2024 TECHNICAL REPORT

Loss-of-Coolant-Accident-Induced Fuel Fragmentation, Relocation and Dispersal with Leak-Before-Break Credit

Alternative Licensing Strategy

PREPARED UNDER THE NUCLEAR QA PROGRAM

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ABSTRACT

Nuclear plant operators are considering the use of higher burnup fuel designs to meet a number of operational objectives. A major technical issue to extending burnup is fuel fragmentation, relocation, and dispersal (FFRD) during loss of coolant accidents (LOCAs).

EPRI evaluated alternative licensing approaches for utilization of higher burnup fuel rods in pressurized water reactors (PWRs), as described in EPRI Report 3002018457 [8]. The final selected approach applies risk insights based on a combination of extremely low likelihood of occurrence, leak-before-break concepts, and reactor coolant system (RCS) main loop piping performance parameters, developed from xLPR probabilistic fracture mechanics analysis. Using this approach, the proposed EPRI alternative licensing strategy (ALS) is able to evaluate the credibility of fuel dispersal during a postulated large-break LOCA (LB-LOCA). Additionally, the EPRI ALS project evaluates the potential likelihood of cladding rupture and fuel dispersal for higher burnup fuel rods susceptible to fine fragmentation during small-break and intermediate-break LOCA conditions [5]. The ALS also includes evaluation of non-piping RCS components. This approach is aligned with various alternatives proposed within the U.S. Nuclear Regulatory Commission (NRC) regulatory basis for higher enrichment rulemaking [10]. The industry's review of the proposed alternatives supported the adoption of an ALS approach [11].

Based on the analysis results presented in this report, the ALS approach provides a pathway for an individual plant licensee to request license amendments to address LOCA-induced FFRD in high burnup fuel. The proposed methodology supports the transition for the bulk of the PWR fleet to higher burnup and also provides a pathway to apply this approach to PWRs that are not specifically evaluated in this report.

Keywords

Alternative Licensing Strategy (ALS)
Fuel Fragmentation, Relocation and Dispersal (FFRD)
Leak-Before-Break (LBB)
Loss of Coolant Accident (LOCA)
Reactor Coolant System (RCS)

EXECUTIVE SUMMARY

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Primary Audience: Fuel managers, technical staff (utility and fuel vendors), and regulators

Secondary Audience: Safety analysis and licensing managers and staff

KEY RESEARCH QUESTION

The potential for a loss of coolant accident (LOCA) to induce fuel fragmentation, relocation, and dispersal (FFRD) is a key technical and regulatory challenge to increasing the maximum allowable burnup of fuel. The customary approach of using empirical data to develop and validate a model of fuel behavior is not anticipated to support the desired timing and near-term U.S. energy demands. The proposed approach provides resolution to the FFRD challenges within the industry's desired implementation schedule. This results in early adoption, providing enhanced operational and safety benefits for the U.S. pressurized water reactor (PWR) fleet.

RESEARCH OVERVIEW

A regulatory approach (called the Alternative Licensing Strategy (ALS)) for addressing LOCA-induced FFRD is established. This report applies risk insights, based on extremely low likelihood of occurrence, probabilistic fracture mechanics, existing leak-before-break (LBB) processes, and conventional LOCA analysis. This analysis supports the operational, economic, and safety benefits of implementing increased fuel burnup for the PWR fleet. Large break (LB) LOCAs were evaluated based on expert elicitation estimates of LOCA frequency, risk insights obtained from probabilistic fracture mechanics of RCS piping systems, existing LBB processes and previously approved LBB applications, and assessment of non-piping component failures. Smaller piping breaks were evaluated for occurrence of burst of high burnup (HBU) fuel clad. This ALS approach was supported by the industry response to the NRC's regulatory bases for rulemaking, "Increased Enrichment of Conventional and Accident Tolerant Fuel Designs for Light-Water Reactors," (NRC-2020-0034). These elements, along with defense-in-depth considerations, provide the basis for LOCA-induced FFRD evaluation.

KEY FINDINGS

- Previous expert elicitation estimates of LB-LOCA frequency are supported by an extensive probabilistic fracture mechanics evaluation using the xLPR code.
- For reactor coolant system (RCS) main loop piping, analysis using xLPR showed that, in the few instances where a leak does progress to piping rupture, there is significant time from

the leak rate reaching to 1 gpm (3.8 L/min) until the LB-LOCA event. This provides more than adequate time for operators to detect the leak, investigate, and shut down and cool down the plant, as required by Technical Specifications. With the plant shut down and at reduced temperature and pressure, the conditions that could lead to a LB-LOCA or to FFRD have been sufficiently mitigated.

- Conventional LOCA analysis for smaller break sizes demonstrated that fuel cladding rupture is not predicted to occur with high probability.
- Failure of non-piping RCS components is adequately supported by design, fabrication, and in-service inspections to preclude consideration as a credible cause of FFRD.

WHY THIS MATTERS

The adoption of higher burnup fuel designs, in combination with higher enrichments, provides significant operational flexibility (longer cycles, power uprates), improved plant economics (reduced fuel costs), and safety benefits (reduced high level waste, operation dose). The ALS analysis addressed one of the more challenging barriers to obtaining these benefits for the PWR fleet.

HOW TO APPLY RESULTS

Licensees desiring to implement increased fuel burnup limits are required submit a license amendment request (LAR) to address regulatory requirements for showing acceptable performance for the range of possible piping break sizes and locations. FFRD has been documented as a potential phenomenon that occurs as fuel burnup increases. FFRD can be addressed for PWRs by referencing an ALS Safety Evaluation Report and confirming the limitations, conditions, and/or applicability requirements are met by the licensee.

The results in this report and other ALS documents are primarily applicable to U.S. PWRs. However, the approach of addressing LOCA-induced FFRD could be considered by international PWRs after pursuing regulatory change in their country.

LEARNING AND ENGAGEMENT OPPORTUNITIES

This report provides a regulatory framework for addressing LOCA induced FFRD for HBU fuel. This framework provides a bases for the use risk insights to address RCS main loop piping system performance along with deterministic evaluations of smaller diameter RCS piping. Additionally, the performance of non-piping RCS components are also addressed. Other related EPRI ALS reports include:

- *Materials Reliability Program: xLPR Estimation of PWR Loss-of-Coolant Accident Frequencies (MRP-480)*. EPRI, Palo Alto, CA: 2024. 3002023895.
- *LOCA Analysis of Fuel Fragmentation, Relocation and Dispersal for Westinghouse 2-Loop, 3-Loop and 4-Loop Plants – Non-Proprietary, Evaluation of Cladding Rupture in High Burnup Fuel Rods Susceptible to Fine Fragmentation*. EPRI, Palo Alto, CA: 2024. 3002028675.

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IMPLEMENTATION CATEGORY: Reference – Technical Bases

ACRONYMS AND ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
AISI	American Iron and Steel Institute
ALS	Alternative Licensing Strategy
AMP	[EPRI] Aging Management Program
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
BTP	Branch Technical Position
BWR	boiling water reactor
CASS	cast austenitic stainless steel
CDF	core damage frequency
CE	Combustion Engineering
CFR	Code of Federal Regulations
CRDM	control rod drive mechanism
CY	calendar year
DBA	design basis accident
DEGB	double ended guillotine break
DiD	defense-in-depth
ECCS	emergency core cooling system
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
EQ	environmental qualification
ESFS	engineered safety features system
FFRD	fuel fragmentation, relocation, and dispersal
FR	Federal Register
GALL	Generic Aging Lessons Learned
GDC	General Design Criteria
GL	Generic Letter
GPM	gallons per minute
GWd/MTU	gigawatt-days/metric ton uranium
GSI	Generic Safety Issue
HBU	high burnup

IB	intermediate break [alternatively, MB for medium break]
ID	inner diameter
IE	increased enrichment
IGA	intergranular attack
IGSCC	intergranular stress corrosion cracking
IMC	[NRC] Inspection Manual Chapter
INPO	Institute for Nuclear Power Operations
ISI	in-service inspection
ISLOCA	interfacing system loss of coolant accident
LAR	license amendment request
LB	large break
LBB	leak-before-break
LB-LOCA	large break loss of coolant accident
LCO	Limiting Condition for Operation
LER	licensee event report
LIC	[NRC document designation for an office instruction]
LOCA	loss of coolant accident
LRD	leak rate detection
LWR	light water reactor
MFC	month fuel cycles
MRP	[EPRI] Materials Reliability Program
MTU	metric ton uranium
MWD	Megawatt days
NDE	non-destructive examination
NEI	Nuclear Energy Institute
NGF	[Westinghouse] Next Generation Fuel
NPS	nominal pipe size
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
NUREG	[NRC technical report designation]
PFM	probabilistic fracture mechanics
PI	Performance Indicator
PORV	power-operated relief valve
PRA	probabilistic risk assessment
PTS	pressurized thermal shock

PWR	pressurized water reactor
PWROG	Pressurized Water Reactor Owners Group
PWSCC	primary water stress corrosion cracking
PZR	pressurizer
RCIV	reactor coolant isolation valve
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	[NRC] Regulatory Guide
RHR	residual heat removal
RIL	[NRC] Research Information Letter
ROP	Reactor Oversight Process
RPV	reactor pressure vessel
RV	reactor vessel
RVH	reactor vessel head
SB	small break
SCC	stress corrosion cracking
SECY	Office of the Secretary; [NRC staff report to Commission designation]
SER	[NRC] Safety Evaluation Report
SG	steam generator
SLR	subsequent license renewal
SR	Surveillance Requirement
SRM	Staff Requirements Memorandum
SRP	Standard Review Plan
SRV	safety relief valve
TBS	transition break size
TGSCC	transgranular stress corrosion cracking
TR	topical report
TS	Technical Specification
TSTF	Technical Specifications Task Force
US	United States
USI	Unresolved Safety Issue
VT	visual testing [inspection]
WCAP	[Westinghouse document designation]
xLPR	extremely low probability of rupture

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1 INTRODUCTION

1.1 Purpose

In support of extension of allowable fuel burnup from 62 to 75 GWd/MTU (metric tons of uranium), this *Loss-of-Coolant Accident-Induced Fuel Fragmentation, Relocation, and Dispersal with Leak-Before-Break Credit* topical report (TR) provides technical justification to exclude consideration of fuel fragmentation, relocation, and dispersal (FFRD) from the core cooling evaluation for a loss of coolant accident (LOCA). This topical report applies risk insights for large break (LB) LOCA induced FFRD. The methodology described herein is referred to as the Alternative Licensing Strategy (ALS).

In conjunction with related submittals listed in section 1.5 and described in Sections 3 and 5, this ALS TR evaluated the credibility of the rupture of HBU fuel cladding caused by a LOCA. This report is supported by evaluations showing that FFRD caused by a large break of piping in the RCS main coolant loops has an extremely low probability of occurrence and that clad burst is avoided for smaller LOCAs.

Nuclear Regulatory Commission (NRC) acceptance of the ALS methodology requested by this TR would address LOCA-induced FFRD to efficiently implement the economic and safety benefits of increasing allowable burnup fuel for pressurized water reactor (PWR) licensees. The proposed methodology supports a transition for the bulk of the PWR fleet and provides a pathway to apply this analysis to PWR Nuclear Steam Supply System (NSSS) designs that are not specifically addressed in this report.

1.2 Safety Benefits of High Burnup Limit Extension

One of the fundamental principles of nuclear fuel design is the inversely proportional relationship between fuel discharge burnup and reload batch size. For a given core energy, the equilibrium or long term reload batch size can be determined based on the batch average discharge burnup. This principle is illustrated by comparing the fuel management results in Tables 2 and 3 from [1]. The batch size is reduced by a factor of 1.27 in Table 2 and 1.20 in Table 3 while the discharge burnup is increased by a factor of 1.25 in Table 2 and 1.20 in Table 3. Note that the small variations in the batch size vs. discharge burnup comparisons are due to core designs not being fully converged to an equilibrium design.

The proposed burnup limit increase from 62 GWd/MTU to 75 GWd/MTU is expected to result in an approximate 20 percent reduction in reload batch size. This produces a corresponding 20% reduction in the number of discharged fuel assemblies, and a significant reduction in the amount of high-level waste over the remaining operating life of the LWR fleet. Assuming an 80-year operating life from Table 8 [1], the implementation of a high burnup PWR limit will reduce dry fuel storage cost by $2574M which corresponds to a reduction of 41,000 discharged fuel assemblies ($2M for a 32 assembly PWR dry cask).

While other regulatory approaches are available to achieve higher burnup design limits, they are all expected to require more time to develop and implement than ALS. For example, development of methods to address fuel dispersal consequences requires additional research and testing along with implementation and NRC review of updated safety analysis methods. Since these options are not fully developed one can only estimate the additional development time relative to ALS. Based on a five to ten year assumed additional development timeline and ~40 years remaining to the end of an 80-year operating life, the ALS approach will produce 12.5 to 25 percent fewer discharged assemblies. Using the lower end of this estimate results in a significant reduction in high level waste.

In order to achieve higher burnup, increased fuel enrichment is needed to produce the same cycle energy. The is illustrated by results in Tables 2 and 3 of [1]. As discussed in [1], higher burnup and higher enrichment will enable all US PWRs to operate 24-month fuel cycles (MFC). This is a significant increase over the current conditions which only supports 20% of the US PWR fleet operating 24-MFC.

The adoption of higher burnup/higher enrichment fuel will result in a number of safety benefits, which are summarized below:

- Smaller discharge batch sizes will reduce the number of dry cask loading campaigns resulting in reduced occupational dose to site workers. Fewer dry casks stored on-site will also reduce the site boundary dose and its potential impact on the public. A reduced number of discharged fuel assemblies will reduce the number of shipments to a high-level waste repository once such a facility is completed, reducing the impact of these shipments on the general public. Additionally, the risk of transportation accidents is similarly reduced.
- As shown in Figure 1 of [1], uranium feed, conversion and fabrication requirements are reduced by using higher enrichment and burnup. This reduces the risk of front end fuel cycle transportation accidents and occupational exposure to workers during mining, milling and fabrication. The radiological impact of transportation is also reduced. As shown in Figure 1 [1], enrichment requirements are increased. This is accomplished by extending the time material remains in each centrifuge stage so that has little to no effect on transportation or worker dose.
- The transition from 18-MFC to 24-MFCs eliminates 1 in 4 refueling outages and provides a corresponding reduction in occupational dose. The reduction in outages also reduces outage related risk that may occur with various systems inoperable due to maintenance activities.
- The improved economics of these fuel designs reduce the risk of early plant shutdowns, thereby supporting US and international environmental goals of reduced greenhouse gases emissions.
- The ALS approach avoids testing, modeling and analysis activities of dispersed fuel. This provides allocation of highly skilled NRC and industry resources to more safety significant projects.

1.3 Advantages of ALS Approach as Basis for Burnup Limit Extension

The advantage of the ALS approach proposed herein is to efficiently evaluate the safety impact of extending the limit on PWR maximum burnup. Timely NRC review and acceptance of ALS would provide the benefits noted in section 1.2. ALS provides a framework permitting individual licensees to submit License Amendment Requests (LARs) and the NRC staff to review them based on an accepted methodology. This approach saves licensee and NRC effort and aligns with the NRC Principles of Good Regulation (efficiency, clarity, and reliability, in particular) [2].

Without ALS, realization of the benefits of increased burnup would be delayed and might be too late to qualify for the provisions of recent U.S. incentives for increased non-carbon generation.

Therefore, benefits of ALS are that it:

- Considers risk insights in its methodology to assess the safety significance of increasing the allowable fuel burnup for existing clad formulations.
- Minimizes licensee and NRC effort necessary to document the basis for judging acceptability of increasing the maximum allowed fuel burnup by providing a methodology that could be incorporated by reference in LARs with minimal plant specific evaluation.
- Avoids the need for additional experimental data and development of qualified analytical models to predict FFRD phenomena, which would delay seeking NRC approval to raise the HBU limit past the desired implementation date.

1.4 Basis for the ALS

The ALS objective is to determine if LOCA-induced FFRD of fuel may be addressed based on:

- The extremely low likelihood of LB-LOCAs makes FFRD of HBU fuel not credible because:
  - Rupture of piping in the reactor coolant system (RCS) main loop is extremely unlikely, as documented in NUREG-1829 [3]. As described in section 5.3, EPRI has performed extensive probabilistic fracture mechanics analysis to confirm the likelihood of rupture remains low through a plant life of 80 years and that piping would leak for an extended period to time before rupture, allowing ample time for operating staff investigation, identification, and corrective action.
  - Although core cooling analysis for ruptures of non-piping component pressure boundaries (e.g., pump casings) is not required, such ruptures are also extremely unlikely as evaluated in Section 6. These component pressure boundaries, including flanged joints secured with threaded fasteners, are designed with margin to rupture and would display a long period of degradation and detectable leakage before they did actually fail.

- Small break (SB) and intermediate break (IB) LOCAs involving ruptures of primary piping smaller than the main loops are also unlikely but have a higher probability relative to LB LOCAs. Therefore, ALS includes an evaluation of the adequacy of the emergency core cooling system (ECCS) for a break of the largest branch lines off the RCS main loop to demonstrate acceptable fuel relocation and no clad rupture of HBU fuel, as described in section 3.3.

The ALS approach uses methods in a manner similar to those previously accepted by the NRC to show that the extent and consequences of fuel dispersal do not need to be included in the design basis.

1.5 Supporting Analyses

This ALS TR is supported by three documents being submitted for NRC review, the first two are bundled in the same submittal as this report. The third report will be submitted separately by Westinghouse directly to the NRC.

1. *Materials Reliability Program: xLPR Estimation of PWR LOCA Frequencies (MRP-48*0). EPRI, Palo Alto, CA: 2024. 3002023895 [4]
2. *LOCA analysis of Fuel Fragmentation, Relocation, and Dispersal for Westinghouse 2-Loop, 3-Loop and 4-Loop Plants - Non-Proprietary: Evaluation of Cladding Rupture in High Burnup Fuel Rods Susceptible to Fine Fragmentation*. EPRI, Palo Alto, CA: 2024. 3002028675 [5]
3. Westinghouse WCAP-18850-P, "Adaptation of the FULL SPECTRUM LOCA (FSLOCA) Evaluation Methodology to Perform Analysis of Cladding Rupture for High Burnup Fuel," February 2024 [6].

Additional LOCA analyses that meet the ALS analysis requirements may be submitted at a future date to address different NSSS configurations, fuel designs, or other vendor-specific LOCA methods.

1.6 Scope of this TR

The scope of this TR is limited to the potential for FFRD to be induced by the occurrence of a LOCA. Other analyses are necessary to complete the safety case for extending allowable PWR fuel burnup to 75 GWd/MTU.

While Westinghouse, Combustion Engineering, and Babcock and Wilcox design plants have been previously authorized to use LBB for large diameter RCS main loop piping, ALS supporting analyses have only been performed for Westinghouse plants two, three, and four loop versions with either large, dry or ice condenser equipped containment. These supporting analyses provide the basis for extension of maximum allowable burnup for Westinghouse plants using Westinghouse fuel. ALS may be applied to plants using fuel and analysis methods from other vendors and to other PWR designs, provided they can meet the criteria of Appendix A.

Approval of the ALS methodology would satisfy Commission direction to expeditiously develop an approach to address FFRD [7].

1.7 Approaches Previously Identified by EPRI

During the first half of 2020, a group of experts two utilities, fuel vendors, and EPRI assessed alternatives to the conventional fuel licensing methodology of developing and qualifying an empirically based evaluation model of FFRD of HBU fuel during a LOCA. The reason for the assessment was to provide a time and resource optimized approach to address this issue [8]. The alternatives considered were:

1. Redefine the worst case LOCA (i.e., double-ended guillotine break (DEGB), worst loss of off-site power, and worst case single active failure) as beyond design basis – build on the extensive work done by the NRC on transition break size (TBS) to show that the LB-LOCA initiating event had a low enough frequency of occurrence to be considered outside the licensing basis. The potential for FFRD to occur and its severity could then be considered on a realistic basis (i.e., for credible accident scenarios).
2. LBB – use LBB to demonstrate that occurrence of a LOCA that could lead to FFRD was extremely unlikely. Application of LBB in accordance with 10 CFR 50 Appendix A General Design Criterion (GDC) 4 would demonstrate the slow progression from a detectable leak rate until a rupture might occur in order to take credit for operator detection and response.
3. Reduce conservatism in analysis of FFRD – although there is considerable uncertainty in interpreting empirical data, determine a means to use a more realistic (rather than bounding) means to estimate the extent and consequences of FFRD (use of more best estimate approaches acceptable because very low probability of event).
4. Perform risk-informed evaluation of LOCA-induced FFRD – implement the Regulatory Guide (RG) 1.174 process to show that the change in core damage frequency would be less than the *de minimus* threshold of 10^{-6} per reactor year, based on prior initiating event frequency estimates for LB-LOCAs.
5. Develop FFRD analysis models and acceptance criteria – use the traditional approach of qualifying a predictive model with adequate conservatism, based on empirical data.

Each of the first four options would take advantage of the exceedingly low frequency of occurrence of large LOCAs (see section 5.2).

These five options were gauged on several criteria, including: need for rulemaking, ability to address pellet fragmentation and fuel rod burst, schedule, etc. A risk-informed approach based on RG 1.174 [9] was ranked ahead of the rest, largely because it appeared to offer the most flexibility and had guidance on NRC expectations, both of which were viewed to be beneficial in maintaining the schedule to support initial operation of fuel to higher burnup.

EPRI did not request a review and the NRC did not provide any formal feedback regarding [8].

In February 2022, based upon review of NRC-approved precedents similar in scope to FFRD, EPRI decided that the LBB-based approach could be more expeditiously developed than a risk-informed, RG 1.174 approach. Therefore, this report does not further discuss use of RG 1.174.

1.8 ALS Technical Approach Comparison with NRC Alternatives

The Commission directed the NRC staff to address FFRD as part of rulemaking to raise the limit of fuel enrichment [7]. In [10], the NRC provided various alternatives for dealing with FFRD and requested industry feedback and data on costs, as this TR was being prepared for publication. The NRC stated that Alternative 5 in [10] was an extension of the planned EPRI ALS approach. The industry provided specific feedback on the proposed NRC FFRD alternatives and endorsed the ALS approach without the extension beyond use for FFRD suggested by the NRC staff [11].

1.9 Regulatory Approach Proposed

For purposes of facilitating extension of the fuel burnup limit and reducing NRC resources for the review, this ALS TR provides justification that can be incorporated by reference in individual plant LARs, provided they meet the applicability requirements defined in Appendix A.

This TR focuses on demonstrating that LB-LOCAs have an extremely low likelihood of occurrence; therefore, FFRD can be excluded from core cooling analyses of LB-LOCAs. The assessment in this TR is not intended to show that LB-LOCAs can be excluded from consideration in general for the purpose of assessing ECCS performance, fuel damage, and fission product release. ALS is, however, intended to remove the need for additional testing to support modeling of fuel dispersal. For LOCA sizes below RCS main loop piping, core cooling analysis is performed to determine if clad burst occurs. By justifying that HBU fuel will not undergo cladding burst, potential consequences of dispersal are avoided.

1.10 Action Requested of the NRC

NRC approval is requested of

- The ALS approach that shows:
  - RCS main loop piping LB-LOCAs causing HBU fuel rod clad burst are not credible and
  - LOCAs in smaller piping will not result in unacceptable pre-burst fuel relocation and HBU fuel rod clad burst, based on core cooling analysis using an acceptable methodology.
- PWR licensee reference to the ALS methodology, subject to the criteria listed in Appendix A, in LARs seeking authorization to extend the limit on allowable burnup.

2 RELEVANT REGULATORY GUIDANCE

This section provides background information on FFRD and conditions under which it may occur and summarizes regulations and guidance related to LOCAs.

2.1 Description of Fuel Fragmentation, Relocation, and Dispersal

FFRD is a fuel degradation mechanism identified in separate effects testing of HBU fuel. The following is a description of the separate phenomena comprising FFRD.

- Fragmentation: Some cracking occurs in uranium dioxide fuel pellets housed in cylindrical cladding during normal operation. The resulting pieces are large and typically interlocked so that the pellets remain essentially as one piece, assisted by the cladding restraint. This level of cracking is referred to as macrocracking. During abnormal and accident conditions, thermal transients occur, and the fuel is subject to temperature excursions, which can lead to fuel fragmentation, where the pellets break into smaller pieces than those that have been created by macrocracking. Fine fragmentation or pulverization corresponds to a fragment size typically less than 1 mm in average diameter as defined by the NRC Office of Regulatory Research [12]. Research on fuel in the mid-burnup range has shown fragmentation by macrocracking [13], but fine fuel fragmentation has been found to occur, especially at high pellet local burnups near or above the current fuel burnup limit [14].
- Relocation: Fuel near the outer surface of a pellet has a substantially higher burnup than the pellet interior because of self-shielding (most neutrons are absorbed before reaching the pellet interior). Therefore, fine pellet fragmentation is more likely to occur in the outer pellet regions, during LOCA-type transients, in which the internal rod pressure increases simultaneously with the cladding and fuel temperatures. The extent of fine fragmentation depends on the evolution of the fuel pellet micro-structure and mobility of fission product gases in the grain boundary. Such conditions lead to cladding outward creep deformation which, if large enough, removes the mechanical cladding constraint on fuel pellets. As a result, the fuel fragments created by the temperature transient can potentially drop axially downward within the fuel rod and into open space created by clad outward creep. Fuel fragments can accumulate in the local, larger areas of cladding deformation, called balloons. The size of a balloon may be constrained by nearest spacer grids and surrounding fuel rods.
- Dispersal: Deformation of the localized balloons and higher temperature caused by the reduced fuel decay heat removal will continue to increase cladding ductility and can lead to clad burst. Depending on the burst opening size, it is possible for the fuel fragments near the burst opening to eject from the fuel rod, assisted by the gas flow blowing from the inside of the fuel rod through the rupture opening. This process is called dispersal, and it has been observed to be enhanced by the fine fuel fragmentation at very high burnup in single-rod integral LOCA tests of in-pile and out-of-pile hot cell conditions [12].

2.2 Relevant Regulations and Policy

This section discusses the regulations and guidance pertaining to LOCAs and core cooling analysis. A key but generic requirement is maintaining a coolable geometry.

Although raised as an issue twice since the 1970s, FFRD of HBU fuel is not specifically required to be addressed in current regulations or NRC guidance. However, the Commission previously asked the staff to evaluate the need for it to be considered [7], and the staff is in the process of obtaining public comments regarding alternatives [10].

2.2.1 LB-LOCA Requirements and Considerations

Selection of ductile materials for the reactor coolant pressure boundary (RCPB) has been an underlying principle of the safety of light water reactors (LWRs) since the beginning of commercial nuclear power. The objective was to avoid abrupt, large failures of the RCPB and to provide a means to limit the consequences should one occur.

The precursor to the NRC, the Atomic Energy Commission, made judgments to screen out consideration of failures deemed not to be credible (i.e., extremely low probability of failure). As a result, the existing design basis for LWRs requires consideration of piping ruptures up to the largest pipe in the RCS, but excludes failures of the pressure boundary of large components, based on providing reasonable assurance against occurrence through use of codes and standards, design margins, operating limitations, overpressure protection, maintenance, periodic inspections, etc. Among others, non-credible events include:

- Rupture of the reactor pressure vessel (RPV) or its head
- Rupture of shell enclosing primary inlet and outlet plena of the steam generators
- Rupture of reactor coolant pump casings
- Displacement of major components (e.g., failure of supports) that break more than one pipe

Each of these failures has preventive measures, such as specific material controls to limit material susceptibility, large design margins, procedural means to minimize the possibility and/or severity of events that might challenge their integrity (e.g., pressurized thermal shock), and periodic inspections to verify no unexpected degradation.

In the early 1970s, detailed regulatory requirements for engineered safety feature system (e.g., ECCS) performance during LB-LOCA were developed to ensure that they were designed to cope with a bounding LOCA: an instantaneous DEGB, compounded by failure of the most limiting single active component and loss of off-site power. Although generally viewed as not credible, the DEGB has also been a surrogate to avoid the need to define what lesser events were within or outside the design basis. The NRC has held to the DEGB as the basis for assessing ECCS and containment adequacy.

In 1971, adoption of 10 CFR 50 Appendix A, “General Design Criteria for Nuclear Power Plants,” provided detailed considerations for plant design. The GDC address both maintaining the integrity of the RCPB and providing mitigation for LB-LOCAs. The current version of those GDC most relevant to the subject of this TR are shown in Table 2-1. These GDC identify principles to minimize the potential for failure of the RCPB and to provide core cooling for the most limiting piping ruptures.

Table 2-1. General Design Criteria Relevant to LOCA-induced FFRD

	Description of GDC Relevant to LOCA-induced FFRD (The GDC wording as of December 2023 is shown.)	**Relevance to ALS**
Definitions—Loss of coolant accidents.	“Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.”	Applicable initiating event
Criterion 4—Environmental and dynamic effects design bases.	“Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.”	Regulatory basis for LBB
Criterion 14—Reactor coolant pressure boundary.	“The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.”	Acceptance criterion for RCPB
Criterion 30—Quality of reactor coolant pressure boundary.	“Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.”	Requires ability to detect source of leakage

Table 2-1 (continued). General Design Criteria Relevant to LOCA-induced FFRD

	Description of GDC Relevant to LOCA-induced FFRD (The GDC wording as of December 2023 is shown.)	Relevance to ALS
Criterion 31—Fracture prevention of reactor coolant pressure boundary.	"The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws."	RCPB design criteria that underlie LBB
Criterion 32—Inspection of reactor coolant pressure boundary.	"Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel."	RCPB inspectability is key consideration for LBB
Criterion 35—Emergency core cooling.	"A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that "(1) fuel and clad damage that could interfere with continued effective core cooling is prevented and "(2) clad metal-water reaction is limited to negligible amounts. "Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure."	Specifies basic ECCS capability

In particular, 10 CFR 50.46(c) states [emphasis added]

> *"As used in this section: (1) Loss-of-coolant accidents (LOCA's) are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from <u>breaks in pipes</u> in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system."*

Therefore, core cooling analyses of non-piping component rupture are not required, and also do not apply to scenarios leading to FFRD.

2.2.2 FFRD Guidance

No regulations specifically address FFRD. Limits on fuel burnup are identified in several regulatory guidance documents (e.g., NUREG-0800 [15] section 4.2 and RG 1.183 [16]) and in NRC approval of fuel designs and analyses.

As the design basis accident (DBA) for LWRs, the DEGB has been the focus of safety analysis of plant design and operations. As defined in licensing guidance, an LB-LOCA occurs without warning (e.g., precursor leakage). Without an ECCS, reduced heat removal causes fuel temperature to rise and internal cladding pressure to increase, which causes fuel to get hot enough to fail, releasing fission product radioactivity. To limit radioactive release from fuel to acceptable values, an ECCS is designed, built, and analyzed to show that the system can sufficiently cool the fuel such that damage is limited and radiological consequences are acceptable. Note that for some fully functional ECCS scenarios, some fuel damage and radioactive release may occur. Because of its importance to the plant safety case, LB-LOCA requirements have been evaluated multiple times, refined by inclusion of more detail, and revised to allow alternative treatments.

2.2.3 Options for Screening LB-LOCA Events

In the past, the NRC has accepted several approaches for screening events from consideration in the accident analysis. These have included quantitative (e.g., extremely low likelihood of occurrence, small change in core damage frequency), qualitative (e.g., non-mechanistic, ability to detect precursors, bounded by events with greater risk), or a combination. Screening criteria were used prior to development of probabilistic risk assessment (PRA) and continue to be used both inside and outside a PRA framework. Several screening options exist in current NRC guidance or were previously considered as means to risk-inform evaluation of LB-LOCAs in recognition of the low probability of large RCS ruptures, notably:

- Extremely low likelihood of occurrence – before the use of PRA, some reactor accident scenarios were deemed not credible and, therefore, to not need to be evaluated.[1]

  The 1986 Federal Register notice revising GDC 4 to use LBB states [18] [emphasis added:

  > *"<u>The definition of 'extremely low probability' of pipe rupture is given as of the order of 10^{-6} per reactor year for PWR primary coolant loop piping</u> when all pipe rupture locations are considered. This is consistent with past NRC decisions relating to other postulated events. This value, which includes the probability of an initiating event occurring (such as an earthquake, abnormal transient or an accident), conforms with the implicit design goal of components and structures*

[1] Failure of an RPV is considered not credible because of stringent design requirements, operating procedures, and inspection/surveillance practice. Research on vessel integrity concluded in 1974 that the probability of an RPV failure is less than 10^{-6} per vessel year, with the most likely failures being within the capability of the plant safety systems [17]. Since then, the NRC has added some requirements for preventive measures (e.g., 10 CFR 50.61 regarding pressurized thermal shock).

that are engineered on a deterministic basis. Research performed at Lawrence Livermore National Laboratory confirmed that the three major or U.S. vendors of pressurized water reactors meet this requirement."

In [19], the Commission stated [emphasis added]

> *"Estimating the probability of extremely unlikely events involves considerable uncertainty when sufficient data are not available to plug into the formula. Therefore, the Standard Review Plan for reactors deems a threshold probability of one in a million (1 x 10^{-6}) to be acceptable where, when combined with reasonable qualitative arguments, a still conservative probability can be shown to be lower. That is, <u>where a conservative estimate shows an event has no greater than a one-in-a-million probability, that event may be ignored in facility design if reasonable estimates result in a lower probability when conservative margins are not factored in</u>."*

Note that NRC guidance on risk-informed decision-making [20] defines a process by which a licensee submittal is evaluated based on the extent that PRA information is used:

> *"Qualitative insights of risk significance or supporting information can be used to support regulatory decisions. Examples include probabilities of failures of equipment, the frequency of initiating events, or other pertinent information."*

The guidance notes that such "likelihood information" should not be called PRA results or input.

- Break exclusion zone – the Standard Review Plan (SRP) permits exclusion of breaks and cracks in large and small high and medium energy piping in containment penetration areas (i.e., between inner and outer containment isolation valves) if the piping in this "break exclusion zone" meets specific requirements defined in SRP Branch Technical Position 3-4 (BTP 3-4) [15].[2]
- TBS[3] – for over 15 years, the NRC and industry pursued the idea that design basis analytical methodology could be less conservative for the lowest probability large breaks. For rupture sizes above the TBS, which was expected to be the pressurizer surge line for PWRs, ECCS performance did not need to consider a single failure or a coincident loss of off-site power.
- LBB – since its initial adoption in the mid-1980s, LBB has been expanded to justify acceptability of exclusion of main loop piping rupture for purposes other than its initial application, removal of pipe whip restraints and jet impingement shields. Although an NRC policy generally prohibits its use for ECCS, containment, or environmental qualification (EQ) relaxations, LBB has been applied to several LB-LOCA issues, as described Section 4.

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[2] For example, in addition to meeting ASME Code Section III, Article NE-1120, design stress and fatigue limits specific to the piping code class must be met. Also, welded attachments should be avoided and the number of piping welded should be minimized.

[3] Also referred to as "50.46a" for the regulation number it was assigned during rulemaking. Note that the designation "50.46a" was subsequently repurposed for the acceptance criteria for RCS venting systems.

- Risk-informed design basis changes – RG 1.174 [9] provides a process for risk-informed changes to plant design provided the increase in risk is limited to a value determined from the current total core damage frequency (CDF) and large early release frequency). Increases of CDF less than 10^{-6} per reactor year are generally acceptable.
- American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA standard [21] defines screening as "a process that eliminates items from further consideration based on their negligible contribution to the probability of an accident or its consequences." The screening process is described as the following three processes:
  - Screening criteria: "The values and conditions used to determine whether an item is a negligible contributor to the probability of an accident sequence or its consequences."
  - Qualitative screening: identify portions of the analysis where contribution to overall risk can be judged negligible without quantitative analysis.
  - Quantitative screening: eliminate further consideration based on preliminary estimates of risk contribution through use of established quantitative screening criteria.

  The ASME PRA standard screening criteria to eliminate initiating events from further evaluation are one of the following:
  - Event frequency is less than 10^{-7} per reactor-year (/ry), and the event does not involve either an interfacing systems LOCA (ISLOCA), containment bypass, or vessel rupture.
  - Event frequency is less than 10^{-6}/ry, and core damage does not occur unless at least two active trains of diverse mitigating systems have independently failed.
  - The resulting reactor trip is not an immediate occurrence. That is, the event does not require the plant to go to shut down conditions until sufficient time has expired during which the initiating event conditions, with a high degree of certainty (based on supporting calculations), are detected and corrected before normal plant operation is curtailed (either administratively or automatically).

The ALS methodology does not rely on PRA. Instead, ALS shows that fuel dispersal is not credible by use of a combination of extremely low likelihood of occurrence and ability to detect a small leak long before rupture. In the ALS approach, through-wall failure is detected and the plant is shut down per current technical specifications well before rupture of the RCS main loop. Smaller LOCAs in connected fluid systems are addressed by the ECCS core cooling analysis [5] summarized in Section 3.

2.2.4 Regulatory Guidance Pertaining to Fuel Fragmentation

Although FFRD of HBU fuel is not specifically addressed in NRC regulations or guidance, another type of fuel fragmentation is. NUREG-0800, the light water reactor Standard Review Plan (SRP) [15] section 4.2, specifies that fuel rod fragmentation must not occur as a direct result of blowdown loads: combined loads on fuel rods and components other than grids must remain below justified strength values. Note that this fragmentation is mechanistically different from FFRD but raises similar concerns regarding coolability and deleterious impact on fission product retention. As such, it applies to fuel of any burnup. The criterion concerns the differential pressure loads caused by the rapid blow down in a LOCA. However, the reason for the acceptance criterion is similar to concerns for FFRD: to maintain a coolable geometry. Additionally, section 4.2 includes the need to show that 10 CFR 50.46 temperature and oxidation limits are not exceeded. Satisfying these criteria is done by performing an ECCS core cooling analysis. Applicability of LOCA acceptance criteria of section 4.2 of NUREG-0800 has been excluded by taking credit for LBB (see section 4.2).

2.2.5 Potential Changes to Regulations

No NRC regulation nor guidance document identifies how to address FFRD, whether it is within or beyond the design basis, the level of conservatism in analysis, etc. The Commission requested that the staff consider the need to address FFRD as part of evaluating rulemaking to allow fuel enrichments in excess of five percent [7]. The increased enrichment rulemaking basis [10], issued in September 2023, discussed a number of alternatives that were still under evaluation when this TR was prepared. In January 2024, the industry provided feedback supporting the development and use of the ALS methodology [11].

Implementation of Higher Fuel Burnup Limit

The implementation of the current 62 GWd/MTU peak rod average limit appears in several NRC guidance documents, such as Regulatory Guides 1.195 (footnote 5) [22] and 1.183 (footnote 10) [16].

The NRC Office of Regulatory Research [12] has recommended a burnup-dependent correlation to conservatively bound the extent of fine fragmentation (i.e., 2 mm and 1 mm diameter fragments). The NRC has not indicated how the criterion from the Research Information Letter (RIL) would be implemented in regulations or by analytical criteria, but it would only affect the ECCS cooling analysis for branch line LOCAs because main loop LB-LOCA is excluded under the ALS methodology. The ECCS evaluation methodology specifies the onset of fine fragmentation.

Increased Enrichment

Regulatory changes are required to allow fabrication of fuel with U-235 enrichments exceeding five percent. This enables fuel to be operated to higher burnups while still meeting core reactivity design requirements. The Commission directed the staff to address FFRD as part of preparing rulemaking to allow increased enrichment [7]. However, rulemaking to allow increased enrichment is not dependent on NRC acceptance of this TR.

2.2.6 Other Potential Fuel Damage Caused by LOCAs

ALS addresses only HBU fuel potentially susceptible to FFRD induced by LOCAs. Damage of low burnup and high burnup fuel from other mechanisms must be addressed separately.

Fuel assemblies with rods at low burnup are generally operated in high power locations in the core and some rods may be predicted to burst. Low burnup fuel generally is subject to only macro-cracking, but the potential for fine fragmentation susceptible to dispersal increases with rising burnup. At fuel burnups below the threshold for onset of fine fragmentation, clad burst is not expected to release much fuel material [13].

2.2.7 LOCA-Induced FFRD in Context of Current Regulatory Framework

As the NRC stated [10], “current precedent is to assume that fuel dispersal does not occur in accident analysis under 10 CFR 50.46.” The NRC notes that the licensing basis of operating plants assumes that fuel remains confined within the rod cladding if regulatory criteria are satisfied.

> *“Licensees may also be able to achieve HBU and corresponding [increased enrichment] IE necessary to go to such burnups by demonstrating that fuel dispersal can be limited or prevented during LOCAs…. The staff would continue to use risk insights, as they do today, in determining whether the current state of knowledge on FFRD and individual licensees’ safety analyses are sufficient to provide reasonable assurance that any potential dispersal would be sufficiently limited to preclude a safety concern without further evaluation.”*

The NRC [10] pointed out that dispersal is separable from fuel fragmentation and relocation:

> *“In summary, there are no regulatory hurdles, or any regulatory action needed regarding fuel fragmentation and relocation, but the staff believes that action may be needed to address and analyze fuel dispersal.”*

2.3 Defense-in-Depth

2.3.1 Extremely Low Likelihood of Scenarios Leading to FFRD

As described in Section 5, [4] documents xLPR evaluation of PWR LOCA susceptibility performed as part of the overall ALS technical basis. This assessment confirms the expert elicitation results in NUREG-1829 [3] are appropriate for piping (see section 5.1) and non-piping (see Section 6) for an extension of plant life to 80 years. These evaluations use different approaches and independently confirm that failures with the potential to cause a LB-LOCA leading the FFRD have an extremely low likelihood of occurrence.

2.3.2 Level of Conservatism

As previously noted, the uncertainties in modeling various FFRD phenomena (dispersal in particular) have led to interpreting test results in a highly conservative manner. Superimposing a highly conservative treatment of burst and dispersal on top of a low frequency event such as LB-LOCA would result in an unrealistically pessimistic assessment of FFRD. In other words, assessing FFRD phenomena on a bounding or even 95/95 basis is inappropriate for possible fuel behaviors instigated by a less than one-in-a-million initiating event. This is consistent with a principle of risk-informed decision-making [23]: [emphasis added]

> *"For screening purposes, the level of conservatism used is generally the minimum required to generate a frequency, consequence, or risk estimate that is below established criteria (i.e., the level of conservatism may have to be reduced in order to screen out an item). When a less-than-bounding but conservative analysis does not result in the screening of an item, it may be necessary to perform a more detailed analysis to either screen the scope item or provide a more realistic estimate for use in the risk-informed application. It is possible that a specific PRA item could be screened using a combination of conservative and best-estimate models and data."*

Although ALS is not a PRA methodology, the above discussion of screening is relevant, supports the use of mean frequency values from NUREG-1829, and supports assessment of consequences on a risk-informed, not bounding, basis.

Similarly, defense-in-depth for DBAs may be done on a best-estimate/realistic basis. For example, the NRC states [24]:

> *"For design basis accidents as described in Guideline 11 [diversity in accidents] (in combination with primary protection system failure), the goal of defense-in-depth analysis using best-estimate methodology is to show that any credible failure does not result in exceeding the 10 CFR 100 dose limits, violation of the integrity of the primary coolant pressure boundary, or violation of the integrity of the containment."*

The suitability of the ALS approach is built upon a number of varied but complementary analytical, procedural, human factor, and other inputs that provide the basis for concluding that LOCA-induced FFRD need not be considered in the design basis of U.S. for PWRs meeting the criteria of Appendix A. Specifically:

- Plant Technical Specifications (TS) include a Limiting Condition for Operation (LCO) that require shutting down with cooldown to Mode 5 (or Mode 4 for a few plants) if unidentified leakage exceeds 1 gpm (3.8 L/min) or if there is any RCPB leakage.
- Plant TS also include a required surveillance to assess RCS leak rate periodically, not less than once every 72 hours, but usually daily.
- In response to Technical Specifications Task Force (TSTF) item 513 [25], Westinghouse revised the standard TS to clearly lay out actions to monitor for, diagnose the causes of, and conservatively respond to indication of RCPB leakage [26] (see section 4.4.3).
- RG 1.45 provides guidance on diverse means for monitoring for RCS leakage, as discussed in section 4.4.2.

- As part of the revised Reactor Oversight Process (ROP) implemented in 2000, the NRC focused attention on key plant operating conditions for monitoring plant safety. RCS leakage is one of the performance indicators (PIs) comprising the Barrier Integrity cornerstone.
  - Actual monitoring for increased unidentified leakage has been shown to be capable of detecting rates at or below 0.05 gpm (0.19 L/min), or one-twentieth of that assumed in the xLPR evaluation. If 0.1 gpm (0.38 L/min) were to be used as the onset of leak detectability, then the time available for operators to identify, evaluate, and respond would increase.
  - Zero RCPB leakage is permitted by PWR plant TS (see section 4.4.3), and unidentified leakage is limited to a maximum of 1 gpm (3.8 L/min). Emphasizing minimal identified leakage makes detecting new or unidentified leakage easier. Because it is a performance indicator (PI), RCS leakage is regularly reviewed by plant operating staff, plant management, licensee internal reviews, Institute for Nuclear Power Operations (INPO) reviews, and NRC inspections. These reviews assess not just the quantity, trend, and potential consequences but also the plant staff and management attention to it.
  - In accordance with NEI 99-02, identified RCS leakage is recorded at least monthly and reported quarterly.
  - The RCS Leakage PI monitors the integrity of the RCS pressure boundary by reporting the maximum monthly value of identified leakage compared to the TS limiting value. The value must be less than 50% to avoid a rating greater than Green.
- The NRC has issued a number of generic communications (e.g., Generic Letters (GLs), Bulletins) to inform licensees, and sometimes direct their action, regarding significant events or trends affecting RCPB. These are discussed throughout this TR and listed in Section 8, References.
- Although not binding on licensees, the NRC has established Inspection Manual direction to NRC staff inspectors to review key topics associated with RCPB degradation and its PI.

In addition to the operational considerations noted above, the deterministic process for LBB applicability includes conservatisms such as assuming that a through wall crack develops with the potential to grow to the point of rupture, that flaw assessed for structural stability will be sized to leak at a rate 10 times the minimum detectable leak rate for the location, and that the least favorable combination of loading and material properties exist. Also, the as-built (vs. design) configuration is used in the evaluation with attention to proper location and functionality of supports and snubbers (Section 3.6.3 of [15]). Most such LBB flaw evaluations identify significant margin to crack propagation. While an important consideration for this evaluation, it is worthy to note that other LBB applications rely on the conservative evaluation of proposed LBB piping to ensure piping rupture will not occur.

2.3.3 Eliminating the Causes of FFRD

If operators detect a leak, shut down and transition to Mode 3 within approximately six hours and Mode 5 (Mode 4 for some plants) within approximately 36 hours in accordance with plant TS LCO, the energy to drive a pipe break (RCS pressure) and the conditions to cause clad burst and FFRD have been removed.

Once at reduced temperature and reduced RCS pressure, the energy and conditions to cause a large pipe rupture are absent: pressure loads, temperature gradients, subcooled water, etc. Table 2-2 summarizes the significance of plant conditions regarding potential for an LB-LOCA. In between reactor shut down and being in Modes 4 or 5, the parameters needed to cause FFRD are significantly reduced or eliminated. Following shutdown, decay heat drops off rapidly and stored energy in the fuel and coolant is reduced. The resultant lower fuel rod temperatures would reduce or preclude rod ballooning. The gas pressure inside the fuel pellets and in the free space inside the rod will be lower, reducing the ability to cause fine fragmentation or clad balloon and burst and to force fuel fragments out. For those plants with TS specifying Mode 4, rather than Mode 5, growth of a small leak would be slowed because of the reduced stresses compared to Mode 1 and, therefore, be unlikely to rupture.

Failure of the plant operations staff to shut down the plant prior to the occurrence of a LB-LOCA is not a credible scenario given the long time between detectable leakage and a LOCA as described in section 3.2.1. As shown in section 4.4.7 and Table 4-7, human performance error resulting in the highly qualified operations crew failing to perform the LBB T/S LCO as required is unlikely, but the entire operations staff not observing onset and growth of leakage multiple times across multiple shifts is not a credible scenario. The existence of the large number of opportunities available to shut down the plant is a key element of the defense-in-depth inherent in the ALS.

Table 2-1. Post-LCO Plant Conditions Affecting Likelihood of LB-LOCA and FFRD

Plant Conditions	Value / Conclusion
Mode	Mode 5 {Mode 4 for some plants}
TS max time to reach Mode 5	Be In Mode 3* within 6 hours and in Mode 5 {Mode 4 for some plants] within 36 hours
Coolant temperature	< 200 F (93 C) {between 350 F (177 C) and 200 F (93 C) for Mode 4}
Coolant pressure	Elevation head {below cooldown curve for Mode 4}
Decay heat vs. time after shutdown (percent of rated power) [27]	6 hours – 0.70% 36 hours – 0.44% 1 week – 0.30% 1 month – 0.21%
LB-LOCA susceptibility	Mode 5 none {Mode 4 negligible}: a) coolant has insufficient energy to drive leakage crack to grow to point of rupture and to offset pipe ends b) coolant temperature is below boiling and absence of pressurizer bubble prevent rapid blowdown c) RCPB stresses low at reduced pressure, temperature, and flow[4]
FFRD Susceptibility	Mode 5 and Mode 4 none a) no rapid blowdown to cause fuel to uncover b) no rapid rise in fuel temperature to cause fission gas driven fragmentation or weakening of clad c) no depressurization transient to cause high clad stress leading to burst d) low rod internal pressure eliminates force to drive clad burst and fuel fragment dispersal
Energy stored in fuel and coolant	Temperatures of fuel and coolant are at least 150 F (66 C) less than while operating at power, which eliminates initial heat up caused by stored heat and provides large heat capacity to moderate heat up should active decay heat removal flow be affect
Fuel uncovered as direct result of leak?	No
Need for operator action	Ample time to restore decay heat removal
Remarks	Mode 5: No pressurizer bubble to push water out
Conclusion	**LOCA-induced FFRD not credible in Mode 5 or 4**
* Plant is only transiently (less than six hours) in Mode 3, so Mode 3 is not evaluated as an initial condition.	

[4] xLPR evaluation concluded 19 months or more from detectable leakage until rupture while at normal operating conditions. Even in Mode 4, the lower stresses will be insufficient to cause rupture of will extend time to rupture.

3 METHODOLOGY

The ALS methodology has been developed to provide an efficient approach, from the perspective of both licensees and the NRC staff, to satisfy safety criteria in support of extending PWR HBU fuel limits. ALS applies approaches similar to those previously accepted by the NRC for other purposes, with the objective of facilitating review and approval on a timetable that supports industry strategic objectives. ALS methodology is comprised of the following parts:

1. This TR which
   - Describes and justifies the overall methodology
   - Provides the basis for assessing if LB-LOCA-induced FFRD is credible
   - Identifies applicability requirements for the use of ALS in licensee LARs
   - Discusses regulatory framework relevant to ALS, including applicable regulations, guidance, and precedents
   - Assesses, for defense-in-depth, the possibility of a large RCPB rupture because of failure of a non-piping component in the RCS
2. Probabilistic fracture mechanics analysis using the xLPR code documented in [4] that
   - Develops LOCA frequency estimates to both complement and compare against similar estimates from NUREG-1829
   - Assesses the time between onset of detectable leakage and occurrence of a LOCA
3. Core cooling analyses for SB-LOCAs and IB-LOCAs documented in [5] that
   - Uses an evaluation method extended for HBU applications, details of which are provided in a topical report [6] incorporated by reference.
   - Evaluates LOCAs of branch lines off the RCS main loop piping to determine if clad burst of HBU fuel will occur.

As items 2 and 3 are described in other documents, this section briefly discusses their results, in addition to the methodology. Refer to [4], [5], and [6] for details.

3.1 Considerations for Why the ALS is Needed Now

Previously, the NRC did not consider action was needed for FFRD. Both the timing of implementing the ALS and its approach of avoiding evaluation of the degree of dispersal are consistent with NRC's position in 2015. As part of rulemaking to issue 10 CFR 50.46c ("Emergency Core Cooling Systems Performance During Loss-of-Coolant Accidents"), the NRC staff concluded that addition of FFRD requirements to 50.46c was neither practical nor appropriate at the time [28] [emphasis added]:

> *"The experimental results have continued to support the hypothesis that FFRD phenomena are primarily a high burnup fuel issue and that the current licensing limits in the U.S. are adequate to prevent dispersal of large quantities of fine fuel fragments."*

"Research has shown that as burnup exceeds 62 GWd/MTU, fuel becomes increasingly susceptible to FFRD."

"Regulations proposed in the draft final 10 CFR 50.46c rulemaking define ECCS performance requirements. During a postulated large break LOCA, fuel cladding rupture may occur early in the transient. <u>Requirements for ECCS performance to prevent or minimize the degree of fuel cladding rupture as a means to prevent or minimize fuel dispersal would likely not be practical. However, fuel performance requirements could be developed as part of a separate regulatory effort to focus on preventing rupture in rods susceptible to fine fuel fragmentation, and therefore susceptible to fuel dispersal, while avoiding unnecessary restrictions on rods that are not susceptible to fine fuel fragmentation.</u> Establishing this boundary condition (i.e. no rupture of fuel rods susceptible to fine fuel fragmentation) addresses one of the Commission's concerns in SRM-SECY-12-0034 by minimizing the likelihood of repetitive costs relative to § 50.46c implementation."

Note that SECY-15-0148 [28] suggests focusing on "preventing rupture in rods susceptible to fine fuel fragmentation," which is the approach used for the ALS. Also, NRC's statement that imposing requirements on the ECCS "to prevent or minimize fuel dispersal would likely not be practical" is consistent with the ALS approach to determine if clad burst can be shown to not occur. Without clad burst, no HBU fuel fragment dispersal can occur.

The ALS approach precludes the need to evaluate dispersal, which is acknowledged as a possibility by the NRC identified alternatives included in the regulatory basis for the increased enrichment rulemaking [10]. In January 2024, the U.S. industry provided feedback supporting the development and use of the ALS methodology [11].

Table 3-1 summarizes the elements of which ALS is comprised.

Table 3-1. Summary of Elements of ALS Methodology

Element	Current Criteria	ALS Approach	Requires Exemption?	Precedents?
Acceptance criteria for core cooling	Meet 10 CFR 50.46 hydrogen, peak clad temperature, clad oxidation limits for full range of LOCAs. Maintain coolable geometry	Current criteria remain, but, for FFRD, LBB and probabilistic fracture mechanics results demonstrate that FFRD induced by a LB-LOCA is not credible. For smaller lines, IB-LOCA and SB-LOCA analyses (which consider fuel relocation) demonstrate that clad rupture will not occur and, therefore, no fuel dispersal will occur.	NRC [10] stated 50.46 is the only regulation affected by fuel dispersal and that LBB and xLPR are not allowed currently to be applied to LOCA analysis.	LBB has been approved by the NRC in regard to 50.46 criteria (see section 4.2)
Extremely low likelihood of occurrence	Initiating events with a frequency of less than 10^{-6} per calendar year (CY) are not credible	Based on NUREG-1829, frequency of occurrence is below 10^{-6}/ CY, which justifies exclusion of FFRD from design basis	No	Yes: RPV failure and Unresolved Safety Issue (GSI) A-2 asymmetric blowdown
LBB applicability for piping	PWR RCS main loop piping is approved for LBB	Show RCS main loop LB-LOCA is not credible based on extremely low likelihood of occurrence, supported by LBB.	GDC 4 allows LBB for dynamic effects	Similar: LBB used to justify not considering 50.46 criteria and clad fragmentation caused by hydraulic forces (see section 4.2)
Non-piping failure	Non-piping failures not analyzed for core cooling	As DiD, non-piping failure is assessed with conclusion that large rupture is not credible	No	No: Non-piping rupture not in licensing basis
NRC policy limits use of LBB for ECCS, containment or EQ	LBB applicability limited – no change to engineered safety feature system (ESFS) design or operation	LBB applied to breaks possibly leading to HBU fuel clad burst but not for EQ, ECCS, or containment design or operation	No: Commission policy, not regulation	Yes: (see section 4.2)
Defense-in-depth (DiD) analysis	Not explicitly required for using extremely low likelihood of failure or LBB.	LBB as confirmed by xLPR analysis shows extended time period available for operator recognition and action	No: no regulations prohibit, but GDC 4 requires Commission approval	Previous LBB justifications have not addressed DiD

To provide complete coverage of the various aspects of LB-LOCA with regard to HBU fuel, the approach is described in the following sections of this TR:

- Section 4 discusses the history, limitations, expansion, and other aspects of LBB in detail.
- Section 5 summarizes results of xLPR evaluations of susceptibility to large piping ruptures.
- Section 6 discusses the possibility of large, non-piping component ruptures to assess defense-in-depth.
- Appendix A identifies the criteria to be met by an individual plant to incorporate the ALS evaluation and results of this TR and [4] and [5] by reference. Appendix A also addresses extending ALS applicability to reactors other than those included in the current Westinghouse analysis [5].

The basis for the ALS is that LB-LOCA ECCS performance evaluation requirements are unchanged, but occurrence of HBU fuel rod clad burst and subsequent fuel fragment dispersal are not considered credible and, therefore, not part of the LB-LOCA analysis. Justification for this involves extremely low likelihood of failure supported by LBB to screen out the occurrence of LB-LOCA-induced FFRD. Thus, a plant applying the ALS as justification for seeking an increase to the maximum allowable fuel burnup would continue to perform core cooling analysis in accordance 10 CFR 50.46 for pipe breaks up to a DEGB, but would not be required to include occurrence of FFRD in its analysis model.

3.2 xLPR Evaluation of Reactor Coolant Loop Piping LB-LOCAs

NRC regulations (10 CFR 50.46) require evaluation of LOCAs caused by breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS. As discussed in section 4.3, the piping in the RCS main loop of U.S. PWRs has been approved for application of LBB per 10 CFR 50, Appendix A, GDC 4.

The xLPR probabilistic fracture mechanics code [29], developed cooperatively by EPRI and the U.S. Office of Nuclear Regulatory Research, has previously been used to generate piping failure frequencies for use in studying whether piping systems meet the extremely low probability of failure requirements of GDC 4. The xLPR code was developed under a rigorous quality assurance program with the objective of using it for decision-making [29]. For the work discussed herein, xLPR is used to develop analytically derived LOCA frequency estimates to both complement and compare against similar estimates presented in NUREG-1829 for a range of PWR piping systems and line sizes, and to rigorously investigate the time between detectable leakage and large break LOCA (LB-LOCA). Key xLPR outputs investigated are:

- The probability of LOCAs (e.g., pipe ruptures) as a function of line size
- Time between detectable leakage (which occurs as a precursor to rupture) and the occurrence of a LB-LOCA event to demonstrate that sufficient time exists to allow for operator detection, diagnosis, and response

For the purposes herein, detectable leakage is defined as a through-wall crack that results in achieving a rate of 1 gallon per minute (gpm)(3.8 L/min). In the xLPR model, this crack is then grown to the point of pipe failure (i.e., LB-LOCA). Note that RCS leakage is detectable at rates below 0.05 gpm (0.19 L/min): 1 gpm (3.8 L/min) is the general LCO upper bound for unidentified leakage, so using it as the initial opportunity for detection conservatively underestimates the time available for operator detection and response.

The xLPR code can determine the likelihood of an LB-LOCA in piping by employing Monte Carlo sampling from probability distributions assigned to user-selected inputs to account for uncertainties in those parameters. Through this randomized selection of input values, the user can generate numerous possible scenarios (referred to as “realizations”) that thoroughly consider the variability of conditions within the plant and individual welds. xLPR analysis cases were developed applying fatigue (driven by plant transients) and/or primary water stress corrosion cracking (PWSCC) as material degradation mechanisms. xLPR models two-dimensional cracks with axial or circumferential orientations and can model individual or multiple flaws present at the start of the simulation or use initiation models to calculate the time to flaw initiation. In either case, flaws of engineering scale were modeled in xLPR. When multiple cracks are postulated in a weld the code also accounts for coalescence of adjacent circumferential cracks.

Sensitivity cases were included to model alternate inputs for parameters such as geometry, loading, weld residual stress profiles, seismic effects, or initial flaw sizes and thereby gain a broader understanding of the problem. The xLPR code can also account for the likelihood of in-service inspection (ISI) finding a developed crack so that it may be repaired or how the sensitivity of leak rate detection (LRD) can preclude rupture.

Results from other studies such as [30] were leveraged when possible and supplemented with additional xLPR analyses as needed. When key assumptions or inputs (e.g., whether a specific location has been mitigated) differ between otherwise comparable cases, reconciliation of these differences and their significance is discussed in [4].

3.2.1 Conclusions of xLPR Evaluation

The xLPR study [4] evaluated potential for a piping rupture (for individual welds) in various piping locations within the RCS main loop. For each location, a base case is initially evaluated using expected conditions of that piping section with local environmental and operating conditions consistent with the best estimate approach for using the xLPR code. Then, sensitivity cases were defined to inform understanding of the base case by investigating inputs known to have influence on xLPR results and modeling decisions made during input development. Consequently, the sensitivity cases are less constrained to maintain fidelity to realistic plant conditions which means piping rupture results for sensitivity cases should be thoroughly investigated to ensure that the results are applicable to the plant configurations of interest.

The results of the xLPR evaluation are described in Section 6 of [4] and summarized below:

- As base case probabilities of rupture for individual welds with 1 gpm (3.8 L/min) LRD were zero through 80 years, plant level probability of rupture could also be taken as zero.
- For the RPV outlet nozzle sensitivity cases with nonzero "occurrence of rupture with ISI and LRD," the 80-year results are on a similar or lower order of magnitude than NUREG-1829 results at 40 years.
  - Notably, the cases exhibiting ruptures while crediting ISI and LRD are sensitivity cases modeling scenarios not representative of current plant conditions and operations.
- No cases indicate any significant probability of a rupture for the operating fleet without being preceded by a detectable leak.
  - To obtain the time from detectable leakage to LB-LOCA, the analysis had to omit credit for either LRD or ISI.
- The distribution of times between detectable leakage and LB-LOCA can be characterized by a lower bound 95/95 one-sided tolerance interval of 19 months before a crack with a leakage rate of 1 gpm (3.8 L/min) progresses to the point of a LB-LOCA.
- For the nozzles of the RPV inlet, the reactor coolant pump (RCP), and the steam generator (SG) inlet (which have all been mitigated to reduce break likelihood), LB-LOCA was not observed to occur or would be highly unlikely.

In summary, the xLPR study [4] results considering ISI and LRD produced 80-year LB-LOCA frequency estimates on a similar order of magnitude to those in NUREG-1829. Supported by additional consideration of time for LRD before rupture, "this collection of work further improves confidence in the NUREG-1829 LOCA frequency estimates for future applications."

3.3 Fuel Clad Integrity for LOCAs Smaller than those Excluded by LBB

As ALS does not consider LBB for piping other than the reactor coolant loop, analysis of core cooling for break sizes up to the largest branch line piping must be performed to determine if clad burst occurs. Analysis has been performed for intermediate and small LOCAs to determine if HBU rod cladding burst would occur. Westinghouse core cooling analysis is described in separate proprietary and non-proprietary TRs [5]. In addition, the detailed core cooling evaluation methodology [6] has been submitted separately for NRC review concurrent with this ALS TR and is summarized below.

For LOCAs up to the size of the largest RCS main loop branch connections (the pressurizer surge line for the hot leg and an accumulator line for the cold leg), the potential for fuel dispersal is addressed by demonstrating that the HBU fuel rod cladding does not fail.

3.3.1 Conclusions of Core Cooling Evaluation

The cladding rupture calculations are described in [5]. Comparisons of peak cladding temperature to the cladding burst temperature show a positive margin for small and intermediate break LOCAs up to the maximum connecting line break sizes. Therefore, all analyses consistently indicate with a high probability that cladding rupture would not occur for higher burnup fuel rods potentially susceptible to fine fragmentation during a LOCA.

4 LEAK-BEFORE-BREAK

Based on experience with boilers and steam systems since the 1800s, nuclear power plant systems have been required to use ductile materials, which can sustain substantial deformation under tensile stress before failure. Experience has shown that ductile materials can be expected to fail incrementally, providing advance indication by leaking for some time before failing.

NRC regulations and guidance require special precautions to protect low ductility components against pressurized thermal shock (PTS) or other conditions that could cause abrupt failure. NRC regulations also require the RCS to be a leak tight barrier and that there be multiple means to detect leakage.

As discussed below, deterministic fracture mechanics methods were developed to implement the concept of LBB. NUREG-0800 [15] section 3.6.3 discusses the use of LBB to exclude certain failures identified in 10 CFR Part 50, Appendix A, GDC 4. Because the application of LBB requires detectability of leaks with a factor of 10 margin (i.e., LBB is demonstrated if a crack were big enough to leak at a rate 10 times that assumed to be detectable), LBB has generally been applied to large diameter piping. ALS credits LBB as part of the basis for screening out occurrence of piping breaks only in the RCS main loop, although some plants may be authorized to use LBB for smaller piping.

4.1 Adoption and Expansion of LBB

LBB is founded upon use of fracture mechanics to demonstrate that high energy fluid system piping is very unlikely to experience large ruptures or their equivalent as longitudinal or diagonal splits. Application of LBB requires [31]:

1. Knowing the loads (internal pressure, deadweight, thermal, etc.) to which piping may be subjected during operation and transients
2. Knowing the geometry
3. Identifying material properties of the piping appropriate for its service life
4. A suitable and accepted method for analyzing piping flaws (i.e., fracture mechanics)

When LBB was first accepted by the NRC for use in the 1980s, uncertainties were addressed by determining an acceptable range and defining minimum required margins. At the time, because of limits of knowledge and analytical techniques, the NRC excluded application of LBB to ECCS design and performance, and EQ of safety-related equipment [32]. In establishing this policy, the NRC noted that lack of identified safety benefits were the primary determinant of whether to expend resources to perform research and rulemaking. At the time, the NRC observed, based on industry input, that safety benefits of applying LBB to these areas could be obtained more expeditiously and efficiently under a revised rule allowing best estimate analysis with quantified uncertainty for evaluating LOCAs.

The NRC-accepted methodology for applying LBB is described in section 3.6.3 of the SRP [15]. This methodology includes conservative criteria (see section 2.3.2). Based on structural conditions (e.g., loading, pipe geometry, fracture toughness), through-wall flaw sizes are postulated which would cause a leak at a rate of ten times the leakage detection system capability of the plant. Generally, large margins for such flaw sizes are demonstrated against flaw instability. As smaller piping must be able to sustain a lower leak rate before failing than a large pipe, leaks in smaller pipes may not be detectable while operating. Therefore, LBB is useful primarily for larger diameter piping, and ALS considers it only for RCS main loop piping.

The development, adoption in regulatory justifications, and continued refinement of LBB has been the most widely used approach to provide flexibility in addressing LB-LOCAs. GDC 4 requires consideration of the dynamic effects of high energy line breaks such as pipe whip. When 10 CFR 50 Appendix A was initially issued, it did not include the concept of using LBB to justify exclusion of dynamic effects.

4.1.1 Advent of LBB – USI A-2

Asymmetric blowdown loads on PWR primary systems by an LB-LOCA, initially identified to the NRC staff in 1975, was designated Unresolved Safety Issue (USI) A-2 [33]. A postulated DEGB could cause previously unanalyzed loads on primary system components, with the potential to alter primary system configurations or damage core-cooling equipment and contribute to core melt accidents. Newer plants had pipe whip restraints, but a number of older ones did not. The resolution of this issue would have required some licensees for operating PWRs to add large piping restraints incurring considerable cost and radiation exposure and inhibit access to piping for ISI. Instead, this issue was resolved by the industry and the NRC staff by the adoption of the LBB approach utilizing advanced fracture mechanics techniques [34].

USI A-2 identified that if an RCS loop piping rupture caused an asymmetric blowdown which displaced the reactor vessel or the core, then the geometry might become uncoolable. If such an event occurred, the core cooling analysis might not be valid, creating uncertainty (e.g., distortion of flow paths) regarding the ability to meet safety criteria. GL 84-04 [34], which contains the NRC Safety Evaluation Report (SER), described resolution of USI A-2 by application of LBB to exclude RCS loop piping failures from the design basis for several Westinghouse plants. Justification included avoided cost and avoided radiation exposure.

In 1986, the NRC revised GDC 4 [18] consistent with GL 84-04 and allowed its use in other plants to omit pipe restraints originally intended to mitigate piping loads. Both GL 84-04 and ALS involve phenomena with the potential to affect coolability.

GL 84-04 stated double-ended pipe breaks of RCS main loop piping need not be considered:

> *"the potential for a significant failure of the stainless steel primary piping was low enough that pipe whip or jet impingement devices for any postulated pipe break locations in the main loop piping should not be required."*

"The staff evaluation concludes an acceptable technical basis has been provided so that the asymmetric blowdown loads resulting from double-ended pipe breaks in main coolant loop piping need not be considered as a design basis for the Westinghouse Owner's Group plants."

The objective of this ALS TR is to provide suitable justification to make an equivalent statement regarding dispersal of HBU fuel during an LB-LOCA:

> The evaluation concludes an acceptable technical basis has been provided so that the potential effects of FFRD resulting from large pipe breaks in RCS main loop are not credible and need not be evaluated for HBU fuel in PWR plants within the scope of the analyses.

Table 4-1 provides a summary comparison of how LBB was applied to USI A-2 and how it is proposed to be applied in this ALS TR.

4.1.2 Expansion and Limitation of Use of LBB

Resolution of USI A-2 in 1984 set the stage for the modifications of GDC-4 which have occurred since to allow the use of LBB for excluding from the design basis dynamic effects of postulated pipe ruptures. While LBB was originally limited to main loop piping in PWRs, its application was extended to qualified high energy piping in PWRs.

The 1986 rulemaking [35] introduced an acknowledged contradiction in treatment of LBB, as noted in its supporting justification [emphasis added]:

> *"…acknowledges…an inconsistency into the design basis by excluding only the dynamic effects of postulated double-ended pipe ruptures in PWR primary coolant loops while retaining this postulated accident for emergency core cooling systems, containments and environmental qualification. <u>The present view is that insufficient technical information is available for applying leak-before-break technology to other aspects of facility design</u>. Further studies must be conducted to develop suitable replacement criteria for the PWR primary coolant loop doubled-ended pipe rupture if this accident is no longer required for containment design, emergency core cooling or environmental qualification."*

At that time, the supporting methodology of fracture mechanics was still relatively new for nuclear plants and was limited to a deterministic approach that was highly dependent on state of knowledge. Hence, the limitation was imposed on broader use of LBB. During the intervening 40 years, the nuclear industry and the NRC have worked to improve LBB methodology. As shown in Table 4-2, the NRC has gradually accepted wider adoption of LBB to specific scenarios. The development of probabilistic fracture mechanics (PFM), and xLPR in particular, has now reached the point of being an accepted methodology based on extensive development and qualification. The concern stated in the 1986 rulemaking regarding "insufficient technical information" should no longer be a constraint upon application of LBB.

Table 4-1. Application of LBB for USI A-2 vs. for ALS

	USI A-2	ALS	Comment
Year	1984	Now	
Use of LBB	Exclude asymmetric DEGB to permit elimination of pipe whip restraints. [33]	Confirm extremely low likelihood of failure of RCS main loop piping to permit fuel to operate to higher burnup.	With 40 years of experience since USI A-2 and better assessment tools such as xLPR, the confidence in relying on LBB is higher.
Plant applicability	Initially: D.C. Cook 1-2, R.E. Ginna, San Onofre 1, H.B. Robinson 2, Surry 1-2, Zion 1-2, Point Beach 1-2, Haddam Neck, Turkey Point 3-4, Yankee, Fort Calhoun (Combustion Engineering (CE) NSSS). [34]	Initially: Westinghouse plants seeking to raise their burnup limit; with criteria to be met specified in ALS Appendix A. Subsequently: Likely extend to CE plants and Framatome fueled plants.	Same general population. Initial USI A-2 resolution extensible to other plants that could show they meet same criteria. This TR includes criteria for applicability of ALS.
Licensing pathway	Request exemption. However, once GDC 4 revised, no exemption needed.	Submit License Amendment Request (LAR) invoking ALS TR.	Both based on referencing a generic analysis.
GDC 4	Revised to allow using LBB and deterministic fracture mechanics. [18].	Applied to RCS main loop piping approved to use LBB per GDC 4.	ALS uses PFM code xLPR to confirm extremely low likelihood and determine time available for operator response
Leak detectability	GL 84-04 established criteria for means and sensitivity of leak detection that were later incorporated into RG 1.45. [34]	ALS uses RG 1.45 leak detection.	Licensees, PWR Owners' Group (PWROG), Westinghouse, INPO, and NRC have emphasized RCS leak detection/response.
Failure	DEGB of piping anywhere in RCS main loop.	Both piping and non-piping failures addressed.	Non-piping failures are not required to be considered.

As part of closing out USI A-2, in 1986, the NRC issued a revision to GDC 4 adding [18]:

> *"However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping."*

NUREG-0800 section 3.6.3, "Leak-before-Break Evaluation Procedures" provides guidance on performing LBB evaluations in a manner that the Commission has found acceptable.

- GDC 4 allows exclusion of dynamic effects of pipe ruptures.
- LBB may be applied only to high energy, ASME Code Class 1 or 2 piping or equivalent, although use for other high energy piping will be considered.
- Exclusion of dynamic effects is obtained for individual piping systems; LBB may not be applied to "individual welded joints or other discrete locations." Application of LBB requires all potential rupture locations be examined.

The Acceptance Criteria for use of LBB are:

- Components important to safety shall be designed to function under environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. Safety-related components shall be protected against dynamic effects of "pipe rupture, such as missiles, pipe whip, and discharging fluids."
- Probability of pipe rupture shall be shown to be extremely low under conditions consistent with the design basis for the piping. A deterministic evaluation[5] of the piping system that demonstrates sufficient margins against failure, including verified design and fabrication, and an adequate ISI program, can be assumed to satisfy the extremely low probability criterion.

The RCS main loop piping in Westinghouse PWRs has already been approved for application of LBB, which has been used to screen out other phenomena, as described in Table 4-2. This TR proposes to extend the applicability of LBB to exclude the phenomenon of FFRD, much like LBB was used to eliminate the effects of asymmetric blowdown loads from design basis analysis.

Per [36], LBB should not be used to:

- Redefine LOCAs that place requirements on safety systems and structures.
- Change design margins in the primary loop heavy component supports.

---

[5] Note that Westinghouse plant RCS loop piping has already been accepted for consideration of LBB based on deterministic evaluation consistent with SRP Section 3.6.3. The use of probabilistic fracture mechanics (i.e., xLPR) as discussed in this TR is for the purposes of confirming predictions of frequency of LB-LOCA and estimating the time available to detect a through-wall crack before rupture.

In 1989, the NRC commissioners issued a policy statement [32] that they had decided not to undertake to extend applicability of LBB to ECCS or EQ at that time, but kept open the possibility in the future rulemaking

> *"The Commission encourages industry to develop quantitative information that could justify the diversion of resources to the rulemaking effort. Primary attention should be given to establishing an appropriate substitute or replacement for the double-ended pipe rupture used in ECCS and EQ evaluations. The Commission will consider modifying its current ECCS and EQ regulations, when adequate technical justification supports the feasibility and benefit, of the proposed modification. In the interim, the Commission recognizes that situations may arise where justification can be developed by the industry for alternative ECCS and EQ requirements. Such justifications, if accepted by the Commission pursuant to the existing exemption process could allow a limited number of case-by-case modification to ECCS and EQ requirements."*

The "LBB Knowledge Base" [36] summarizes key considerations in implementation of LBB through its publication in mid-2007. It notes that the NRC staff classifies the regulatory positions on application of LBB into five categories:

(C1) Qualification of high energy piping systems for LBB analysis is required. LBB may not be applied to:

1. ASME Code Class 3 piping,
2. Piping individual welded joints or other discrete locations,
3. Piping repaired by overlays,
4. Piping support by masonry walls,
5. Piping susceptible to water hammer, creep, erosion, corrosion, and fatigue.

(C2) Allowable licensing/design basis changes via LBB applications

When LBB is approved for a particular piping system, applicants are to exclude from the design basis only local dynamic effects associated with postulated pipe ruptures in that system in the nuclear power unit. The local dynamic effects are:

- Missiles
- Pipe Whipping
- Pipe Break Reaction Forces
- Discharging Fluids

The permitted plant activities are, in the order of local dynamic effects:

- Remove jet impingement barriers or shields.
- Remove pipe whip restraints.

- Redesign pipe connected components, their supports and their internals, and other related changes.
- Disregard jet impingement forces on adjacent components, decompression waves within the intact portion of the piping system, and dynamic or non-static pressurization in cavities, sub-compartments, and compartments.

(C3) Limitations on applying LBB to containment design, ECCS, and EQ of safety related electrical and mechanical equipment.

LBB may not be applied to containment design, ECCS design, and EQ. Specifically:

- *"For Containments. Global loads and environments associated with postulated pipe ruptures, including pressurization, internal flooding, and elevated temperature.*
- *"For ECCS. Heat removal and mass replacement capability needed because of postulated pipe ruptures.*
- *"For EQ. Pressure, temperature, flooding level, humidity, chemical environment, and radiation resulting from postulated pipe ruptures."*

These limitations stem from the 1989 Policy Statement [32].

(C4) Recent applications requiring interpretation of regulatory positions on LBB.

(C5) Future position changes regarding LBB applications.

4.2 Precedents for Application of LBB

The following sections discuss the relevance of ALS to some specific uses of LBB that were accepted by the NRC and one that was not. These and a few similar LBB uses are summarized in Table 4-2, which includes the following information:

- Application: why LBB was used.
- Year: Year of NRC action.
- NRC Action: Approved or Rejected.
- Description: safety significance.
- Timing of Effects: timing of undesirable effects if an LB-LOCA were to occur.
- Technical Area: principal type of effect if an LB-LOCA were to occur.
- Structures, Systems and Components (SSCs) Affected: the SSCs that could be damaged if an LB-LOCA were to occur.
- DiD: Was defense-in-depth addressed in application of LBB?
- Non-piping: Were non-piping component failures considered in addition to piping?
- Effects of LOCA: Were consequences of damage assessed if an LB-LOCA were to occur?

4.2.1 Broken Baffle-Former Bolts

The Westinghouse TR and associated NRC SER credit LBB to exclude the effects of broken baffle bolts on core cooling during an LB-LOCA [37]. This use of LBB is quite similar to that proposed for FFRD: a large piping rupture at one of the same locations could lead to fuel rod fragmentation. The NRC accepted this use of LBB stating:

> *"As with current analyses of the reactor vessels and internals, LBB exclusions allowed under GDC-4 will be credited when selecting the break location and size."*
>
> *"The break locations considered in the analysis of each bolting configuration will be the two largest lines not excluded or exempted from consideration under GDC-4, one of which will be on the RCS cold leg, the other on the RCS hot leg. Typically, where the main loop piping is excluded, these breaks will be in the three main branch lines, the accumulator line on the cold leg and the pressurizer surge line or RHR line on the hot leg."*

The NRC SER states that the methodology uses the best-estimate WCOBRA-TRAC code for intermediate pipe break size to establish the two-phase loads. The Westinghouse methodology establishes the best-estimate depressurization transient, excluding ruptures of the RCS main loop piping on the basis of LBB.

Table 4-2. Significant Approved and Rejected Applications of LBB

Application of LBB	Year	NRC Action	Description	Timing of Effects	Technical Area	SSCs Affected	DiD	Non-piping	Effects of LOCA
USI A-2	1986	Approved	DEGB loads could alter plant geometry	Blow down	Mechanical	RPV	No	No	No
Pipe Whip/Jet Impingement	1986	Approved	Removal of pipe whip restraints	Blow down	Mechanical	Piping supports	No	No	No
Baffle-former-bolt breakage	1998	Approved	No fuel fragmentation Meet 50.46 Control rod insertability	Blow down	Fuels Thermal Mechanical	Fuel RPV internals	No	No	No
ECCS cross-connect valve[6]	2003 to 2007	Approved	Eliminate pipe whip that could fail both trains of ECCS	Post blow down	Mechanical	ECCS: low pressure injection	No	No	No
Control rod insertion	2008	Approved	Exclude LB-LOCA blowdown forces	Blow down	Mechanical Nuclear	Control rods	No	No	No
Next Generation Fuel (NGF) structural	2008	Approved	No fuel fragmentation Meet 50.46 Control rod insertability	Blow down	Fuels Thermal Mechanical	Fuel RPV internals	No	No	No
GSI-191 sump blockage	2010	Rejected	Eliminate debris generated by LB-LOCA	Post blow down	Many	ECCS: recirculation	Yes	No	No
ALS	2024	TBD	HBU fuel dispersal precluded by LBB for LB-LOCAs and by no clad burst for smaller LOCAs	Blow down and post blow down	Fuels Thermal Mechanical	Fuel	Yes	Yes	No

[6] Involved resolution of Differing Professional Opinion that found application of LBB was suitable [38].

4.2.2 Westinghouse 17x17 NGF Fuel

Westinghouse invokes LBB in the core reference report for a new fuel design [39]. In addressing the acceptance criteria of section 4.2 of NUREG-800, LBB is used to exclude consideration of LB-LOCAs in piping approved for application of LBB, noting [emphasis added]:

> *"Currently, all Westinghouse designed US PWR primary coolant main loop piping has been excluded from consideration for dynamic effects associated with postulated pipe rupture under Reference 61 [GL 84-04 [34]] or subsequent LBB analyses. As a result, <u>all current fuel qualification analyses are performed on the basis of postulated rupture of branch lines connected to the primary coolant loop</u>."*

Elsewhere in [39], Westinghouse discusses exclusion of RCS main loop piping breaks [emphasis added]:

> *"The primary success criteria for the baffle bolting program are the same as those documented in SRP Section 4.2 discussed above: i.e., no fuel fragmentation, 10 CFR 50.46 criteria continue to be met, and control rod insertability is maintained. These analyses were also based on LBB exclusion of the main coolant loop piping."*
>
> *"In 1999, MULTIFLEX 3.0 was used again in conjunction with control rod insertability analyses performed … which were reviewed and approved by the NRC, and included acceptance of the use of MULTIFLEX 3.0. … provided results for both main coolant loop piping breaks and branch line breaks, only the branch line breaks not covered by LBB are considered in the licensing basis. As a result of this analysis, … could credit control rod insertion for addressing boron dilution issues post-LOCA on the basis of branch line break LOCA loads."*

This precedent is considered the closest match to ALS: LBB is applied to exclude evaluation of LB-LOCAs in RCS main loop piping to simplify analysis of fuel proposed for use in a PWR. In [39], all fuel is excluded from consideration of hydraulic fragmentation, whereas ALS only excludes HBU fuel from need to evaluate FFRD.

4.2.3 GSI-191

The NRC did not accept industry proposals to use LBB to close GSI-191, which concerned the possibility that the jet issuing from a broken high-energy pipe could erode insulation and coatings from nearby pipes and structures, with the resultant debris being deposited in the ECCS recirculation sump. The nature and quantity of the debris have the potential to block recirculation flow, leading to fuel overheating.

NRC reasons for not agreeing to apply LBB to GSI-191 were discussed in [40], which included those listed in Table 4-3.

Table 4-3. NRC Staff Reasons for Not Allowing Use of LBB for GSI-191

LBB for GSI-191	Relevance to LBB for ALS
Applying GDC 4 beyond removal of pipe whip constraints and missile barriers is inappropriate without a deliberate rulemaking process that permits further staff evaluation while also considering stakeholder input.	The NRC issued for public comment the IE rulemaking basis document [10], which outlined alternatives for addressing FFRD, one of which is an expanded version of the ALS. Nuclear industry comments were provided in [11].
Applying LBB to avoid modifying sumps or piping insulation is a reduction in defense-in-depth.	Evaluation using xLPR demonstrates that there is ample time for operators to detect leakage and respond, providing DiD (see section 2.3).
The NRC did not intend GDC 4 to be used as an equivalent alternative to the ECCS regulations.	ECCS requirements would be unchanged. The ALS provides the basis for excluding FFRD in core cooling analysis of LB-LOCAs.
Adoption of LBB would reduce regulatory requirements with no perceived safety benefit.	ALS safety benefits are identified in section 1.2.
A regulatory inconsistency with risk-informed ECCS regulation 50.46a could be created. A Statement of Considerations is needed to minimize the chance of unintended consequences	50.46a was discontinued.
Most PWR owners are still addressing PWSCC of Alloy 82/182 dissimilar metal welds in large piping approved for LBB.	PWSCC of dissimilar metal welds has been resolved, and there are no known degradation mechanisms with the potential to cause an LB-LOCA.

4.3 Summary of Use of LBB for ALS

Since its initial acceptance by the NRC in 1986, LBB has been authorized to be applied to the largest diameter piping in PWR plants, the RCS main loop piping.[7] The NRC criteria for applying LBB has evolved over the time. By NRC policy, LBB should not be used for modification of ECCS, containment design, and for relaxation of environmental qualification parameters without Commission review and approval.

Applying LBB in support of ALS is justified because it:

- Provides significant safety benefits (see section 1.2).
- Implements the principles of good regulation (see section 1.3).

[7] Some plants have evaluated and been authorized to apply LBB to piping smaller than that in the RCS main loop. However, for ALS, LBB is only credited for main loop piping.

- Considers risk-insights consistent with Commission policy to modernize the agency.
- Is based on a sound understanding of plant design, operations, aging, etc.
- Is capable of being accomplished by plant operating and maintenance staff using existing equipment and procedures.
- Is consistent with previously accepted precedents.
- Considers operating experience.
- Is consistent with the industry response to the NRC Increased Enrichment Rulemaking Regulatory Basis [10].
- Continues to provide reasonable assurance of adequate protection of the public's health and safety.

The NRC has approved a number of specific cases where LBB has allowed exclusion of RCS main loop piping LB-LOCAs and eliminated the need for demonstrating that the ECCS acceptance criteria of 10 CFR 50.46 and NUREG-0800 section 4.2 are met. ALS is consistent with these LBB precedents, but provides additional technical justification by:

- Performing a probabilistic fracture mechanics analysis of the possible piping failures (Sections 3 and 5)
- Addressing defense-in-depth (section 2.3), such as evaluating non-piping failures equivalent to an LB-LOCA (Section 6)

4.4 RCS Leak Detection and Response

Crediting LBB for the purpose of ALS can consider more than the plant operators on watch because of the long time available to detect leakage. While those on duty are responsible for monitoring the plant and responding to indications of abnormal and annunciator alarms, many other groups have the ability to observe indications of RCPB leakage: operators coming on watch question changes from their last shift, plant management sets expectations and reviews plant status, non-licensed staff can note signs of leakage during maintenance and upkeep, radiological monitoring personnel may note discrepancies in local surveys, and engineering reviews plant status reports and evaluates discrepant conditions. Even the site warehouse and procurement staff would have an opportunity to note increased soluble boron usage associated with additional makeup to the plant.

With so many people attentive to the integrity of the RCPB, signs of RCS leakage at a rate far below the 1 gpm (3.8 L/min) assumed in the xLPR evaluation will not be missed over a period of days, much less months.

4.4.1 Integrity of Reactor Coolant Pressure Boundary

NRC regulations and guidance have always emphasized the importance of maintaining the integrity of the RCPB and minimizing leakage. The two most relevant GDC from 10 CFR 50 Appendix A are:

> "*Criterion 14—Reactor coolant pressure boundary*. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture."
>
> *"Criterion 30—Quality of reactor coolant pressure boundary*. Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage."

Leakage past non-structural barriers (e.g., pump and valve seals, flanged joints) needs to be minimal to limit spread of radioactivity, avoid deposits of boric acid that can cause corrosion, and facilitate detection of abnormal leakage. As directed in RG 1.45 [41], leakage through expected pathways is collected and quantified so that it does not obscure abnormal leaks.

U.S. PWR TSs prohibit continued operation with known RCPB leakage (see section 4.4.3). RCPB leakage is from a non-isolable fault in the material comprising a portion of the RCS such as a pipe wall, component shell, weld, or flanged closure. Even though excessive leakage through valve packing, an unseated relief valve, or pump packing requires a plant shutdown, these leak pathways cannot progress to an LB-LOCA.

4.4.2 Regulatory Guidance for Leakage Monitoring

The SRP [15] section 5.2.5 "Reactor Coolant Pressure Boundary Leakage Detection," details NRC expectations for acceptable leakage detection. The acceptance criterion for GDC 30 is meeting the guidelines of RG 1.45 [41], which provides guidance on monitoring and responding to RCS leakage. Operating PWRs were built and originally licensed to Revision 0 of RG 1.45, which was subsequently updated in 2008, so many operating plants still implement Revision 0. The version of RG 1.45 in a plant licensing basis does not matter for ALS, provided leakage detection criteria in Appendix A are met.

- RG 1.45 stipulates the use of leakage detection systems with a response time of no greater than one hour for a leak rate of 1 gpm (3.8 L/min) (excluding transport time per revision 1).
- RG 1.45 R0 specifies at least three separate detection methods with two of them being sump level/flow monitoring and airborne particulate radioactivity and with the third being one of air cooler condensate flow or airborne gaseous activity. However, improvements in fuel integrity since the initial issue of Revision 0 in 1973 have made the gaseous radiation monitors less effective of detecting leakage within a reasonable period of time. Although a delay of many hours would not meet RG 1.45 criteria, it would still be small compared to

the time available before a rupture would occur (i.e., 19 months is over 13,000 hours). Revision 1 acknowledges that gaseous radioactivity monitoring may no longer be useful and suggests alternatives. It specifies that there be at least two independent and diverse detection methods, especially the three noted at the start of this paragraph.

- Revision 1 identifies that unidentified and identified leakage trends should be periodically analyzed.
- Indications and alarms for each leak parameter monitored are to be provided in the main control room.
- Leak monitoring is not only diverse, but leakage indications also are visible to multiple operators and other plant personnel, so detection of RCS leakage is not dependent on just one or a few individuals.

Per the PWR TS [43]. RCS leakage is categorized as either:

- <u>Identified leakage</u>:
  - Leakage into closed systems, such as pump seal or valve packing that is captured, measured, and directed to a collection location
  - Leakage to the atmosphere of containment from sources that are both specifically located and known either not to interfere with the operation of unidentified leakage monitoring systems or not to be from a flaw in the RCPB
- <u>Unidentified leakage</u>, which is everything else.

If leakage is detected but cannot be ascribed to an allowed source (e.g., pump seals), it is designated as unidentified leakage, which requires plant shutdown if the rate exceeds 1 gpm. If a point of through-wall RCPB leakage is found, continued operation is not permitted until it is satisfactorily repaired. Upon finding the source of unidentified leakage, it may be treated as identified if allowed.

RG 1.45 Revision 1 specifies consideration of other means to detect and monitor leakage, even if not sufficient to meet the quantitative criteria, such as:

- Airborne gaseous radioactivity
- Humidity of the containment
- Temperature of the containment
- Pressure of the containment
- Acoustic emission
- Video surveillance

The multiple and diverse detection methods are summarized visually in Figure 4-1. The NRC Inspection Manual, Chapter (IMC), IMC 2515, Appendix D [42] requires that licensees monitor RCS leakage and respond appropriately. Industry implementation of these NRC monitoring criteria is discussed in below.

Monitored over a period of days or weeks, some monitoring methods are sensitive down to a few hundredths of a gallon per minute.

The allowable time during which leakage detection instruments may be out of service should be specified, even for those not required by the TS. At least one of the TS-required leakage monitoring systems should function following any seismic event not requiring plant shutdown. Additionally, leakage monitoring systems should have provisions for calibration and testing during plant operation to ensure proper operation. Finally, RG 1.45 identifies that alarms from leakage monitoring systems are to be provided in the main control room.

The multiplicity and variety of leak detection methods provides diversity that helps ensure that RCS leakage is not overlooked because of the characteristics of a particular leak or the mindset of a few personnel.

4.4.3 Technical Specification LCO and Surveillance for RCS Leakage

The Westinghouse standard TS leakage criteria and associated required actions are presented in Table 4-4 and Table 4-5. The TS LCOs and Surveillance Requirements (SRs) for each plant with a few exceptions follow the standard plant TS. Differences include an end state of Mode 4, vice Mode 5, within 30 hours (vice 36 hours).

LCO 3.4.13 requires shutting down and transition to Mode 5 (Mode 4 for some plants) if any RCPB leakage exists, if unidentified leakage exceeds 1 gpm (3.8 L/min), or if identified leakage exceeds 10 gpm (38 L/min). As noted in the previous section, the detectable leak rate for many plants is as low as 0.05 gpm (0.19 L/min), a factor of 20 margin to the 1 gpm (3.8 L/min) value requiring shutdown of the plant.

The periodicity of SR 3.4.13.1 is every 72 hours or in accordance with the Surveillance Frequency Control Program, if implemented by a plant. The RCS Leakage SR timing becomes every 24 hours if one of the monitoring instruments is out of service, as described in the LCO for leakage monitoring (LCO 3.4.15 in [43]). In actual practice, plants have multiple RCPB leak detection methods that provide updated information varying from continuously to weekly. For example, many plant process computers are programmed with continuous RCPB leak rate calculation. As this is an operator aid, it is not required to be available for continued plant operation.

Further instructions regarding actions to take in response to certain indications are detailed in plant-specific procedures.

Table 4-4. Westinghouse Standard Technical Specifications LCO for RCS Leakage [43]

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13	RCS operational LEAKAGE shall be limited to: a. No pressure boundary LEAKAGE, b. 1 gpm unidentified LEAKAGE, c. 10 gpm identified LEAKAGE, and d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. AND B.2 Be in Mode 5.	6 hours 36 hours

Table 4-5. Westinghouse Standard Technical Specifications RCS Leakage Surveillance [43]

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	------------------------------NOTES----------------------------- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. --- Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.	[72 hours OR In accordance with the Surveillance Frequency Control Program]

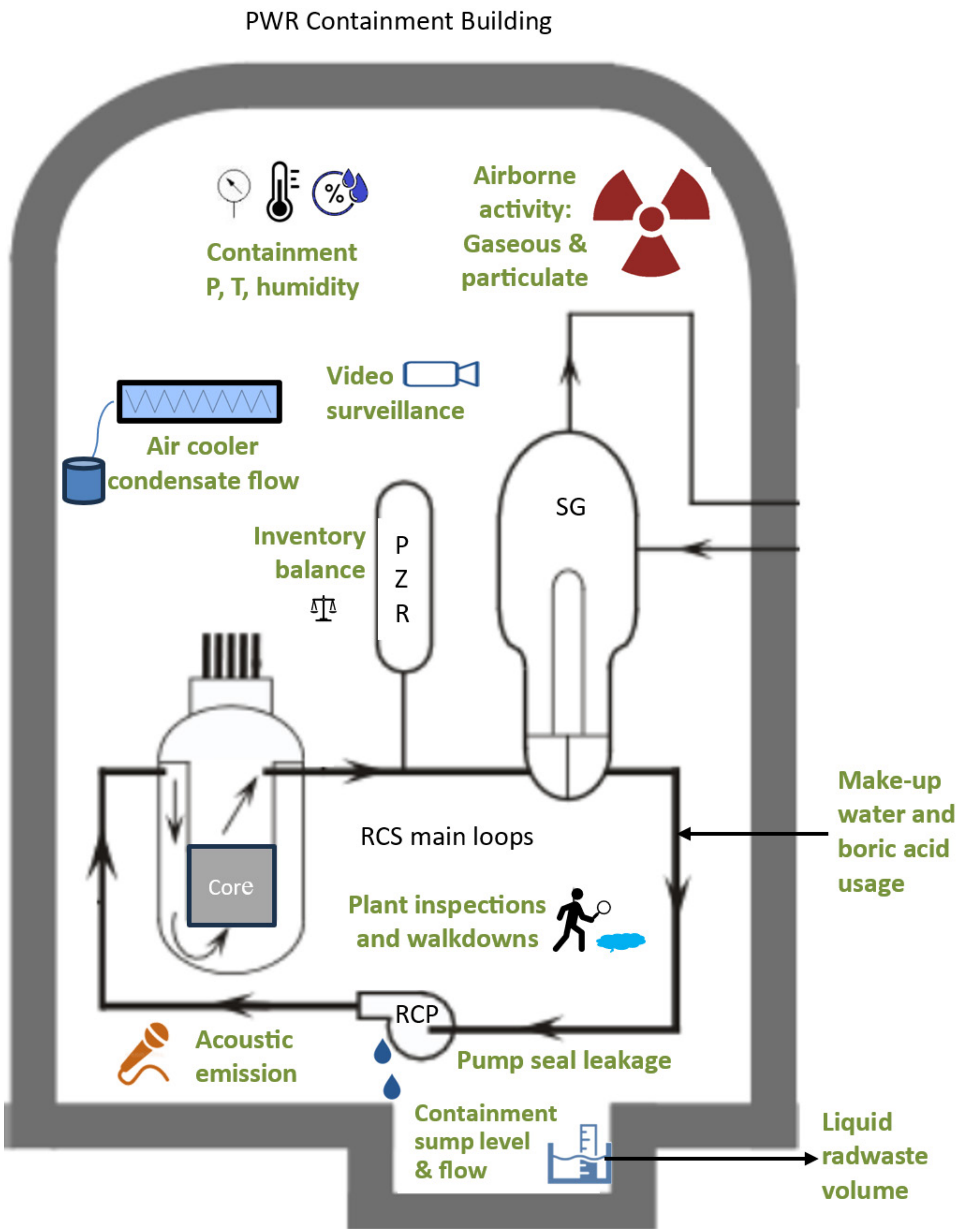

Figure 4-1. Multiple, Diverse Leak Detection Methods

4.4.4 Diversity in Leak Identification

As backup to continuous monitoring systems, RG 1.45 states that leak rate trends should be periodically analyzed. If leak rate increases noticeably from the baseline leakage rate, the plant should evaluate the safety significance of the leak and determine the rate of increase to verify that plant actions can be taken before the plant exceeds TS limits. This requires assessment before reaching TS RCS leak limits, which leads to plant operating staff performing leak determination activities proactively.

To ensure that RCS leakage is not overlooked because assessment criteria are too narrow, Westinghouse developed for the PWROG a set of three tiers of criteria for identifying possible RCS leakage [44]. Each tier addresses successively higher rates for Modes 1 through 4. Their purpose is to provide standard action levels and response guidelines consistent with the intent of NRC Inspection Manual, IMC 2515, Appendix D, Attachment 1 [42]. These tiers are summarized in Table 4-6. The lowest tier quantities were selected to provide detection of very small leaks that are well below TS limits. Although details for some plants may differ slightly, each operating PWR has a leak detection process that has been reviewed and accepted by the NRC.

In addition to detection by personnel, the instrumentation and control system will alert operators by actuation of annunciator alarms should a monitored parameter be exceeded. For leak rates exceeding the total capacity of charging pumps, an Emergency Operating Procedure (EOP) would be initiated.

The PWROG guidelines [44] use a statistical approach by establishing values for RCS unidentified leakage that are updated on at least a quarterly basis. Criteria use a baseline mean and standard deviation to determine limits and necessary actions. The mean is calculated by averaging valid leak rate values for a calendar quarter. In addition to comparison to the action criteria, the daily unidentified leak rate is visually screened for discrepancies, and the quarterly data set is checked for normality. Although details are left to the licensee, the guidelines recommend that the baseline should be updated:

- Quarterly using data from the just completed quarter
- Upon return to operation following a refueling outage, or
- After performing maintenance activities on the RCS or connected systems

A fixed unidentified leak rate limit is included to guard against the possibility that a slowly increasing small leak would go undetected by gradually raising the baseline mean.

Table 4-6. PWROG Standard RCS Leakage Action Levels and Response Guidelines [44]

Tier	Standard Action Levels	Standard Criteria for Assessing Leak Rate	Response for Exceeding an Action Level
1	Unidentified leak rate	a) One 7-day rolling average daily unidentified rate > 0.1 gpm (0.38 L/min) b) Nine consecutive daily unidentified leak rates > baseline mean	1) Perform the following a. Confirm indication. b. Evaluate trend of affected parameter. c. Evaluate trend of associated Tier One parameters. d. Run confirmatory leak rate calculation. e. Check for abnormal trends for other leak indicators. 2) If the indication of leakage is confirmed a. Increase monitoring of leakage indicators. b. Initiate a Condition Report. 3) Notify cognizant system engineer(s) to obtain input/help.
2	Deviation from baseline mean	a) Two consecutive daily unidentified rates > 0.15 gpm (0.58 L/min) b) Two of 3 daily unidentified rates > mean +2σ c) 30-day total unidentified leakage > 5,000 gal.(19,400 L) (0.116 gpm (0.45 L/min) average over 30 days)	1) Perform Tier One response 2) Commence a leak investigation: a. Review recent plant evolutions to determine any "suspect" source(s). b. Evaluate changes in other leak detection indications. c. Initiate outside containment walk-downs of various portions of potentially affected systems. 3) Identify the source of the increase in leakage: a. Check any components or flow paths recently changed or placed in service, shutdown, vented, drained, filled, etc. b. Check any maintenance activity that may have resulted in increasing leakage. 4) Check any filters recently alternated or changed, etc.

Table 4-6 (continued). PWROG Standard RCS Leakage Action Levels and Response Guidelines

Tier	Standard Action Levels	Standard Criteria for Assessing Leak Rate	Response for Exceeding an Action Level
3	Total unidentified leak rate	a) One daily unidentified rate > 0.3 gpm (1.14 L/min) or > mean +2σ b) Long term (operating cycle) total unidentified leakage > 50,000 gal. (190,000 L)	1) Perform Tier One and Tier Two responses 2) If increased leak rate is indicated inside containment, then: a. Begin planning for a containment entry while carrying out other actions. b. Obtain a containment sump sample (during pump out) and analyze for activity, a larger than expected boric acid concentration and other unexpected chemicals. c. Evaluate other systems for indications of leakage (Component Cooling Water, etc.) d. Obtain a containment atmosphere sample for indications of RCS leakage. 3) Identify source of the leak. 4) Quantify the leakage. 5) Initiate plan to correct the leak. 6) Monitor containment airborne radiation levels as well as area radiation monitors. Sample containment atmosphere for indications of RCS leakage. 7) Monitor other containment parameters (temperature, pressure, humidity, etc.). 8) Implement portions of RCS leak investigation procedure (plant specific) to identify potential leak sources. 9) If the leak source is found and isolated or stopped, re-perform RCS leak rate calculation.

The PWROG criteria do not consider detection by atmospheric radiation monitors because of the differences among plants in equipment, RCS radioactivity, fuel cycle, etc.

Because of the variation in density of water with temperature, RCS leak rate is normalized to the nearest 0.01 gpm (0.04 L/min) at 70˚F (21 C) and 14.7 psia (10.1 kPa) (i.e., density of 62.30 lbm/ft3 (997.97 kg/m3)). A typical standard deviation is about 0.055 gpm (0.208 L/min) [44].

Each plant should have an RCS leakage monitoring program that details the plant parameters monitored, expected values, equivalent leak rates for those measurements using other physical parameters (e.g., containment atmosphere radiological activity, containment sump pump frequency). The program should assess trends and make recommendations to plant management, periodically (or for emergent changes) present results at plant staff meetings, and provide input to posted plant metrics.

The overall goal of the RCS Leakage Monitoring Program is to provide early leak detection and minimize the consequences associated with RCS leakage by:

- Maintaining leakage of reactor coolant at the lowest attainable values.
- Providing assurance that the plant will not be operated with RCPB leakage.
- Monitoring RCS leakage trends for earliest possible detection and evaluation of new or increasing RCS leakage.
- Prompt notification of management personnel of new or increasing RCS leakage, even if the leakage is well within TS thresholds for action.

4.4.5 Leak Investigation

If there appears to be a leak from the inventory balance or other evidence, operators determine if unidentified leakage exceeds 1 gpm (3.8 L/min) or if any RCPB leakage is indicated. If either is true, then the operators are required by the RCS Leakage LCO to verify there is no RCPB leakage and reduce the unidentified leakage within four hours. If this cannot be done, then the plant must be in Mode 3 within six hours and be in Mode 5 (cold shutdown) or Mode 4 within 36 hours.

4.4.6 Detection by Other Means

If operations staff and management cannot confirm no RCPB leakage or that the leak rate is stable, then they will shut down the plant to ensure the LCO is not exceeded. If operation continues, other indications will provide reinforcement of the need for action, backing up the TS surveillance requirement. “Other means” includes tripping an annunciator alarm or a diverse indication being exceeded, such as high sump level or high airborne/radiation.

Also, there are indirect means to cue plant staff to a problem ranging from high liquid waste processing volumes, exceeding normal usage of boric acid or RCS chemical additives, abnormal make-up flow rates, etc. Note that at 1 gpm (3.8 L/min), the amount of water leaked is equivalent to one large tank truck (about 10,000 gallons (37,850 L)) in one week, which is an amount of misplaced water that would be almost impossible to overlook.

4.4.7 Response to Indication of Abnormal Leakage – Human Reliability

The time available for response and the number of personnel who would have visibility of multiple indications of an RCS leak make overlooking the leakage or not responding to it implausible. As this evaluation is not a PRA, a quantitative human reliability analysis is not required. However, an evaluation of factors that could adversely affect operator performance was carried out. Table 4-7 presents the results of this qualitative assessment. In general, the ALS situation results in few factors that would lead to degraded human performance, of which none are considered to result in reactor operation continuing with a leak to the point where a rupture occurs.

In addition, Table 4-8 identifies equipment issues that could adversely affect operator action. Again, the long time and multiple detection methods available would make the impact of these errors temporary and not invalidate the reliance on detection, diagnosis, and response to indication of RCS leakage.

Table 4-7. Factors Potentially Affecting Operator Response

Factor	Effect	ALS
Time urgency	Time pressure increases likelihood of error	With an xLPR-calculated time of 19 months from 1 gpm (3.8 L/min) leakage to possibility of rupture [3], ample time is available to recognize, characterize, and respond to the leak and to recover from poor attentiveness or misdiagnosis
Lack of urgency	Importance or attention to activity may be reduced	RCS leak rate is a PI, giving it high visibility
Environmental conditions (e.g., noise, heat)	Affects cognition and operator attention to details	Control room conditions apply (except for walkdowns)
Poor safety culture, inappropriate mindset, and groupthink	Bias of an individual or group of operators causes them to dismiss indications of leakage	Because of ample time to respond, operating staff, plant management, engineering, INPO, and NRC personnel will have visibility of leak indications. Questions from outsiders on reported leak trends would cue need for re-evaluation.
Fitness for duty	Personnel performance degraded	Involves multiple operator crews over extended time period, so not dependent on one or two operators
Infrequently used and/or complex procedures	Operators may be confused or incorrectly interpret procedures	Leak detection is performed in some form daily and using multiple means. Multiple opportunities for many operators to recover from errors in performing procedures.
Experience and training	Unfamiliarity may cause operators to respond late or incorrectly	RCPB importance and high importance of preventing and detecting and responding to RCS leak is a major focus

Table 4-8. Equipment Issues Potentially Affecting Operator Response

Factor	Effect	ALS
Common cause failure of leak detection	Design, maintenance, miscalibration errors, etc. could prevent or delay detection of leakage	Multiple, diverse means are available to detect RCS leakage (see Figure 4-1). Procedures require checking multiple indications.
Failure of mitigation systems	Response and recovery actions must be performed with less familiar equipment and procedures	No safeguards system (e.g., ECCS) functionality is required

Having determined that RCS unidentified or RCPB leakage exists, the operators must follow the procedure to shut down and cool down. Once power production has ceased, decay heat drops off rapidly and will no longer be capable of causing clad burst. In addition, as the plant is cooled down and depressurized (i.e., be in Mode 5 or 4 within 36 hours), the possibility of an LB-LOCA or of FFRD drop rapidly and become negligible. With the plant shut down to investigate the abnormal leakage, resumption of operation without finding and correcting the leakage is considered highly unlikely, although there have been a few events where operating staff misdiagnosed the reason for abnormal leakage trends and returned the reactor to operation only to find the abnormal leakage persisted, requiring another shut down.

For ALS, high likelihood of recovery from human error comes down to ample time, operator backup, high priority for operator attention, emphasis in operator training, redundant indications, and simplicity of actions. There is clear procedural guidance on which operators frequently train. Even should operators be slow to recognize an abnormal increase in leakage, there are several annunciator alarms on particular, monitored parameters that provide direct response procedures to actions to taken in response to the alarm.

Whereas operator actions in PRA scenarios may need to occur in a short period of time, under stressful conditions, in accordance with infrequently used procedures, and/or with little opportunity for backup or recovery by other operators, ALS has none of those factors. As shown by xLPR results, there is ample time before a detectable leak could reach the point of an LB-LOCA for operators to detect, diagnose, shut down, investigate, and ensure the plant is in a safe, stable condition.

5 PIPING RUPTURES

Assessment of vulnerability to LOCAs that could cause FFRD requires evaluation of various types of failures in different locations in the reactor coolant and connected systems.

5.1 Extremely Low Likelihood of Occurrence

As discussed previously in this report, acknowledgement of extremely low likelihood of abrupt failure of large piping and components in U.S. PWR plants has been the basis for excluding certain LB-LOCA consequences from the licensing basis of various plants. In section 2.2.3, quantification of “extremely low” was shown to generally correspond to 10^{-6} per year.

5.2 Expert Elicitation Estimates

The objective of the expert elicitation documented in NUREG-1829 [3] was to establish improved frequency values for various size LOCAs. The results are summarized in Table 1 of Volume 2 of the NUREG, the PWR section of which is shown below with the addition of the columns for LOCA Size Category and LOCA flow rate in L/min.[8] LOCA event frequencies were developed for six rupture size categories with the smallest having an effective break size from ½ to 1⅝-inch (12.7 to 9.53 mm) diameter double-ended, which corresponds to flow rates ranging from 100 to 1,500 gpm (5,678 L). Note the following points:

- Break frequencies of smaller size categories include all larger breaks. The table frequencies are cumulative: to find the frequency of just LOCA Size Category 3 breaks, the Category 4 frequency should be subtracted from the Category 3 entry in the table.
- The frequency values in the table are total for a plant. In other words, the number of welds of a given size were considered, and piping and non-piping frequencies were merged.
- For piping, the assumption is usually a DEGB for a given nominal pipe size (NPS).
- Failure frequency values at 5%, 50% (mean), median, and 95% confidence were given for both PWRs and BWRs and for a fleet average age of 25 years and 40 years. Based on the current age of Westinghouse plants, the 40-year fleet average operation values are considered most appropriate. Mean values are used for comparing probabilities. Therefore, only the mean 40-year values are discussed in this TR, as they are most relevant for comparison to PFM results.
- The experts’ estimates take credit for ISI in accordance with the ASME Code and for leak detection by operating staff.

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[8] The first column also identifies the LOCA size in terms of the historical groupings of large break (LB), medium break (MB) (or intermediate break (IB)), and small break (SB) LOCAs. These historical labels are not precisely defined and will have different break size thresholds in different references.

- Although NRC regulations define LOCAs as associated with piping ruptures, NUREG-1829 uses the term LOCA for leak and ruptures associated with failures of components. However, lifting of relief/safety valves and active failures are not included.
- A combination of operating experience and fracture mechanics is used to demonstrate that the conditional probability of a rupture, given a leak, decreases as pipe diameter increases. The size of detectable cracks and leaks remains relatively constant regardless of pipe size. In other words, a similar size detectable leak (i.e., crack) in a large diameter pipe will have more safety margin because it represents a smaller fraction of the pipe pressure boundary (i.e., circumference) that needs to fail.
- Results were adjusted (increased frequency) for expert optimism bias.

The frequency values for a given LOCA size are cumulative: rupture frequency of a size group includes the possibility of larger ruptures (e.g., a Category 4 size LOCA with a rupture size of 7 to 14-inch (17.8 to 35.6 cm) diameter and corresponding flow rate range of 25,000 to 100,000 gpm (94,635 to 378,540 L/min) has frequency of 3.6×10^{-6} per CY, which also includes the occurrence rate of Category 5 and 6 LOCAs of 1.4×10^{-6} per CY). Note that the flow rates given in the table assume a double-ended rupture, but that the frequencies are based on pipe size and, therefore, the same for double and single-ended.

As previously discussed, below a flow rate equivalent to a surge line or accumulator line rupture (single-ended), ECCS analysis using design basis analysis methods has demonstrated that rod clad burst does not occur (Section 3).

Of course, NUREG-1829 is a generic assessment. The number of RCS loops determines the number of welds in a plant. With more RCS loops in a plant, a higher frequency would be expected if all other conditions were the same. Similarly, the six LOCA size categories do not correspond to every pipe size used in the plant. Pipe diameter varies slightly among Westinghouse plants, with the most common RCS main loop inside diameters being 27.5 inches (70 cm) for inlet piping, 29 inches (73.7 cm) for outlet piping, 31 inches (78.7 cm) for the reactor coolant pump (RCP) suction piping [26].

For a power curve fit to the 40-year mean data in Table 5-1, a 27-inch ID pipe yields a maximum mean frequency of about 9×10^{-8} per CY and includes the possibility of larger pipe breaks. This value is below most of the various thresholds identified for screening initiating events as part of either a deterministic or probabilistic approach (see section 2.2.3). As discussed in the remainder of this section, such a rupture should never occur because of the very long period of detectable leakage allowing plant staff to detect, diagnose, and respond.

Table 5-1. NUREG-1829 PWR LOCA Frequencies by Size Category

LOCA Size Category	LOCA Flow rate (gal/min)	LOCA Flow rate (L/min)	Eff. Break Size (inch)	Current-day Estimate (per CY) (25 yr fleet average operation) 5th Per.	Median	Mean	95th Per.	End-of-Plant-License Estimate (per CY) (40 yr fleet average operation) 5th Per.	Median	Mean	95th Per.
1 (SB)	>100	378	½	6.9E-04	3.9E-03	7.3E-03	2.3E-02	4.0E-04	2.6E-03	5.2E-03	1.8E-02
2 (SB)	>1,500	>5678	1 5/8	7.6E-06	1.4E-04	6.4E-04	2.4E-03	8.3E-06	1.6E-04	7.8E-04	2.9E-03
3 (IB)	>5,000	>28K	3	2.1E-07	3.4E-06	1.6E-05	6.1E-05	4.8E-07	7.6E-06	3.6E-05	1.4E-04
4 (LB)	>25K	>94K	7	1.4E-08	3.1E-07	1.6E-06	6.1E-06	2.8E-08	6.6E-07	3.6E-06	1.4E-05
5 (LB)	>100K	>378K	14	4.1E-10	1.2E-08	2.0E-07	5.8E-07	1.0E-09	2.8E-08	4.8E-07	1.4E-06
6 (LB)	>500K	>1.89M	31	3.5E-11	1.2E-09	2.9E-08	8.1E-08	8.7E-11	2.9E-09	7.5E-08	2.1E-07

5.3 xLPR Assessment

The xLPR assessment of PWR RCS main loop piping [4] determined that only the analyses that modeled crack growth due to PWSCC resulted in ruptures, while fatigue alone did not lead to leaks or ruptures. Therefore, the discussion below is focused on the more limiting dissimilar-metal welds that are susceptible to PWSCC.

Results from xLPR did not predict any piping ruptures over an 80-year plant life, except the few cases that model factors that are not representative of actual plant conditions and operations. The 80-year rupture frequency for cases that did experience rupture with ISI and LRD are on a similar or lower order of magnitude than NUREG-1829 results at 40 years. Notably, the cases exhibiting ruptures while crediting ISI and LRD are sensitivity cases modeling scenarios not representative of current plant conditions and operations. Although xLPR analysis results are on a per-weld basis, plant level frequency of rupture with 1 gpm LRD can also be taken as zero because no ruptures with ISI and LRD occurred through 80 years in realistic cases.

The time between detectable leakage and LOCA was also characterized using xLPR. The assessment of time between detectable leakage and the occurrence of a LOCA event (which occurs as a precursor to rupture) is used to demonstrate that sufficient time exists to allow for reactor shutdown and the reduction of decay heat generation, and thereby likely preclude progression to a LOCA event. For the reactor vessel inlet nozzle, the reactor coolant pump nozzle, and steam generator nozzles, LB-LOCA was either not observed to occur or was determined to be highly unlikely. For the reactor vessel outlet nozzle, the xLPR results showed that LB-LOCA does not occur when crediting ISI and LRD, and the distribution of times between detectable leakage and LB-LOCA can be characterized by a lower bound 95/95 one-sided tolerance interval of 19 months. Therefore, despite being unrealistic to assume piping ISI is not being performed and operators are oblivious on a continuing basis to multiple indications of RCS leakage, it is not credible for the operating staff to not respond to a leak exceeding the limit for continued operation for over a year and a half.

Collectively, these results provide a robust technical basis that sufficient margin is available for timely identification of an RCS leak and subsequently placing the plant in a safe condition in accordance with plant Technical Specification Limiting Conditions for Operation to prevent pipe rupture.

5.4 Summary

RCS main loop piping rupture is extremely unlikely. NUREG-1829 expert elicitation and PFM modeling of a large number of cases support the conclusion that cracks with detectable leakage will precede the possibility of a piping rupture. If a leak with rupture potential did develop, xLPR predicts over a year and a half (19 months) of leakage at 1 gpm (3.8 L/min) or higher before the leakage crack could become an LB-LOCA [4]. As discussed in Section 4, the multiple sensors available, many continuous or frequent leak rate determinations performed, visibility of leakage measurements to the entire operating staff and plant management (see section 4.4.7), quarterly (minimum) reporting of RCS leakage as a PI, operations and records reviews prior to periodic third-party audits/inspections, etc., there is high confidence that a 1 gpm leak would be successfully recognized, diagnosed, and resolved long before a rupture would occur.

6 NON-PIPING RUPTURES

NRC regulations (10 CFR 50.46(c)) require evaluation of core cooling for

> *"…breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system."*

This regulatory excerpt limits the scope of LOCAs to piping ruptures. Non-piping component sources of RCS leakage are addressed in the mechanical design requirements, supporting analysis, and ISI programs for those components. Piping is not the only potential location for a failure with reactor coolant loss rates equivalent to a LB-LOCA, but the regulations require thermal-hydraulic core cooling analysis only for piping breaks.

Piping has historically been considered for LB-LOCAs, not because a pipe segment is expected to fail in mid-span but because there are a number of welds joining pipe segments together. These welds are locations for potential flaws caused by materials, processes, operating conditions, and external stresses. The piping welds have usually been made in the field, where control of welding conditions and performance of inspections are more difficult than in a factory. Such welds were sites of leakage in the early days of commercial nuclear power before materials specifications, welding, and inspections techniques were refined based on experience and improved technology.

For components, design methods, fabrication processes, and inspection capabilities have been improved and implemented to make occurrence of a failure equivalent to an LB-LOCA of piping extremely unlikely. Components generally have supplemental design requirements that make occurrence of a large, abrupt rupture of the RCPB even less likely. Also, welds in the pressure boundary of components are minimized. When welds are needed, they are performed in the fabrication shop under controlled conditions, may be heat-treated/stress-relieved, and can be inspected closely. These processes reduce the potential for rupture during operation over life.

Core cooling analyses are not performed for postulated ruptures of component pressure boundaries. Consistent with this, as shown in Table 4-2, previously approved LBB applications have not addressed non-piping component failures as part of exclusion of LB-LOCAs in PWR main loop piping. However, NUREG-1829 does include evaluation of "LOCAs" caused by non-piping failures. Therefore, for defense-in-depth purposes, this section provides the basis for concluding that a component rupture large enough to result in fuel dispersal has an extremely low likelihood of occurrence. Because the measures put in place to preclude component ruptures are assessed as part of license renewal for each plant, the evaluation of this section need not be performed for each plant implementing the ALS.

Particularly relevant operating experience discussed in various references is briefly described in the subsections for specific components. Results of a search for more recent relevant operating experience are discussed in Appendix B.

6.1 Non-Piping Components Considered

Although most components have some similarities to piping (i.e., materials selected, designed to ASME Code guidance, and exposure to similar operating conditions), the LBB methodology has not been developed to specifically assess failures other than piping. Alternative approaches are employed to assess the vulnerability to a component rupture large enough to potentially cause FFRD. Possible non-piping component failures and locations are:

- Component shells, bodies, and casings
  - Reactor pressure vessel
  - Pressurizer shell
  - Steam generator (SG) shell in contact with reactor coolant
  - RCS isolation valve (not in all plants) bodies
  - Reactor coolant pump (RCP) casings
- Bolted/threaded closures
  - Reactor vessel head
  - SG primary side manways
  - SG and pressurizer hand holes
  - Valve bonnets
  - Pump motors to casings
  - Pressurizer manway
- Penetrations through the RCPB
  - Control rod drive mechanism (CRDM)
  - Pressurizer heaters
  - Instrumentation
  - RCP shaft seals
- Interfacing systems in which a failure could cause an RCS leak rate equivalent to an LB-LOCA
- Active component (valve) actuation
  - Reactor coolant system relief and safety valves

The possibility of these non-piping failures suffering a rupture large enough to potentially lead to FFRD and subsequent fragment dispersal is a function of several factors. This section assesses vulnerability to non-piping ruptures considering the location-specific failure mechanisms discussed in [45], evaluation of operating experience and inspections, reports of expert elicitation of frequency of failure, evaluations of component integrity and aging, regulatory guidance, etc. Fleet-wide aging assessments of the large RCS main loop components have been performed, such as the Generic Aging Lessons Learned (GALL) program (Section 6.4.1) Also, each individual component is evaluated as part of life extension.

Note that there is no requirement nor physical reason to postulate non-mechanistic ruptures in which randomly sized and located breaches occur without a physical feature, such as a weld causing a stress concentration or susceptibility to corrosion or other degradation phenomenon (e.g., stress corrosion cracking, fatigue).

Some non-piping component pressure boundary materials are susceptible to non-ductile behavior under certain conditions later in life unless mitigating actions are implemented. NRC regulations and guidance and plant licensing bases include provisions to minimize brittle behavior through careful material selection, testing, design, inspection, and operational controls. Although the PWR fleet applies many different vintages of the ASME Code, regulatory guidance, and other codes and standards, analyses in support of license renewal involves ensuring appropriate margin to onset of non-ductile behavior are implemented.

As previously noted, although analysis of core cooling for non-piping ruptures is not required, this section considers the potential for non-piping ruptures as defense-in-depth. Potential component failures have been evaluated to determine which of the following justifications is relevant. Figure 6-1 is a color-keyed representation of the location distribution of these failure types in one loop of a PWR.

1. Failure of components screened out of existing licensing basis (purple in Figure 6-1) for general purposes, not just core cooling – historically, the likelihood of some specific failures has been recognized as extremely low. These include reactor vessel brittle fracture and steam generator lower head and shell failure. To justify excluding these events from the deterministic design basis, various preventive measures are implemented:
   - Conservative design rules were developed by the NRC, ASME, and other organizations to provide additional margin against failure compared to piping.
   - Operating restrictions and precautions are imposed on plant conditions and scenarios where the failures might occur. These include operation within pressure-temperature limits, prevention of PTS, and bounding aging effects (e.g., radiation and thermal embrittlement).
   - ISI in accordance with Section XI of the ASME Code involves periodic inspection of key component pressure boundaries to verify that unexpected degradation is not occurring.
   - Neutron irradiation surveillance capsules installed near walls of the RPV assist in accurate estimation of neutron embrittlement.
2. The main loop piping is addressed in Section 5 (cyan in Figure 6-1).
3. For ruptures or active failures of specific non-piping components, effects on fuel cooling are bounded by the core cooling analyses described in Section 3 (green in Figure 6-1) that show clad burst does not occur – these analyses include the largest branch lines off the RCS main loop. As clad burst is prevented, any rupture in connected systems with equivalent or smaller leak rates in corresponding locations in the RCS can also be concluded to not cause clad burst with subsequent HBU fuel dispersal. This may apply in either of two ways:
   a. The failure itself will result in a rupture size smaller than those shown not to cause clad burst. Examples are a SG handhole threaded closure or blowing out an RCS valve shaft.

b. Although flow area of the rupture may be larger, separation from the RPV by smaller piping or other flow restrictions limits the maximum leakage rate from the RPV. An example is that, for any rupture in the pressurizer, the RCS loss rate is limited by the surge line flow resistance and, therefore, addressed by pressurizer surge line break analysis.

Failures of active components involve different causes. The most impactful is an inadvertent lift and failure to close of a pressurizer relief valve. This is an event required to be analyzed by SRP section 15.6.1. As the relief valves are upstream of the pressurizer surge line connection to the RCS main loop, the pressurizer surge line break is bounding.

4. Remaining components (orange in Figure 6-1)
    a. Large, bolted closure – a qualitative assessment is made of the potential for gross failure and relevant mitigations based on past assessments. Examples are RPV/RVH main flange bolting closure, the RCP casing, and the SG lower head (primary plenum) manway (the pressurizer manway is type 3.b.).
    b. Large cast component such as a valve body or pump casing – because LBB is not qualified for non-piping components, an alternative justification for protection for rupture is presented in this section.

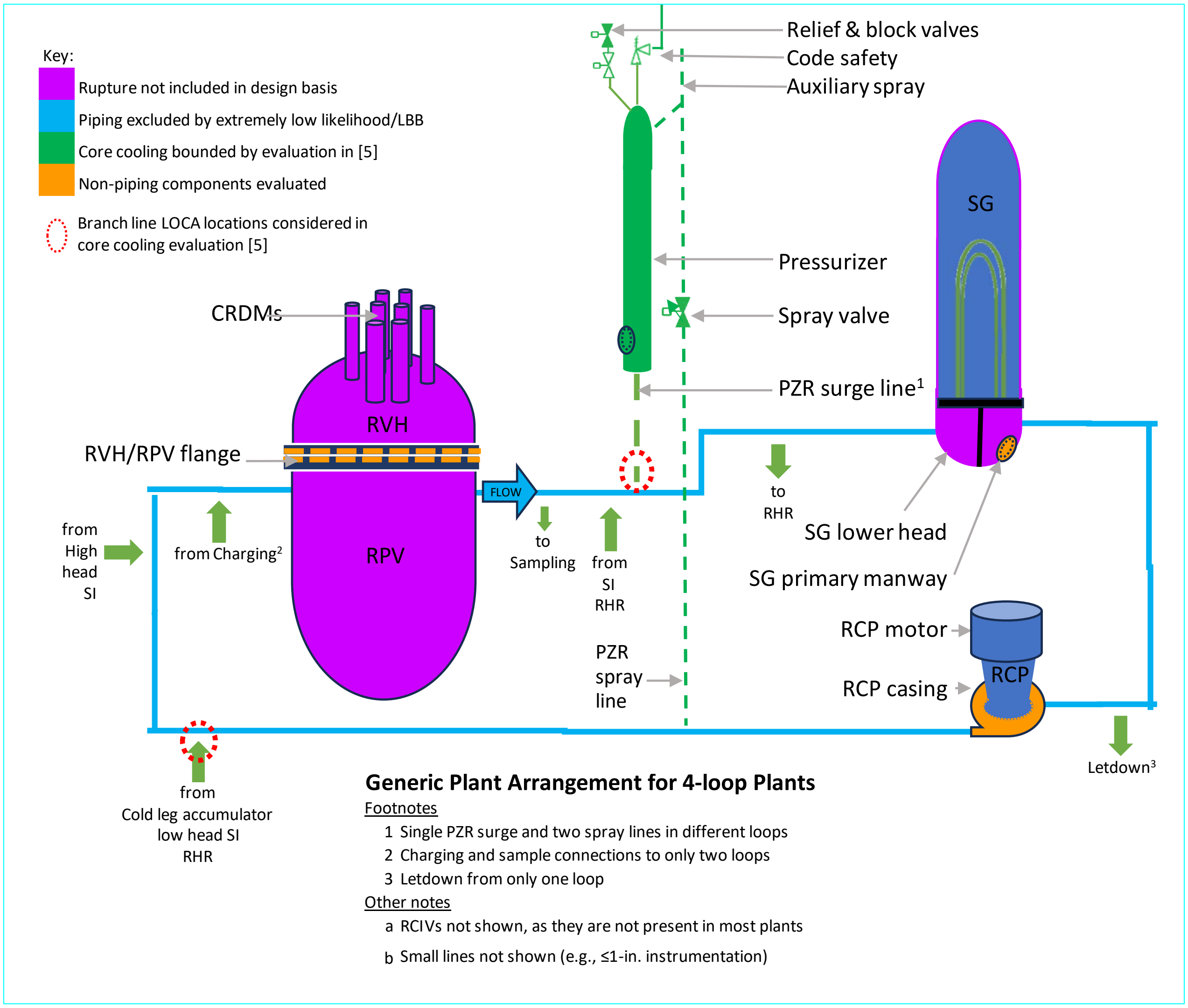

Figure 6-1. Application of LB-LOCA Exclusion Approach by Location in Plant

Table 6-1. Rationale for Acceptability of Various Non-piping Failures

Potential Non-piping Failure	Discussed in Report Section	ISI per ASME Sec. XI	Not in Licensing Basis	Leak Rate Below Burst Threshold	Additional Justification
Component shells, bodies, and casings					
RPV	6.4.2	√	√		Initial design margins, avoid PTS, ISI, adhere to pressure temperature limits, monitor and limit irradiation effect on ductility
Pressurizer shell	6.4.2		√	√	Upstream of surge line so satisfactory analysis of surge line rupture is bounding
SG shell wetted by reactor coolant	6.4.2	√	√		Initial design margins, avoid PTS, adhere to pressure temperature limits
RCS isolation valve (not in most plants) bodies	6.4.4		√	√	More required design margin than piping
RCP casings	6.4.3	√	√	√	More required design margin than piping. Thick-walled to avoid distortion affecting pump functionality.
Bolted closures					
Reactor vessel head	6.5.4	Bolts √	√		EPRI NP-5769 [46] analysis shows LBB-like behavior
SG primary side manways	6.5.5	Bolts √	√		
RCP casing to motors	6.5.6	Bolts √	√		
Valve bonnets	6.5.7	Bolts √	√		
Pressurizer manway	6.2.1		√	√	Leak rate limited by surge line
Penetrations through the RCPB					
CRDM	6.5.9		√		Core cooling analysis not required
PZR heaters	6.2.1			√	Leak rate limited by surge line
RPV instrumentation	6.2.1		√		Core cooling analysis not required
RCP pump shaft seals	6.4.3			√	Maximum flow rate less than LOCA category 1
Active component (valve) actuation					
Primary relief/ safety valves	6.5.10			√	Maximum flow rate within small break LOCA category

6.2 Screened (Excluded) Failures

The original design and licensing basis of LWRs excluded certain major component failures from consideration in accident analyses. Historically excluded failures include:

- Brittle failure of heavy forgings including the RPV, its head, the SG shell, and the pressurizer shells. Exclusion was based upon:
  - Use of conservative codes and standards (e.g., the ASME Code) for design of components: required materials, design methodology and margins, overpressure protection
  - Periodic inspections in accordance with the ASME Code and periodic aging assessments
  - Procedural restrictions on plant conditions/operations such as PTS that could possibly lead to rupture
- Blow out of openings in the RPV and RPV head (e.g., CRDM and bottom-mounted instrumentation penetrations) are excluded from ECCS core cooling analysis.
- Sudden RPV head separation from vessel flange by enough to cause a rapid loss of RCS coolant. Exclusion was based upon:
  - Elastic behavior of bolting
  - Use of ASME Code for design of bolting: required materials, design methodology and margins, overpressure protection, periodic inspections
  - Corrosion control programs

6.2.1 Rupture Flow Bounded

The ability to demonstrate that a piping LOCA or equivalent will not lead to clad burst depends on many parameters, including both the size and location of the break. For a given break size, the limiting location for potential breaks within the reactor coolant system is in the reactor coolant pump discharge line (typically referred to as a cold leg break). The largest connecting line to the cold leg piping is the accumulator line. However, for some plant designs, the largest connecting line to the primary reactor coolant system piping is the pressurizer surge line (which is connected to the hot leg). Postulated LOCAs at various locations and sizes (up to the diameter of the largest connecting lines) can be bounded by analysis of breaks in the RCS piping at the accumulator line and pressurizer surge line connections.

Westinghouse has performed core cooling analyses considering breaks up to the largest connecting lines to the cold leg and hot leg and has shown that clad burst does not occur in accordance with the methodology described in Section 3 within the envelope of applicability described in [6]. Without clad burst, HBU fuel fragment dispersal is precluded. Therefore, RCPB failures with a smaller effective break size than the connecting lines considered in the Westinghouse core cooling analyses showing no clad burst can also be concluded to preclude dispersal.

The threshold break sizes for which Westinghouse plant LB-LOCAs have been shown to avoid HBU fuel rod burst are [47]:

- 2-Loop
  - Maximum cold side piping inner diameter (ID): 10.2 inches (25.9 cm)
  - Maximum hot side piping ID: 8.8 inches (22.4 cm)
- 3-Loop
  - Maximum cold side piping ID: 10.5 inches (26.7 cm)
  - Maximum hot side piping ID: 11.2 inches (28.4 cm)
- 4-Loop
  - Maximum cold side piping ID: 8.8 inches (22.4 cm)
  - Maximum hot side piping ID: 11.5 inches (29.2 cm)

For non-Westinghouse plants, comparison of potential non-piping ruptures should be made on an equivalent basis, in accordance with Appendix A.

Based on ECCS analysis showing acceptability of breaks of the pressurizer surge line and cold leg accumulator lines (see Section 3), the following non-piping LOCAs may be concluded to be satisfactory (i.e., not cause burst of HBU fuel):

- Any ruptures in the pressurizing system such as its manway, shell, heaters, spray valve, isolation valve, power-operated relief valves, safety valves, and their interconnecting piping.
- Any ruptures in systems connected to the RCS such as safety injection, charging, letdown, and residual heat removal.
- Inadvertent opening of valves in systems connected to the RCS, such as the code safety valves.
- After identification of LOCAs bounded by the pressurizer surge and accumulator lines, the only components needing to be further evaluated for LB-LOCA are:
  - Reactor coolant isolation valves
  - RCP casings
  - SG manways

6.3 Assessment of Non-Piping Component LB-LOCA Vulnerability

As the LBB methodology used for piping has not been developed and authorized by the NRC for application to non-piping components, acceptability of possible ruptures of the RCPB of components cannot use NUREG-0800 section 3.6.3 nor xLPR. Even so, the RCPBs of non-piping components are:

- Made of ductile material similar to nearby piping
  - Except for threaded fasteners, which are generally made of high strength materials that are less ductile than plant piping.

- Designed with more margin to failure than nearby piping.
- Subject to similar degradation mechanisms.
- Subject to in-service inspection.
- Assessed for aging mechanisms.

Note that for purposes of this TR, only degradation mechanisms that could lead to the equivalent of a piping LB-LOCA need be considered, as smaller piping LOCAs are evaluated to not result in cladding rupture. For example, failure of shaft seals or the RCP heat exchanger will result in leakage at a rate in the small break LOCA size range.

6.4 Component Bodies, Shells, and Casings

In accordance with 10 CFR 50.46, the maximum reactor coolant rupture size to be analyzed is that of a DEGB of the largest RCS loop pipe, which is 27.5 to 29-inch (69.9 to 73.7 cm) inside diameter (ID) for piping in the cold leg [26].[9] To provide equivalent assurance against clad burst as done for piping, this section assesses whether ruptures larger than the single-ended surge line rupture or single-ended accumulator line rupture could occur for components: reactor coolant isolation valves (RCIVs), steam generator shell and manway or cold leg (RCIVs, RCPs, steam generator shell and manway).

6.4.1 Aging and Life Extension Evaluations

This section describes industry and NRC programs that apply across various components to justify continued acceptability as plants are operating beyond their original design basis.

Generic Aging Lessons Learned Program

RCP casings and RCS valves were evaluated as part of the NRC's GALL program. Long-term exposure of cast austenitic stainless steel (CASS) to normal RCS operating temperature can cause some loss of ductility via thermal embrittlement (i.e., loss of Charpy V-notch energy) but has only a minor effect on other mechanical properties (e.g., tensile strength, fatigue resistance). As described in item IX.M12 of NUREG-2191, Volume 2 [48], the NRC assesses the potential loss of fracture toughness from thermal aging embrittlement of CASS non-piping components (except valve bodies). The aging management program (AMP) determines the significance of thermal embrittlement.

[9] Piping diameter varies slightly by plant. The RCP suction piping is 31 inch (78.7 cm) ID, but is separated from the RPV by the slightly smaller diameter cold leg piping.

Factors that increase susceptibility to embrittlement are [49]:

- Casting method – static, as opposed to centrifugal
- Material composition having
  - High molybdenum
  - High delta-ferrite – the CF-8, CF-8M, and CF-3 grades typically have a volume fraction in the range of 8 to 20 percent [50]
- Higher temperature and longer duration of exposure

Where aging of a CASS component is potentially significant, visual inspections, ultrasonic testing, or a flaw-tolerance evaluation is required. The NUREG-2191 GALL report states

> *"For pump casings, as an alternative to the screening and other actions described above, no further actions are needed if applicants demonstrate that the original flaw tolerance evaluation performed as part of Code Case N-481 implementation remains bounding and applicable for the subsequent license renewal (SLR) period or the evaluation is revised to be applicable for 80 years. For valve bodies, based on the results of the assessment documented in the letter dated May 19, 2000, from Christopher Grimes, U.S. Nuclear Regulatory Commission (NRC), to Douglas Walters, Nuclear Energy Institute (May 19, 2000 NRC letter), screening for significance of thermal aging embrittlement is not required. The existing ASME Code, Section XI inspection requirements are adequate for valve bodies."*

In the GALL report, the NRC also states

> *"For valve bodies greater than 4 inches nominal pipe size (NPS), the existing ASME Code, Section XI inspection requirements are adequate…*
>
> *"For pump casings, as an alternative to screening for significance of thermal aging, no further actions are needed if applicants demonstrate that the original flaw tolerance evaluation performed as part of Code Case N-481 implementation remains bounding and applicable for the SLR period, or the evaluation is revised to be applicable to 80 years."*

ASME Code Case N-481 [51] provided a basis for replacement of periodic volumetric in-service inspections by a combination of visual (VT-1) examination of the external surfaces of the RCP most susceptible to thermal aging and a flaw tolerance evaluation of the most critical locations in the pump's casing. Examinations of the outside surface of pump casings during hydrostatic testing (VT-2) and of internal surfaces when the pump is disassembled (VT-3) are also performed.

Based on performing a stress analysis of the pump casing that considered material properties, including fracture toughness, a postulated one-quarter thickness reference flaw with a length six times its depth, flaw stability, a review of operating history, and the effects of thermal aging, embrittlement, and other degradation mechanisms, mandatory periodic pump inspections can be eliminated [52].

Following incorporation of its provisions in Section III of the ASME Code, Code Case N-481 was annulled in the year 2004.

NUREG-2191 Item XI.M12 [48] concludes:

> *"For valve bodies, and other 'not susceptible' CASS piping components, no additional inspection or evaluations are needed to demonstrate that the material has adequate fracture toughness."*

The NUREG-2191 [48] provision for eliminating CASS component inspections is indicative of a low level of concern regarding growth of flaws and subsequent component progressing to pressure boundary rupture.

EPRI Materials Reliability Program

The EPRI Materials Reliability Program (MRP) evaluates known and potential degradation phenomena in PWR RCSs (e.g., [45], [66]). ALS-relevant activities include RPV integrity, fatigue management, and surveys of inspection data.

6.4.2 Reactor Pressure Vessel, Pressurizer Shell, and Steam Generator Shell

As noted in Section 6.2, these major vessel pressure boundary ruptures are beyond design basis and do not need to be considered for ALS.

6.4.3 Reactor Coolant Pump Casings

The pump casing contains and supports the hydraulic internal structures in the pump and its externally mounted motor. The casings have excess thickness relative to the connected piping to withstand vibratory and dynamic characteristics of pump operation, which provides margin to withstand normal loads while maintaining internal dimensions that are critical to pump functionality. The construction of the pump casing(s) also provides margin to contain an internals failure event.

As part of the RCPB, the casing is stainless steel casting(s) welded to nozzles for joining to the RCS piping. The casing is a large, thick-walled, asymmetric piece comprised of two cast segments welded together with openings for the intake, discharge and impeller (Figure 6-2 and Figure 6-3).[10] Because the casing welds are shop-made, they should be high quality and less likely to have initial flaws than field-made welds like those joining to the piping, but they do have high residual stresses where cracks might initiate in service. An average of 8.3 and 4.5 radiographic/liquid-penetrant-identified defects in small and large feet castings, respectively,

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[10] The pump design described is that used in Westinghouse PWRs. Westinghouse plants utilize a Type F (see Figure 6-2), of which there have been several models. Although some RCPs may differ in dimensions, have a single or two piece casing casting, etc., the discussion of RCP characteristics vis-à-vis pressure boundary rupture is generally applicable.

required weld repair to an average depth of 3.9 inches (9.9 cm). The highest incidence of indications was near the flange, which is the thickest and last poured part, while the inlet and outlet nozzles had a low number of indications [53].

The main flange is a thick piece of metal that restrains and supports the pump internals and the motor. The impeller assembly is bolted in place and trapped between the casing flange and motor; thereby, filling most of the apparently empty casing with the impeller, shaft, bearings, diffuser, and flow guides. Tapering nozzles are welded to piping. The casing includes internal structures to ensure precise positioning of the impeller and consistent internal flow.

A flow rate of several gpm from the chemical and volume control system provides water to the radial bearings. Several seals on the impeller shaft minimize reactor coolant leakage which is collected and monitored so that it does not count against the plant allowable leakage LCO. Pump oil level, vibration, and electrical start and run current are monitored to obtain information on pump performance.

Continued integrity of RCP casings has been evaluated in detail as part of plant license renewal and life extension. Susceptibility to specific degradation mechanisms has been addressed.

- The ASME Code Section III allowable primary membrane plus bending stress for the pump pressure boundary requires the casing design be limited to a stress considerably lower than that of the pipes to which it is attached.[11]
- RCPs have significant structural requirements to ensure proper operation. Casings are thicker than required to meet ASME Code stress criteria and have wall thicknesses typically set by deflection limitations rather than stress. The non-axisymmetric casing's pressure-induced deflections have to be kept small enough that the multiple fine running clearances between the pump rotor and the stationary internals stay in proper alignment. RCP wall thicknesses are four to eight times that of attached piping.
- Evaluation of aging effects of thermal embrittlement in accordance with guidance in former Code Case N-481 provided a process to justify elimination of periodic volumetric inspections required by ASME Code Section XI. The same high assurance used to justify eliminating the inspections also supports the conclusion that a large rupture is extremely unlikely.
- A PWR Owners' Group evaluation [54] of RCP casings confirmed that fracture mechanics evaluation was applicable through 80 years and that continued visual, in lieu of volumetric, inspection was justified.

[11] ASME Code Section III, Division 1: Primary membrane + bending stress allowable

- Pipe fittings: 3.1 S_m and 4.5 S_m allowable for Levels C and D, respectively
- Piping: 2.1 S_m and 3.0 S_m for allowable for Levels C and D, respectively

NB 3545: Valve body primary membrane stress due to internal pressure S_m allowable

- Numerous evaluations have concluded that stress corrosion cracking (SCC) (and intergranular attack (IGA)) of CASS piping and components is not a concern.
  - Design and fabrication processes prohibit use of sensistized austentitic stain steel For Class 1 piping and components. Heat input during welding is controlled to avoid sensitization. As stated in [52]

    > *"For Class 1 piping and components manufactured from austenitic stainless steel, the effects caused by SCC and IGA do not degrade the Class 1 piping and associated components intended function."*

  - Because of the relatively high delta ferrite content, CASS piping is not susceptible to this mechanism [66].
  - According to [67], SCC has not been noted for CASS components in actual operation and would only be a concern if the material were thermally aged and embrittled to the maximum extent at 400°C [752 F] and subjected to severe oxygen transients. Restoration of a reducing chemistry arrested SCC progression.
  - SCC in sensitized weld area is limited by material specifications and plant chemistry procedures.
- The major source for high cycle fatigue is operation of the pump. Vibration, thermal gradients, system pressure, and fluid hydraulic cyclic loading can lead to flaw growth, especially in areas with high weld residual stresses. Measurements of stresses in the piping elbow below the pump found them to be very low (2 to 3 ksi (13.8 to 20.7 MPa)) and well below the fatigue endurance limit [55]. A fatigue evaluation was performed for conservatism because the loading can exceed 10^{10} cycles. Because of excess wall thickness, low cyclic stresses, and damping provided by connection to the RCS piping, Westinghouse Type F pumps have been judged to have insignificant crack growth from vibration [53]. Monitoring of stator and bearing temperatures and vibration sensors provides continuous indications of pump behavior, giving advance warning of a mechanical failure that could affect structural integrity.

A summary of possible aging-related concerns for RCPs noted the following regarding RCPs [emphasis added] [56]:

> *"Based on best information to date, the RCP body castings are considered the most critical pump component with regard to license renewal. The most likely failure mode for a pump casting would be through-wall leakage of primary coolant water. In the unlikely event that thermal embrittlement (long-term aging) ever becomes a problem, unstable ductile tearing of the pump body during a design transient would be a potential failure mode. <u>The RCP body fatigue life is usually conservative and is not considered to be a limiting factor for any license renewal</u>."*

Considering the design criteria, material, operating conditions, etc., a through-wall rupture in aged/embrittled material during a transient is the only mechanism for a pump casing failure that could produce a LOCA large enough to lead to FFRD. Note that the piping flaw evaluation procedures of ASME Code article IWB-3641(b)(1) and fracture toughness for CASS piping materials per IWB-3641(c) may be applied to CASS pump casings and weld bodies, as the most limiting lower-bound fracture toughness for piping and CASS components is similar. Thus, the process for monitoring for potential aged/embrittled material is common between RCP casings and the connected piping.

The RCPs are in the reactor coolant loop cold legs, which reduces the rate of temperature-sensitive degradation mechanisms. They are equipped with temperature, vibration, and seal leak rate monitoring to assess TS criteria and indications of need for further evaluation of expected degradation. Although an RCP is an active component, its casing is a passive structure in which degradation would develop slowly, as in piping.

The core cooling analysis described in Section 3 shows that cold leg LB-LOCAs in a 4-loop plant hot leg up to the size of an 8.8-inch (22.4 cm) inner diameter, single-ended opening can be shown to not cause clad burst. A 180-degree circumferential crack in the pump casing weld would need to open suddenly to over a 1-inch width to cause a leak rate exceeding that analyzed and shown not to cause clad burst. If the crack were at the smaller diameter outlet nozzle to pipe weld, the sudden crack would need to be even wider. Sudden cracking and crack opening this large is considered extremely unlikely, as there is no physical failure mechanism to go from no detectable leakage to that large a crack width in a short period of time.

The following points summarize evaluation of Type F RCPs. [Quoted statements in italics are from NUREG/CR-4731 [56] unless otherwise marked.]

- Operating experience – "No pressure retaining boundary leakage and/or cracking problems in castings and/or casting repairs or fabrication welds. This experience is based on nonnuclear as well as nuclear service."
- Fabrication Process – Westinghouse RCP bodies now in service in the United States were fabricated with statically cast austenitic stainless steel.
- Welds in pump bodies – Type F pump bodies were originally cast in two pieces (although more recently fabricated units used single-piece castings): a suction section and an impeller/outlet section (see Figure 6-2). Assembly was:

  > *"Made using electroslag welding for the fabrication welds (owing to the circular geometry) with no postweld heat treatments; a postweld heat treatment is not an ASME Code requirement (nor is it prohibited) for stainless-steel welds. Without a postweld heat treatment, some high residual stresses (close to yield strength level) may be introduced hooked on the heat-affected zones near the weldments in Type F pump bodies."*

- Thermal embrittlement of base metal – CASS used in RCP bodies is subject to degradation by thermal embrittlement, which results in a slow loss of fracture toughness of base metal over extended periods of time, as noted in section 6.4.1. [56] notes:

  "Investigations into thermal embrittlement effects, using fracture mechanics methods at low deformation rates (approximately five orders of magnitude lower than those used in impact testing), show that the embrittlement problem does not seem to compromise the safety and integrity of the reactor coolant pump body materials."

  Also, [52] states [emphasis added]:

  Valve bodies and pump casings are adequately covered by existing inspection requirements in Section XI of the ASME Code, including the alternative requirements of ASME Code Case N-481 for pump casings. Screening for susceptibility to thermal aging is not required during the period of extended operation because the potential reduction in fracture toughness of these components should not have a significant impact on critical flaw sizes. Accordingly, the current ASME Code inspection requirements are sufficient.

  AMP-3.7 provides aging management for RCP casings through the demonstration of compliance with Code Case N-481. [52]

  The welds in the pump body are not particularly sensitive to thermal embrittlement.

- Fatigue Crack Growth – RCP bodies are subject to thermal and mechanical fatigue damage caused by the system operating transients and pump vibrations. The welds in Type F pumps (such as the electroslag welds) are susceptible to fatigue damage because of high residual stresses. In addition, the presence of any microfissures in low-ferrite (<3%) welds may adversely affect the fatigue strength of the pump body and should be taken into account in estimates of fatigue damage. Fatigue evaluations have concluded the pumps in particular plants are satisfactory for operation to 80 years. Specifically, [52] states:

  "According to WOG, detailed fatigue analyses of RCP casings were not required because the ASME Code conditions specified in NB-3222.4(d)(1) through (6) were met. The ASME Code does not require an explicit fatigue analysis if these limits are satisfied."

- Flaw growth – Tearing modulus analysis (J-integral) has shown that large, final flaw sizes will not lead to fracture.

- Stress corrosion cracking –

  "Cast stainless steel pump bodies and their welds have excellent resistance to stress corrosion cracking. Generally, the [delta] *ferrite content in the welds is >5%. However, if low levels of ferrite are present in any of the welds because of the filler material and weld procedures used, those welds could be sensitized and become susceptible to environmentally induced stress corrosion cracking."* [56]

- Fasteners – "The degradation of the pump closure fasteners caused by borated water corrosion will not lead to the failure of cast stainless steel pump bodies..." [56]

- Shaft failure – "Failure of a pump shaft will not compromise the integrity of the pressure boundary." [56]
- Pump internals failure – *"Failure of pump internals, for example, shafts and bearings, will not compromise the integrity of the pressure boundary."* [56]
- Inservice inspection – ASME Section XI inservice weld inspection requirements were originally developed for the Type F pump bodies, because of their high residual stresses at the welds. NRC has accepted elimination of volumetric examination of pump casing welds, based on experience and improved analytical methods. Section XI requirements for visual inspections of external surfaces during hydrotest and of internal surface when accessible still apply.

To obtain the rapid depressurization that is necessary to cause HBU rod burst, prior degradation must have weakened the pressure boundary nearly uniformly so that a flaw/crack reaches the point where the entire circumference abruptly fails. Figure 6-3 shows the asymmetry of the casing and that its thickness considerably exceeds that of the pipes to which it is attached. This geometry and ISI make a large, abrupt rupture extremely unlikely. In addition, to accommodate thermal expansion and avoid high restraint stresses, RCPs are within a network of structural supports because of their weight and potential for operating vibrations, as shown in Figure 6-4.

Note that separation of pieces and blowdown would be impeded by pump structure and internals, connected piping, and component supports (see Figure 6-4).

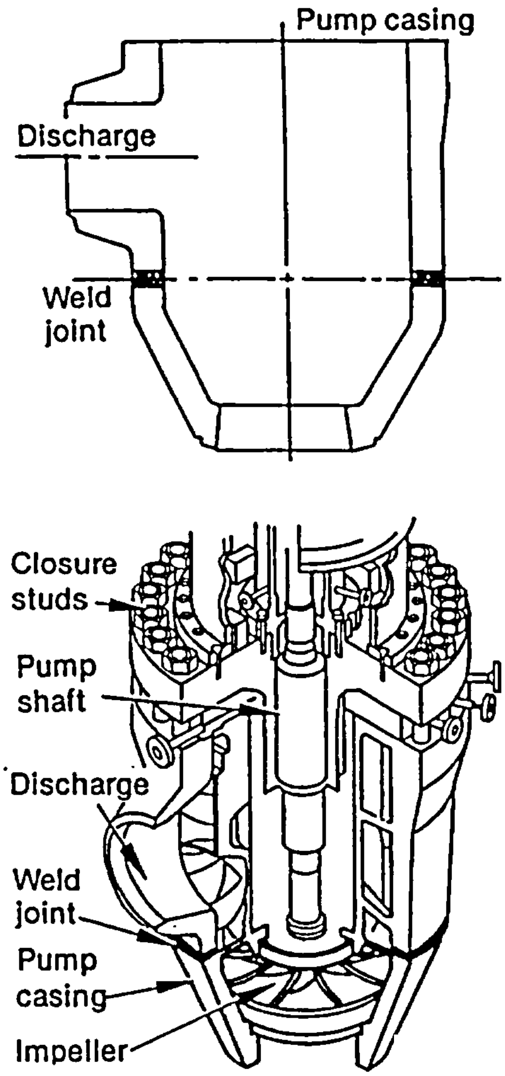

Figure 6-2. Westinghouse (Type F) RCP Casing [50]

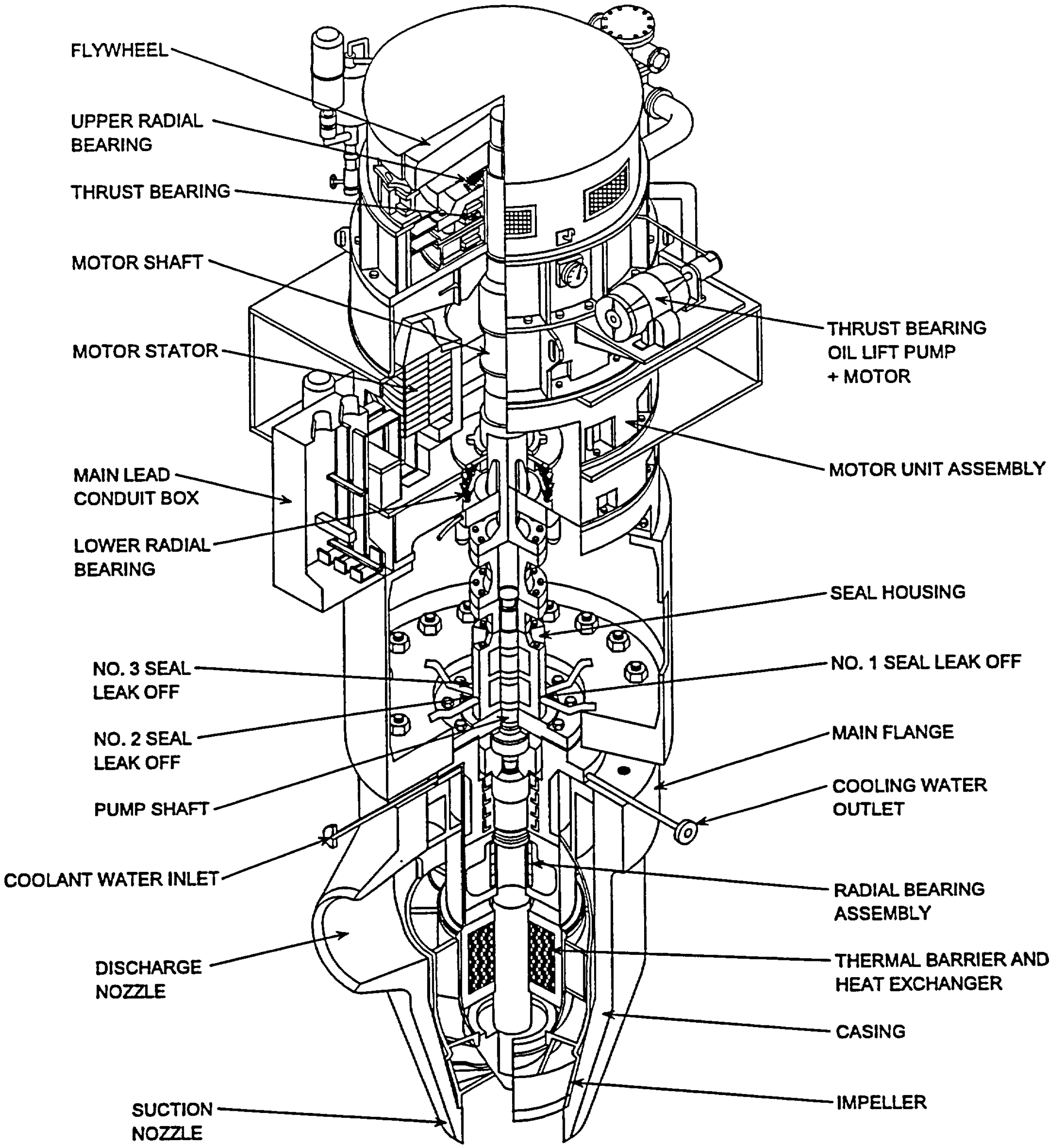

Figure 6-3. Cutaway of Westinghouse RCP [57]

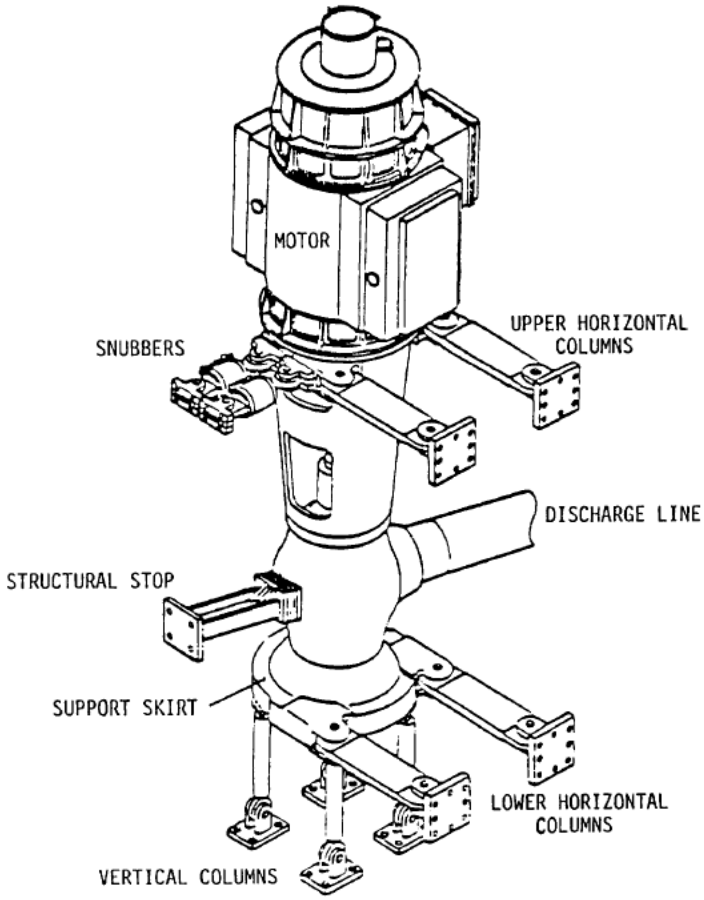

Figure 6-4. Representative Arrangement of RCP Supports [46]

In summary, a sudden circumferential rupture RCP casing has extremely low likelihood of occurrence for the following reasons:

- No instances of gross failure of the RCP RCPB have occurred.
- Fatigue crack growth is satisfactory to 80 years of service.
- The most likely failure mode would be through-wall leakage
- Large flaws will not lead to fracture.
- Thermal embrittlement occurs but at a rate that does not challenge the integrity of the pressure boundary.
- Degradation of fasteners, failure of a rotor shaft, or failure of pump internals will not cause a loss of integrity of the casing pressure boundary.

6.4.4 Reactor Coolant Isolation Valve Bodies

Some Westinghouse plants have RCIVs, which are used for isolating a loop during maintenance but do not have an operating plant function. For plants with these valves, the maximum credible leak rate for a valve failure should not exceed that for the accumulator line.

Although the valve body wall thickness is not set by conditions other than pressure loading, the ASME Code current design rules are indicative of the additional margin required for these components. The asymmetric cross-section of the valve body would not be susceptible to a degradation phenomenon that could cause uniform weakening around most of the circumference setting up a situation where an abrupt and complete separation can occur.

6.5 Bolted/Threaded Closures

Bolted (also known as threaded) closures are used where periodic access inside a system is needed, as repeated welding and cutouts at a location degrade pressure boundary integrity and are costly from a time and radiation exposure standpoint. These joints consist primarily of flanged parts that are secured to a flange on the other side using threaded fasteners. The fasteners are torqued (tensioned) in one of several ways to resist the force of internal pressure pushing the two sides apart and to prevent failure with seismic and accident loads (i.e., faulted conditions).

To provide sufficient strength under faulted loads, fasteners are usually made out of an alloy with higher tensile strength than the steel used for piping. As a result, the fastener material are generally more susceptible to degradation in service (e.g., boric acid corrosion) than the stainless steel used for piping.

Preventing or minimizing leakage is usually accomplished by including a single or dual seal/gasket of appropriate material between the two flange faces with appropriate compression and position. If degradation or improper installation of fasteners reduces the joint preload, the seal may be damaged or pushed out of shape, allowing leakage. Once leakage past the seal occurs, it may grow because of erosion or accelerated corrosion of fasteners exposed to the leakage. In some cases, where prevention of leakage is paramount, bolted closures may have a seal weld applied to provide another barrier should the flange seal not be sufficient.

For ALS, potential for occurrence of a joint separation caused by failure of bolting is evaluated qualitatively to demonstrate defense-in-depth. A properly assembled and maintained bolted joint cannot leak at a rate sufficient to cause fuel rod clad rupture; to reach that high a flow rate requires failure of substantial fraction of the fasteners of a joint.

6.5.1 Definitions

The terms associated with threaded fasteners are sometimes used interchangeably although they represent different mechanical components.

Bolt – a threaded rod with an integral head at one end.12 The head is used to hold the bolt for tightening and bears against the flange to apply the preload. The bolt may be secured in a threaded hole or with a nut.

Stud – threaded rod with no head. The stud may be installed with a nut at both ends or in a threaded hole at one end.

Nut – an internally threaded piece screwed onto a bolt or stud to secure it.

Threaded fastener or bolting – general term for any of the above.

Seal – a compressible insert placed between flange surfaces, sometimes in a groove to block leakage along imperfections in flange surfaces.

Galling – wear of fastener threads when making or unmaking threaded connections.

Preload or tensioning – an axial force developed through threaded fastener installation to maintain joint closure. Elastic expansion of the fastener or fastener group is typically developed through fastener expansion (e.g., heating of the fastener), external axial force (e.g., hydraulically-driven elongation of the fastener), or external torque on the fastener head and/or its nut(s).

6.5.2 Causes of Leakage and Failure of Threaded Closures

Failure of bolted closures or threaded fasteners that are part of the RCPB was evaluated. Although structurally dissimilar from piping and component welds, bolted closures can also be assessed by an LBB-like process. However, because of requirements for high strength, bolting is usually more susceptible to non-ductile failures, and embrittlement effects must be considered. Even if some form of degradation or pre-existing condition leads to the failure of one or more bolts, bolted closures often have sufficient margin to withstand one to several already failed bolts before those remaining intact are no longer capable of maintaining joint closure.

[12] Some of the length of a bolt or stud may be unthreaded. Diameters may differ.

Bolted closures are potentially susceptible to certain failure types:

- Human activities – Human error is the most likely cause of threaded fastener failures. As bolted components are on occasion disassembled by plant maintenance personnel. Errors may occur, such as:
  - Reinstalling with flaws (in-service or storage/handling damage)
  - Incorrect material (and/or paired with the wrong parts or lubricant)
    - Incorrect fastener material – incorrect specifications, supplier error, or maintenance worker selection of the wrong items from inventory could lead to unexpected degradation.
    - Incompatible lubricant – to minimize galling, a lubricant may be used on threads. The lubricant might contain ingredients that are not chemically compatible with the fastener material, thereby causing corrosion.
  - Installation outside of specifications (improper preload, galling, sequence)
- Failure progression caused by load transfer – after human error, the most likely sequence is failure of a bolt caused by some degradation mechanism with transfer of load to other bolts. If there is continuing degradation that causes other bolts to fail, then distortion of the bolting flange shape/alignment, yielding of remaining bolts, and extrusion/damage of seals allows leakage to occur long before degradation progresses to the point of allowing separation, but the rate of coolant loss is still far below that assumed for a licensing basis LB-LOCA.
- External corrosion – usually made of high-strength material instead of stainless steel, threaded fasteners are sensitive to degradation mechanisms different from the rest of the RCPB. In the 1970s and early 1980s, boric acid deposits on fasteners caused rapid wastage. This problem was discussed in EPRI 1000975, *Boric Acid Corrosion Guidebook* [58] and improvements in cleanup of deposits and elimination of leaks have substantially reduced this vulnerability.
- Fatigue – repeated load or temperature cycling of fasteners can cause a crack to form, grow, and break through surface, eventually leading to an abrupt failure of the fastener.
- Design error – these can include sharp contours that produce localized stress concentration, not properly accounting for all relevant stresses, not allowing for differential thermal expansion, neglecting aging effects, etc.
- Defective fabrication – manufacturing errors such poor material quality, exposing material to deleterious substances, incorrect manufacturing, inspections not performed or inadequate to find problems, etc.

Note that bolted closures may leak for reasons not associated with fastener degradation. Most of these involve problems with the seal such as incorrect alignment, insufficient compression, wrong material, damage, scratched/dirty seal surface, etc. that do not affect the structural capability of the joint except for allowing persistent leakage and possible exposure to corrosive substances. The continued leakage, though small, can then become the instigator for corrosion of the bolts actually holding the joint together. In that case, operating staff must recognize the indications of leakage, investigate, and mitigate before structural integrity is lost. This period of detectable leakage before failure of the joint should be many months to years, based on evaluations and operating experience.

6.5.3 Previous Bolted Closure Issues and Response

USI A-12 was established to assess the potential for low fracture toughness of component support materials, including bolting and threaded fasteners. The NRC issued procedures and evaluation criteria in NUREG-0577 in 1979 [59]. In late 1981, the Advisory Committee on Reactor Safeguards (ACRS) recommended additional attention be focused on SCC of high-strength, low-alloy steel bolts. The NRC staff issued IE Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants" in response [60]. The bulletin directed certain actions to assess the integrity of threaded fasteners. A task group on bolting was formed shortly thereafter to develop a coordinated industry response and consider bolting issues of NUREG-0577. The scope of this task group's actions was expanded to 19 tasks that became the Generic Bolting Program. [61]

While the task group was working, the NRC established a new, high priority generic issue B-29 regarding bolting degradation and failure. The NRC noted an increasing number of failures of Class 1 component bolting. Common aspects involved materials that with excessive hardness that was out of specification, sustained high tensile stresses, incorrect preload, an aqueous environment caused by high humidity, primary water leakage, and borated water leakage.

The NRC assigned the failures of bolting to four groups; only the two relevant groups are discussed below. [61]

- Group I – Degradation or Failure of Pressure Boundary Bolting due to General Borated Water Corrosion (Wastage or Erosion/Corrosion). This type is the subject of NRC IE Bulletin 82-02. Degradation due to wastage was viewed primarily as a maintenance problem because no significant corrosion would occur without RCS leakage. Methods for minimizing leakage can be applied. While the materials now in use in flanged connections are fully adequate for the intended application, material changes can also mitigate this problem.
- Group II – Degradation or Failure of Pressure Boundary Bolting due to SCC. These failures can compromise integrity of the RCPB. Some bolts in flanged joints have failed due to SCC. The cause of these failures can be attributed to an undesirable combination of stress, environment, and material condition. Generally, these types of failures can be tied to leaking gaskets and certain lubricants or sealants. The failure of pressure boundary bolts can be eliminated through proper use of tensioning techniques, lubricants, and sealants. For bolting currently in service, assessment of material condition may be accomplished by non-destructive examination (NDE) inspection. Out-of-specification material was not implicated as a cause of SCC of pressure boundary materials.

The Generic Bolting Program work included corrosion evaluations, fracture mechanics studies, development of examination techniques, development of codes and standards, maintenance actions, and training. For the above groups, EPRI developed a Generalized Closure Integrity Model analogous to LBB to [46]:

> *"… demonstrate that a degraded joint (due to wastage, cracking, etc.) has ample margin against catastrophic failure when the leakage from the joint reaches levels that have a very high probability of detection,"*

This objective aligns exactly with the ALS objective to detect, diagnose, and respond to detectable rates of unidentified leakage before a rupture of the RCPB occurs. EPRI evaluation of failures of bolted closures concluded [46]:

> *"Satisfying a leak-before-break criterion is an effective strategy for assuring closure integrity, while at the same time balancing the demands on NDE and minimizing personnel radiation exposure. Preliminary analyses of various primary pressure boundary closures (steam generator and pressurizer manways, RCP main flanges, and check valves) suggested that integrity can be assured by monitoring closure leakage in excess of operational limits. Large leak rates were calculated when a few fasteners were assumed to have failed."*

The EPRI guidebook on boric acid corrosion [58][13] summarizes the applicability of LBB to threaded fasteners:

> *"The leak-before-break model developed NP-5769 assumes that the fastener degradation starts at one fastener and then spreads around the joint in a progressive manner. This is illustrated in Figure 8-12a* [shown below as Figure 6-5]*. Before applying this criterion, it must be confirmed that the degradation mechanism will not affect many fasteners simultaneously…. Simultaneous degradation could result from SCC or from leaking borated water collecting around the entire joint region. Although several incidents involving fastener failures of the type illustrated in Figure 8-12b have exhibited leak-before-break, it cannot be shown that all of these cases would have met ASME Code margins of safety. For example, if a joint were held in place by only a few fasteners uniformly distributed around the flange and stressed to the material yield strength, any small additional amount of corrosion would result in detectable leakage and some small amount of yielding, but the margin of safety on yielding would only be 1.0. This type of joint would not fail catastrophically, but it would not meet the ASME Code specified margin of safety. Results of EPRI flange tests in Section 4 can be used to determine the type of degradation distribution that should be expected."*

[13] Boric acid corrosion guidance was updated and issued as Revision 2 in July 2012 as EPRI document 1025145.

The guidebook also flowcharts the LBB approach for threaded closures as shown in Figure 6-6.

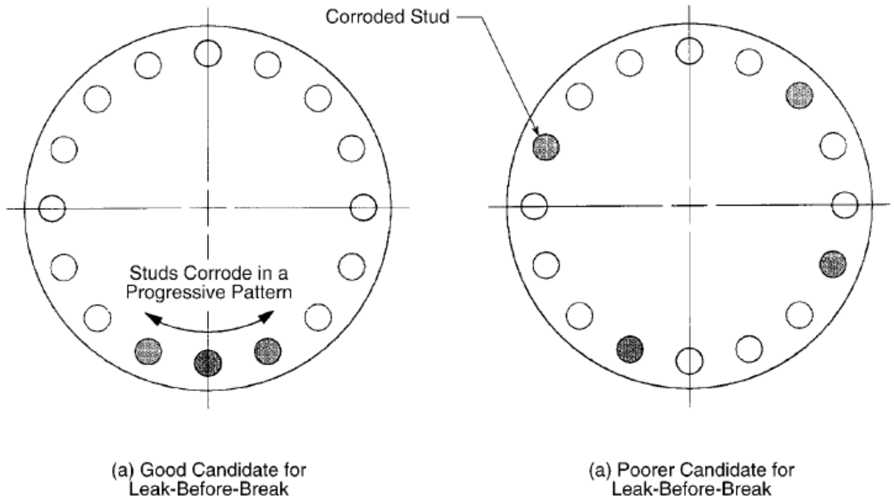

Figure 6-5. Suitability of LBB Depending on Bolt Failure Progression [46]

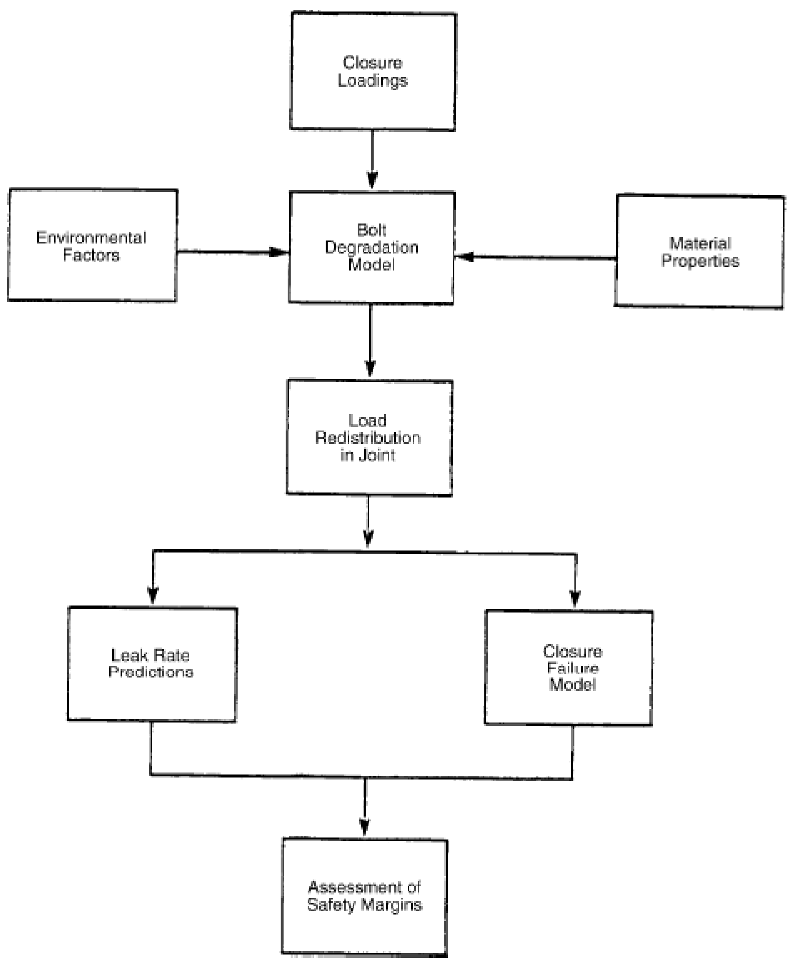

Figure 6-6. LBB Approach for Threaded Closures [46]

Therefore, although an explicit leak rate calculation is not performed for possible bolted joint failures, the same monitoring for unidentified leakage credited for piping LBB should provide timely detection of leakage to limit the occurrence of failure of bolted closures. The EPRI NP-5769 report also noted a benefit if DEGBs could be eliminated, which would allow bolted joint preload to be reduced, which would in turn reduce the failure probability of bolts.

The following tables and figures present key information from EPRI NP-5769. [46]

- Table 6-2 (Table 3-3 of NP-5769) shows that a significant percentage of the studs in various RCPB closures would have to fail in order for leak rate to exceed 1 gpm (3.8 L/m) and that, even so, there would still be a considerable margin to complete failure. Note that RCIV closures were not analyzed.
- Figure 6-7 (Figure 3-13 of EPRI NP-5769) provides the results of EPRI modeling of expected RCS leak rate vs. the number the contiguous bolts failed for various RCS closures. Leak rates are detectable even when the number of broken bolts is low.
- Table 6-3 is derived from Figure 6-7 and lists the approximate leak rate expected for various bolted closures if two, three, or four contiguous bolts are failed. Figure 6-8 (Figure 3-4 of EPRI NP-5769) plots the stress ratio of new load to original load in nearby studs for up to eight contiguous failed studs in a 20-stud SG manway. The bolts with increased loads still have margin to failure.

These results demonstrate that, like LBB for piping, threaded closures can be expected to fail incrementally. Low rate leakage will provide ample advance warning to allow time for detection of the abnormal leakage, evaluation of leak rate, and completion of shutdown and cooldown actions before a sudden failure would occur, causing a large opening with rapid loss of reactor coolant.

Table 6-2. Structural Margins of Bolted Closures at 1 GPM (3.8 L/m) Leak Rate

Closure	Percentage of Failed Studs for 1 gpm (3.8 L/m)	Factor of Safety at 1 gpm (3.8 L/m)
16 Stud SG Manway	15.9	3.2
20 Stud SG Manway	14.5	3.0
RCP Flange	7.8	3.3
10-inch Check Valve	17.8	2.6

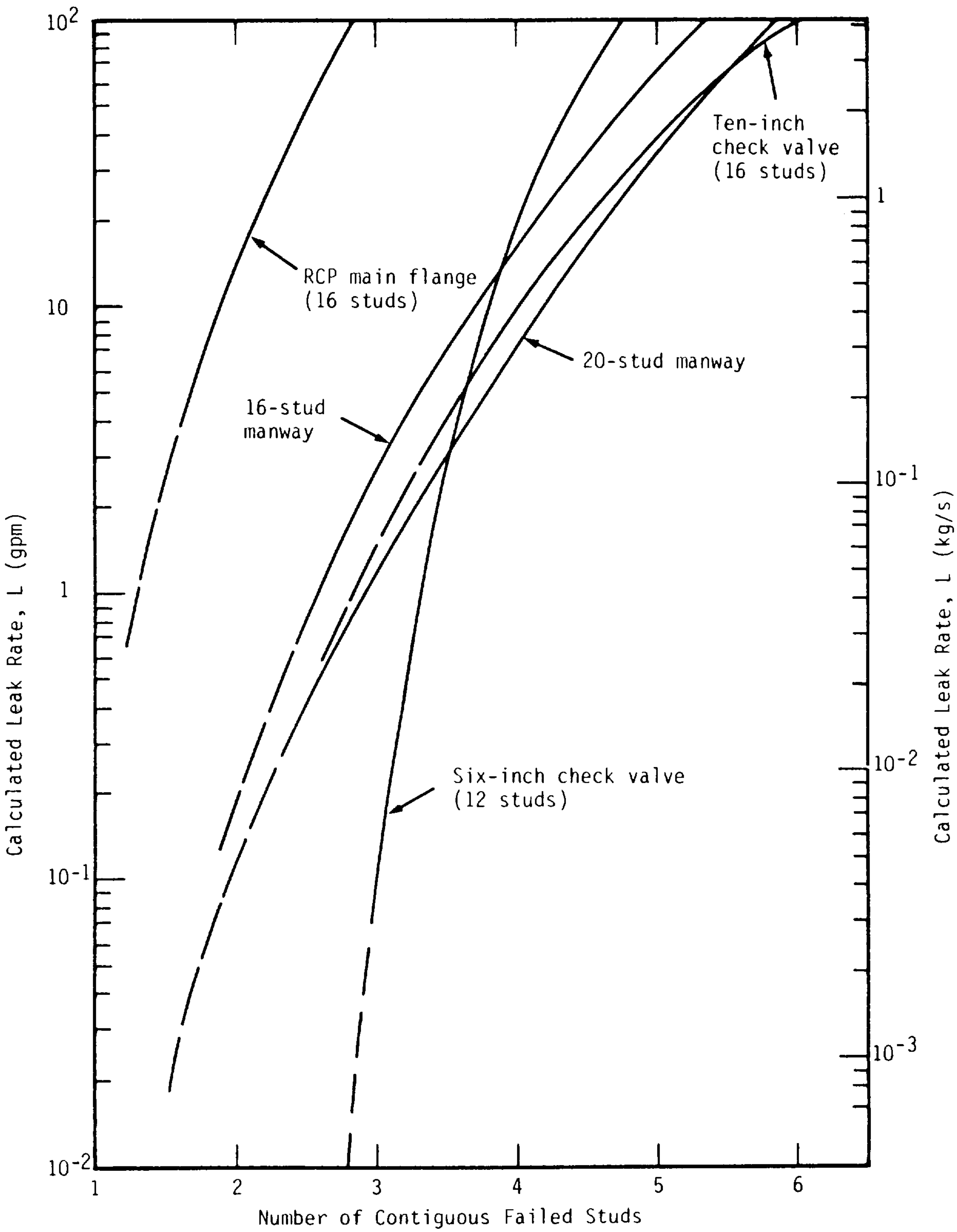

Figure 6-7. Leak Rate Predictions for Different Primary System Closures [46]

Table 6-3. RCS Leak Rate (gpm (L/min)) vs. Number of Contiguous Failed Bolts

	No. of Contiguous Failed Bolts		
Component	2	3	4
RCP main flange	15 (56.8)	>100 (>379)	-
16-stud manway	0.2 (0.8)	3 (11.4)	15 (56.8)
20-stud manway	0.1 (0.4)	1 (3.8)	7 (26.5)
10-in. check valve	-	1.7 (6.4)	10 (38)

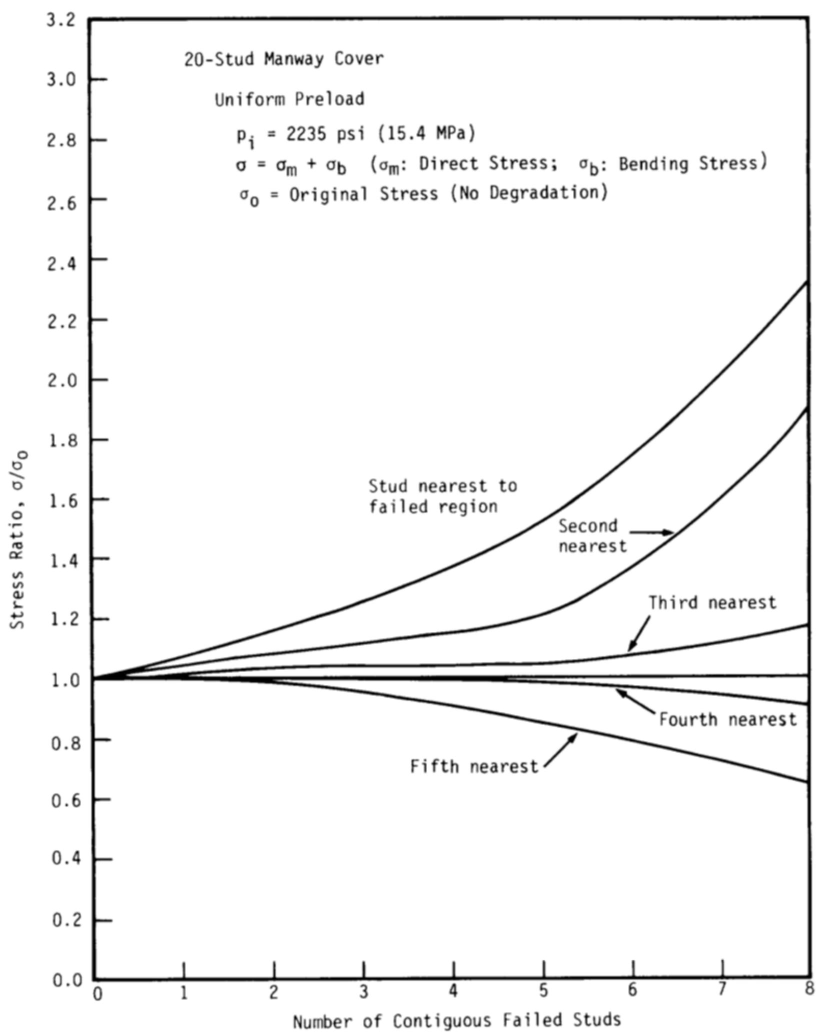

Figure 6-8. Load Redistribution to Nearest Studs vs. Number of Failed Studs for 20-Stud Manway [46]

Boric Acid Corrosion

Boric acid corrosion of the RCPB is both an indicator and a potential initiator of RCPB leakage. For it to occur, some RCS leakage must exist, and the presence of the boric acid on external surfaces of some components can lead to future degradation of the RCPB.

In response to the Davis-Besse head corrosion event in 2001, the NRC required PWR licensees to provide the basis for concluding that leakage and monitoring of RCS leakage and associated boric acid corrosion were satisfactory. Several fleet-wide initiatives were undertaken in response, including:

- EPRI updated its *Boric Acid Corrosion Guidebook* [58], including improved methods for reducing leaks and guidelines for inspecting flanged joints. This guidance stresses the importance of "reducing leakage rather than refining methods to deal with the consequences of leakage." The primary focus is bolted joints and valve packing, based on reported incidents.
- As identified in [62], INPO issued a comprehensive list of areas to be reviewed to evaluate the integrity and maintenance of the RCPB and other borated systems.
- Westinghouse provided guidance on inspection practices to find signs of reactor coolant leakage, boric acid deposits, and corrosion [62].

In 2019, EPRI presented the results of a study to optimize in-service volumetric inspections of bolting greater than 2-inch diameter [63] by reviewing relevant degradation mechanisms. The study concludes that fatigue and SCC are aging phenomena applicable to non-leaking locations.

> *"Issues related to SCC have been addressed by preventive measures that limit the yield strength of bolt material and by prohibiting the use of lubricants that can promote SCC. These issues have been implemented by all U.S. and most overseas plants and are addressed in aging management programs documented in applications for extended plant operation. Issues related to leaking connections, which could lead to BAC* [boric acid corrosion] *and steam cutting are readily identified by visual observation, including operator observation and maintenance activities and are typically addressed in plant BACCPs."*

EPRI concluded that fatigue was the aging mechanism requiring continued attention for non-leaking joints. Growth estimates were developed, and conservative stress estimates were calculated and input into fracture mechanics analyses, that included conservative allowance for uncertainties in loading and fracture toughness of the bolts. The results showed "significant margin on flaw growth to reach a limiting flaw size." provided six criteria are met:

1. Closure bolt material has a yield strength of no more than 150 ksi (1 GPa).
2. No leakage at the joint.
3. Maximum bolt stress bounded by assumed value.
4. The maximum number of low-cycle thermal and pressure transients for plant life is 8000.

5. Bolts are tensioned less than 80 times over their life.
6. Thread lubricant used is in accordance with guidance of the Bolting Integrity Program.

This 2019 study reaffirms the EPRI NP-5769 conclusion that failure of bolted closures is extremely low likelihood, provided threaded fasteners are properly maintained and inspected.

6.5.4 RPV/RVH

The RPV head and its bolting should demonstrate the same behavior as other bolted closures. The preload imposed by tensioning bolts is required to be sufficient to maintain leak tightness of the joint for loading conditions combining RPV head dead weight, seismic accelerations, and RCS pressure.

EPRI [64] performed an evaluation of the usefulness of RPV "threads in flange" examinations that involved the following activities.

- Conducting a survey of 168 nuclear plant units to evaluate past inspection results of these components.
- Evaluating potential degradation mechanisms for "threads in flange" – the only potentially active cause identified was mechanical/thermal fatigue.
- Performing a flaw tolerance evaluation assuming the presence of an initial ASME Section XI IWB-3500 acceptance flaw – fatigue was addressed in a flaw tolerance evaluation using a configuration of a typical PWR plant to determine the time for a postulated flaw to challenge the integrity of the RPV head joint. The allowable flaw size was determined to be at least 77% of the component thickness. A fatigue crack growth analysis was performed with an initial postulated flaw corresponding to the ASME Section XI acceptance standards flaw. Crack growth was determined to be insignificant over 80 years of plant life (i.e., 40 years of plant life extension). This indicates that the integrity of the RPV head to vessel flange joint would not be challenged by any potential degradation mechanism.
- Considering operating events such as an inoperable stud – analysis of redistribution of stresses with one stud out of service in a PWR concluded that:

  > *"the maximum stud stress and the maximum average service stress in the closure studs adjacent to the out-of-service closure stud would still be less than the ASME B&PV Code limit due to the increased loading. The cumulative fatigue usage factor of the closure studs remained below the ASME B&PV Code allowable limit for the rest of the operational life of the reactor vessel. RPV flange separation at the O-ring gaskets was also evaluated with the finding that the O-rings will remain sealed during reactor operation, given the increased load in the closure studs adjacent to the out-of-service closure stud."*

- Considering regulatory interactions such as the anticipated transient without scram (ATWS) rule – evaluation of ASME Service Level C pressure identified that other components in PWR plants were more limiting than the RPV flange.

- Considering beyond design basis events such as severe accidents
- Considering a bounding generic risk impact assessment

Based on the above considerations, it was concluded that the Threads in Flange examinations as mandated in ASME Code Section XI could be eliminated without increasing plant risk or posing any safety concerns for the RPV.

Operating Experience

In 2011, an unusual event occurred at one plant when unidentified leakage exceeded 10 gpm (38 L/m) during a startup. The cause of the leakage was insufficient tensioning (i.e., preload) of the closure head studs such that increasing RCS pressure lifted the head far enough to allow leakage. This scenario is bounded by the ALS because leakage occurs and is detectable during startup. Once the RCS pressure is high enough to unseat the head, leakage will prevent further pressurization. Therefore, an abrupt increase to a high leak rate would not occur. Even if it did, high reactor power and other precursor conditions for FFRD would not exist during startup, and the head would reseat (although leakage would likely continue past displaced seals) as soon as pressure dropped low enough to be insufficient to lift the head against its own weight plus the low preload.

6.5.5 SG Primary Manways

The primary side manway cover on the inlet and outlet plenum of a Westinghouse SG is a 27-inch (686 mm) diameter circular plate over a 16-inch diameter opening [46]. The cover is 5.75 inches (146 mm) thick and secured by 16 1.875-in. (47.6 mm) studs made of AISI 4340 steel according to either American Society for Testing and Materials (ASTM) A193-B7 or A320-L43 specifications. A 20-stud manway cover of similar geometry is also used by one PWR vendor. As part of NP-5769 work, an EPRI three-dimensional finite element model was developed to evaluate cover separation. It showed that several neighboring stud failures were insufficient to lead to cover separation, but that leakage would occur for a second stud failure, providing an early warning of degradation.

Service experience shows that boric acid wastage is associated with leakage of primary coolant, typically at a single location in the circumferential direction (the bottom stud). This pattern reduces the likelihood of several studs having equivalent degradation.

EPRI NP-5769 [46] identified that failure rate depended on vendor, not years in service, with Westinghouse SGs having the lowest rate, possibly because of differences in design and/or maintenance. The most common cause for stud rejection was boric acid corrosion, which constituted more than a third. Rejection rate in terms of bolt-years ranged from about 2.5% for Westinghouse plants to nearly 10% for Combustion Engineering plants.[14] The top three causes

[14] CE plant data was skewed by one instance at the Calvert Cliffs 2 plant where 61 bolts were rejected. These were characterized as caused by pitting or removal damage and are not included in determining the top three causes. [46]

for manway bolt rejection were boric acid corrosion, mechanical damage (e.g., galling), and stress corrosion cracking. Also, the specific thread lubricant had some effect on the rate of particular types of damage.

The EPRI Generalized Closure Integrity Model predicted that for a 16-bolt manway, about three contiguous bolts must fail to get a 1 gpm (3.8 L/min) leak rate and that the increase in stress in adjacent bolts was less than 30%.

Operating Experience

- IE Bulletin 82-02
  - Main Yankee (1982) – six of 20 SG manway studs were found failed and another five had cracks detected by ultrasonic inspection. The cause was identified as SCC promoted by primary leakage past an incompletely compressed gasket.
  - Calvert Cliffs 2 – rejected (almost all) SG manway fasteners because of pitting during operation and mechanical damage during removal.14 (See Appendix B)
- EPRI [58, Appendix A] summarizes incidents involving significant degradation of steam generator or pressurizer manways.
  - St. Lucie 1 (1978) – An increase in containment gaseous activity was traced to a minor leak from a pressurizer manway. Five of twenty 1½-inch (3.8 cm) diameter SA-540 Grade B24 Class 3 studs were corroded. The maximum corrosion depth was ⅛-inch (3.2 mm).
  - Arkansas Nuclear One 1 (1981) – Leakage of about 3.4 gpm (12.9 L/min) from a SG manway resulted from erosion of the bearing surfaces between the generator flange and manway cover plate and degradation of several 2-inch (51 mm) diameter SA-320 Grade L43 studs.
  - Calvert Cliffs 2 (1981) – Leakage was discovered from a pressurizer manway gasket. Two of the 20 SA-540 Grade B24 Class 3 studs were replaced due to boric acid wastage.
  - Arkansas Nuclear One 2 (1982) – Leakage of about 3 gpm (11.4 L/min) from a SG manway. Eight of the 20 studs had been damaged by steam cutting and there was some evidence of minor boric acid attack. The maximum depth of the damage was about ⅛-inch (3.2 mm).
  - Indian Point 2 (1983) – Leakage from three SG manways (two hot leg and one cold leg) caused pitting and corrosion of the closure parts.

  The EPRI report [58] noted that there had been no events involving boric acid corrosion of primary manways since the initial issue of the Guidebook in 1994, an indication that utilities "are doing a better job of reducing leaks from these important joints and dealing with any resultant leaks before there is any significant degradation." The report also provided an assessment of the progression of manway leaks:

  > *"… these incidents have all involved a leak-before-break type behavior. Manways are typically installed at an orientation where the leakage does not tend to form a pool and corrode all studs simultaneously. Finite element analyses in Section 8*

show that significant leakage will occur before risk of failure of the remaining closure bolts. The main incentives in preventing leaks from these joints are to avoid forced outages and the need to make repairs."

- NRC Inspection Reports
  - Byron 1 (2002, inspection report #2002003) – failure to follow the procedure for the installation of the 1B SG manway cover. NRC inspectors identified that the installation of the 1B SG hot and cold leg manway covers was not completed in accordance with applicable maintenance procedures. The failure to properly install the manway covers adversely affected RCS integrity. This finding was determined to be of very low safety significance because the failure did not result in an increase in the likelihood of a significant loss of reactor coolant. A Non-Cited Violation of Technical Specification 5.4.1.a, for the failure to follow the maintenance procedure associated with SG manway closure installation was identified.

6.5.6 RCP Closures

Table 3-12 of [46] summarizes RCP flange bolt rejection rates at eight plants. The average number of years in service at time of rejection ranged from slightly over 4 years to slightly over 11 years with boric acid wastage as the principal cause. The average rejection rate was less than 3% for a population of 3752 bolts.

Operating Experience

- IE Bulletin 82-02 – In May 1980 at Fort Calhoun, wastage was attributed to boric acid attack as a result of leakage at flexitallic gasketed joints between the pump casing and pump cover. These closure studs are 3.5 inches (89 mm) in diameter and are manufactured of SA 193-B7 (AISI 4140) low-alloy, high-strength steel. Accordingly, the NRC issued Information Notice No. 80-27 on June 11, 1980, to all PWR licensees about the potential for undetected boric acid corrosion wastage and emphasized the need for supplemental visual inspection of pressure-retaining bolting in pump and valve components. Subsequently, similar but limited occurrences of corrosion wastage from borated water leakage were identified at other PWR plants, as discussed in the IE Bulletin.
- EPRI [58] summarizes incidents involving significant degradation of RCP closures:
  - Ft. Calhoun (1980 and 1981) – As introduced above (and documented in IN 80-27). leakage from the spiral-wound gaskets caused adjacent studs (three on one pump and four on another) to be severely corroded. The corrosion created an hourglass shape over a region extending about 3.75 inches (95 mm) above the top of the pump casing with diameters of the worst case studs reduced from the original 3.5 inches (89 mm) to 1.0–1.5 inches (25.4–38.1 mm), which is reduction to less than 20% of the original cross-sectional area on worst case studs. There was no corrosion of the stainless steel pump casings and flanges. The original pump design included two concentric gaskets with a leak-off port in between to detect leakage past the inner gasket. However, the leak-off ports on the Fort Calhoun pumps had been plugged, and the leakage was not detected. The pumps were repaired by replacing damaged parts, including 11 studs with less

severe corrosion. During the 1981 refueling outage, some new corrosion of studs on the same pumps had occurred. This corrosion was attributed to condensate from a component cooling water line dripping onto the flange region that still had boric acid residue from the previous incident. The maximum corrosion depth in this case was 0.25 inches (6.4 mm). A total of 14 studs were replaced during this outage.

- Oconee 3 (1981 and 1982) – Leakage had been occurring from the closure flange of all four RCPs for about four years, with 0.5 gpm (1.9 L/min) the maximum from one any pump. During the 1981 refueling outage, the most corroded stud was found to have been reduced from the original 3.84 inches (97.5 mm) to less than the vendor's minimum allowable 3.25-inch (82.6 mm) diameter. This stud was replaced. The diameter of the next most severely corroded stud was 3.52 inches (89.4 mm). During the 1982 refueling outage, an additional five studs in two pumps were found to be less than the vendor's minimum diameter. These studs were replaced with new studs machined from SA-540 Grade B23 material rather than the original SA-193 Grade B7 material.
- Oconee 2 (1981) – Four studs on one RCP were found corroded by leakage from the pump closure gasket. One of the studs was corroded in excess of the vendor-specified 3.25-inch (82.6 mm); the one stud below minimum diameter was replaced. It was concluded that continued operation would have led to increasing leakage that would have provided warning to conduct a safe plant shutdown.
- H.B. Robinson (1995) – Leakage from a high-pressure tap flange gasket and the main pump flange gasket allowed borated water to come into contact with and corrode the main pump flange bolts. The pressure tap flange gasket leak was attributed to insufficient bolt preload. The main flange gasket leak was attributed to loss of bolt preload over time. Corrective action consisted of replacing degraded studs and retorquing all of the studs to the specified preload.
- Callaway (1996) – During shop maintenance, blistering and linear indications were found on the two opposite flanges of the thermal barrier and the number 1 seal. The material for both was SA-182, grade F304 stainless steel, and had been in service for about 12 years.

These events demonstrated that attention to RCS leakage is important and provides a means to detect and correct leakage prior to degradation progressing to the point of structural failure associated with significant RCS leakage much less an LB-LOCA.

6.5.7 RCIV Bonnets

EPRI has not evaluated bolting failure of RCIVs, which have an internal diameter of about three times the largest valve assessed (10-inch (25.4 cm) check valve) in the NP-5769, but the conclusions would be the same as for other components:

- More than one bolt failure would be needed to cause detectable leakage.
- There is adequate margin to failure of remaining bolts after reaching the point of detectable leak rate.

Considerable time would be available for operators to detect and respond to leakage before complete failure of bonnet bolting occurs.

Operating Experience

None noted.

6.5.8 CRDMs

For Westinghouse and Combustion Engineering design PWRs, the CRDMs are threaded into an insert in the reactor vessel head and then seal welded. Although this threaded pipe configuration differs from flanged closures with multiple small fasteners, gross failure of the CRDM to reactor vessel head penetration (leading to a non-isolable blowdown from the top of the RPV) is outside the design basis and is not included in core cooling analysis. Such a failure requires widespread degradation of the threads or failure of the insert to head weld. Therefore, the CRDM threaded connections do not need to be considered for purposes of FFRD for HBU fuel at Westinghouse and Combustion Engineering plants.

For Babcock and Wilcox (B&W) design PWRs, the CRDM nozzles were a concern for boric acid leakage as described in the following subsection. Mitigating actions have been identified and implemented. The B&W CRDM nozzle concern has been resolved.

Operating Experience

The industry concern regarding CRDM leakage at B&W plants was realized in 2001. The reactor vessel head (RVH) at Davis Besse had lost considerable material around a leaking CRDM penetration. The Davis-Besse head corrosion incident is well known as an organizational failure to address abnormal indications and respond to signs of degradation of the RCPB. As part of industry corrective actions for the Davis-Besse incident, a major focus has been to inspect for and correct leakage leading to boric acid corrosion of plant materials. Requirements for monitoring for unidentified leakage were strengthened, and the need for investigating low leak rates has been emphasized.

6.5.9 Penetrations

CRDMs are the majority of reactor vessel head penetrations. Other potential sources of leakage are not considered in core cooling analysis and, therefore, are also not applicable to the potential for LOCA-induced FFRD of HBU fuel.

6.5.10 Active Component Failures

Depressurization and coolant loss may also occur as a result of inappropriate opening or failure of a valve connecting the RCS to another system. Although not considered a LOCA, inadvertent opening of a pressurizer pressure relief valve must be evaluated as an abnormal operating occurrence in accordance with standard review plan [15] section 15.6.1.[15]

The largest pathways involving flow from the RCS through valves in intact systems are bounded by the core cooling analyses for pressurizer and accumulator pipe failures, per SRP Section 15.6 [15].

6.6.11 Other Operating Experience

In [44], Westinghouse provided a review of RCS leakage event operating experience that had been undertaken to evaluate the benefit of the action levels. The review was limited to events from 1/1/1995 to 2/25/2005 (see Appendix B for results of a search of similar OE since 2005).

Of the 87 PWR RCS leakage events found for this ten-year period, 32 were identified during full power operation. Those identified during startup and refueling shutdowns and those associated with Shutdown Cooling System operation are considered to not be relevant to ALS, as fuel and plant conditions (e.g., low decay heat, low fuel and clad temperature, low RCS pressure) will not lead to an event that uncovers the fuel, causing clad burst and HBU fuel dispersal.

Leaks during fueling were discovered because of boron inspection programs. Small leaks associated with deposits generally were not detected by other methods during operation. Leaks detected during startup were usually noted during plant heatup and pressurization by visual inspection during walkdowns. As many of these had been caused by maintenance during the prior outage, the inventory balance method did not detect them because they were present during and part of the baseline determination. In summary, operating experience showed:

- Leaks as small as 0.01 gpm (0.038 L/min) were detected while operating
- Only two RCS piping welds have had leaks

6.6 Factors Affecting Non-Piping Component Integrity

The data in the two tables described below are excerpted from Appendix F (boiling water reactor (BWR) entries omitted) of a 2002 NRC study of risk-informing 10 CFR 50.46 (table numbers from reference are shown in parentheses), as part of early work for TBS [65]. The tables have been expanded to include comparable ALS entries (light shading) for comparison.

- Table 6-4 lists possible PWR component failures and whether the failure was included a contemporaneous NRC staff elicitation (not NUREG-1829). The two shaded columns on the right denote whether the failure is addressed in this TR and provide clarifying remarks.
- Table 6-5 summarizes the failure mechanisms contributing to the LOCA frequency reported in [65] and those included in this evaluation.

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[15] NUREG-1829 lists failure of safety relief valve piping as a major contributor to category 4 piping LOCA frequency.

Table 6-4. Non-piping Component Initiating Events Considered (Table F.4 of [65])

PWR		Consider for ALS	Remarks for ALS
Component Failure	Elicit?		
CRDM	Y	N	Not analyzed for core cooling in current licensing basis
Secondary Side Failures (Dynamic Effects)	N	N	Not relevant
External Events	Y	Y	Only seismic loading is relevant; components are design for seismic loadings
RCP Body/Casing	N	Y	Evaluated in section 6.4.3
Bolted Flange Connection	Y	Y	Evaluated in section 6.5
Valve Bonnets	N	Y	Evaluated in section 6.5.7
Reactor Coolant Pump Seal	Y	Y	Leak rate bounded by core cooling analysis
Power-Operated Relief Valve (PORV)/ Safety Relief Valve (SRV)	Y	Y	Leak rate bounded and location consistent with core cooling analysis
ISLOCA	Y	Y	Leak rate bounded and location consistent with core cooling analysis
RPV Failure	N	N	Not analyzed for core cooling in current licensing basis
Support Fatigue	N	N	Not specifically considered
Class 2 Pipe and Valve	N	N	Leak rate bounded and location consistent with core cooling analysis
CRDM Housing	Y	Y	Not analyzed for core cooling in current licensing basis
Steam Generator Tube Rupture	Y	Y	Leak rate bounded and location consistent with core cooling analysis
Steam Generator Manways	N	Y	Evaluated in section 6.5.5
RPV Head Degradation	N	N	Not analyzed for core cooling in current licensing basis. Opening of RPV/RVH flange evaluated in section 6.5.4

Table 6-5. PWR Failure Mechanisms Considered (Table F.3 of [65])

PWR Mechanism	PWR Elicit?	Consider for ALS?	Remarks for ALS
Intergranular SCC (IGSCC)	N	N	Has not been a problem in actual service. Design, fabrication, and operation is controlled to avoid occurrence, as discussed in section 6.4.3.
Thermal Fatigue/Striping	Y	Y	Considered as part of life extension
Vibration Fatigue	Y	Y	
Water Hammer/Steam Hammer	N	N	Precluded by proper design and operational controls
Seismic	N	Y	Considered in component loading.
Transgranular SCC (TGSCC)	Y	N	See IGSCC above.
Creep	N	N	Maximum RCPB temperatures below point at which effect of creep would have effect. "Creep is not a concern for austenitic allows below 1000˚F [538˚C]" [52]. Bolting at PZR temperature might have slight reduction of preload.
Thermal Aging	Y	Y	Addressed for CASS (see section 6.4.1)
Crane Drop	N	N	Crane operations in vicinity of RCS usually limited when plant at conditions of temperature and pressure where FFRD is a possibility.
Overpressure	N	N	ASME overpressure protection (i.e., safety valves) and design rules obviate concern. Coolant loss rate through stuck open relief or safety valve is bounded by PZR surge line cooling analysis.
Human Error	N	Y	Some errors (e.g., wrong bolt, lubricant) are discussed in Section 6
External Chloride SCC	N	N	No source of chloride expected in containment
PWSCC	Y	Y	Considered as part of life extension
RPV Cleavage Failure	N	N	Beyond design basis

As part of its 2002 evaluation, the NRC noted that recent experience had shown vulnerability to non-piping component degradation, pointing to examples such as CRDM housings (Oconee), BWR stub tubes (Hamaoka), and RPV head degradation (Davis Besse) that were not included in initiating event estimates. NRC went on to note that there could be "surprise failure mechanisms" as plants were operated longer. However, for the past 20 years since the study was prepared, operational experience has not run into such unexpected phenomena, and plant integrity programs have appeared to be effective.

6.7 Non-Piping Component Summary

Although core cooling analysis of non-piping component failure is not required, an evaluation was performed of their impact for defense-in-depth purposes. Evaluation of the potential for failure of non-piping components in the RCS main loops of PWRs concludes that occurrence of a rupture capable of causing FFRD comparable to a LB-LOCA is not credible.

NRC regulations and guidance do not identify an LBB-like process to exclude consideration of ruptures of non-piping components as is done for piping, even though the ductile materials and design methods used assure a similar LBB behavior. In fact, other design considerations for these components impose requirements for more margin than for the piping to which they are attached. Plant component aging is periodically evaluated to ensure that phenomena such as irradiation and thermal embrittlement do not cause unexpected or unacceptable degradation of the RCPB. Licensees must address component aging as part of license renewal.

Early plant licensing established criteria for design of certain components and mitigation of adverse phenomena (e.g., PTS) to justify omitting certain ruptures (e.g., RPV) as not credible. Consistent with this practice, these failures are also screened out, as are failures that would be bounded by the core cooling analyses for failure of the largest branch lines (e.g., pressurizer surge line, accumulator injection line).

Finally, this evaluation of non-piping components is not required by regulations for LOCA analyses, nor has it been performed for other justifications to use LBB to exclude breaks in the RCS main loop. The evaluation was performed as a defense-in-depth measure, and it demonstrates that rupture of non-piping components does not present a potential for HBU fuel clad burst that must be further analyzed or mitigated.

Although intended to be generic, this section does focus on Westinghouse two, three, and four loop plants. Since protection for non-piping component leakage is primarily based on the application of industry standards for design, analysis, and inspections, including extensions to longer operating life, the conclusions are applicable to other NSSS configurations which employ these or equivalent standards. Use of higher burnup fuel and consideration of FFRD has no direct impact on non-piping component leakage or failure rates. Therefore, performance of a similar non-piping component assessment is not needed.

7 SUMMARY AND CONCLUSIONS

7.1 Purpose and Scope

The purpose of the ALS is to provide a near term, efficient means for U.S. PWRs to assess acceptability of adopting an increased allowable fuel burnup. The major impediment to raising the limit is satisfactorily dispositioning concerns that a RCS rupture could precipitate FFRD. Because limited testing data indicates FFRD as the result of a LOCA could lead to fuel damage and release of HBU fuel fragments, the ALS avoids the need to estimate the extent and consequences of fuel dispersal.

This Topical Report presents for NRC review and approval, in conjunction with [4] and [5], the Alternative Licensing Strategy for assessing the acceptability of LOCA-induced fuel fragmentation, relocation, and dispersal in support of extension of the PWR fuel burnup limit extension above 62 GWd/MTU.

7.2 Premise for the ALS

The ALS provides a basis for excluding fuel dispersal for PWR LOCA events. For the RCS main loop, the ALS demonstrates that LB-LOCA induced FFRD is not credible when risk insights are applied. For smaller LOCA scenarios up to the largest branch line off the RCS main loop, core cooling analyses using acceptable extensions to approved evaluation models [6] demonstrate no dispersal occurs by showing that clad burst does not occur. Additionally, an assessment of RCS non-piping components based on design, fabrication, procedure, ISI requirements and operating experience is performed for defense-in-depth.

7.3 Parts of the ALS

The ALS is comprised of the following:

1. Loss-of-Coolant-Accident-Induced Fuel Fragmentation, Relocation and Dispersal with Leak-Before-Break Credit Alternative Licensing Strategy topical report (this report) – provides overall methodology, summarizes results of xLPR and ECCS reports noted below, provides supplemental justification (e.g., non-piping LB-LOCA assessment, bounding analysis of consequences of LOCA-induced FFRD), and regulatory assessment.
2. xLPR fracture mechanics analysis supplemental topical report [4] – documents analysis showing extremely low probability of piping failure through a plant life of 80 years and a long period between the point of detectable leakage and a LOCA event, providing sufficient time to ensure detection and response to RCS leakage before possible rupture.
3. Westinghouse-specific LOCA analysis supplemental topical report [5] – provides results of ECCS core evaluation analysis of HBU fuel for LOCAs not precluded by extremely low probability and LBB.

4. LOCA methodology updates topical report [6] – prepared by Westinghouse, describes methodology used for ECCS core cooling analysis (previous item).

See Appendix A for information on applying ALS to PWRs other than the Westinghouse designs that are described in this report.

7.4 Regulatory Framework for ALS

FFRD is relevant for core cooling analyses, which are required only for piping LOCAs.

There are no current regulations specifically requiring consideration of FFRD. The Commission requested the staff to address FFRD as part of rulemaking [10] to allow for increasing enrichment of light water reactor fuel above five percent. The NRC staff evaluation of comments received on the rulemaking basis is being performed concurrently with preparation of ALS documentation. Staff alternatives discussed in [10] and industry comments in response [11] are considered in selecting the ALS approach.

Despite the noted absence of FFRD regulatory guidance, the following points are relevant:

- Although core cooling criteria of 10 CFR 50.46 would apply if LB-LOCAs were analyzed, use of LBB for RCS main loop piping eliminates LB-LOCAs and the need to perform an FFRD analysis, consistent with past precedents for extremely low likelihood of failure and for LBB.
- Application of LBB to RCS main loop piping has already been approved per 10 CFR 50 Appendix A GDC 4, and exclusion of evaluation of specific phenomena of RCS main loop LB-LOCAs has been allowed in several cases.

7.5 Precedents for ALS

ALS is consistent with NRC prior acceptance of the following:

- Eliminating events with an extremely low probability of occurrence from consideration because they are not credible.
- Eliminating the effects of an RCS main loop LB-LOCA from consideration provided the design of the ECCS and the containment are not changed. While similar LBB applications have been approved by the NRC, clarification of the existing LBB policy related to applicability to ECCS system is needed. This activity is expected to occur as part of the on-going rulemaking, previously described. These anticipated clarifications will be confirmed to be applicable in site specific LAR submittals, per Appendix A.

7.6 Enhanced Justification for ALS

The justification for ALS includes additional information not provided for the precedents noted above:

- Assessment of non-piping failures.
- Estimation of time available for RCS unidentified leakage detection, diagnosis, and response.
- Assessment of defense-in-depth included.
- Safety benefits for adopting ALS identified.

7.7 Assessment of Dispersal of HBU Fuel Fragments

NRC stated [10]:

> *"Licensees may also be able to achieve HBU and corresponding IE necessary to go to such burnups by demonstrating that fuel dispersal can be limited or prevented during LOCAs."*

The ALS approach documented in this TR shows HBU fuel dispersal is prevented because:

1. RCS main loop piping LB-LOCAs causing HBU fuel rod clad burst are not credible and
2. Smaller LOCAs will not result in unacceptable pre-burst fuel relocation and HBU fuel rod clad burst based on ECCS analysis using acceptable extensions [6] to previously approved methodology.

Keeping HBU fuel rod cladding intact avoids the need to develop capability to model fuel fragment dispersal and averts potential consequences such as radiation dose changes from redistribution of fission products.

The ALS methodology considers the effects of potential clad ballooning and fuel fragment relocation on core coolability for smaller LOCAs up to those bounded by failure of the largest branch lines off the RCS main loop.

The ALS approach thereby demonstrates that the allowable burnup of fuel in PWRs meeting the criteria of Appendix A can be raised, per fuel vendor specific analysis results, without explicit consideration of the occurrence and effects of HBU fuel dispersal in the design basis.

7.8 Assessment of Likelihood of Occurrence

In 10 CFR 50.46 (c)(1), a LOCA is defined as having a loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system. The largest piping of the RCS main loop – full break of this piping has an extremely low probability of occurrence, as quantified in NUREG-1829. The frequency predicted by the experts for LB-LOCAs in NUREG-1829 are supported by the results of the xLPR evaluation.

As confirmed and supplemented by LBB, which is already approved by NRC for use for this piping, analysis using the joint NRC/EPRI probabilistic fracture mechanics code xLPR, sudden breaks of large piping are not expected. Instead, if degradation of the piping occurs, a crack may grow to the point of leakage with substantial time before reaching the point where the pipe actually might break. This lengthy period of leakage provides ample time for operators to identify the leak using the multiple monitoring methods available to them. Once they have determined that leakage has definitely increased and could exceed TS LCOs, operators will take actions to shut down and cool down the plant, as required by Technical Specifications.

With the plant shut down and cooled down (Mode 5 or 4), the energy to drive a pipe break, blow down the coolant, and uncover the core is reduced or no longer present and the conditions necessary for FFRD are removed. Decay heat will continue to decrease and RCS temperatures will follow to levels that would preclude cladding rupture, even if a piping rupture did occur. This condition is expected to be reached well before the xLPR predicted time for a leak to develop to the point of rupture.

As LBB and xLPR apply only to piping, failures of non-piping portions of the reactor coolant pressure boundary (e.g., bolted flanges) have also been evaluated, based upon the design of these components and use of similar ductile material as in piping.

7.9 Limitations and Extensibility of ALS

ALS is specific to exclusion of LOCA-induced FFRD and does not provide a basis for modification of ECCS, containment, or environmental qualification requirements or design.

This TR is applicable to plants meeting the requirements described in Appendix A, which also establishes criteria for extension to other plant and fuel designs.

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A REQUIREMENTS TO APPLY ALS TO SPECIFIC PLANTS

This appendix identifies criteria that would permit adoption of the ALS conclusions regarding LOCA-induced FFRD to a specific plant.

To use the ALS methodology in a LAR without supplemental plant specific analyses and justification, the following criteria should be met:

1. Confirm compliance with any limitations and conditions imposed by the NRC in the Safety Evaluation Reports for this TR and [4] and [5].
2. Use of LBB to exclude rupture of RCS main loop piping has been authorized by the NRC.
3. An NRC-approved, LBB-compliant leak rate monitoring program is in place.
4. Analyses approved by the NRC that demonstrate HBU fuel clad will not burst for LOCAs in piping smaller than the RCS main loops. Such analyses are provided for Westinghouse two, three, and four loop plants in [5]. If [5] is determined not to be applicable to a given plant, an approved alternative analysis can be substituted.

Note that main loop non-piping components (i.e., RCP casing, RCIV body, SG shell, and RPV) have been accepted for continued use in accordance with NRC criteria for demonstrating acceptable integrity for a service life through the subsequent license renewal process. Therefore, additional evaluation of non-piping components is not required.

B OPERATING EXPERIENCE

In various places in the main report, examples of operating experience (OE) from references are given. In some instances, the OE references are dated and, therefore, do not include more recent events that occurred since the reference was written. Therefore, a search was performed using the NRC Licensee Event Report (LER) database for recent OE relevant to the focus of this report; namely, leakage or degradation with the potential to lead to a LB-LOCA in RCS main loop piping or failure of a non-piping component.

License Event Reporting

As required by 10 CFR 50.73(a)(2), nuclear reactor plants with an operating license are required to submit an LER for the any event of the type listed in the regulation, relevant examples of which are:

- The completion of any nuclear plant shutdown required by the plant's TSs.
- Any operation or condition prohibited by the plant's TSs with some exceptions.
- Any event or condition that resulted in:
  - The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded
  - The nuclear power plant being in an unanalyzed condition that significantly degraded plant safety
- Any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to:
  - Shut down the reactor and maintain it in a safe shutdown condition
  - Remove residual heat
  - Control the release of radioactive material
  - Mitigate the consequences of an accident

LER Search

The NRC has a contractor that maintains the database of submitted LERs and provides the ability to search the LER database using various metadata. This search utility can be accessed at https://lersearch.inl.gov/. For the purposes of this report, the LER database was reviewed for events occurring January 1, 2005 to 2024, inclusive. Other filter parameters were selected to focus on RCS leakage and RCPB degradation in PWRs, as shown in Figure B-1.

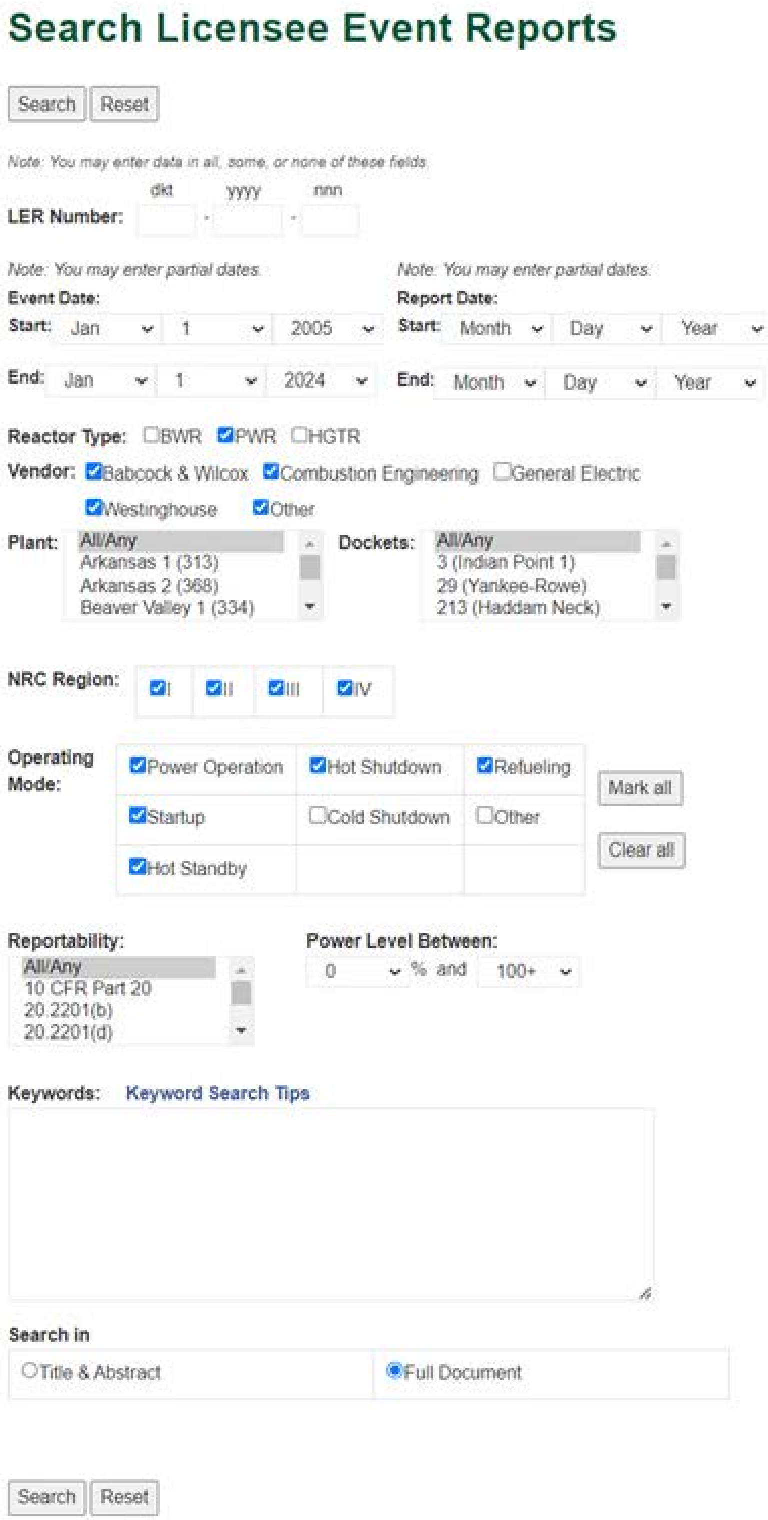

Figure B-1. LER Search Parameters

Note that all PWR vendors were included, as were most plant conditions, and that the keyword search was performed on the full document.

Using the above metadata and the search terms noted below. The search is not case sensitive, so capitals are used to show Boolean operators. Italicized text in parentheses provides explanation of the reason for the search term.

- surveillance requirement AND 3.4.13.1 (*Westinghouse standard TS surveillance requirement number for RCS leak*)
- 3.4.13 (*Westinghouse standard TS LCO number for RCS Operational Leakage*)
- rcs operational leakage (*title of TS LCO 3.4.13*)
- unidentified leakage (*LCO threshold for RCS leakage*)
- rcs leak
- rcs leakrate
- rcs piping AND crack
- rcs AND inventory balance
- rcpb leak
- rcpb crack
- rcp AND casing AND crack
- failed stud
- failed bolt

Results

Some of the searches returned no hits. For those that did have hits, the title and abstract of the LERs were reviewed for relevance to the ALS. No events indicated an on-going or new problem with RCPB boundary integrity.

Conclusion

A search and review of the NRC LER database for operating experience relevant to ALS since January 1, 2005, did not find any additional events that would invalidate or alter the conclusions of this topical report.

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PROGRAM

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