA). The RIM Expert Panel identifies plant-level risk and reliability targets for the RIM program. The overall plant risk and individual reliability targets for passive components within the RIM scope are based on the PRAs used to ensure overall regulatory compliance.

As discussed in the recently released ASME BPVC, Section XI, Division 2, the RIM program at a high level consists of the following seven concepts:

(1) Identify the SSCs within the scope of the RIM program for the plant of interest.

(2) Assess and summarize the degradation mechanisms for each SSC to enable inspection for cause for the plant (could be LWR or ANLWR).

(3) Develop and allocate the plant and the SSC reliability target, which is the acceptable risk of failure.

(4) Identify and evaluate methodologies used in the RIM program to meet reliability targets and goals, such as monitoring and nondestructive examination (NDE) (MANDE).

(5) Evaluate and summarize uncertainties in SCC reliability performance.

(6) Implement the RIM program.

(7) Regularly evaluate and update MANDE in formal reviews.

Some further discussion of the reliability targets in the third and fourth items is in order. Reliability targets are a key aspect of the RIM program. At this point, it is not known how these reliability targets will be established, but they clearly need to support the PRA assumptions. Some questions to be considered include the following:

- Will the reliability targets just be an extraction from the PRA?
- Are PRAs explicit enough to extract this information?
- How will the reliability targets be demonstrated (e.g., will probabilistic fracture mechanics evaluations be conducted to demonstrate that the reliability targets will be met)?

There are also rules to address the treatment of degradation mechanisms for each reactor type. Presently, there are mandatory appendices for Generation 3 (or above) LWRs and HTGRs. Mandatory appendices for liquid metal reactors, MSRs, Generation 2 LWRs, and fusion plants are under preparation, and this is considered a gap at present. The expected degradation mechanisms for each plant type are laid out in these appendices. For example, the HTGR mandatory appendix lists some of the following degradation mechanisms that must be considered:

- carburization (an issue in the UK AGRs because the coolant, carbon dioxide (CO_2), was considered essentially “inert” during plant design but was later discovered to be otherwise; carburization is unlikely in proposed HTGRs as helium is the intended coolant)
- creep (an active degradation mechanism)
- crevice corrosion (similar to pitting corrosion, occurs in confined spaces where protective films break down)
- erosion–cavitation corrosion caused by corrosive fluid flow on the metal
- external chloride stress-corrosion cracking (SCC) (often caused by coolant contamination)
- flow-accelerated corrosion

- flow-induced vibration causing large local stress cycles (a frequent failure mechanism in the ANLWR operating experience summarized in Section 3.1)
- high-temperature cracking
- intergranular SCC
- loose parts interacting with passive or active components
- mechanical fatigue
- microbiologically induced corrosion
- particle erosion-corrosion
- pitting corrosion
- radiation embrittlement of the material
- SCC
- self-welding of rubbing parts and fretting fatigue
- thermal aging
- thermal stratification cycling and striping
- thermal fatigue
- transgranular SCC
- thermal transients (caused during upsets or plant startups and shutdowns)
- vibration fatigue
- water-hammer-induced damage
- other unknown damage mechanisms (which may occur after operation due to many effects, including aging)

Many of these degradation mechanisms appeared in the operating experience (Turk et al., 2019) and are summarized in the gaps discussed in Section 3.1 of this report. As ANLWR plants are built and operated, additional degradation mechanisms will become apparent that must be addressed and will be added to this list.

The RIM program addresses the plant life cycle and will be updated continually based on operating experience. MANDE is based on the active degradation mechanisms, the reliability target defined for the plant, and operating conditions. For ANLWRs, traditional NDE methods (e.g., ultrasonic testing (UT), liquid penetrant testing) may be replaced or supplemented by techniques better suited for the particular ANLWR and for degradation mechanisms not presently addressed by ASME BPVC, Section XI, Division 1, for LWRs. Some of these methods might include the following:

- online acoustic emission monitoring
- periodic surveillance specimen testing
- active leakage measurement systems

Authorized nuclear inservice inspectors carrying out inspections would follow the specific RIM program developed for the ANLWR under consideration. Section 3.2 includes more details on inspection.

Nuclear power is moving toward such developments as new designs and miniaturization. RIM is intended to accommodate these changes while maintaining long-term safety and reliability. Like the Japanese SBC system, the new RIM program should reduce code inflexibility and overlap of margins. The next two examples help illustrate the RIM program.

2.1.1. Example System Based Code or Reliability and Integrity Management Program for Monju

The paper by Takaya et al. (2015) discussed the application of the SBC process (similar to RIM described in ASME BPVC, Section XI, Division 2) to the determination of ISI requirements for an SFR and includes an example for the prototype SFR at Monju. This work was meant to realize effective and rational ISI by properly accounting for plant-specific features. The proposed process consisted of two complementary evaluations, one focusing on structural integrity and the other on the detectability of defects before they would grow to an unacceptable size. If defect detection were not feasible, structural integrity evaluation would be required under a sufficiently conservative hypothesis.

Compared to LWRs, SFRs operate at elevated temperature and low internal pressure. Structural components experience almost negligible corrosion in pure sodium. The SBC concept is expected to be of great use for determining suitable ISI requirements for SFRs. As discussed above, Asada, Tashimo, and Ueta (2002a), Asada, Tashimo, and Ueta (2002b), and Asada (2006) proposed the SBC concept for the development of Japanese SFRs in the 1990s. One of the key concepts is margin optimization, which provides a new framework intended for optimum allocation of margins on the structural integrity of components encompassing various technical aspects of a plant life cycle, such as material, design, fabrication, installation, inspection, and repair and replacement. By taking full account of these technical characteristics, the SBC concept can improve reliability and economy while meeting safety goals.

Asada, Tashimo, and Ueta (2002a), Asada, Tashimo, and Ueta (2002b), and Asada (2006) considered a two-stage evaluation. Stage 1 is a design-basis evaluation of structural integrity for a specified component. As part of the evaluation, all failure modes to be considered are identified, then the probability of failure is assessed for the component based on potential active failure modes. This includes consideration of accident scenarios in which the maximum permitted break size of the component is defined. If the reliability assessment meets component life goals, the evaluation proceeds to the Stage 2 assessment. On the other hand, if the Stage 1 assessment goals are not satisfied, a modified design or a change in operation conditions is necessary.

Stage 2 assessment is safety oriented and the possibility of a break is assessed. For example, the leak-before-break (LBB) concept (in modified form) can be applied to the sodium boundaries to demonstrate the detectability of such a break, and the continuous leakage monitoring will become an ISI requirement. However, when such a break cannot be detected, additional requirements must be met. For the latter case, the requirements must be correlated to the most important degradation mechanism.

An example of this process for a Monju component follows. The upper core structure was considered for this assessment (Figure 1). The cylindrical vessel is about 14 meters (46 feet) high with a diameter of 2.6 meters (8.6 feet), and the material is equivalent to American National Standards Institute Type 304 stainless steel with an operating temperature of 540 degrees C (100 degrees F) where creep can occur. The sodium coolant is in the bottom portion of the vessel covered by an argon shielding gas. The control rod drive mechanisms are expected to perform in case of a large earthquake. Note the “bucket-like” structure (Figure 1) near the sodium/cover gas interface. This was introduced to limit the development of creep-fatigue damage in the shell region of the vessel due to thermal fluctuations. The bucket shields the vessel from numerous thermal transients that can occur as the temperature level fluctuates.

For the Stage 1 assessment, the degradation mechanisms in the upper core section of the vessel were first identified using a list of degradation mechanisms to be considered in the aging management of LWRs published by the Atomic Energy Society of Japan (2008), “Code on Implementation Review of Nuclear Power Plant Aging Management Programs.” These degradation mechanisms included wall thinning, cracking (e.g., fatigue, corrosion, creep), aging, and creep deformation, among many others (Table 1 of Takaya et al., 2015). The assessment did not include many of these because they were not applicable. For example, it excluded SCC because this degradation mechanism has not been observed in purity-controlled liquid sodium. Other degradation mechanisms specific to SFRs were included. Cracking caused by creep-fatigue interaction damage was identified as the key degradation mechanism to consider as detailed in Takaya et al. (2015).

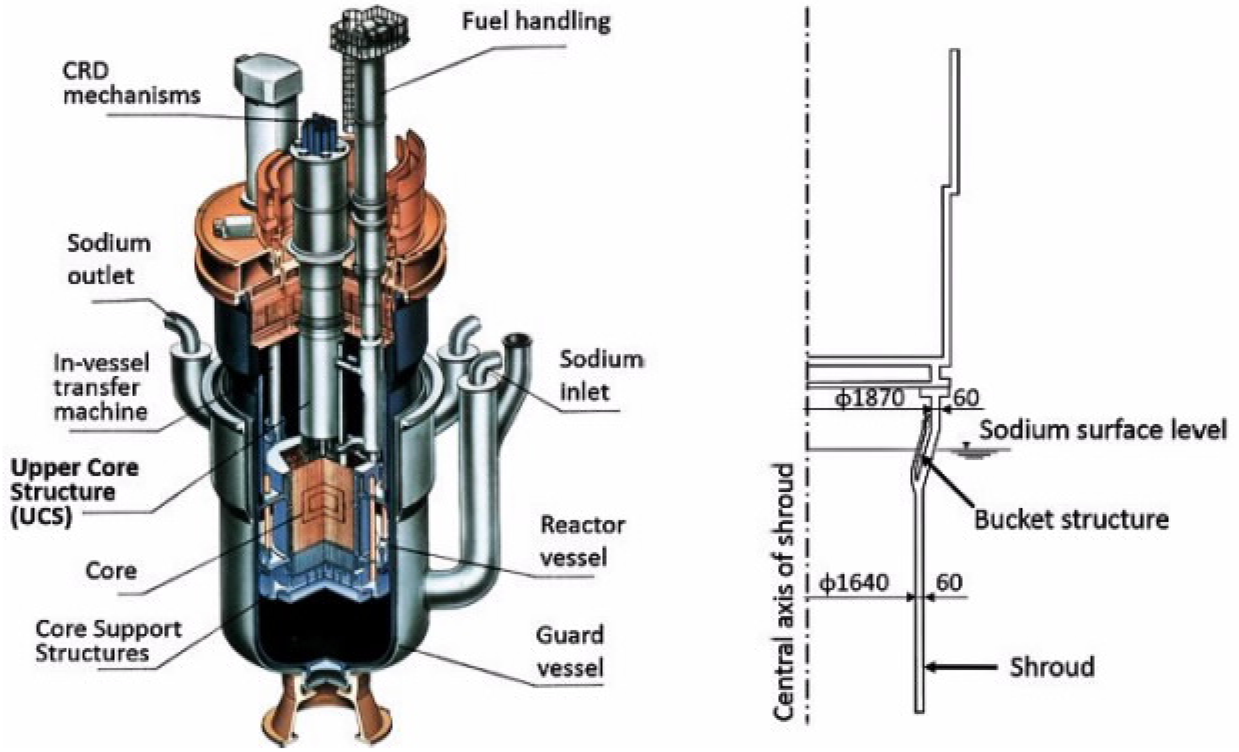

Figure 1 Component considered for Monju SBC (RIM)-type assessment (adapted from Figures 3 and 4 from Takaya et al. (2015))

The Stage 2 evaluation considered a fully circumferential 30-millimeter (1.2-inch)-deep flaw (half the wall thickness) at the sodium level of the shroud part. This is very conservative, as the flaw would most likely be a partial arc surface crack before growing the full 360 degrees. As an added conservatism, it was assumed the flaw could not be detected. This led to the consideration of additional requirements and to the development of the bucket structure (Figure 1) placed on the internal diameter of the shroud vessel to reduce axial thermal stresses on the upper core structure. As an additional requirement, the assessment assumed the loss of the bucket structure entirely. Hence, a probabilistic crack growth and fracture assessment was made, assuming a full 30-millimeter (1.2-inch)-deep circumferential crack at the sodium/argon interface. It was also assumed the protective bucket structure was not present. The Monte Carlo risk assessment using creep-fatigue fracture damage analysis led to the conclusion the failure probability of the upper core structure was less than 1×10^{-9}/30 years, or less than the component level reliability requirement of 1×10^{-7}/30 years (with 30 years as the design life). This represents an SBC assessment of an SFR component and is similar to the RIM program in ASME BPVC, Section XI, Division 2.

Finally, the pressure for an SFR is generally around atmospheric pressure. With this low pressure, it may be possible to conduct a generic evaluation (or plant-specific evaluation) and conclude that the impact of allowing an active degradation mechanism is minimal and will not result in any substantial pipe whipping. Of course, sodium leakage must be prevented or minimized.

2.1.2. Example Reliability and Integrity Management Program for the Pebble Bed Modular Reactor

The RIM program for PBMR helium pressure boundary components was considered by Fleming, Gamble, and Gosselin (2007) and summarized in Fleming et al. (2008). The methodologies for the RIM assessment of passive metallic components for the South African PBMR were the focus of this study, which was inspired by RG 1.178. The RIM methodologies investigated included design elements, leak detection and testing, and NDE. The South African PBMR was beginning construction when government funding was cut in 2010. The opportunity to influence the reliability of passive metallic components during the design stage was an important conclusion obtained from the pilot study.

The general approach to the PBMR pilot study consisted of the following steps:

(1) Determine the scope of SSCs to be considered in the RIM PBMR study. The scope of the study is passive metallic SSCs, and, while important, the PRA addressed the development of special treatment for active SSCs. The RIM program considered the reactor pressure vessel (RPV), including all nozzles, penetrations, bolted connections, and structural supports. In addition, it considered the HPB system, including piping.

(2) Evaluate and list applicable SSC damage mechanisms to consider in the RIM study. These damage mechanisms included those applicable to LWRs and additional mechanisms listed in ASME BPVC, Section XI, Division 2, including active mechanisms such as creep.

(3) Determine the plant and SSC reliability requirements for the RIM study. The RIM study established goals to address passive components, as summarized by Fleming et al. (2008). For example, the frequency of event sequences involving loss of RPV structural integrity for the control of core heat removal and core heat generation shall be less than 1×10^{-8} per reactor year. These goals were established in part from PRA studies.

(4) Develop and evaluate RIM methodologies to achieve the reliability targets for SSC assessment. The methodologies considered in the pilot study included eliminating or reducing the damage mechanisms identified in Step 2, use of an online leak monitoring system, and periodic NDE, combined with repair and replacement. These are all designed to reduce the probability of degradation mechanisms that might lead to pipe rupture. For example, a specification was developed to ensure that a leak in the HPB could be identified within 24 hours with a probability of detection of at least 90 percent in the PBMR. Moreover, double-ended guillotine breaks of large piping need to have a very low probability of occurrence and be supported by LBB studies.

(5) Evaluate and determine the uncertainties in reliability performance. The prediction of passive component reliability for any ANLWR, including PBMR, involves very large inherent uncertainties and is clearly a major gap in the RIM program. The reliability assessment of the HPB for this pilot study addressed uncertainties in the predicted failure rates and rupture frequencies using established LWR assessment methods. The failure rate versus rupture size for the large carbon steel inline welds in the HPB were

calculated with risk-informed assessment methods used in the past for LWRs (Figure E-3 of Fleming et al., 2008). A very large uncertainty was calculated for these pipes. For example, the difference between the predicted probability of failure for a 500-millimeter (19.7-inch)-diameter break size between the mean and 5th percentile was four orders of magnitude, while the difference between the mean and 95th percentile break size was less than one order of magnitude. Recognizing the failure probabilities of passive components is a gap, defense in depth was used to address these large uncertainties. The use of LWR assessment procedures, which do not have the impact of creep or creep-fatigue (and other ANLWR damage mechanisms), is considered a gap at present.

(6) Determine the scope and parameters of the RIM program. Some of these parameters ensured a high degree of HPB reliability and reduced the potential for degradation mechanisms. In addition, the location and number of examinations for RPV shell and nozzle welds were based on the reliability goals of the PBMR design. The locations for the volumetric examinations of piping welds were determined by using multiple guidelines. Examples of the guidelines include the following: (1) 10 percent of piping welds are to be examined, (2) examinations will be performed at selected weld locations identified by the degradation mechanisms and where the consequence of a postulated pipe rupture would result in high risk, and (3) nearly 100 percent of the examination locations must be accessible.

(7) Monitor SSC reliability performance and update the RIM program. This requires repeating Steps 1–6. This step is the same as that required by the NRC in risk-informed ISI programs under RG 1.178.

Further details, along with full results, can be found in Fleming et al. (2007). This represents a typical application of the new ASME BPVC, Section XI, Division 2, RIM program.

2.2. RCC-MRx Code

Électricité de France (EDF), Commissariat à l'Énergie Atomique, and AREVA developed a complete set of codes and standards for nuclear power plants inside AFCEN (Faidy, 2011; AFCEN, 2016). RCC-MRx (Design and Conception Rules for Mechanical Components of High Temperature Nuclear Islands and Experimental Reactors) is specific to ANLWRs. RCC-MRx was developed initially for mechanical components of SFRs, research reactors, and fusion reactors, but it can also be used for components of other types of advanced reactors (Faidy, 2013).

The scope of application of RCC-MRx exclusively covers mechanical components for high-temperature nuclear installations classified as vessels, pumps, valves, piping, bellows, box structures, or heat exchangers and their supports. These codes were improved or modified based on the Phénix and Superphénix experience and will be used in future designs. The ASME BPVC and French RCC-MRx are independent of one another, but Faidy (2016) continues to provide informational updates to RCC-MRx at ASME BPVC meetings.

RCC-MRx has a complete set of rules for flaw assessment. Some of these rules are being included in the developments of the ASME Working Group on High Temperature Flaw Evaluation (WG-HTFE), along with R5 procedures. These rules are needed to perform some of the RIM assessments of ASME BPVC, Section XI, Division 2.

Lee (2015) compared RCC-MRx and other elevated temperature design codes and described the Korean position on high-temperature design. The Korean position on Gen IV high-temperature reactor design considers ASME BPVC, Section III, Division 5, rules for design and RCC-MRx A16 for defect assessment.[1] Lee, Won, and Huh (2019) developed a computer code for high-temperature defect assessment based on the RCC-MRx code.[2] Such computer codes, which Lee (2019) named "HITEP_RCC-MRx," can be of use for future applications of the RIM program.

Korea plans to have an SFR prototype plant by 2028 and an HTGR demonstration plant by 2026 and provided application of the rules for the SFR and HTGR under design in Korea (Lee, 2015). The summary by Lee (2015) describes the materials and design goals for both plants.

2.3. R5 Fitness for Service Procedure

R5 (2003) was originally developed to support the high-temperature Magnox CO_2 gas-cooled reactors in the United Kingdom. Many of the high-temperature fracture mechanics developments used to manage these plants were created during this period by Webster (1994) and subsequently added to R5. This procedure was later used to help manage the AGRs and experimental ANLWRs in the United Kingdom. R5 has a complete set of rules for high-temperature design, including rules for creep and creep-fatigue crack growth. Section 2.5 outlines the R5 procedure and past and current improvements.

2.4. Relationship Between RCC-MRx and R5

British Energy was the historical developer of R5 and maintained its development and rule changes over the years. In 2009, EDF acquired British Energy. This is a brief summary of the interaction of R5 with RCC-MRx and current developments.[3]

R5 continues to be developed and improved. EDF Energy leads this development, overseen by an industrial panel that includes Rolls-Royce, Wood Group (UK), Frazer-Nash Consultancy, National Nuclear Laboratories, EMPA (Swiss laboratory) and ANSTO (Australian laboratory). R5 was last updated in November 2014 (R5, Issue 3, Revision 002) (EDF Energy, 2014). A new update is due in early 2020. While there are some technical links between R5 and the RCC-MRx developers, the two codes remain separate with no intent of combining them at present. EDF Energy in the United Kingdom and EDF in France remain quite separate organizations at a working level. EDF Energy is not currently able to sell the high-temperature

1 A16 is the appendix that deals with crack assessment.
2 RCC-MRx is quite similar to R5. Some of the main differences are material property definitions, a stress intensity factor solution compendium, and reference stress definitions.
3 The following discussion came from a private interaction with the current R5 chair, Dr. Marc Chevalier, of EDF Energy on May 29, 2019.

R5 procedure. R5 was only available in paper copy format, but when the code is sold again, the intent is to have electronic versions available. The latest version of R5 includes substantial updates compared to earlier versions.

EDF Energy has a separate material data handbook, called R66. R66 is considered proprietary and EDF Energy does not share or sell it. It is also very focused on UK applications (i.e., the UK AGRs). R5 does contain sections on the required materials data and, at a high level, how these data can be acquired. The expectation is that stakeholders will generate their own materials data or obtain them elsewhere. Brust et al. (2010) summarized high-temperature flaw evaluation data for many materials obtained from the open literature.

EDF Energy has started a project to generate a guidance document for modular high-temperature gas-cooled reactors (mHTGRs). The project is cofunded by a UK government grant, EDF, and the industrial partners. The objective of this project is to provide a guidance document for advanced modular reactor (AMR) vendors (noting all AMR designs being considered in the United Kingdom are high-temperature reactors) on the strengths and weaknesses of the available design codes (focusing on ASME BPVC, Section III, Division 5, and RCC-MRx) and the supplementary role the R5 and R6 procedures could play to support an AMR through the UK regulatory generic design assessment. This guidance document will apparently determine the parts of R5 and R6 that are appropriate for AMR design and the necessary improvements and additions. It will also highlight where future developments will be required to make the overall approach acceptable to the regulator (e.g., accounting for environmental factors in some Gen IV plants). EDF Energy will work with the UK regulator on this project.

R5/R6 will help fulfil a role with regard to defect-tolerant safety arguments required by the UK regulator for safety critical components. This is not a requirement in the United States at present, although such procedures will be necessary to use the ASME BPVC, Section XI, Division 2, RIM program. Furthermore, EDF Energy expects this work to demonstrate the benefits of using R5 over current design codes for significant creep-fatigue loading.

The following summarizes this discussion:

- EDF Energy, the owner of the former British Energy, continues to develop R5.
- R5 and RCC-MRx remain separate with no links at present.
- R5 is currently not for sale. This will change, but EDF Energy has not specified when R5 will be available for sale. Previous versions of R5 were on paper, but the future version may be electronic.
- EDF Energy is involved in a program to establish guidance for mHTGRs in preparation for licensing activities.

2.5. Ongoing Enhancements to R5 and the Relationship to ASME BPVC, Division 5

In preparation for new mHTGR reactor development in the United Kingdom, major modifications and enhancements to R5 are underway. As discussed above, this development is led by EDF Energy and overseen by a six-member industrial panel.[4]

Four of the panel members are UK organizations, one is Swiss, and one is Australian. A new update is due in early 2020, but it is still not known when R5 might be available for use outside the industrial panel group.

Numerous enhancements to R5 have been made since its latest revision but only appear in the proprietary version at present. A summary of the new and ongoing enhancements to R5 was detailed in a recent paper presented at the 2019 ASME Pressure Vessel and Piping (PVP) conference (Hughes, Chevalier, and Dean, 2019). The operational feedback from more than 30 years of operation of the UK AGR fleet is the key driver for the R5 enhancements.

Volumes 2 and 3 of R5 have procedures for addressing defect-free structures (crack initiation) in initially defect-free structures. These procedures are similar to those in ASME BPVC, Section III, Division 5. It should be noted that Section III, Division 5, does not cover flaw evaluations, as they are discussed in Section XI. Section III of the ASME BPVC is based on preventing crack initiation. In the 1980s, the need to address defect tolerance, which the UK design codes did not explicitly cover, was recognized. This led to the development of R5 procedures for addressing creep and creep-fatigue crack growth and instability, as well as to the development of Volumes 4 and 5 of R5. Therefore, the R5 assessment procedure is divided into two stages: (1) crack initiation of the initially defect-free component and (2) assessment of the time to grow a crack to a critical size. Many of the materials in the UK AGRs are subjected to creep-fatigue loading, resulting in relatively high thermal stresses during transients and startup/shutdown cycles.

2.5.1. Lessons from Operation of the United Kingdom Advanced Gas-Cooled Reactor Fleet

The text below lists some of the critical lessons learned from AGR operation, which led to improvements to R5 for future high-temperature reactors being designed today (Hughes, Chevalier, and Dean, 2019):

- High-temperature cracking typically occurs in the heat-affected zone of weldments. This is true for all relevant high-carbon grades of austenitic steels used in AGRs for a wide range of thicknesses, regardless of whether the welds were postweld heat treated. Postweld heat treatment refers to solution annealing. The problem is usually associated with weld designs that produce high tensile weld residual stress (WRS). It is important to note that WRS relaxation, even if it does not produce cracking, does induce some amount of damage and can reduce the life of the component. SCC is driven in part by

[4] A High Temperature Center involving Imperial College, University of Bristol, Oxford University, the University of Manchester, Open University, Loughborough University, and Korea University also helps direct research development of R5.

tensile residual stress, so reducing residual stress will help with this damage mechanism. However, creep damage can be enhanced.

- Secondary stresses, such as thermal transients and WRSs, often lead to cracking.
- Cracking tends to occur in materials with low creep ductility, such as Type 316H welds.
- Carburization is an issue with AGRs because of the CO_2 gas coolant. This will not be an issue for designs using helium as a coolant.
- Probabilistic approaches have been useful in managing the lifetimes of large populations of similar components.

2.5.2. Current R5 Procedures

The R5 procedures provide a structural integrity assessment of components whose life might be limited by elastic-plastic overload, creep rupture, ratcheting, creep deformation, creep and creep-fatigue initiation, and creep-fatigue crack growth. Other issues that must be addressed include corrosion, fretting, high-cycle fatigue, and thermal aging, among others. This report provides a short overview of the current R5 procedures and relationship to ASME BPVC, Section XI.

Crack Initiation. Volumes 2 and 3 of R5 provide procedures to estimate the cyclic stresses and strains leading to creep or creep-fatigue crack initiation. The procedures estimate the number of cycles to create a crack of a defined size. The procedures use simplified and conservative methods based on elastic stress analysis. R5 has other features to address shakedown, ductility exhaustion (including multiaxial effects) for estimating creep damage, plastic collapse and ratcheting, creep rupture, creep-fatigue interaction, and negligible creep.

The R5 crack initiation approach is based on the construction of stress and strain hysteresis, in addition to dwell time effects using simple approaches. Hughes, Chevalier, and Dean (2019) included a high-level summary of the approach. Fatigue and creep damage are linearly accumulated until crack initiation is predicted. The initial flaw size is assumed to be equal to the depth of the cyclic plastic zone or one-tenth of the section thickness for the subsequent flaw assessment, discussed next. The actual rules for R5 are complex and cannot be summarized here in detail. Many of these methods were developed before the advent of computer modeling technology and are often overly conservative.

Crack Growth. Volumes 4 and 5 of R5 address the growth and instability of the crack of a given size predicted using crack initiation procedures. The ASME BPVC does not include the high-temperature crack growth process, but this is currently under development and is following the lead of R5. The methods used in RCC-MRx follow procedures similar to R5.

The crack growth predictive procedures are based on fracture mechanics parameters called the C* integral (for steady-state creep conditions) and C(t) (for transient creep crack growth). The information needed for crack growth assessments requires the defect type; a list of material data, including elastic-plastic and creep properties; and crack growth properties for creep and fatigue, along with the loads and service history. Simplified equations have been developed to estimate the creep characterizing parameters (C*, C(t)) based on the elastic stress intensity factors, component and crack geometry, and a reference stress that is related to the limit load. These assumptions are conservative and permit a simple assessment of creep and creep-fatigue crack growth. In fact, the estimates of C(t) for transient conditions have resulted in overly conservative predictions leading to the improvements discussed below. The methods for crack assessment at high temperatures, detailed in Webster and Ainsworth (1994), cannot be summarized in this report.

2.5.3. R5 Improvements for 2020 Release

Significant improvements to R5 are being made based on operating experience, and some of these are discussed below. Many of these improvements are due to one of the main architects of the R5 approach, Ainsworth, Dean, and Budden (2011) and Ainsworth et al. (2015).

As summarized by Hughes, Chevalier, and Dean (2019), extensive improvements are being introduced to R5 to address combined primary and secondary loading. In particular, the procedures to estimate the transient high-temperature fracture parameter, C(t), have been modified significantly. For primary and secondary loading, plasticity and creep interact in the estimations of C(t); Hughes, Chevalier, and Dean (2019) outlined these developments, and they will be part of the next release of R5. The new approach provides the C(t) estimates based on a time-dependent J-integral and has the same general form as the original R5 estimate with C(t) being the steady-state C* value modified appropriately by simple functions of reference stress and strain and material parameters. This will have an important effect on reducing the conservatism that currently exists for crack growth assessments in R5.

Hughes, Chevalier, and Dean (2019) extensively discussed carburization, and its impact on creep-fatigue degradation mechanisms is of significant concern. The next version of R5 will account for this degradation mechanism. It is now understood that carburization occurs at the surface of stainless steels in AGR environments due to radiolysis of the CO_2 coolant. The CO_2 cooling gas was originally considered to be relatively inert. However, because no proposed ANLWRs are expected to use CO_2 in the primary loop, it is not expected to be an issue.

In summary, major enhancements will appear in the next release of R5 in late 2020. These mainly consist of enhanced procedures to account for short-term creep crack growth in the transient regime and assessing creep-fatigue damage in carburized components. Eventually, this release will be available for purchase; however, the release date is not yet known.

3. POSSIBLE ASME BPVC, SECTION XI, DIVISION 2, CODE RECOMMENDATIONS

Several task groups in recent years, led by Oak Ridge National Laboratory (ORNL), Argonne National Laboratory (ANL), INL, and the NRC (Corwin et al., 2008), have identified and suggested some ASME BPVC design needs for next-generation ANLWRs (Gen IV). Multiple DOE, NRC, and national laboratory reviews of high-temperature materials and structural integrity issues addressing some of the code improvement needs include Corwin et al., 2008; Huddleston and Swindeman, 1993; O'Donnell, Hull, and Malik, 2008; and McDowell et al., 2011 (and many references cited therein). In addition, Turk et al. (2019) identified operating experience issues for ANLWRs. This section briefly summarizes code recommendations for ASME BPVC, Section XI, Division 2. The ISI requirements for ANLWRs within Section XI, Division 2, will require flaw evaluation procedures. These are currently being developed based on the R5 and RCC-MRx procedures discussed above. The NRC has not yet endorsed Division 2. Moreover, it is not known at this time whether ANLWR designers plan to use Division 2, regardless of NRC endorsement. The recommendations below are judged to provide enhancements that may improve the Division 2 rules.

3.1. <u>Recommendations Identified from ANLWR Operating Experience</u>

3.1.1. Recommendation 1—Fracture and Crack Growth Considerations

ASME BPVC, Section XI, does not have high-temperature crack growth and fracture rules. Historically, cracks are not acceptable in either the original ASME BPVC, Subsection NH (ASME, 2016), or the new rules under ASME BPVC, Section III, Division 5. If a flaw was detected during fabrication, it was repaired. However, undetected fabrication flaws may serve as initiation sites for creep and fatigue-driven crack growth. Weld fabrication flaws were a major source of operating experience issues in both SFRs and HTGRs (Turk et al., 2019).

R5 was developed in the United Kingdom and continually improved for the management of the Magnox and AGRs. The theory behind the high-temperature fracture mechanics methods used in R5 is based on classical asymptotic analysis techniques originally developed for elastic-plastic fracture (Rice and Rosengren, 1968; Hutchinson, 1968), where the "strength" of the asymptotic field represents the fracture parameter (for example, J-integral or C*-integral). These methods were subsequently expanded into the creep regime, where a parameter called C* (and its nonsteady-state form) represented the fracture parameters. Riedel (1987) summarizes the entire theory behind these asymptotic solutions.

Webster and Ainsworth (1994) and their team later developed simple engineering methods based on these theories to support the British HTGRs. Other consensus design codes using these methods include RCC-MRx (AFCEN, 2016) and American Petroleum Institute (API)-579, "Fitness for Service" (2017). The consensus design code methods have slight differences, but the underlying theory and techniques are related.

A joint ASME BPVC, Section III and Section XI, WG-HTFE is currently working to implement similar methods within ASME BPVC, Section XI. Brust et al. (2009, 2010) summarized plans for the WG-HTFE. In large part, the plan follows the crack assessment methodology developed in R5. This methodology was substantially adopted in API-579 and RCC-MRx. Unlike the ASME BPVC, the RCC-MRx requires the designer to consider flaws during the design phase.

Recommendation: A crack assessment procedure similar to the 13-step process outlined in Appendix II is recommended for ASME BPVC, Section XI. The details of the methods are similar for R5, API-579, and RCC-MRx design code and are cited in the potential ASME *approaches in RCC-MRx (2016), API-579 (2017), and Brust et al. (2009, 2010). The material properties used in the recommended crack assessment procedure must be qualified for all ASME BPVC materials for high-temperature applications. High-temperature flaw evaluation procedures are currently being developed in a joint Section XI/Section III working group, starting with a Code Case.*

3.1.2. Recommendation 2—Reheat Cracking and Residual Stress Relaxation due to Creep

Residual stresses often develop during plant fabrication processes. Most often, these are caused by weld shrinkage (i.e., WRS) at joins, and they can be present from other causes, such as metal forming of pipe. In LWRs, residual stresses can contribute to SCC and affect fatigue crack growth rates. In ANLWRs, additional, high-temperature damage mechanisms can affect components; for example, residual stresses in components operating in the creep regime will relax and redistribute as creep straining occurs. This relaxation causes inelastic strains (creep strains) resulting in damage. In a WRS field, this relaxation and possible damage can be complex to account for without using numerical methods.

For some materials, this relaxation of WRS can lead to reheat cracking, which occurred in several ANLWRs (Turk et al., 2019). Reheat cracking was determined to be the cause of cracking in a number of reactors, especially in vessel headers that operated at high temperature. These included the Prototype Fast Reactor in the United Kingdom and Phénix in France. As a result, WRS must be accounted for in RCC-MRx and R5 crack assessments. Often, these reheat cracking issues were corrected by changing the material, as some materials have better resistance to reheat cracking. Even if stress relaxation does not lead to reheat cracking, creep damage may still occur in a component. This may affect the high-temperature performance and reduce the life of the component. Reheat cracking is a concern for both HTGR and SFR components operating at high temperatures. Some types of austenitic stainless steel (e.g., Type 321 and 316H) are significantly more prone to reheat cracking than other austenitic stainless steels.

Finally, if a defect is found during fabrication, it is often repaired using a weld. Weld repairs should be carefully managed because they may give rise to very high tensile residual stresses in the local repair region. This could increase the potential for reheat cracking.

Recommendation: The developers of ASME BPVC, Section XI, should consider rules accounting for the assessment of WRS relaxation damage and the possible effect on cracking.

3.1.3. Recommendation 3—Thermal Expansion Stresses

Thermal expansion mismatch stresses, particularly in areas of structural constraint, must be carefully considered. These stresses have often been the source of structural integrity issues in ANLWRs, especially SFR operation. This has also been a problem during startup and cooldown transients. Phénix experienced fretting fatigue in the fuel bundles from thermal expansion effects. In addition, Phénix experienced cracking and crack growth in the intermediate heat exchanger due to a combination of thermal expansion mismatch stresses and a poorly designed joint until a redesign increased the joint flexibility, remedying the problem. The ASME BPVC, Section XI, Division 2, RIM program must account for this load mechanism.

Recommendation: Turk et al. (2019) clearly summarized the failures of some ANLWR components caused by mishandling thermal expansion stresses. Therefore, this item is included here as a precautionary note for analyses of crack growth and instability conducted as part of a RIM program. The guidance that the NRC provides for thermal fatigue should be adequate for ANLWRs. There should also be opportunities to identify thermal expansion issues, such as during startup testing.

3.1.4. Recommendation 4—Sodium Flow-Induced Thermal Striping

Similar to Recommendation 3, thermal creep-fatigue crack growth (thermal striping) caused by mixing sodium flows at different temperatures is a significant issue in SFRs. Because of the high thermal conductivity of sodium compared to water as a coolant, this issue is more likely to occur and must be managed.

Recommendation: Turk et al. (2019) clearly summarized the failures of some ANLWR components caused by thermal striping. Therefore, this item is included here as a precautionary note for the analysis of crack growth and instability in areas prone to thermal striping as part of a RIM program.

3.1.5. Recommendation 5—Buckling Limits for Time-Independent Conditions

The National Aeronautics and Space Administration (NASA) recently reevaluated the buckling rules for space vehicles used since the 1960s (Hilburger et al., 2018) through a large combined testing and modeling program. The program showed that modeling the buckling process (both global and local) using modern finite element codes (mainly ABAQUS) with an initial imperfection matched buckling test data extremely well—even for complex stiffened structures. This resulted in the rewrite of the NASA buckling standard, reducing the excessive conservatism.

Recommendation: The developers of ASME BPVC, Section XI, Division 2, should consider a modeling-based assessment of buckling for high-temperature situations using advanced finite-element procedures. The NASA experience shows that such nonlinear modeling approaches can be quite accurate with modern computational codes.

3.1.6. Recommendation 6—Leak Before Break

For LWRs, NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” establishes the LBB framework (NRC, 2007). A leak through a crack is to be detected before a break leading to a loss-of-coolant accident. Changes to Section 3.6.3, “Leak-Before-Break Evaluation Procedures,” of NUREG-0800 may be necessary to support licensing of ANLWRs. LBB issues are important for HTGR designs but have not been part of the ASME BPVC. Engineering Mechanics Corporation of Columbus, OH (Emc^2), has been extensively involved in LBB assessment for the NRC for LWRs. In particular, leak-rate models for coolants other than water need to be expanded.

Zhang et al. (2004) summarized LBB considerations for the Chinese HTR-10 that may apply to next-generation HTGRs. Section 3 of Zhang et al. (2004) provides a good overview of LBB considerations for HTGRs. As with LWRs, if LBB cannot be satisfied in an HTGR, a piping break must be postulated, and appropriate protection against the dynamic effects of the break must be provided for the safety-related SSCs. LBB analyses allow for the elimination of pipe whip restraints, jet impingement barriers, and other safety features. Formerly, the LBB methodology could not be applied to piping that was degraded by an active degradation mechanism such as SCC or creep cracking. The development of the xLPR code (NRC and EPRI, 2019) to place the active degradation mechanism of primary water stress-corrosion cracking (PWSCC) in bimetallic welds in LWRs into a probabilistic framework may permit LBB considerations to bypass this requirement.

A similar development could also be made for active degradation mechanisms such as creep, creep-fatigue, and corrosion in ANLWRs. Enhancing the xLPR code to account for these degradation mechanisms is a logical step for the assessment of ANLWR licenses. (The references in Chapter 7 provide more details on the LBB application in an ANLWR.)

Recommendation: ASME BPVC, Section XI, Division 2, or guidance for ANLWRs should incorporate LBB assessments. Since the liquid metal reactors operate at such a different pressure than LWRs, General Design Criterion 4, "Environmental and Dynamic Effects Design Bases," in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the Code of Federal Regulations *Part 50, "Domestic licensing of production and utilization facilities," should probably be relaxed in some way for ANLWRs that use liquid metals as coolants. A generic evaluation should be performed that addresses whether pipe whip constraints are needed.*

3.2. ANLWR Nondestructive Examination Considerations

Bishop et al. (2011) examined the NDE and ISI technologies for high-temperature reactors in depth. This work was performed under Task 12 of the ASME Standards, LLC, series of studies supporting Gen IV high-temperature reactor technology. The ASME BPVC, Section XI, Division 2, RIM program listed some of the NDE methods. The approach recommended in Bishop et al. (2011) reflects the RIM program under development for ASME BPVC, Section XI, Division 2 (ISI Code for HTGRs), which has been expanded and now includes an appendix for HTGRs. This effort focuses on HTGRs, although the methods may also be applied to SFRs.

ANLWRs are expected to accommodate both outage-based and online monitoring and examination. To this end, Bishop et al. (2011) introduced the concept of nondestructive monitoring (NDM). NDM targets online monitoring of active degradation mechanisms at susceptible regions (this could be online sensors that detect leaks or temperature spikes, for example). Appendix A to Bishop et al. (2011) identified the active degradation mechanisms and included those listed in the RIM program in ASME BPVC, Section XI, Division 2. The NDE and NDM methods must be capable of detecting these mechanisms. The Bishop et al. (2011) report has two parts: Part 1 conducted a technology assessment of advanced monitoring, diagnostics, and prognostics systems, and Part 2 identified new inservice NDE methods, such as acoustic emission and ultrasonic methods. The operating conditions for next generation HTGRs include an outlet temperature of up to 900 degrees C (1,652 degrees F) and a steel RPV operating temperature of 300–400 degrees C (572–752 degrees F) at helium coolant pressures of 5–9 megapascals (0.73–1.3 thousand pounds per square inch). The outage frequencies were expected to be in the range of 18–60 months, depending on the design.

3.2.1. Assessment of Past High-Temperature Gas-Cooled Reactor Nondestructive Examination Experience

The operating experience report (Turk et al., 2019) discussed NDE experience. Much of the past HTGR experience is limited by designs that are not relevant to today’s proposed configurations. Many of the past HTGR vessels were constructed as prestressed concrete pressure vessels that also enclosed helium circulators and heat exchangers. This made access to the components difficult even for traditional inspection methods. Monitoring methods included the following:

- pressure testing
- remote visual inspections that were often recorded
- helium leak monitoring—methods defined in ASME BPVC, Section XI, that can be directly applied today in HTGRs
- weld inspection using x-ray or ultrasound during shutdowns
- moisture detection systems
- gas incursion detection systems

3.2.2. High-Temperature Gas-Cooled Reactor Nondestructive Examination Methods

Bishop et al. (2011) considered three modern HTGR designs for the assessment. The choice of the ISI strategy to use for each component depends, to some extent, on the damage mechanism that needs to be detected and the material and operating conditions of the component. For each component, the degradation mechanism is identified.

The types of NDE inspections are first identified for the various components in the HTGR. NDE techniques serve as the basis for ISI programs and help determine possible degradation effects at critical locations along the pressure boundary. NDE is used primarily to detect and size degradation (e.g., wear scars, cracks, corrosion, deformations). Some of these include the following (details can be found in Bishop et al. (2011) and are not included here):

- Volumetric—Inspection methods that can identify damage and flaws internal to the component. Methods appropriate for high-temperature situations include radiography, ultrasonic (including time of flight diffraction and phased array), noncontact laser, and eddy current.

- Surface—Identifies surface or near-surface damage. Methods include magnetic particle, liquid penetrant, eddy current, magnetic flux leakage, and laser UT Rayleigh waves.
- Visual—Damage observation of the component surface. This includes direct fiber optics and remote TV (used in Phénix) infrared monitoring, and pattern image correlation analysis (used for creep monitoring in high-temperature fossil plants).
- Other needs for NDM include vibration and loose part monitoring, acoustic emission, leak monitoring, and displacement monitoring using laser profiling (distortions can affect performance). Current methods used for LWRs should be explored for application to HTGRs.

For each of the NDE methods listed above, Tables 2, 3, and 4 in Bishop et al. (2011) described the current state of the art and further development needs for application to ANLWRs. For example, improvements in sensors and robotics are necessary for high-temperature use of ultrasonic and eddy current techniques. Ultrasonic and eddy current techniques are expected to be available for ANLWRs in the short term, but other methods, such as laser UT Rayleigh wave techniques, may require significantly more development time.

LWR operators encountered difficulty in finding flaws using UT for various phenomena, such as intergranular SCC and PWSCC. This difficulty led to the creation of the performance demonstration initiative to create flawed examination specimens, detailed NDE procedures, and personnel testing to increase the probability of detection of flaws. Even with these measures, ISI missed some flaws. Based on this experience, this is a gap that must be addressed. It is recommended that creation of specimens, procedures, and personnel testing be a necessity for the ANLWRs in revisions to current standards.

ASME BPVC, Section XI, which provides rules for inservice inspection, examination, and testing of the reactor coolant pressure boundary components, also addresses repair and replacement activities in nuclear power plants. These represent a mandatory program to provide adequate safety and manage deterioration and aging effects for LWRs, which are also needed for ANLWRs.

While Section XI, Division 1, contains these rules for LWRs, the 2019 Edition of Division 2 contains rules for implementing a RIM program. The NRC has not reviewed and accepted Section XI, Division 2, at this time. Division 2 has some limited NDE requirements for ANLWRs in Appendix V, which refers to Appendix VII for the acceptance standards. Appendix VII only includes information for the Gen III LWRs and for HTGRs and refers to the acceptance standards in Division 1 for LWRs. While this may be appropriate for Gen III LWRs, it may not be appropriate for HTGRs, as Appendix VII-3.1(c) limits the design temperatures to 370 degrees C (700 degrees F) for ferritic materials and 426 degrees C (800 degrees F) for austenitic materials over the design life of the component. Rules for liquid metal reactors, MSRs, and fusion reactors are being prepared. The development of inspection requirements and acceptance standards for liquid metal reactors and MSRs is considered a gap in the standards. Justification of the use of LWR acceptance criteria for HTGRs is also considered a gap in the standards at this time.

3.2.3. Degradation Mechanism and Associated Nondestructive Examination/ Nondestructive Monitoring Method

Certain NDE or NDM techniques are preferred for identifying degradation, depending on the mechanism involved. Some of these are summarized below for HTGRs:

- *Radiation embrittlement*—Currently radiation embrittlement detection is accomplished with surveillance specimens placed in the core and subjected to irradiation. These are removed periodically and tested to obtain material data changes due to damage. Similar concepts can be designed for use within the ANLWRs. Material condition monitoring techniques are also being developed to monitor this degradation nondestructively using acoustic and electromagnetic techniques and sound velocity attenuation. This has potential application in ANLWRs but further development is needed. Ultrasonic inspection from outside the vessel using phased array (and other) methods are also becoming possible and can be used in ANLWRs.

- *Thermal stratification cycling and thermal striping*—Visual and eddy current methods are suggested to monitor this damage mechanism in ANLWRs. Local geometry changes can be monitored using eddy current gap methods, and infrared camera monitoring can identify unexpected temperature field distributions during plant operation.

- *Flow-induced vibration*—This damage mechanism has led to problems in ANLWRs in the past (Turk et al., 2019). Eddy current and ultrasonic angle beam inspection methods such as phased array should be used with remote robotic tooling to detect cracking.

- *Mechanical fatigue*—Damage should be inspected by remotely operated robotic tooling using visual inspection, magnetic particle, and liquid penetrant. For detailed characterizations of fatigue cracking, including crack sizing, ultrasound methods, including phased array and laser UT, are preferred. Acoustic emission monitoring can be used to follow crack progression.

- *SCC*—Damage can be determined for exposed surfaces using magnetic particle or eddy current methods since they can be applied using remote tools.

- *Creep and creep-fatigue*—Visual monitoring, combined with accurate local deformation measurement methods can be performed using lasers, eddy current gap measurements, or strain gauges.

- *Nondestructive monitoring*—Some methods for online monitoring are available, although development in sensor technology is necessary for prolonged use at high temperatures. In addition to new monitoring methods, ANLWRs will continue to use traditional methods, such as surveillance samples. These specimens will continue to be traditional compact specimens. However, ANLWRs may use single-edge notch, since the toughness obtained is more appropriate for deep surface cracks. Recent work (Wilkowski et al., 2019) shows toughness measured with single-edge notch specimens for deep cracks is lower than compact tension toughness values.

Bishop et al. (2011) also discussed advanced material characterization methods for HTGRs. Improperly conducted thermal treatments, inhomogeneous physical properties, creep, and residual stresses can now be detected by changes in the acoustic and electromagnetic properties. For example, magnetic Barkhausen methods can be used to observe transient pulses across a search coil placed around ferromagnetic material undergoing a change in magnetization. This method shows promise for qualitative evaluation of irradiation damage. Other methods, such as laser ultrasonic methods and electromagnetic acoustic transducers, are under development and should be practical in the near future for ANLWRs. Mechanical testing using microsamples is currently under development.

Bishop et al. (2011) presents a road map for determining and improving advanced methods and their requirements for preservice and inservice NDE. The status of these efforts is currently not available to the authors.

4. COMPUTATIONAL CODES FOR CONFIRMATORY ANALYSES FOR MATERIALS DEGRADATION AND COMPONENT INTEGRITY RECOMMENDATION IDENTIFICATION SUMMARY

Computational mechanics involves computer-based solutions and the corresponding tools to solve engineering mechanics problems. For ANLWR applications, this embraces thermal hydraulics, stress analysis, dynamics, damage, and fracture analysis.[5]

Some computational codes and computational platforms can play an important role in the NRC's licensing verification, validation, and confirmatory analyses. ANLWR vendors will use computational methods to aid in the design, performance assessment, maintenance, life prediction, and safety assessments of the reactors. This report describes some of these computational codes, along with an assessment of possible gaps and need for improvements. These codes represent commercial codes, open source codes, government codes (often developed by DOE), and codes developed by the NRC or its contractors currently used to assess reactors. The report includes some limited discussion of other computational simulation tools being developed outside the United States.

This chapter first discusses the computational codes that may play a role in ANLWR assessments, including the gaps that must be addressed to properly use these codes for licensing assessments. Next, Chapter 5 summarizes a roadmap for enhancing the codes to address ANLWR licensing needs. This final section focuses on several paths forward, including the possible enhancement of the DOE code GRIZZLY, which shows promise for future ANLWR assessment if proper enhancements are made.

[5] Computational mechanics methods are actually forms of the weighted residual methods of mathematics, where the errors to certain integral representations of the governing equations are minimized. The most popular form of computation mechanics used today is the finite-element method, although other methods, such as finite difference and control volume techniques, are also used.

Finally, the report makes some recommendations on the computational codes and their improvement to permit the NRC staff to ensure component reliability. These recommendations are intended for near-term and long-term use.

4.1. Description of Computer Codes and Possible Recommendations

The sections below list some computer codes commonly used to evaluate the structural integrity of U.S. reactors. This list focuses on computer codes used by the NRC and its contractors for structural integrity assessment of nuclear reactor components. Some industry codes (such as the probabilistic code PRAISE) are not readily available to the authors of this report. In addition, Flanagan, Mays, and Madni (2014) listed some codes used for many aspects of SFR simulation, including reactivity, thermal hydraulics, safety analysis, and others. The codes discussed below include commercial codes, NRC codes, open source codes, and government codes. These codes are either finite-element based, or special-purpose analytically based codes used for nuclear structural assessment.

4.2. Commercial Computer Codes

4.2.1. ABAQUS

The commercial code ABAQUS (Dassault Systèmes, 2019) is considered by the report's authors to be the most complete commercial computer code for handling nonlinear mechanics issues for nuclear systems and is the most significant code for NRC structural assessments. The code is very flexible, as modelers can write USER[6] subroutines to account for material laws not in the ABAQUS library (e.g., cracked pipe elements (used for seismic assessment), concrete damage models). The NRC and its contractors are intimately familiar with ABAQUS and have a library of USER subroutines. The NRC staff uses a version of the Emc^2 code VFT,[7] which is tied to ABAQUS, for weld modeling of three-dimensional (3-D) structures. ABAQUS has very sophisticated fracture and crack growth modeling features that are simple to use, including the Extended Finite Element Method (XFEM) (which can be used to assist in fracture parameter calculations such as stress intensity factor), and has excellent mapping features using different meshes (often required for multiphysics solutions and crack growth modeling in WRS fields). The solution process on multiple processors is very efficient.[8] ABAQUS has features for special

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7 USER subroutines are computer programs written by the person using the finite-element code. These are usually written in FORTRAN and are proprietary to the organization using the code (ABAQUS, for example, does not have access to these USER subroutines), and they are invoked and compiled during the analysis. Special USER subroutines are necessary to perform analyses using ABAQUS that are not in the computer code library. One example is a creep material law that is not in the ABAQUS library of available material laws. The user writes this code to model unique features. For ANLWRs, this might be a complex creep material law not available in ABAQUS.

7 The VFT (Virtual Fabrication Technology) code is a finite-element code used to predict the WRSs and distortions caused by the welding process. These WRSs are then often used to perform crack growth analyses (such as corrosion crack growth), where the crack is driven by these stresses. VFT consists of a thermal solver (to predict temperatures caused by welding), utility subroutines, and a weld-specific material USER routine to address complications caused by material melting and resolidification. It interfaces with ABAQUS for the structural portion of the solution.

8 For finite-element solutions that require large models, in conjunction with the inclusion of nonlinear effects, the solutions may take a very long time. ABAQUS has very efficient procedures for running the solution on

purposes, such as the structural stress fatigue modeling procedures and optimization routines. Leak-rate modeling and NDE (such as ultrasonic) modeling are also possible with ABAQUS.

ANLWR Gap Assessment—ABAQUS is a powerful general-purpose code regularly being updated. ABAQUS also permits multiphysics solutions and the addition of different deterministic damage mechanisms. USER subroutines will be necessary to model corrosion damage, creep, and irradiation-induced swelling damage. The ABAQUS code can be prohibitively expensive, especially if access to additional models, such as advanced manufacturing modeling and fatigue modules, is necessary. Section 5.5 provides a roadmap for detailed development to address ANLWR structural assessment needs.

4.2.2. ANSYS

ANSYS (2019) is a powerful general-purpose, finite-element code similar to ABAQUS. Many nuclear plant owners, builders, and contractors use it for structural integrity assessment of nuclear components. Like ABAQUS, users can write FORTRAN subroutines to add additional capability.[9] ANSYS also has fracture and crack growth modeling features but may be more limited than ABAQUS. The solution process on multiple processors is efficient and similar to the ABAQUS performance. It also has features for special purposes, such as the structural stress fatigue modeling procedures (n-Code), in addition to optimization routines. ANSYS permits easy ASME code assessments. For example, the ASME BPVC classifies stresses according to primary, secondary, and peak stresses, and ANSYS does this automatically, while other codes such as ABAQUS do not. The acquisition of the FLUENT fluid modeling code by ANSYS gave it powerful fluid and fluid structure interaction capabilities. The ANSYS code has undergone a nuclear quality assurance evaluation, while ABAQUS has not. ANSYS design analysis software is certified under International Organization for Standardization (ISO) 9001. Product development, testing, maintenance and support processes also meet the NRC's quality requirements, as they have for nearly four decades.

ANLWR Gap Assessment—ANSYS could be used for further NRC development to address ANLWR needs described in Chapter 5. However, the NRC staff and contractors have more experience with ABAQUS, and many NRC and contractor personnel have developed USER routines for ABAQUS. Therefore, a choice to adopt ANSYS to perform confirmatory analyses would require an investment of time and resources on the part of the NRC staff and its contractors. However, many utilities use ANSYS.

4.2.3. Other Commercial Codes

Several other commercial finite-element codes could be used for ANLWR structural, life prediction, and fracture assessment. These include MSC/NASTRAN, ADINA, COMSOL, and LS-DYNA, among others. This report does not discuss them as they are not used for nuclear applications as often as ABAQUS and ANSYS.

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many of the computer's processors or cores. For example, using 20 cores for a particular problem might reduce the computer time by factor of 15 to 20. Some finite-element codes are not as efficient.

9 USER subroutines are written in a computer language such as FORTRAN, C++, Python, and possibly others. FORTRAN is often the choice, since it runs very efficiently with ABAQUS, but other languages are also used.

4.3. NRC-Developed Computer Codes

4.3.1. Extremely Low Probability of Rupture (xLPR), Version 2

NRC licensing and maintenance assessment procedures focus on the use of risk-based assessment of LWRs. The Extremely Low Probability of Rupture (xLPR) code has been developed under a memorandum of understanding between the NRC and the Electric Power Research Institute (EPRI). It is a probabilistic code to assess a variety of structural integrity issues related to nuclear reactors, including active degradation and LBB of piping systems (NRC and EPRI, 2019). A Monte Carlo method is used to estimate the probability of adverse events (including, but not limited to probability of crack occurrence, leak occurrence, rupture, and loss-of-coolant accidents) based on many deterministic runs. The code can also be used deterministically.

The xLPR code might be used to predict the possibility of a loss-of-coolant accident in an RPV nozzle subjected to PWSCC. A PWSCC crack might initiate, grow due to PWSCC, and become unstable or begin leaking. The possibility of the crack being detected and repaired is also considered. Each of the mechanisms discussed above has a separate FORTRAN module predicting the deterministic process. Section 4.3.5 describes some of the deterministic codes, including crack-stability and leak-rate models. For each mechanism, there are variables, including loads and the probability of crack detection before and after leaking, among others. The xLPR code consists of a number of modules, along with a probabilistic engine (called GoldSim) to predict the probability of a loss-of-coolant accident. This code became a necessity once the active degradation mechanism of PWSCC began to occur in plants. ANLWRs likewise experience additional active degradation mechanisms, such as creep crack growth, and the risk-based assessment procedure of xLPR may be an important tool for use in evaluating the safety risks of ANLWRs.

To run a large number of realizations in xLPR,[10] the deterministic modules are used, rather than a finite-element-based approach, so rapid risk assessments can be made.[11] The modules use analytical approximations based on accepted analysis procedures such as fracture mechanics methods (J-tearing theory, weight function methods (for stress intensity calculation)) and a Henry-Fauske model for leak-rate predictions that are necessary for LBB assessments. These methods have been independently verified and validated for each individual module using finite-element methods or fitting with observed data. The joint NRC-EPRI team has also verified and validated the framework, assembling all modules into a coherent ensemble and using ISO 9001 standards, as well as the selected Monte Carlo procedures (including Latin hypercube sampling, discrete probability distribution, and importance sampling).

10 A realization is a ‘run’ with a given set of parameters for each of the deterministic modules. For example, a realization might use a set of yield stresses, elastic moduli, fracture parameters, or crack growth parameters, each value of which is determined from random sampling of the available data. Thus, for a particular risk assessment, there might be 100,000 realizations included in a Monte Carlo-based simulation to predict the probability of rupture or leakage. The xLPR code has about 500 random variables, of which 40 or so play an important role in the failure probabilities.

11 If each of the deterministic models were finite-element based, the solution time would be excessive.

The xLPR code was designed to be modular so improvements and changes to each deterministic module can be performed easily. Moreover, it is possible to add deterministic modules to account for different damage mechanics, such as creep or irradiation damage. Each aspect of the xLPR code has undergone a full nuclear quality assurance evaluation.

ANLWR Gap Assessment—xLPR is a powerful risk-based nuclear piping assessment and LBB analysis code now becoming available for use. It may be convenient to implement ANLWR damage mechanisms into the code as different modules. Modules could be developed to evaluate the following:

- creep and creep-fatigue high-temperature crack growth and stability
- leak rates for sodium and healing leakage in SFR and HTGR piping, respectively
- effectiveness of inspection
- irradiation embrittlement
- concrete damaging cracking

At the present time, xLPR could theoretically include nonlinear finite-element modules, but the solution time may be too long for practical use. To ensure convergence for low-probability events, the xLPR code must often perform hundreds of thousands of simulations. However, surrogate modeling methods may be able to account for the results of numerous finite-element solutions. Section 5.4 provides a roadmap for detailed development to address ANLWR structural assessment needs.

4.3.2. PROMETHEUS

PROMETHEUS (Kurth et al., 2019) is a probabilistic code to assess a variety of structural integrity issues related to nuclear reactors, including active degradation and LBB of piping systems. This code can also be used deterministically. The code uses the same modules present in the xLPR code but includes other damage mechanism models for fatigue and SCC damage. For deterministic analyses with the same damage models as in xLPR 2.0, PROMETHEUS produces identical results but is 60 to 150 times faster. This code was developed using Emc^2 internal research and development funds to implement the adaptive sampling method for Monte Carlo simulations and then funded by the NRC to become a preprocessor for xLPR 2.0. The standalone software package PROMETHEUS (which the NRC has access to) contains numerous deterministic modules, including modules to evaluate the following:

- stress corrosion initiation and crack growth
- fatigue crack initiation and growth
- leak-rate prediction module
- stability predictions for both surface and through-wall crack in pipe
- inspections for crack detection and leak detection

The evaluations are all within a probabilistic framework permitting risk assessment of nuclear piping systems to be performed. PROMETHEUS is also modular, so additional deterministic modules that account for different damage mechanics such as creep or irradiation damage are possible. An advantage of not having the probabilistic framework tied to GoldSim is that modifications are much easier to perform and runtimes are one to two orders of magnitude faster than the xLPR code. However, the PROMETHEUS code did not undergo a nuclear quality assurance process.

ANLWR Gap Assessment—ANLWR damage mechanisms could be implemented into PROMETHEUS as different modules. In addition, many other ANLWR damage modules could be developed and used in PROMETHEUS for ANLWRs. At the present time, PROMETHEUS could theoretically include nonlinear finite-element modules, but the current solution time would be impractically long. However, surrogate modeling methods may be able to account for results of numerous finite-element solutions. Section 5.4 provides a roadmap for detailed development to address ANLWR structural assessment needs.

4.3.3. FAVOR_

FAVOR (Fracture Analysis of Vessels, ORNL) is a probabilistic structural integrity code for vessel assessment for LWRs (Bass et al., 2016). ORNL has been developing this code for the NRC for 25 years. Analysts from the nuclear industry and regulators at the NRC have applied FAVOR, including the 2017 release, v16.1, to perform deterministic and probabilistic fracture mechanics analyses to review, assess, and update regulations designed to ensure the structural integrity of aging, and increasingly embrittled, nuclear RPVs.

Early releases of FAVOR were developed primarily to address the pressurized thermal shock issue; therefore, they were limited to applications involving pressurized-water reactors subjected to cooldown transients with thermal and pressure loading applied to the inner surface of the RPV wall. Current versions of the FAVOR code encompass a broader range of transients (heatup and cooldown) and vessel geometries, addressing both pressurized- and boiling-water RPVs. This includes improvements to the consistency and accuracy of the calculation of fracture mechanics stress-intensity factors for internal surface-breaking flaws and shallow flaws. Those improvements were realized in part through implementation of the ASME BPVC, Section XI, Appendix A, A-3000 curve fits into FAVOR. FAVOR has recently been coupled to the INL GRIZZLY code as discussed in Section 4.6 (see Spencer, Hoffman, and Backman, 2019). In this version, some of the modules use finite-element solutions and some use the analytical solutions already within GRIZZLY. Section 4.6 discusses the potential use for GRIZZLY in much more detail.

ANLWR Gap Assessment—The FAVOR code was specifically developed to predict the probability of failure of LWR pressure vessels. ANLWR damage mechanisms can be implemented into FAVOR as different modules for thermal shock assessment of SFRs. There are several areas of concern for thermal shock in SFRs, including the RPV, intermediate heat exchangers, and steam generators. Irradiation damage mechanisms specific to SFRs and HTGRs can be added as modules for these specific applications, as they already exist for LWRs. Graphite damage modules could be added for HTGR systems. As discussed in Section 4.5.1, FAVOR has already been tied to the GRIZZLY code. Section 5.3 provides a roadmap for development to address ANLWR structural assessment needs, with FAVOR tied to GRIZZLY.

4.3.4. ALT3D

ALT3D is a fracture mechanics code based on the finite-element alternating method (FEAM) (ALT3D, 2016). FEAM alternates between the closed-form solutions for crack face tractions on an elliptical crack in an infinite body and the finite-element solution for the uncracked body. The crack can be any part of the ellipse intersecting the finite-element mesh. It is somewhat analogous to the XFEM method, except FEAM uses a closed form solution instead of enriching elements near the crack tip with the singular crack displacement field. Arbitrary loading is possible (e.g., displacements, pressures, forces, WRS, thermal gradients). Advantages of FEAM over XFEM are that FEAM only requires very coarse meshes while XFEM uses very fine meshes for accuracy. The FEAM solution time is superior to XFEM. There is no use of contour integrals to obtain K, as these are automatically evaluated using the singular crack field. XFEM can handle more general crack shapes than a partial ellipse. Emc^2 has used FEAM for crack growth and fracture assessments for the NRC on numerous occasions over the last 20 years. An example is for crack growth and leakage assessment of control rod drive J-welds in the RPV head (Brust et al., 2011).

ANLWR Gap Assessment—ALT3D is a deterministic code that can be used for ANLWR fracture assessments. Creep crack growth methods are based on use of the elastic stress intensity factor solution along with the reference stress (Webster and Ainsworth, 1994; Hughes, Chevalier, and Dean, 2019). Thus, ALT3D can be added to a framework for creep fracture assessment or SCC assessment. Moreover, ALT3D could be coupled to the xLPR, PROMETHEUS, or GRIZZLY codes as a convenient way to obtain stress intensity factors for complex geometries. ALT3D also has a nonlinear modeling capability for calculating J (and C* for creep) based on the reference stress method in Webster and Ainsworth (1994). However, this method within ALT3D needs to be verified for ANLWR conditions.

4.3.5. NRCPIPE, NRCPIPES, SQUIRT, and Related Codes and Modules from xLPR

NRCPIPE, NRCPIPES, SQUIRT, and similar codes are used for fracture assessment and leak-rate modeling. Many of the subroutines served as a starting point for the deterministic modules in xLPR and PROMETHEUS. These are essentially “standalone” codes used for LBB assessments. NRCPIPE is a through-wall crack elastic-plastic stability code for pipe based on numerous J-estimation schemes. NRCPIPES is like NRCPIPE except it analyzes for surface cracks in piping (as opposed to through-wall cracks). SQUIRT is the standalone leak-rate code. Other codes use Ramberg-Osgood fitting subroutines, J-resistance fit subroutines, and others. Updated versions of these codes serve as deterministic modules for xLPR, as discussed in Section 4.3.1.

ANLWR Gap Assessment—These commonly used codes for NRC LBB assessments are dated and should be modernized. If the codes are updated, they can be made to accommodate ANLWRs and their damage mechanisms.

4.4. Open Source Codes

4.4.1. Introduction

Numerous open source codes may help the development of ANLWR analyses with appropriate modifications. This report only discusses WARP3D for the following reasons:

- The authors have extensive experience using WARP3D and have helped improve or modify the code to permit features such as computational weld modeling and have helped the WARP3D architect (Dr. R. Dodds) add and improve features. These include features such as a radial return plasticity algorithm (to improve thermal plasticity solution convergence) and testing out the new contour subroutines for J-calculation, including the stress history (e.g., residual stresses, plastic strains—something commercial codes currently lack). WARP3D can perform computational weld analyses using the VFT software with UMAT, among others.
- WARP3D has extensive features that can be used at present for ANLWR damage assessment and crack growth analysis.

4.4.2. WARP3D

WARP3D is an open source production-quality code, originally developed by Dr. R. Dodds and his collaborators at the University of Illinois, Champaign. WARP3D can perform many of the same fracture or weld model assessments that can be made with ABAQUS. The code is undergoing continuing development to meet the challenges of large-scale, 3-D solid simulations for focused investigations on fatigue and fracture mechanisms and behavior in metallic components and structures. While Dodds and his collaborators have made many of the developments, it is an open source code, so others make improvements as well. WARP3D provides an alternative computational resource for this narrower class of simulations, in comparison with the expansive, general-purpose commercial codes (e.g., ABAQUS, ANSYS) and evolving families of the nonpublic, multiphysics-driven codes at the national laboratories (e.g., SIERRA, GRIZZLY, MOOSE, DIABLO). The WARP3D source code, ready-to-run executables, extensive technical and user documentation, verification and example problem suites, postprocessors, and documented workflows are provided under the University of Illinois, National Center for Supercomputing Applications, open source license (allows free unrestricted use, modification, redistribution, commercialization).

Several unique features of the code that support challenging simulations to understand fatigue and fracture processes for ANLWRs include the following:

- The code has a comprehensive library of solid and interface-cohesive elements, all supporting finite-deformation behavior.
- The code has an array of constitutive models for metals, including temperature, strain-rate, creep and finite-deformation effects; extensive models for crystal plasticity simulations of face-centered cubic, body-centered cubic, hexagonal close-packed, and single-slip systems with multiple options for slip-rate relationships; 3-D nonlinear

cohesive constitutive relationships with mode I-II-III interactions; boundary cavitation/slip models; nonlocal effects on cohesive behavior; and finite interface separations-rotations. All are implemented in a robust finite deformation formulation based on decompositions of the deformation gradients **F**.

- The computation of the 3-D *J*-integral includes the combined effects of residual strains/stresses, crack-face loading, thermal loading, inertia, and functionally graded and anisotropic materials with arbitrary orientations. The capabilities introduced in fall 2018 support thermomechanical process simulations with extensive plastic deformations (e.g., bead-by-bead weld simulation using VFT) before insertion of one or more cracks. This unique and complete *J*-integral formulation in the code provides path (domain) independent values. The code has interaction integrals to compute K_I, K_{II} , K_{III} and *T*-stress components for linear elastic solutions.

- Options to grow cracks during a 3-D simulation include (1) node release through manual or automatic tracking of the crack tip opening displacement/crack tip opening angle along a front and (2) manual or automatic tracking of element damage/degradation and extinction (Gurson-Tvergaard, SMCS) and 3-D triangular/quadrilateral interface elements with degrading, mixed-mode local/nonlocal cohesive behavior. Crack growth drivers adaptively govern the global solution processes to avoid the truncation of highly local damage evolution along crack fronts.

- The code has general 3-D mesh tying, rigid body contact, and an extensive library of model loading capabilities.

- An exceptionally robust, globally implicit-iterative solver with multilevel adaptive sub stepping is combined with line-search and extrapolation. WARP3D employs an industry-leading, high-performance Paradiso equation solver (free in Intel's MKL), allowing models with millions of nodes or elements on single workstations and clusters.

- The code has plug-compatible interfaces with ABAQUS USER subroutines and UEXTERNALDB to incorporate existing, highly specialized constitutive models and external data access.

- Multiple output schemes include support for the visualization software ParaView.

ANLWR Gap Assessment—WARP3D is a powerful code that can immediately be applied to ANLWR assessments. Some additional capabilities are recommended. The development of a WARP3D-specific mesh generator and input file format code options listed would improve the capabilities of WARP3D. WARP3D does not have SCC growth modeling based on stress intensity factor solutions or irradiation damage models, and these would require development. WARP3D can model creep and creep crack growth. However, additional constituent laws will be required to model ANLWRs. Modifications can be made directly in the code, since it is open source. Section 5.5 provides a roadmap for detailed development to address ANLWR structural assessment needs, along with ABAQUS.

4.5. Government Codes

The sections below discuss the evolving families of nonpublic, multiphysics-driven codes at the national laboratories (e.g., GRIZZLY, SIERRA, MOOSE, DIABLO). The comments were based on reviewing numerous technical reports and publications and on discussions with several experts working at the DOE laboratories who have experience using these codes.

4.5.1. GRIZZLY

GRIZZLY is a powerful code that has recently been tied to nuclear vessel analysis and the FAVOR code. The main development of GRIZZLY occurs at INL with some other DOE laboratories also participating, particularly ORNL. Spencer et al. (2013 and many references cited therein) are the main users at INL for the RPV assessment. GRIZZLY operates within the MOOSE object-oriented finite-element framework for the development of tightly coupled multiphysics solvers. It is a comprehensive framework into which specialists can add in their specific analysis codes and needs or use the finite-element codes, material subroutines, and others available in the MOOSE library. This environment allows for the addition of new physics modules in a simple fashion, permitting the convenient and rapid development of GRIZZLY. For example, one combined fluid, thermal, and structural problem can be solved using GRIZZLY. The meshes for all three problems (fluid, thermal, and structural) can be different, and the mapping algorithms transfer the necessary analysis data between codes for the solution of the different multiphysics problems. The MOOSE environment handles transfers between individual solvers (e.g., iterative, direct) for each problem. The MOOSE framework is not simple to use and requires experience, compared with commercial codes such as ABAQUS or ANSYS. In addition, solution speeds within the MOOSE framework may be slower than commercial codes and WARP3D because of the convenience of adding physics modules seamlessly.

GRIZZLY recently supported light-water sustainability efforts but will also be used to evaluate ANLWR designs. GRIZZLY can be used for characterizing the behavior of nuclear power plant SSCs subjected to a variety of age-related degradation mechanisms. GRIZZLY simulates the progression of aging processes as well as the capacity of aged components to safely perform as these modules are added to the framework. GRIZZLY includes capabilities for engineering-scale, thermomechanical analysis of RPVs and will ultimately include capabilities for a wide range of components and materials. GRIZZLY is in a state of constant development, and future releases will broaden the capabilities of this code for RPV analysis, as well as expand it to address degradation in other critical nuclear power plant components.

GRIZZLY can model the effects of neutron fluence, irradiation embrittlement modeling, and thermal aging in materials, including molecular dynamics simulations and chemical kinetics modeling. It can model thermal heat transfer, heat flux, conduction, and thermal stresses. The structural portion of the solutions can handle thermal expansion and multiple materials, along with arbitrary loads and boundary conditions. It has 3-D submodeling capabilities. GRIZZLY can also perform fracture mechanics calculations of fracture-based integral parameters, along with the recent addition of the XFEM method for easy fracture assessment. GRIZZLY permits the use of both elastic and ductile damage modeling with a Gurson model and Beremin model for brittle response, in addition to the capability for cohesive zone modeling, including the possibility of ductile-to-brittle transition modeling. GRIZZLY has been used to model heat transfer, moisture diffusion, creep, and fracture processes in concrete (Huang and Spencer, 2016). Some recent applications of GRIZZLY include modeling RPV performance within a probabilistic framework provided by the FAVOR code (Dickson et al., 2013; Bass et al., 2016). The MOOSE/GRIZZLY codes are perhaps the most useful suite of U.S. Government codes for ANLWR assessment at present, with users at INL, ANL, and ORNL.

ANLWR Gap Assessment—GRIZZLY is constantly being updated. The MOOSE multiphysics environment permits the addition of different deterministic damage mechanisms to GRIZZLY (e.g., corrosion, creep, crack growth). More importantly, the GRIZZLY architecture permits links to analytically based modules alongside the finite-element base solutions modules. This was done with the FAVOR module to speed up the probabilistic solutions for some modules. This can permit the inclusion of some xLPR modules to speed up solution times. The MOOSE environment is convenient for the purpose of adding different ANLWR degradation mechanisms but requires a computational modeling specialist to ensure accurate solutions are obtained. Section 5.3 provides a roadmap for detailed development to address ANLWR structural assessment needs.

4.5.2. SIERRA

SIERRA is Sandia National Laboratories' (SNL's) engineering mechanics simulation code suite (SNL, 2019). SIERRA is similar to GRIZZLY (a multiphysics code within an analysis framework), and much of what was written above for GRIZZLY applies to SIERRA. The SIERRA code suite is older and has many of SNL's finite-element codes already resident within the suite (e.g., PRONTO and JAC). This suite includes coupled simulation capabilities for thermal, fluid, aerodynamics, solid mechanics, and structural dynamics. These simulation capabilities are used to predict the performance of a system in normal operation, as well as the response of a system in abnormal environments, such as a fire, which can be modified and used for ANLWR assessments.

ANLWR Gap Assessment—The GRIZZLY assessment also applies to SIERRA. However, SIERRA is older and perhaps further along in development. It is not known if its use is more suitable for ANLWR assessment.

4.5.3. DIABLO and ALE3D

The DIABLO code (Parsons et al., 2007; Hodge, Ferencz, and Solberg, 2013) developed at Lawrence Livermore National Laboratory (LLNL) uses implicit, Lagrangian finite-element methods for the simulation of solid mechanics and multiphysics events over moderate-to-long time frames. It is integrated into the SHARP multiphysics toolkit to support coupled neutronics and thermo-fluid-structural calculations, among other capabilities. Extensive features have been added recently to perform high-fidelity modeling of the additive manufacturing process.

ALE3D, another LLNL-developed code, is a multiphysics numerical simulation software tool using arbitrary-Lagrangian-Eulerian (ALE) techniques. The code is written to address both two-dimensional (plane and axisymmetric) and 3-D physics and engineering problems using a hybrid finite-element and finite-volume formulation to model fluid and elastic-plastic response of materials on an unstructured grid. The ALE and mesh relaxation capabilities broaden the scope of application in comparison to tools restricted to Lagrangian-only or Eulerian-only approaches, while maintaining accuracy and efficiency for large, multiphysics and complex geometry simulations. For some applications, ALE can deliver accuracy similar to Eulerian techniques using as few as one-tenth the number of mesh elements and a reduction in memory requirements. Beyond its foundation as a hydrodynamics and structures code, ALE3D has multiphysics capabilities integrating various packages through an operator-splitting approach. Additional ALE3D features include heat conduction, chemical kinetics, species diffusion, incompressible flow, a wide range of material models, chemistry models, multiphase flow, and magnetohydrodynamics, which can be used in numerous combinations for long (implicit) to short (explicit) time-scale applications.

ANLWR Gap Assessment—The LLNL suite of finite-element codes permits high-fidelity modeling of many of the features necessary for modeling ANLWR structural and damage mechanisms. However, its use for practical damage modeling of ANLWR systems in a timely fashion may be a challenge for many structural analysts.

4.6. Non-U.S. Simulation Codes

Other countries also have computational development programs. The ASTRID program is a collaboration between France and Japan to develop and build an SFR and develop a suite of simulation codes. Devictor (2018) summarized the collaborative French/Japanese SFR simulation program. The simulation suite will consist of predictive multiphysics and multiscale simulation tools and optimized design tools to support all aspects of ASTRID development. The suite of codes will be able to simulate all situations (normal operation, incidental, and accidental situations) to justify the design and support a robust safety demonstration. The simulation suite will have provisions for examining the coupling of computational fluid dynamics and thermomechanical modeling to assess the behavior of structures of the entire nuclear island within the framework of a high-performance computing environment. Simulation topics include sodium leakage, severe accident analysis, and optimization of the RCC-MRx and JSME design codes, among others. This suite of codes is expected to be available to support licensing decisions in late 2020. French and Japanese SFR operating experience will be used to validate the computational codes, along with new tests being planned. In particular, the severe accident analysis simulation process is a very important part of this program.

4.7. Computational Graphite Modeling

This section is a short summary of computational modeling efforts and available codes related to graphite. Efforts to evaluate graphite are particularly strong in the United Kingdom. Many of these modeling efforts use either a commercial code or an open source code with special graphite damage mechanisms implemented either with the use of USER Material subroutines or directly as subroutines into open source codes. For crack growth, XFEM crack modeling and cohesive zone modeling of damage development have been used. The major degradation issues of concern include acute oxidation (from accidents), normal operational oxidation, irradiation and creep (and swelling) degradation, wear and abrasion, and fracture initiation. The porous behavior of graphite, along with material property changes that occur during high-temperature service and irradiation exposure, are challenges. Burchell (2018) summarized graphite manufacturing and degradation mechanisms, as well as the temperature dependence of graphite thermal and structural properties. Windes, Rohrbaugh, and Swank (2017) at INL provide a more complete summary of graphite properties and the change of these properties under ANLWR operating conditions.

Windes, Davenport, and Burchell (2019) summarized the ongoing development of codes that handle the degradation mechanisms of graphite as part of the INL and ORNL Advanced Reactor Technologies program. The group is developing an extensive database as part of that program, using a number of different grades of modern graphite to be found in future ANLWRs. The database also involves subjecting the graphite specimens to extensive irradiation to categorize the damage development with dose. This will lead to enhanced understanding of graphite degradation, improved graphite degradation models, and computer codes for use in ANLWR assessment.

In a critical review paper, Marsden et al. (2018) discussed the three most important graphite properties that change with exposure to radiation—namely, dimensional change, irradiation creep, and thermal expansion—and the difficulty in modeling these issues. These three effects must be properly included in the reactor design for the lifetime of the reactor. Graphite-moderated reactors have been used mainly in the United Kingdom and Russia, and this is the reason the United Kingdom has produced many graphite studies. An excellent example in Marsden et al. (2018) illustrates the finite-element modeling challenges with graphite (finite-element code not specified). A bore hole in a brick with irradiation fluence within the hole is considered. The fluence and temperatures are largest at the bore ID and reduce away from this region. This causes the brick to shrink faster at the bore than at the outside, and properties vary across the block. This results in tensile hoop stresses at the bore, which then reverse with time near the end of reactor life. The irradiation creep tends to relax stress in graphite, which is advantageous. When the reactor is shut down and the graphite cools to ambient temperatures, the stresses change significantly. The design must consider irradiation-induced stresses and stresses generated by thermal expansion and contraction. It must also consider stresses created by the thermal expansion differential between graphite and the metallic components in contact.

4.7.1. ABAQUS (Multiple Graphite Damage USER Subroutines)

The NRC and ANL (Mohanty et al., 2012; Mohanty, Majumdar, and Srinivasan, 2013) developed a finite-element approach using ABAQUS and multiple USER subroutines to model the structural integrity of graphite core components under extreme temperature and irradiation conditions. HTGRs use numerous graphite blocks for reflecting neutrons and containing fuel. Irradiation-induced dimensional changes and creep strongly influence structural integrity and are considered in this modeling study. A coupled thermal-irradiation multiphysics structural analysis methodology is implemented into ABAQUS and used to solve several problems in this effort. In the ANL model, an ABAQUS UMAT[12] was developed that calculates flux, neutron fluence,

[12] A UMAT is a specific type of USER routine that represents a special material routine. For example, when

material properties based on both temperature and fluence, and creep strains at each integration point.

The code was used to model a hexagonal fuel brick geometry in an HTGR environment. Predictions of principal stresses versus time were calculated and compared with the tensile and compressive strength of irradiated graphite. Predicted compressive strength was about four times that of tensile strength, which was observed in measurements validating the procedure. Moreover, it was predicted that the maximum stress level after 7 years of operation approximated the irradiated ultimate strength, with a safety factor of 1.4. The results were discussed in terms of design and development of a graphite replacement schedule.

4.7.2. ABAQUS with UMAT

A collaborative effort among the University of Nottingham, the University of Bristol, and the University of Manchester in the United Kingdom (Kyaw et al., 2014) involved implementation of a proprietary mechanical constitutive USER routine (or UMAT) in ABAQUS to model crack growth in graphite. The effort used an ABAQUS USER routine, combined with a damage model, to simulate crack growth within the graphite bricks. The damage model is a linear traction separation model commonly used with ABAQUS and with the ABAQUS XFEM feature to model crack initiation and growth. Cracks were predicted to initiate in the vicinity of the keyway roots. Oxidation occurs and was modeled, followed by subsequent porosity changes and weight loss. This results in significant changes in the material properties of graphite and creates residual stress fields within graphite bricks. The results were qualitatively compared with observations in the field but a direct quantitative comparison was not provided.

4.7.3. ASTER (Open Source Code)

A recent Ph.D. thesis, funded by EDF Energy for the UK HTGR, summarized recent efforts to model graphite and included a comprehensive literature review of graphite modeling (Crump, 2017). It showed how the use of graphite damage introduced by the development of a USER material routine that permits irradiation and creep damage to be accounted for correlated well with experimental data. In addition, it created special USER subroutines to account for graphite damage development during crack initiation and growth.

Crump (2017) implemented XFEM into the open source ASTER code. This implementation included cohesive zone modeling within the framework of XFEM, which is a new approach, and the use of a USER material routine (UMAT). One focus of Crump's work was to model a graphite cracking mechanism called prompt secondary cracking specifically in the Hunterston-B AGR plant. This cracking mechanism occurs near an existing crack and is caused by stress waves traversing the graphite bricks. The damage mechanisms in UMAT accounted for velocity toughening (graphite toughness increases with crack speed due to microcracks ahead of a main crack), irradiation-induced material degradation, creep and material aging effects with multiple 3-D dynamic crack initiations, and propagations and arrests into a single model. The implementation was benchmarked experimental results. The approach was able to successfully

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one is performing a computational weld analysis, a UMAT is necessary to handle the specific issues associated with weld modeling (e.g., material melting/resolidification) that are not in the library of material laws available in the code.

model prompt secondary cracking and give insight into features that could not be investigated previously, including finer-scale heterogeneous effects on a dynamic crack profile, comparison between primary and secondary crack profiles, and 3-D crack interaction with a graphite block hole, including insight into possible crack arrest.

4.7.4. Computer Codes in the United Kingdom for Graphite Modeling

The United Kingdom has an advanced set of numerical codes for use in assessing many aspects of graphite use in AGRs. This came from the need to maintain the UK AGRs over the last 30 years. Marsden (2019) listed the computational codes summarized in the table below for both static and seismic analysis with the code modeling purpose briefly described. Details of the codes are publicly available (Energy, 2019), and all of these codes are available for lease through Wood Engineering.

Table 1 List of Numerical Codes Used in the United Kingdom for Assessing Graphite in Nuclear Applications

Code	Purpose
PANTHER	Gas pressures and temperatures
MCBEND	Fast neutron fluence and energy deposition
OCTANT	Gas pressures and temperatures
ABAQUS UMAT	Graphite component stresses
FEAT-DIFFUSE	Radiolytic weight loss
FEAT-GRAPHITE	Graphite component stresses
EIM	Graphite irradiated property equations
SABRE	Probabilistic prediction of bricks stresses
AGRIGID	Core displacements static
GCORE	Core displacements seismic
LEWIS	Control rod entry
Manchester UMAT	Graphite component stresses

The section above discussed the Manchester UMAT code development with the use of ABAQUS. In addition, EDF Energy has some in-house codes that are not available for review. The FEAT computational codes are particularly noteworthy and considered to provide good predictions of graphite damage development within a reactor environment.

4.8. Key Modeling Topics

The following describes some key high-level modeling needs and goals for the computational codes key to successful licensing and operation of future ANLWRs:

- *Material data*—The computational tools are limited by material inputs. Complete material databases characterizing both material data and damage mechanisms are very important. These data must be identified and developed simultaneously with the computer code development.

- *Creep, creep-fatigue, corrosion, and crack growth*—The codes must be able to model these damage mechanisms with high fidelity within a reasonable time. The results should also be compared with the simpler ASME BPVC rule design predictions.

- *Severe accident modeling capabilities*—Simulation of severe accident scenarios is a key aspect of these computational codes. These codes can model the possibility of severe accidents and safe reactor shutdown. They can simulate the combined thermal/fluid and structural performance of components during these accidents. Test data are also needed to validate the model performance, to build confidence that the tools can be used with confidence outside their range of validation.

- *Structural performance under transients*—Many of the operating experience events (Turk et al., 2019) were caused by ANLWR transients. The computational tools available during the early years of ANLWR operation were not adequate to properly simulate the effect of these transients on structural and materials performance. The advancement of these nonlinear computational tools and their ability to obtain solutions for the highly nonlinear physical processes on multiple cores with rapid turnaround time is very important.

5. ROADMAP FOR ANLWR COMPUTATIONAL CODE DEVELOPMENT

This section presents a possible roadmap for development of computational codes to aid in licensing ANLWR plants. The numerous codes discussed above can be used to model the many degradation mechanisms that must be considered for assessing structural integrity in these plants. This section provides possible plans for further development of these codes. The three codes that are useful for further development are the GRIZZLY code, the xLPR code, and ABAQUS for special analyses.

5.1. <u>ANLWR Damage Mechanisms To Be Addressed</u>

The ASME BPVC, Section XI, Division 2, RIM program provides a rational approach for assessing degradation of each ANLWR component. In particular, the process listed in Steps 1 through 7 in Section 2.1.2 is a typical assessment process. For each component to be considered, the damage mechanisms in ANLWRs must first be determined. Section 2.1 listed these for the RIM program. While all these mechanisms must be addressed, depending on the component, this report gives the most significant mechanisms for NRC ANLWR assessments and those that should be addressed early in computational code development. Others can be included and implemented later. The following discussion focuses on these degradation mechanisms to limit the development strategy for each code.

The discussions below consider the following degradation mechanisms, which were presented as part of the RIM program in Section 2.1 (the numbers of the degradation mechanism correspond to those in Section 2.1):

Mechanism 2 creep (a time-dependent active degradation mechanism)

Mechanism 3 creep-fatigue (must consider interaction between creep and fatigue damage)

Mechanism 8 flow-induced vibration causing large local stress cycles

Mechanism 10 intergranular SCC

Mechanism 16 radiation embrittlement of the material

Mechanism 19 self-welding of rubbing parts and fretting fatigue

Mechanism 20 thermal aging

Mechanism 21 thermal stratification cycling and striping

Mechanism 28 reheat cracking (from creep relaxation of residual stresses)

Besides these damage mechanisms, some of which are unique to ANLWRs, the codes must be able to address traditional damage mechanisms observed in conventional LWRs (e.g., crack instability (for both surface and through-wall cracks), LBB, and leak-rate prediction). The computational codes must also be flexible enough to incorporate modules for considering additional damage mechanisms that will not be known until ANLWRs are operating.

5.2. ANLWR Computational Code Recommendations

The recommendations for code improvements to meet NRC licensing goals are presented below for GRIZZLY, xLPR (and PROMETHEUS), and ABAQUS (and WARP3D). The authors believe a path forward using all three computational code processes would best serve NRC interests. With the current emphasis at the NRC on risk-informed, performance-based safety assessments, the use of finite-element-based solutions in a probabilistic code will be a challenge because obtaining convergence within a probabilistic framework for low-probability events often requires tens or hundreds of thousands of realizations, as is currently observed for LBB assessments of LWRs using xLPR.

5.3. GRIZZLY Recommendations and Road Map

As discussed in Section 4.5.1, GRIZZLY within the MOOSE framework is a very powerful alternative to commercial codes such as ABAQUS to solve thermal structural problems. It is recommended that the NRC consider using this code and begin adding damage models, many of which already exist and can be chosen for particular analyses within the MOOSE framework. This code has already been coupled with the probabilistic RPV code FAVOR (see the discussion in Section 4.5.1). GRIZZLY is a modular finite-element code based on the open source MOOSE multiphysics finite-element framework, which provides a set of interfaces that make it straightforward to develop applications for adding physics models in a modular fashion. These physics modules could be finite-element based or consist of other analytical codes, such as the deterministic modules within the xLPR or PROMETHEUS codes. Most importantly, a mixture of finite-element modules combined with analytical modules can be synergistic.

5.3.1. Stage-by-Stage Plan

Stage 1—Become familiar with MOOSE and GRIZZLY

The NRC staff and contractors have extensive experience using the commercial code ABAQUS. This includes writing USER subroutines for material laws, damage development, special elements, and other factors. ABAQUS is easy to use for both pre- and post-processing and performing analyses within the computer-aided engineering (CAE) framework or in batch mode using USER subroutines. However, the learning curve with using the MOOSE system is reportedly significant, and solving problems likely requires more staff hours compared with ABAQUS or WARP3D. For the NRC staff with a computational mechanics background, this should not be a problem except that time must be set aside for extensive training and learning.[13] The NRC staff and contractors should become very conversant with the operation of the GRIZZLY analysis process before continuing to the next steps. This includes familiarization with its solvers (iterative and direct), since iterative solvers cannot be used for highly nonlinear problems such as computational weld modeling. In addition, the users should become familiar with the recommended mesh generator (CUBIT from SNL) for building meshes. Alternatively, the necessary finite-element meshes for components being analyzed could be built using another generation package (such as ABAQUS CAE) and then the mesh can be written out in the format necessary for a GRIZZLY-based solution. The NRC users should become familiar with the open source Paraview visualization computer code (open source code), which is quite different from the ABAQUS CAE viewer, for example.[14] INL provides tutorials on the use of MOOSE and Grizzly, and the NRC Office of Nuclear Regulatory Research staff recently had some initial training. However, the NRC users should become intimately familiar with the code through problem solving with it.

Stage 2—Determine modules necessary for GRIZZLY-based ANLWR assessments

For each problem, whether thermal, structural, or combined, GRIZZLY requires an analyst to define the required blocks of input in an ASCII text input file. This includes the mesh, unknowns (e.g., temperature, displacement, velocities), kernels (the modules necessary to solve the problems of interest), materials (e.g., elastic, plastic, creep, irradiation effects), boundary conditions, solver (iterative or direct), contact definition (if desired), dynamic effects or static, and outputs. This text input file is similar to an ABAQUS input file when performing batch solutions within a Linux environment and is similar to batch solutions in ABAQUS (as opposed to those who solve ABAQUS problems within the CAE “black box” framework).

The solution modules need to be defined for each component structural analysis. As an example, for performing a creep-fatigue analysis of a component, the modules must be chosen

13 Discussions with computational modeling specialists from DOE Laboratories, who are intimately familiar with using both ABAQUS and WARP3D and who have recently begun using GRIZZLY, verified that the learning curve, while time consuming, is not a problem for someone with a strong computational background.

14 This report’s authors frequently use Paraview for visualization of VFT-based weld model solutions using WARP3D. Paraview is very powerful and is not difficult to learn.

from the MOOSE library or developed if the module does not yet exist. For example, if one plans to perform an LBB analysis of a sodium pipe in an SFR where corrosion effects can be neglected, the modules required might include the following:

- *Meshes, material definition (elastic-plastic-creep), boundary conditions, and others:* These modules exist within the MOOSE framework and can be chosen and defined within the input file.

- *Creep crack initiation* (coalescence and transition): This module needs to be developed and can be based on R5 or RCC-MRx models at present.

- *Crack growth* (creep and fatigue): This involves modeling growth of the crack (perhaps using XFEM), cohesive zone modeling, or simply growth of the surface crack at the depth and surface points, as is done in the xLPR code. This module will likely need to be developed for creep and fatigue crack growth.

- *Crack stability* (J-tearing theory is adequate for both surface and through-wall cracks): This can be chosen from the MOOSE framework for calculation of J-integral. Alternatively, the xLPR modules for crack stability could be introduced and used.

- *Crack opening displacement and leak-rate modules*: To predict leakage through the crack, crack opening displacement can be extracted from the crack analysis, but the leak-rate module within xLPR will likely be needed to calculate leak rates through the cracks and modified for sodium coolant rather than water leakage.

- *Weld residual stresses*: This can be neglected but likely plays a key role for reheat cracking from creep and affects the fatigue crack growth through the contribution to mean stress.

- *K-solutions module*: Estimation of crack growth parameters for SCC and creep crack growth parameters such as C* and C(t) integrals can be calculated within a GRIZZLY finite-element module. Alternatively, as was used with the GRIZZLY/FAVOR solutions, a weight function approach (and module) for obtaining K-solutions is available already within the FAVOR code (Spencer et al., 2018).

The solution modules for the analysis of ANLWR components and structural assessments for licensing purposes should be part of an NRC library. Modules such as for creep response, irradiation damage, and plasticity are already available within the MOOSE system. Analytical FORTRAN subroutines (e.g., those within xLPR) can be extracted and used for a GRIZZLY analysis and modified as needed for ANLWRs. This stage involves determining the modules required for an ANLWR solution. These could be modules already available, separate analytical subroutines, or modules requiring development.

Stage 3—Develop modules necessary for GRIZZLY-based ANLWR assessments

Once the necessary solution modules are defined in Stage 2, some module development will be necessary. Modules for creep response, irradiation damage, plasticity, thermal solutions, and others are already available within the MOOSE system. These must be identified for NRC use, as numerous modules are available. Analytical FORTRAN subroutines, for example, already existing within xLPR can be extracted and used for a GRIZZLY analysis. Modules that are not yet available within the MOOSE system or from other NRC codes have to be identified and written (e.g., creep crack growth subroutines based on C* and C(t) parameters). The modules that need to be developed cannot be determined until the full library of GRIZZLY modules is known. Developing a crack growth module based on C* or C(t) should take on the order of several months, including validation of the code. Probabilistic modules, while existing within the MOOSE framework and implemented for the GRIZZLY/FAVOR code, may need to be developed to handle importance sampling. Unknown degradation mechanisms will have to be addressed as operating experience continues, so this database of modules will continually be updated. The library of modules and developmental efforts to add modules needs to be prioritized. Moreover, for a probabilistic assessment, the finite-element modules to be used may need to be carefully chosen because the solution speed of each module is critical to producing a practical risk-based modeling system—especially if tens of thousands of realizations are required to obtain a low-probability event, as is seen with xLPR analyses. Even though GRIZZLY can efficiently perform on multiple processors (of massively parallel hardware architecture), it is not known if the NRC staff will have access to such supercomputer resources.

Stage 4—Associate necessary modules with ANLWR component assessments and document

Within the ASME BPVC, Section XI, Division 2, RIM program, the degradation mechanisms that must be accounted for are first identified for the component of interest. This involves eliminating some of the possible damage mechanisms that do not need to be addressed for this component (see the example in Section 2.1.2). For each ANLWR component (e.g., reactor vessel, graphite blocks, intermediate heat exchanger, primary piping), the most likely degradation mechanism that must be accounted for should be catalogued with the modules placed within the library of GRIZZLY modules identified in Stage 3.

Stage 5—Training

The NRC will require training sessions for using the GRIZZLY system for ANLWR assessment.

Stage 6—Update and Improve Modules

Modules will need to be added as more degradation mechanisms for ANLWRs are discovered. Moreover, the existing modules will need continual maintenance and improvement.

5.3.2. Challenges with Using GRIZZLY/MOOSE

The GRIZZLY Jacobian Free Newton Krylov iterative solver was written using very abstract object-oriented features in C++ and is reported to have less than outstanding efficiency unless significant numbers of processors (hundreds or thousands) are available. For certain classes of highly nonlinear problems (such as the computational weld problem), a direct solver should be chosen to ensure efficient convergence. The average NRC user may not have access to such resources. The learning curve using the MOOSE system is reportedly significant, and solving problems likely takes more staff hours compared to using ABAQUS. Finally, MOOSE/GRIZZLY appears to be a research-oriented code and may not have the requisite NRC auditing or certifications provided by such commercial codes as ABAQUS and ANSYS. Finally, it may be useful to develop a special graphical user interface for use with MOOSE/GRIZZLY analysis of ANLWR components to make it more like a commercial code for easier use. Such a simple interface does not appear to exist at present. Otherwise, the users may have to be computational specialists.

5.4. xLPR and PROMETHEUS Code Recommendations

xLPR and PROMETHEUS are similar codes, except xLPR uses a package called GoldSim for the probabilistic modules, while PROMETHEUS has its own set of probabilistic subroutines. These codes have a number of damage modules within them representing important degradation mechanisms that occur in LWRs. These codes can be modified to include degradation mechanisms in ANLWRs. These degradation modules will be analytically based, which is convenient for risk-based codes and provides rapid solutions. The xLPR code was designed to be modular, so improvements and changes to each deterministic module can be made, albeit with possible large effort. Moreover, it is possible to add more deterministic modules to account for different damage mechanics such as creep or irradiation damage. Each aspect of the xLPR code has undergone a full nuclear quality assurance evaluation, and this process is time consuming (however, the same can be said for the GRIZZLY modules).

5.4.1. Recommended Enhancements for xLPR and PROMETHEUS

The stages for enhancement of these codes for ANLWR use are similar to those discussed with regard to GRIZZLY above.

Stage 1—Determine modules necessary for xLPR-based ANLWR assessments

This will be component specific. The solution modules need to be defined for each component structural analysis. The following lists some of these; some of these developments may take considerable effort:

- Creep and creep-fatigue high-temperature crack growth and stability modules are necessary.
- Leak rates for sodium and helium leakage in SFR and HTGR piping, respectively, are necessary to perform LBB assessments. Modifications to analytically based leak-rate modules within xLPR (LEAPOR and SQUIRT) are necessary.
- Effectiveness of ANLWR inspection methods and probability of detection are necessary. They may take time to develop until after ANLWRs have operating experience. The probability of detection within xLPR for LWRs can be modified for the ANLWR plants.

- Irradiation embrittlement must be considered for many components. Analytical modules such as those within FAVOR can be used as starting points.

- Development of graphite damage modules is necessary.

These modules do not all have to be developed at once but can be done in stages, considering the resources available. Once these necessary modules are determined, the next stage can begin.

Stage 2—Develop modules necessary for xLPR-based ANLWR assessments

Once the necessary solution modules are defined, module development will be necessary. The modules to be written and included within xLPR can be developed along the lines of Lee (2015) and Lee, Won, and Huh (2019) and are discussed in Section 2.2. It is anticipated the creep crack initiation and crack growth modules based on C* and C(t) will be developed first, using the simple estimation methods based on the stress intensity factor as summarized by Lee, Won, and Huh (2019). This is the R5 approach and is used in the proprietary R5 flaw assessment software.

Stage 3—Modify xLPR global control software

The global control portions of the xLPR code will require modification to accommodate ANLWR degradation modules. Additionally, input file definitions, containing parameters for uncertainty, WRS fields, material parameters, and others, will require modification. This may require a quality assurance evaluation during development and testing. Developing uncertainty parameters for ANLWR degradation mechanisms will be time consuming, as it requires large databases.

Stage 4—Training

Training sessions for using the xLPR and PROMETHEUS codes for ANLWR assessments will be necessary. The NRC staff is already familiar with xLPR analysis procedures, so training may be accomplished seamlessly.

Stage 5—Update and improve modules

Modules will need to be added as more degradation mechanisms for ANLWRs are found. Moreover, the existing modules will need continual maintenance and improvement.

5.4.2. Challenges with Using xLPR and PROMETHEUS

Because the deterministic modules in the ANLWR-based xLPR code are analytically based, they can produce rapid results. As a result, realistic ANLWR structural risk results could be obtained within a reasonable amount of time. Assessments of the probability of rupture of piping systems subjected to PWSCC damage can take very large numbers of realizations. However, the new ANLWR modules will not be as accurate as GRIZZLY-based finite-element solutions because of the inherent assumptions underlying analytically based solution modules.

5.5. ABAQUS and WARP3D Code Recommendations

Because ABAQUS and, as noted above, WARP3D are well recognized and are used extensively by the NRC staff and contractors, these codes should continue to play a role for ANLWR component assessments in the future. The stages of development are similar to those for GRIZZLY and are briefly summarized below. Both ABAQUS and WARP3D are included here, since both can play an important role, but only ABAQUS enhancements are summarized below.

Stage 1—Determine modules necessary for xLPR ABAQUS-based ANLWR assessments

The solution modules need to be defined for each component structural analysis listed for the GRIZZLY enhancements above. This effort involves identifying the features that already exist in ABAQUS that can be chosen for ANLWR assessments (e.g., creep material laws and fracture modeling features). For features that do not exist, USER subroutines will need to be developed (e.g., for graphite damage assessment, along the lines of Marsden et al. (2018)). WARP3D can use the same USER subroutines that are appropriate for ABAQUS.

Stage 2—Develop modules necessary for xLPR-based ANLWR assessments

Once the necessary solution modules are defined, USER routines will need to be developed to account for damage mechanisms not currently present within ABAQUS.

Stage 3—Training

Training sessions for using the ABAQUS and WARP3D codes for ANLWR assessment will be necessary for the NRC staff who will perform licensing assessments. The NRC staff is already familiar with the standard ABAQUS software.

Stage 4—Update and improve modules

Modules will need to be added as more degradation mechanisms for ANLWRs are found. Moreover, the existing modules will need continual maintenance and improvement.

5.6. Summary for Computational Code Recommendations

The recommendations and road map presented in this section are aggressive and may stretch technical resources and staff hours during development. Deterministic assessments may be necessary using GRIZZLY or ABAQUS during the licensing process, and probabilistic assessments may be a challenge for some time to come. Additional routines for GRIZZLY and USER routines for ABAQUS will take more time to develop but will aid in validating the developments in the ANLWR xLPR enhancements. Finally, the GRIZZLY developments will be the most powerful for ANLWR structural component life assessments when complete but will take the longest to develop and require the most time to train staff for use.

[15] The discussion of xLPR applies equally to PROMETHEUS.

The rationale for pursuing development of xLPR (or PROMETHEUS, or both),[15] ABAQUS, and GRIZZLY in parallel is as follows. The authors believe that the readiness level for use of these codes is in the order of xLPR first, ABAQUS, and then GRIZZLY.[16] xLPR is currently scheduled for release in May 2020. Adding ANLWR damage mechanisms to xLPR is straightforward. They can be added to xLPR due to the modular nature of this code and will be the fastest approach for permitting risk-informed assessments of ANLWRs. On the other hand, the NRC already uses ABAQUS, and USER modules can be developed and applied for specific deterministic ANLWR assessments for various damage mechanisms. ABAQUS can also be used to validate some of the new xLPR modules. Hence, in the authors' opinion, the readiness level of ABAQUS is close to that for xLPR. Finally, GRIZZLY is a code that DOE is currently developing independently and could eventually be used in the same manner as the GRIZZLY/FAVOR code. Therefore, GRIZZLY is judged to be the code with the lowest readiness level at present for the NRC staff, since there is a significant learning curve for those who are not computational specialists.

5.7. Specific Recommendations

The text below summarizes the specific recommendations for developing computational codes to assess ANLWR safety issues during NRC licensing reviews. Some of the other codes discussed above can also be enhanced to support the following recommendations:

(1) **Probabilistic ANLWR Code Development (HT-xLPR).** The NRC currently prefers a risk-informed decisionmaking process to assess nuclear plant safety to ensure public safety. The NRC, along with EPRI, is completing the xLPR code to provide risk-informed assessments of LWRs. This code has undergone a formal quality assurance process and is in the final stages of validation at present. In addition, the NRC by itself developed a similar code, called PROMETHEUS, which essentially performs the same PRAs as xLPR, but it did not undergo the same formal quality assurance process as xLPR. However, PROMETHEUS uses the same quality assurance process for deterministic modules that are part of the xLPR code. Moreover, results produced by both PROMETHEUS and xLPR for LWR assessment are essentially identical. The advantage of using PROMETHEUS is that the probabilistic routines are not tied to third-party software, while these routines within xLPR are tied to a software system called GoldSim. It is recommended that both codes be developed to include the ANLWR damage mechanism deterministic subroutines discussed above. The level of effort required for these enhancements to both codes to permit risk-informed decisionmaking of ANLWRs is estimated to be on the order of four to five full-time equivalent staff members over 3 years. Additional effort would be required for training and personnel development.

(2) **Finite-Element Code Development.** The ABAQUS finite-element code has many powerful features that permit assessment of the damage mechanisms present in ANLWRs, including impressive fracture mechanics features. The text above

[16] Reportedly, the industry is developing a probabilistic code for ANLWR assessment called BLESS. However, the stage of development of this code is not known at this time.

summarizes the damage mechanisms that need to be addressed. Moreover, as discussed above, ABAQUS permits development of USER routines to allow analysis of damage mechanisms not currently part of the ABAQUS code. In fact, many such routines are already available to the NRC staff, such as creep damage USER routines that include coupled and uncoupled damage within the constitutive routine. ABAQUS would be used in this fashion to perform selected deterministic assessments of ANLWR damage mechanisms and for confirmatory analysis for licensing purposes. In addition, some of the high-temperature deterministic subroutines that are part of the high-temperature xLPR code can be validated using ABAQUS. The level of effort required for these enhancements to both codes to permit risk-informed decisionmaking of ANLWRs is estimated to be on the order of two full-time equivalent staff members over 3 years. It is recommended that the GRIZZLY code be followed for potential future use, as DOE is continuing developments for this code to some extent. This code is not as user-friendly as ABAQUS, so additional training would be required. No estimate can be provided for these developments at this time.

6. CONCLUSIONS AND RECOMMENDATIONS

6.1. Consensus Codes

This report identifies gaps in the consensus code ASME BPVC, Section XI, Division 2, based on the operational experience issues compiled and summarized in the Task 1 report (Turk et al., 2019). This includes high-temperature damage mechanisms and anticipated issues of concern for NRC licensing of future ANLWRs. The operating experience described in Turk et al. (2019) will help detect potential service issues that should be considered by the various code bodies, including ASME and international groups. The first portion of the report focuses on rules for ANLWRs being developed for Section XI, Division 2.

Section 2.2 summarizes the RIM program recently implemented into ASME BPVC, Section XI, Division 2. The RIM program was inspired by experience with the Japanese SBC concept, which was the product of ISI requirements for Monju, a prototype fast breeder SFR. Examples showing the application of the RIM program to Monju and the South African PBMR illustrate the use of the process. The methods are meant to be applicable to both LWRs and ANLWRs, although specific processes for some advanced reactors are still being written at present. The numerous potential damage mechanisms are listed for each nuclear plant type. Some degradation mechanisms are not known at present and will be added later.

Section 2.3 discusses the R5 and RCC-MRx code. Recent enhancements to R5 used in the United Kingdom for ANLWR structural component integrity analysis are summarized and contrasted with current ASME BPVC, Section XI, Division 2, methodologies. R5 is currently undergoing enhancements, including the consideration of carburization (which is unique to the CO_2-cooled UK AGRs) and improved nonsteady-state creep crack assessments. These enhancements should reduce conservatism in flaw assessments considerably. R5 incorporates high-temperature fracture mechanics assessment procedures not available in Section XI, although these procedures are being developed.

Section 3.2 also summarizes NDE and ISI technologies for high-temperature reactors. ANLWRs are expected to accommodate both outage-based and online monitoring and examination. This report summarizes the concept of NDM, where NDM targets online monitoring of active degradation mechanisms at susceptible regions. Section 3.2 also discusses the operating conditions for next-generation HTGRs that require improvement to sensor devices to accommodate higher temperature operation.

6.2. Computational Codes

This report includes an examination of computer codes applicable to the design and assessment of ANLWRs, along with additional features necessary to model ANLWR damage mechanisms. It reviews some of the available commercial, open source, government, and NRC computer codes. The report discusses the perceived needs for improvements (or computer code gaps) for each of the following computer codes:

- **Commercial finite-element computer codes** discussed include ABAQUS, ANSYS, and other commercial codes.

- **NRC-developed analytically based computer codes** covered include version 2 of xLPR, PROMETHEUS, FAVOR, ALT3D, NRCPIPE, NRCPIPES, SQUIRT, and related codes and modules from xLPR, developed initially for assessing LWRs. The first three of these are probabilistic codes used to assess the uncertainties associated with inservice damage to reactor piping.

- **Open source codes** examined include WARP3D, which has many features applicable to ANLWR material degradation and fracture assessment. In addition, since it is open source with extensive documentation, additional features can be added. Other ABAQUS USER subroutines (for example, creep USER subroutines) can be used with WARP3D with minor modifications.

- **Government codes** discussed include GRIZZLY and MOOSE, SIERRA, DIABLO, and ALE3D. These are multiphysics-based finite-element codes developed by INL, SNL, and LLNL. The GRIZZLY code is being developed and used for nuclear applications at present.

- **Graphite Computer Codes** are covered through a limited summary of computational graphite modeling efforts and available codes. Many of these modeling efforts in the United Kingdom use either a commercial code or an open source code with special graphite damage mechanisms implemented either with the use of USER material subroutines or directly as subroutines into open source codes.

This report presents recommendations and a possible road map to make the codes applicable for ANLWR assessment. The necessary improvements may require significant technical resources and staff hours during development. Deterministic assessments may be necessary, using GRIZZLY or ABAQUS during the licensing process, and probabilistic assessments may be limited by available computational resources. Additional subroutines for GRIZZLY and USER subroutines for ABAQUS will take time to develop but will aid in validating the developments in the ANLWR xLPR enhancements. Finally, the GRIZZLY developments will be the most powerful for ANLWR structural component life assessments when complete, but they may take the longest to develop. Performing PRAs based on GRIZZLY or ABAQUS solutions directly will be a challenge because assessing rupture in piping systems susceptible to PWSCC requires tens or even hundreds of thousands of realizations to ensure convergence for low-probability events. Hence, use of surrogate models based on the solution space obtained with GRIZZLY or ABAQUS may be necessary.

Some of the damage mechanisms that need to be addressed within the ASME BPVC, Section XI, Division 2, RIM program include the following (others were discussed above):

- creep and creep-fatigue high-temperature crack growth and stability
- leak rates for sodium and helium leakage in SFR and HTGR piping, respectively
- effectiveness of inspection
- irradiation embrittlement
- concrete damaging cracking

These damage assessment features in ANLWRs need to be addressed in ABAQUS, GRIZZLY, and xLPR.

In addition, the probabilistic codes xLPR and PROMETHEUS can be improved to account for unique ANLWR damage issues for risk-based assessment of safety issues. The xLPR and PROMETHEUS codes are modular, making modifications convenient to implement. Moreover, it is possible to add additional deterministic modules to account for different damage mechanics such as creep or irradiation damage. xLPR and PROMETHEUS are both powerful, risk-based nuclear piping assessment and LBB analysis codes. It would be convenient to implement ANLWR damage mechanisms into these codes as different modules. Modules could be developed to evaluate ANLWR structural and materials issues in a risk-based environment.

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Appendix I: Summary of Significant Operational Experience Issues

Some of the significant findings from Technical Letter Report TLR-RES/DE-CIB-2019-01, "Advanced Non-Light-Water Reactors Materials and Structural Operational Experience," issued March 2019 (Agencywide Documents Access and Management System Accession No. ML18353B121), about sodium-cooled fast reactors (SFRs) and operating experience include the following:

- Corrosion of immersed stainless steel components is not a concern if sodium purity is maintained. The sodium can become contaminated with several unwanted chemicals, such as water and oil from valves and pumps.
- Sodium heat-transport systems have experienced a significant number of leaks because of poor quality control and difficulty with welds. Residual weld stresses, excess weld metal, and weld constraints should be minimized, and direct tube-to-tube-plate welds should be avoided entirely in SFRs.
- Stresses induced by thermal expansion, particularly in areas of structural constraint, must be carefully considered. Thermal expansion stresses have often been the source of structural integrity issues in SFR operation.
- Some steels used in SFRs were particularly susceptible to reheat cracking. Moreover, even if reheat cracking does not occur, partial creep damage can develop in components as residual stresses relax due to creep at high temperature, potentially reducing component life.
- Sodium contamination, and the consequent formation of sodium oxide, can lead to binding of rotating machinery and control rod drives.
- Inadequate or unreliable leak detection systems have resulted in sodium contamination, excessive sodium leaks, and consequent fires, resulting in extensive shutdowns.
- Thermal fatigue is a much more significant issue than for light-water reactors, particularly in regions where hot and cold sodium flows mix and interact.
- Some SFR designs did not adequately anticipate the potential for high thermal stresses during transients due to sodium's high thermal conductivity.
- The startup and cooldown transients in SFRs should be managed to control vibration, thermal expansion loads, and possible fatigue.
- Possible valve failures (all system valves, especially those operating at high temperature) are a concern for SFRs. Valve reliability under operating conditions should

be accurately determined. Valve failures include membrane tears and valve component fractures, among others.

- Secondary measurement devices (e.g., thermocouples) must be properly designed to prevent leaks. Flow-induced vibrations and complex fluid flows in these areas can cause failure of the device and leaks of sodium.

- Seismic and external dynamic loading events of SFR reactors need to be scrutinized during the licensing process. During an emergency shutdown (SCRAM), the intermediate heat exchanger may experience thermal shock, caused by the influx of cold sodium. This could lead to buckling and structural issues, amplified by an external loading.

Turk et al. (2019) made the following significant findings related to high-temperature gas-cooled reactors (HTGRs):

- Accurate prediction of core temperatures is problematic, as some HTGR cores operated at higher than design temperature. This was especially true for pebble bed reactors.

- Moisture ingress and leakage events are a reoccurring problem with HTGRs. HTGRs should be designed to accommodate and mitigate moisture ingress.

- Coolant flow issues have resulted in flow-induced vibrations and fatigue concerns.

- Structural risks exist with the intermediate heat exchanger and hot gas valves that isolate the primary loop gases from the secondary gases.

- Management of thermal stresses is important in HTGRs, as thermal expansion stresses can cause large loads and creep.

- HTGRs should be designed to minimize sources of graphite dust (e.g., fretting) and include filters or other mitigating measures to address it.

- Oil-based lubricants should be avoided entirely.

- Thermal cycling during the testing in the startup phase can be significant enough to cause material or component failure.

- Pumps, seals, and compressors have a history of poor reliability.

- HTGRs need to ensure the structural integrity of the reactor pressure vessel and of the connecting vessels, especially under low helium flow and loss-of-forced-convection conditions, as creep buckling may occur.

Turk et al. (2019) summarized these and many other operational experience materials-based and structural integrity incidents, along with the corresponding timeline for the SFR and HTGR plants examined.

Appendix II: Proposed Creep-Fatigue Assessment Procedure

This appendix describes a step-by-step procedure whereby a component containing a known or postulated defect can be assessed under creep-fatigue loading. This type of procedure is being proposed for inclusion in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) in the future but is still under development. The general 13-step approach shown in Table II-1 is similar to the R5 approach (R5, 2003 and 2014) and is well accepted in the technical community. The procedure addresses both continuum damage accumulation and crack growth and includes insignificant creep and insignificant fatigue as special cases. The procedure may be applied to a component in the design stage, or where it has already experienced high-temperature operation, as in an operating plant where damage has been observed or is postulated. To address an aging nuclear component, guidance is provided on the assumed defect initiation time. Continuum damage failure (creep rupture) of an uncracked body may be considered as a special case by omitting the steps covering crack growth and cyclic loading. However, ASME BPVC, Section III, Division 5, Subsection NH, already addresses this issue. The steps in the procedure are listed below with a description. Brust et al. (2009, 2010) and R5 (2003) detail additional examples. While this follows the R5 approach, other high-temperature crack assessment procedures are quite similar.

Table II-1 Creep-Fatigue Crack Growth Assessment Procedures

Step	Analysis Process	Notes
1	**Establish expected crack cause**	Defect type, bounding size, incubation period, etc.
2	**Define load conditions**	Pressures, dead weight, thermal expansion, seismic, etc.
3	**Define materials**	Creep, fatigue, uniaxial data as functions of temperature. Is corrosion important? Toughness may be reduced due to creep damage.
4	**Analyze basic stress (operation extremes)**	Shakedown assessment. If shakedown does not occur, inelastic analysis is necessary.
5	**Assess time-independent stability**	ASME BPVC, Section XI, type assessment. If failure is predicted, analysis is complete.
6	**Check whether creep, fatigue, or both are important**	Division 5 rules checked to determine significance of creep or fatigue. Check whether creep-fatigue is important.
7	**Calculate creep rupture life based on initial defect size**	Limit load solution (no crack singularity effect). Finite-element ductility exhaustion methods may be necessary.
8	**Calculate initiation time prediction**	It is conservative to ignore this, and it often is ignored. Crack initiation prediction methods are not robust at present.
9	**Calculate crack growth for desired lifetime or to next inspection period**	Integrated creep and fatigue crack growth expressions. Parameters change as crack grows.
10	**Recalculate creep rupture life for final crack size after growth**	Conservative to base this assessment on final crack size. Finite-element-based continuum damage predictions may be applied also.
11	**Check time-independent stability for final crack size**	This is actually performed at each step during crack growth modeling.
12	**Assess significance and determine whether repairs or design changes are needed.**	Margins need to be established.
13	**Prescribe inspection intervals, etc.**	For maintenance.

STEP 1—Establish the Expected Crack Cause

Establish the cause of the cracking to ensure the applicability of the procedures. Identify the defect type, position, and size. For a creep-fatigue design crack growth assessment, the expected crack size and location can be determined from the stress analysis where the highest stresses occur. The size would be the limit of the nondestructive examination confidence. Suitable sensitivity studies (Step 12) should be performed to address uncertainties. The defect should be characterized by a suitable bounding profile amenable to analysis. Defects not of a simple Mode I type should be resolved into Mode I orientation.

STEP 2—Define load conditions

Resolve the load history into cycle types suitable for analysis. This includes all design cycles for inservice assessment, the historical operation, and the assumed future service conditions. Define the service life. For the case of a defect-free component at the start of high-temperature operation, an estimate of the defect formation time (e.g., the crack nucleation time) can be determined. It is conservative to neglect this time. Perform suitable sensitivity studies to address uncertainty for the defect formation time.

STEP 3—Define materials

Define and collect the material data. Define the materials relevant to the assessed feature, including, in the case of weldments, the weld metal and heat-affected zone structures. The material properties must be appropriate over the temperature range and in the correct cyclically conditioned state. The effects of thermal aging may also need to be considered for some time-independent material properties required for the stability analyses performed in Steps 5 and 11. Fracture toughness properties (e.g., J-resistance curves, K_{Ic}) are required for creep-damaged material. If these properties are not available, they must be conservatively estimated from creep undamaged material.

STEP 4—Analyze basic stress

Analyze elastic stress of the uncracked feature for the extremes of the service cycles. In the case of cyclic loading, perform a shakedown assessment of the uncracked feature following the procedures in ASME BPVC, Section III, Division 5, Subsection NH. If a shakedown cannot be demonstrated, justify the use of the methods of this appendix. For example, inelastic analysis methods, including finite-element analysis, may be required. If the shakedown is demonstrated, the crack depth should be insufficient to affect the structural integrity of the component. Although at present, if a crack is found in a Section III, Division 5, component, it is to be repaired within the current standard, this will change as the procedure is developed.

STEP 5—Assess time-independent stability

Check the cracked component to ensure time-independent mechanisms under fault or overload load conditions at the initial defect size do not occur. This can be performed using ASME BPVC, Section XI, procedures or a J-tearing assessment. If failure occurs due to time-independent effects alone at this step, the assumptions in the analysis should be revisited and remedial design action taken. Only if sufficient margins can be justified is it permissible to continue to Step 6 to justify future service life or the design.

STEP 6—Check whether creep, fatigue, or both are significant

Check for insignificant creep using the procedures in ASME BPVC, Section III, Division 5, Subsection NH. If creep is insignificant, the assessment becomes one of fatigue loading alone and Steps 7 and 10 below are omitted. Conversely, if fatigue is judged to be insignificant, the assessment becomes one of steady creep loading alone and further consideration of cyclic loading is not required. If it is not, simplified summation rules for combining creep and fatigue crack growth increments may be adopted (Step 9).

STEP 7—Calculate creep rupture life based on the initial defect size

Calculate the time to continuum damage failure (creep rupture) based on the initial crack size from Step 1. If this is less than the required service life, it may not be necessary to perform crack growth calculations, and the current ASME BPVC, Section III, Division 5, procedures for high-temperature design alone suffice. The estimate of rupture life is based on a calculated limit load reference stress and, for predominately primary loading, the material's creep rupture data. For damage due to cyclic relaxation and to the relaxation of welding residual stresses, ductility exhaustion methods are more appropriate. The particular requirements for defects in weldments are also addressed. For the case of short defects close to stress concentrations, such as notch radii or weld toes, special considerations must be followed to ensure the reference stress is conservatively calculated.

STEP 8—Calculate initiation time prediction

Typically, it takes some time for a crack in a nuclear component to begin growing. For some components, crack initiation may consume the bulk of the life, and when crack growth commences, failure occurs quickly. The crack nucleation or incubation time is the time from the start of the of high-temperature operation to crack growth start. Depending on the cause of cracking, its location within a weldment, and the type of loading, it may be possible to calculate a nonzero incubation time. *It is always conservative to ignore this period and assume crack growth occurs on first loading*. The cause of cracking will influence the determination of an incubation time. For example, a naturally occurring creep defect, such as some weld defects, may not experience an incubation period before macroscopic crack growth. There are several procedures for calculating crack incubation time, including the preferred two-criteria approach within RCC-MRx.

STEP 9—Calculate crack growth for the desired lifetime or to the next inspection period

Calculate the crack size at the end of the design period of operation, following the procedures of R5 based on K, C*, reference stress, and the appropriate estimation schemes, which are well accepted as a conservative approach. These procedures and estimation schemes are currently being defined within the ASME Boiler and Pressure Vessel Working

Group on High Temperature Flaw Evaluation with the development of a Code Case. Finite-element analysis can also be used. This is done by integrating the appropriate creep and fatigue crack growth expressions. This incremental process is simplified in some cases, depending on the outcomes of the significance creep and fatigue tests determined in Step 6. The calculations should include changes in reference stress due to crack growth. Integration is required because all parameters (K, C*, C(t))[17] and reference stress change with time as the crack proceeds.

STEP 10—Recalculate creep rupture life for final crack size after growth

Recalculate the time to continuum damage failure, taking into account the increased crack size from Step 9. Do not perform crack growth calculations in practice beyond an acceptable rupture life. It is conservative to base the estimate of rupture life on the final crack size, as this neglects slower accumulation of creep damage when the crack size is smaller during growth.

STEP 11—Check time-independent stability for final crack size

In practice, this step is carried out in conjunction with the crack size calculations of Step 9. Do not perform the crack growth calculation steps beyond a crack size at which failure by time-independent mechanisms is conceded at fault or overload load levels using ASME BPVC, Section XI.

STEP 12—Assess significance and determine whether repairs or design changes are needed

Assess the uncertainties; for example, in loads, material properties, or defined crack location. Assess the sensitivity of the results of the preceding steps to realistic variations in loads, initial flaw size and location, and material properties as part of a sensitivity study. Various modeling assumptions previously made can be revisited with the intent of reducing conservative assumptions in the analysis if unacceptable margins are determined. If this still fails to result in an acceptable crack growth life, consider changing the design, reducing future service conditions, or repairing or replacing the defective components. For U.S Nuclear Regulatory Commission needs, this may require placing the procedure within a probabilistic framework.

An alternative to the quantitative assessment of margins using the deterministic approach of this section is to use probabilistic methods to directly determine failure probabilities.

STEP 13—Final actions, such as prescribing inspection intervals and modifying loads

The results of the assessment, including margins determined, and the details of the materials properties, flaw size, loads, stress analysis calculations, and other factors used in the assessment should be comprehensively reported. This facilitates both verification of the particular assessment and repeatability in future assessments.

[17] Section 3.1 of the summary report defines these fracture parameters.