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Reassessing Double-Ended Guillotine Break Requirements: Evidence-Based Analysis of Regulatory Assumptions After Five Decades of Nuclear Operation

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EXECUTIVE SUMMARY

After five decades of nuclear power operation encompassing more than 20,000 reactor-years across 35 countries and 647 reactors, zero double-ended guillotine breaks (DEGBs) have been documented in commercial reactor coolant systems—despite DEGB being the fundamental design-basis assumption driving Emergency Core Cooling System (ECCS) sizing, structural protection requirements, and containment design specifications.

This report examines the basis for DEGB requirements in nuclear power plant design. The DEGB postulate assumes the instantaneous, complete circumferential severance of the largest diameter pipes in reactor coolant systems, driving major design requirements under 10 Code of Federal Regulations 50.46, General Design Criterion 4 and containment design specifications.

The United States (4,880 reactor-years) and France (2,505 reactor-years) contribute the largest operational datasets. Probabilistic assessments estimate direct DEGB occurrence probabilities with extremely low event frequencies, far below the 10^{-5}/reactor-year thresholds typically used to define non-credible events in nuclear-safety analyses; i.e., events with probability this low fall into beyond-design-basis events. Current material-science knowledge demonstrates that the ductile steel materials used in nuclear piping systems exhibit stable crack-growth behavior fundamentally incompatible with instantaneous severance. International regulatory experience, particularly Germany's comprehensive break-preclusion implementation, and successful leak-before-break (LBB) applications in almost all of U.S. pressurized water reactor units validate that alternatives can maintain safety performance while reducing economic burden.

Current DEGB protection systems impose estimated lifetime costs of hundreds of millions of dollars per unit, over the life of a plant across the nuclear industry (including ongoing costs), representing substantial resource allocation toward scenarios with extremely low probability. Although this report acknowledges uncertainties regarding long-term aging effects, potential synergistic degradation mechanisms, and site-specific seismic considerations that warrant continued evaluation as regulatory policy evolves, there remains no documented evidence that a DEGB has occurred as a consequence of the conditions or mechanisms described in this report.

This report acknowledges the Nuclear Regulatory Commission's (NRC's) recent efforts—outlined in the draft Interim Staff Guidance (ISG) NRC-DSS-ISG-2025-XX ("Treatment of Certain Loss-of-Coolant Accident Locations as Beyond-Design-Basis Accidents Draft Interim Staff Guidance")—to reduce overly conservative requirements for large-break loss of coolant accidents through technical justifications and exemptions. However, extensive operating experience and validated methodologies—such as LBB and in-service inspection programs—demonstrate that the probability of a DEGB in reactor coolant-loop piping is extremely low, even under seismic conditions. The authors and reviewers of this report recommend that DEGB be removed as a design-basis event through formal rulemaking, rather than case-by-case exemptions, to better reflect credible failure modes, align with current data, and align with modern, risk-informed safety analysis.

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CONTENTS

FIGURES

TABLES

ACRONYMS

ASME	American Society of Mechanical Engineers
ASN	Autorité de sûreté nucléaire
BWR	Boiling-water reactor
CNSC	Canadian Nuclear Safety Commission
DEGB	Double-ended guillotine break
ECCS	Emergency Core Cooling System
EDF	Électricité de France
EDG	Emergency diesel generator
EPR	European Pressurized Reactor
EPRI	Electric Power Research Institute
GALL	Generic Aging Lessons Learned
IAEA	International Atomic Energy Agency
IGSCC	Intergranular stress corrosion cracking
INES	International Nuclear and Radiological Event Scale
ISG	Interim Staff Guidance
LBB	Leak-before-break
LLNL	Lawrence Livermore National Laboratory
LOCA	Loss of coolant accidents
NEA	Nuclear Energy Agency
NPV	Net Present Value
NRC	Nuclear Regulatory Commission
OECD	Organisation for Economic Co-Operation and Development
PCT	Peak cladding temperature
PWR	Pressurized water reactor
RIDM	Risk-informed decision making
RSK	Reaktor-Sicherheitskommission
SCC	Stress corrosion cracking
SSE	Safe Shutdown Earthquake

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Reassessing Double-Ended Guillotine Break Requirements: Evidence-Based Analysis of Regulatory Assumptions After Five Decades of Nuclear Operation

1. INTRODUCTION AND REGULATORY CONTEXT

Nuclear power plant regulations currently mandate comprehensive protection against double-ended guillotine break (DEGB), a postulated event involving the instantaneous, complete circumferential severance of the largest reactor-coolant system piping (see Figure 1). There are zero-DEGB recorded events both in domestic and international reporting systems.

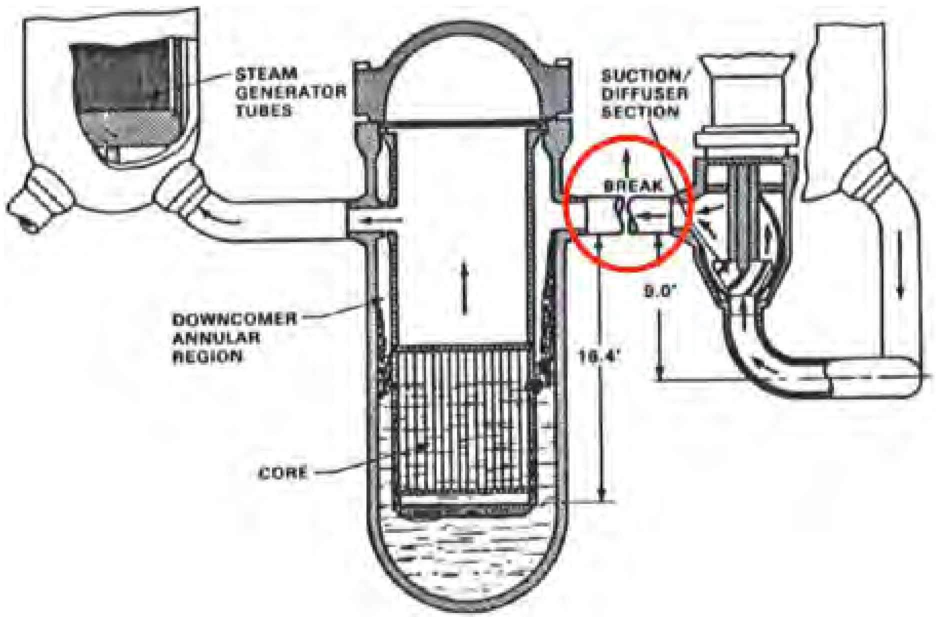

Figure 1. Schematical representation of a postulated DEGB (adapted from Figure 73 of Zhang et al., INL/EXT-19-55888).

This fundamental assumption permeates nuclear plant design requirements, from Emergency Core Cooling System (ECCS) capacity under 10 Code of Federal Regulations (CFR) 50.46 to massive structural-protection systems mandated by General Design Criterion (GDC) 4, 13, 19, 38, and 50 (10 CFR 50.46, 10 CFR 50, APP A). The regulatory framework treats DEGB as a hypothetical design basis that provides for defense-in-depth and margin in core- and containment-cooling design in the face of uncertainty, but does require extensive protective measures throughout nuclear facilities.

The origins of DEGB requirements trace to the early development of commercial nuclear power during the 1970s, when understanding of fracture mechanics remained limited, and operational experience was virtually nonexistent (U.S. Atomic Energy Commission 1971). Regulatory authorities faced the challenge of ensuring adequate safety for an emerging technology with significant potential consequences, but minimal empirical data. The conservative approach adopted during this era emphasizes deterministic safety margins and maximum-credible-accident scenarios, with DEGB representing a bounding assumption to envelope any conceivable piping-failure mode.

DEGB regulatory philosophy reflected appropriate caution given the state of knowledge and societal expectations for nuclear safety. The Atomic Energy Commission and its successor Nuclear Regulatory Commission (NRC) established DEGB as a fundamental design basis primarily to ensure plant survivability under the most severe postulated accidents, without fully accounting for the actual probability of such failures (U.S. NRC 1975). The logic was straightforward: if nuclear facilities could withstand instantaneous rupture of their largest pipes, they could withstand any realistic piping failure that might occur during operation.

While 10 CFR 50.46 establishes DEGB as the upper bound of the postulated loss-of-coolant-accident (LOCA) spectrum, requiring evaluation of "a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties," the regulation does not explicitly justify why instantaneous double-ended rupture should define this upper bound. The subsequent five decades have produced considerable data and information that confirm the overly conservative approach of plant designs to account for a non-credible instantaneous rupture of the largest pipe.

Comprehensive operational databases now document actual failure modes across thousands of reactor-years of experience (Organisation for Economic Co-Operation and Development [OECD] Nuclear Energy Agency [NEA] 2012). Even early probabilistic assessments provided quantitative estimates of DEGB occurrence likelihood (Harris et al. 1989). Enhanced understanding of materials science and crack-growth mechanisms offers detailed insights into actual failure behavior (Anderson 2017). International regulatory experience demonstrates successful implementation of alternative approaches that maintain safety while reducing economic burden (Schulz 1997). Recent regulatory guidance for licensing-basis development, the Licensing Modernization Project outlined in Nuclear Energy Institute (NEI) (NEI 2019) and accepted by the NRC for non-light-water reactors (NRC 2020), clearly places events with frequency lower than 10^{-4}/plant-year outside of the design-basis event category, as seen in Figure 2. While the approach has only been accepted for non-light-water reactors, it still demonstrates that low-frequency events like DEGB should be categorized as beyond-design basis.

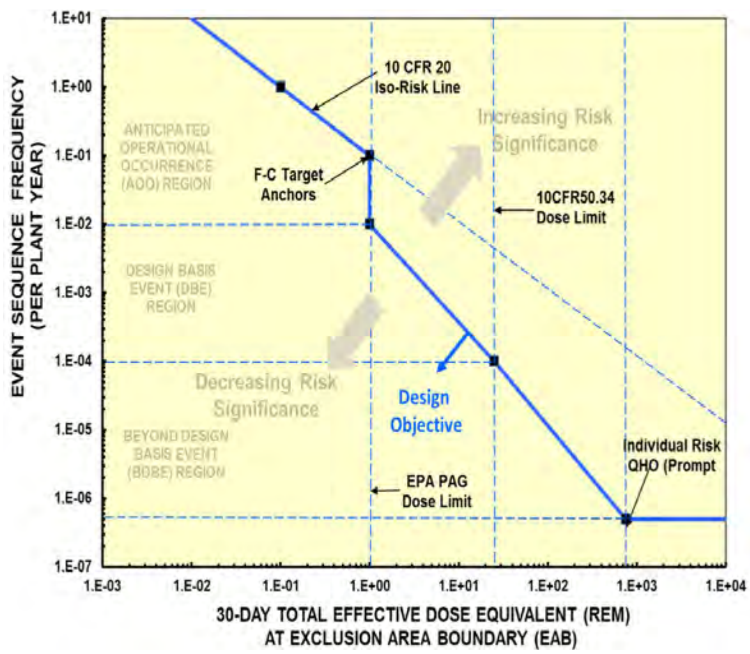

Figure 2. Frequency-consequence target from risk-informed performance-based technology-inclusive guidance for non-light-water reactor licensing-basis development (NEI 2019).

This accumulated evidence creates an opportunity for evidence-based reassessment of regulatory requirements that originated during an era of limited understanding. The policy question is not whether nuclear safety should be maintained—that remains paramount—but whether current requirements represent the most-effective allocation of safety resources based on five decades of learning and technical advancement.

2. TECHNICAL EVIDENCE AND ANALYSIS

The evaluation of DEGB credibility rests on multiple independent lines of evidence that have evolved substantially since the original regulatory framework was established. Operational experience provides the most-definitive real-world assessment of actual failure modes while probabilistic analysis quantifies occurrence likelihood, and materials science explains the mechanisms governing piping-failure behavior.

2.1. Operational Experience: The Foundation of Evidence

The most-compelling evidence regarding DEGB credibility comes from extensive global nuclear operating experience. The International Atomic Energy Agency (IAEA) Power Reactor Information System confirms 19,751 reactor-years[i] of cumulative operating experience through December 2023, exceeding 20,000 by September 2025 (IAEA 2024). Multiple international databases validate the zero-DEGB operational record: the OECD Component Operational Experience, Degradation and Ageing Programme (CODAP) 2022 database contains over 5,100 failure records from 12,505 reactor-years with no DEGB events documented (OECD NEA 2024) while the predecessor Piping Failure Data Exchange (OPDE) FINAL database documented 3,800 events from 2002–2009 without any DEGBs (OECD NEA 2012). Comprehensive reviews of probabilistic-failure studies across reactor-coolant pressure-boundary piping similarly report no documented DEGB occurrences despite extensive analysis of piping failure mechanisms (Krishnan 2019). The U.S. NRC event-notification systems and licensee event reports contain no documented primary system DEGB occurrences. This convergent evidence across multiple independent databases and analytical reviews provides ample documentation of actual failure modes under realistic service conditions. Despite regulatory assumptions requiring protection against such failures, instantaneous complete circumferential severance of nuclear piping has simply not occurred in documented commercial nuclear experience. This highlights a notable difference between regulatory assumptions and operational reality that merits serious consideration in policy development.

The operational record encompasses piping failure experience across diverse systems and degradation mechanisms. The predecessor OECD OPDE project documented approximately 3,800 failure events covering service-induced wall thinning, part through-wall cracks, through-wall cracks with and without active leakage, and instances of significant degradation affecting pressure-boundary integrity (OECD NEA 2012). This analysis spans primary coolant systems, auxiliary safety systems, and secondary systems across multiple reactor designs, including pressurized water reactors (PWRs), boiling-water reactors (BWRs), and other commercial nuclear technologies.

Documented unacceptable inspection flaws encompass various degradation mechanisms, including stress corrosion cracking (SCC), fatigue cycling, flow-accelerated corrosion, and thermal aging effects. These mechanisms represent the actual threats to piping integrity observed in commercial nuclear operation, providing empirical validation developed through laboratory research and theoretical analysis. Importantly, operational experience demonstrates that pressure-boundary piping does not experience instantaneous rupture behavior across all documented failure mechanisms, with piping systems exhibiting detectable leakage well before any approach to catastrophic failure.

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[i] The precise figure is 19,751 reactor-years through December 2023, distributed among major nuclear nations as follows: United States, 4,880 years (24.7%); France, 2,505 years (12.7%); Japan, 1,735 years (8.8%); United Kingdom, 1,668 years (8.4%); and Russia, 1,485 years (7.5%). These five countries account for 62.1% of all global nuclear operating experience (IAEA 2024).

The significance of this operational database is underscored by the longevity of U.S. nuclear operations. Nine Mile Point, Unit 1, the oldest currently operating commercial reactor in the United States, began operation in December 1969 and has accumulated more than 55 years of operational experience without any DEGB events (U.S. NRC 2025a). The average age of the 92 operational U.S. reactors[ii] is 41.6 years, making the U.S. fleet among the world's oldest (U.S.NRC 2024). This extensive operational history across aging reactor fleets provides particularly valuable evidence regarding long-term piping integrity and failure modes; these older reactors have experienced the full range of operational stresses, thermal cycling, and potential degradation mechanisms over multiple decades. Piping inspections have been continuously updated over the same time period to detect, assess, and repair these degradation mechanisms.

The significance of this zero-DEGB record is reinforced by robust international reporting systems that would virtually guarantee documentation of such events. The IAEA's International Nuclear and Radiological Event Scale (INES) requires member states to report events at Level 2 and above within 24 hours, and a primary system DEGB would easily qualify as Level 4+ (defined as "Accident with Local Consequences" or higher), triggering mandatory international notification (IAEA 2013). Although the IAEA Power Reactor Information System includes Chinese operating experience[iii], where China data transparency post-Fukushima is improved, the independent verification of Chinese piping failure data through international databases (Component Operational Experience, Degradation and Ageing Programme [CODAP], OECD Pipe Failure Data Exchange [OPDE]) is more limited. However, major events such as DEGB would be extremely difficult to conceal given: (1) IAEA operational reporting requirements, (2) Chinese participation in World Association of Nuclear Operators peer reviews, (3) extensive international technical collaboration on Chinese nuclear projects (particularly AP1000 and European Pressurized Reactor [EPR] units), and (4) China's own regulatory-reporting systems which, while not fully transparent internationally, would nonetheless document major safety events. Nevertheless, the consistent absence of DEGB events across multiple independent databases—each covering different reactor populations—offers strong, convergent evidence that is unlikely to be influenced by regional reporting variations (Lam et al. 2022).

However, the operational record does document significant piping incidents that provide important insights into actual failure mechanisms and appropriate protective measures. The 1986 Surry Nuclear Plant incident involved an 18-inch feedwater-line rupture due to erosion-corrosion that resulted in four worker fatalities (Lochbaum 2022, U.S. NRC 1987). Investigation revealed that wall thickness had degraded from 0.5 inches to as little as 0.05 inches over an extended period, representing a detectable degradation process rather than instantaneous failure. The regulatory response established mandatory erosion-corrosion monitoring programs that could have prevented this incident through proper implementation of existing inspection technologies (NUREG-0933). Similarly, the 2004 Mihama Nuclear Plant incident resulted from a 22-inch secondary-system pipe rupture due to flow-accelerated corrosion that killed five workers (U.S. NRC 2005). The failed section had wall thickness degraded from 10 to 1.4 millimeters over 28 years without adequate inspection. This incident also involved detectable

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[ii] Note: this does not include the recent (2023–2024) addition of Vogtle Units 3 and 4.

[iii] China represents >2,100 reactor-years of the total global total.

degradation processes and prompted regulatory reforms that established comprehensive pipe-thickness management programs and enhanced inspection standards.

In addition, SCC was observed by Électricité de France (EDF) in the primary system piping in at least six of their operating PWRs. Beginning in late 2021, EDF identified SCC in sections of stainless steel piping associated with safety-related systems at several French PWRs, initially at Civaux Unit 1 and subsequently at additional units. The cracking was primarily located in welded sections of auxiliary primary-connected piping, such as safety injection and residual heat-removal lines, and was attributed to a combination of material susceptibility, residual stresses from fabrication, and local thermal-hydraulic conditions. The affected piping remained intact, with the cracking detected through inspections rather than as a result of in-service failures. EDF, under oversight by the French Autorité de sûreté nucléaire (ASN), implemented an extensive inspection, repair, and replacement program, along with design and manufacturing-process reviews. While the issue led to significant operational impacts and extended outages, no injuries, fatalities, or loss of primary system integrity occurred, and the events were managed within the regulatory safety framework (U.S. NRC 2022). This experience demonstrates that even when degradation mechanisms affect multiple units, proper inspection programs detect issues before they threaten primary-system integrity, and repairs can be implemented through orderly processes—further validating LBB principles.

While these incidents demonstrate the reality of piping failures in nuclear plants, their occurrence in secondary systems using carbon-steel materials limits their relevance to primary-system DEGB credibility. The robust stainless-steel and nickel-alloy materials used in primary systems exhibit fundamentally different degradation characteristics than the carbon-steel components that failed at Surry and Mihama, with enhanced fracture toughness and corrosion resistance that make instantaneous circumferential severance highly unlikely under normal operating conditions.

2.2. Probabilistic Assessment: Likelihood of a Quantifying Occurrence

Probabilistic assessment provides quantitative estimates of DEGB occurrence probability that complement operational experience with theoretical analysis based on fracture mechanics principles. Studies from Lawrence Livermore National Laboratory (LLNL), commissioned directly by the NRC, represent the most-authoritative probabilistic assessments available for regulatory decision-making (Krishnaswamy 1989). These comprehensive evaluations covered Westinghouse, Combustion Engineering, and B&W reactor designs using sophisticated analytical techniques to evaluate all potential failure pathways.

The LLNL analysis employed the PRAISE computer code with Monte Carlo simulation techniques to comprehensively evaluate parameter uncertainties. The studies incorporated material-property variability, loading uncertainties, and inspection effectiveness to provide robust uncertainty quantification. The analytical framework considered crack initiation from fabrication flaws, service-induced degradation, fatigue cycling, environmental effects, and seismic loading to assess all potential pathways to DEGB occurrence.

The quantitative results consistently demonstrate extremely low probabilities of DEGB occurrence under most scenarios. The LLNL studies consistently characterize direct crack-induced DEGB probabilities as “extremely low,” well below thresholds used to define non-credible events in nuclear safety analysis (Krishnaswamy 1989). These probabilities reflect the stable crack-growth characteristics of ductile nuclear-piping materials and the effectiveness of inspection programs in detecting developing flaws before they reach critical dimensions.

However, the analysis also identified seismically induced DEGB as a pathway with higher occurrence probability, estimated at 10^{-7} to 10^{-6} events per plant-year for conservative scenarios. This pathway represents the highest calculated probability mechanism for DEGB occurrence and approaches the 10^{-6} to

10^{-7} per-year thresholds sometimes used to define non-credible events in nuclear safety analysis. It should be emphasized that, while the evaluation of seismically induced DEGB is inherently site-specific, there have been no documented instances in which a seismic event has produced a DEGB in any nuclear power plant piping system (NUREG-1061). For example, the ground motions generated by the 2011 Mineral, Virginia earthquake exceeded the North Anna Nuclear Power Station's Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake design criteria. Despite these exceedances, post-event inspections confirmed that safety-related structures, systems, and components remained within their functional limits, and the U.S. NRC subsequently authorized reactor restart (U.S. NRC, 2011).

Another relevant event is the Great East Japan Earthquake that occurred in March of 2011. A magnitude 9.0 earthquake off the coast of Tohoku, near the Oshika Peninsula of Miyagi Prefecture in Japan, generated a tsunami with maximum wave heights of 40 meters in certain coastal areas of Miyako (Iwate Prefecture) and around 38 meters at Aneyoshi Bay. The average tsunami height was 10–15 meters along much of the northeastern Honshu coastline, where the Fukushima Daiichi and Daini nuclear plants are located. The peak ground accelerations recorded at the Fukushima plants were between 0.52 and 0.56 g horizontally (depending on the unit) and lasted for roughly 150–170 seconds at the plant sites (U.S. NRC 2011a, U.S. NRC 2011b). This exceeded parts of the plant's original seismic design basis, which was about 0.45 g for some components. The core and fuel damage that occurred at the Daiichi site, resulting in the release of fission products, was due to station blackout from the tsunami. The emergency diesel generators (EDGs) for Units 1–4 were located in the basement or lower floors of the turbine buildings of the plants and were rendered inoperable after the tsunami. Units 5 and 6 had some EDGs located in higher or watertight locations (e.g., in the reactor building or in separate protected areas), which allowed them to remain functional after the tsunami. In fact, post-event inspections of Units 5 and 6 confirmed no primary or secondary piping failures from seismic loading. While comprehensive inspection of Units 1–3 was prevented by tsunami damage and subsequent hydrogen explosions, available evidence—including plant instrumentation data and engineering assessments—indicates that the seismic event itself did not cause primary-system piping failures, with core damage resulting exclusively from loss of cooling following station blackout. To summarize, the nuclear-fuel failure and core damage at Fukushima Daiichi were due to the failure of the of the EDGs as a result of the tsunami, not the failure of piping due to the earthquake. Although this is a single example, and more limiting because the plants are BWRs, it further supports the statement that no DEGB has occurred as the result of a seismic event. Thus, previous analyses appear to have been overly conservative, should be re-evaluated based on current data, and should be inclusive of plant inspections.

The seismic qualification requirements established under General Design Criterion 2 and Appendix S to 10 CFR 50 mandate that safety-related piping systems and their supports withstand the SSE while maintaining pressure-boundary integrity. Post-Fukushima seismic walkdowns conducted across the U.S. fleet confirmed that component supports generally maintain substantial margin to their design capacity (NUREG/KM-0017). This suggests that the "indirect" DEGB pathway through support failure—identified by Holman (1984) as having potentially higher probability—can be effectively precluded through maintenance of seismic qualification rather than requiring protection for the consequences of a DEGB.

Recent comprehensive reassessments have updated these foundational studies using modern analytical capabilities and expanded operational experience. Probabilistic fracture-mechanics studies performed for the NRC using earlier LLNL probabilistic fracture-mechanics codes estimated direct DEGB frequencies on the order between 10^{-12} and 10^{-13} per reactor-year. These methodologies form the technical foundation for the joint NRC and Electric Power Research Institute (EPRI) Extremely Low Probability of Rupture (xLPR) Version 2 code (NUREG-2247), which provides a modern implementation of the same underlying fracture-mechanics principles as the previous codes. Although probabilistic analyses earlier than 1989 did not consider SCC, xLPR V2 does. These studies consistently support extremely low rupture frequencies when inspection and leak-detection programs are credited, with some sensitivity cases yielding frequencies on the order of 10^{-12} per reactor-year under realistic conditions.

International regulatory experience demonstrates that break-preclusion decisions need not rely solely on probabilistic analysis. German regulatory authorities eliminated DEGB requirements based primarily on deterministic fracture mechanics demonstrating stable crack growth and LBB behavior (Ali 2010) while recent German probabilistic assessments have provided additional support for break-preclusion concepts (Heckmann 2015). This precedent illustrates that multiple analytical approaches can provide convergent evidence supporting DEGB reassessment.

The consistency of results across multiple independent studies using different analytical approaches provides validation of the extremely low DEGB-probability estimates for direct crack-induced failures. However, uncertainties remain regarding aging effects over extended plant lifetimes, potential synergistic effects between multiple degradation mechanisms, and site-specific factors that could affect probability estimates. These uncertainties suggest that regulatory decisions should consider both the probability estimates and their associated confidence intervals when determining appropriate design and operational requirements.

2.3. Materials Science Perspective: Understanding Failure Mechanisms

Fundamental material-science principles provide compelling evidence that the instantaneous complete circumferential severance assumed by DEGB scenarios is extremely unlikely under normal operating conditions for the ductile steel materials used in nuclear piping systems. Nuclear reactor-coolant pressure-boundary components employ materials specifically selected for high fracture toughness and resistance to brittle-fracture behavior (American Society of Mechanical Engineers (ASME) BPVC II A, ASME BPVC II B). These materials include low-alloy steels such as SA-508, Grade 3, for reactor pressure vessels, stainless steels including Types 316L and 316NG for piping systems, and nickel-base alloys such as Alloys 690, 152, and 52 for steam-generator components.

The ductile nature of these materials fundamentally determines their crack-growth characteristics under realistic loading and environmental conditions. These materials exhibit exceptional fracture-toughness characteristics that preclude instantaneous failure (ASME BPVC III). The 316L and 316NG stainless steels used in primary-coolant piping show strong resistance to SCC in PWR primary-water conditions, with crack-growth rates of only 10^{-9} to 10^{-8} millimeters per second (Zhang et al. 2018). This strong resistance to SCC, coupled with high fracture toughness, ensures that crack growth occurs through stable extension with plastic deformation, rather than the brittle-fracture mechanisms that could theoretically produce instantaneous severance.

The actual progression of crack growth in nuclear piping materials follows a well-understood sequence that precludes DEGB occurrence under realistic conditions. Initial crack-like defects may be introduced during fabrication or develop through such service-degradation mechanisms as SCC, fatigue cycling, or thermal aging. These initial flaws grow slowly under applied stresses and environmental conditions, with crack-growth rates that can be predicted through established fracture-mechanics relationships, including Paris law for fatigue-crack growth (Paris and Erdogan 1963) and SCC models for environmental degradation.

As cracks approach through-wall penetration, the stable crack-growth behavior of ductile materials ensures that small through-wall flaws develop before any possibility of unstable propagation. These through-wall cracks produce detectable leakage that can be identified through properly designed leak-detection systems well before cracks reach dimensions that might threaten structural integrity. Even if leakage goes undetected or unaddressed, continued crack growth occurs in a stable manner with gradual opening of the crack, rather than instantaneous circumferential propagation.

The ultimate failure mode of ductile piping materials involves gradual opening of the crack in a "fish-mouth" configuration rather than the clean circumferential severance assumed by DEGB analysis (Wilkowski et al. 1985). This failure mode produces substantially different mass- and energy-release characteristics than those assumed in DEGB scenarios, typically involving much-smaller effective break areas and more-gradual pressure equalization than the instantaneous blowdown assumed in current regulatory analyses.

Documented degradation mechanisms affecting nuclear-piping integrity have been extensively studied and demonstrate characteristics fundamentally incompatible with DEGB scenarios. SCC in BWR stainless-steel environments exhibit predictable crack-growth rates and morphologies that enable effective management through inspection and mitigation programs (NUREG-0313). Primary-water SCC in PWR nickel-base alloys follows well-characterized patterns that have been successfully addressed through materials improvements and water-chemistry control (NUREG-0691). Flow-accelerated corrosion in carbon-steel secondary systems produces, as described earlier in this section under "Operational Experience," wall thinning that is readily detectable through established modern ultrasonic monitoring techniques (Chexal et al. 1986). None of these documented mechanisms provides a credible pathway to the instantaneous catastrophic failure postulated by DEGB scenarios.

Furthermore, nuclear operators seeking initial or subsequent license renewals are subject to the stringent requirements of NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," drawn from NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," to establish appropriate aging-management programs. Such programs specify applicable ASME requirements and Code-Case use guidelines to maintain the presumption of LBB methodology or leak-detection system throughout the period of extended operation. The commitments associated with these submissions assure the continued applicability of the gradual ductile failure mode mentioned above and further preclude DEGB failures in those plants operating outside of the original 40-year license.

However, material-science analysis also reveals areas of uncertainty that merit continued attention. Thermal-aging effects in cast stainless-steel components can produce gradual toughness reduction over extended operating periods; however, this occurs gradually, rather than suddenly (Chopra 2016). In addition, thermal aging will also occur in austenitic stainless-steel welds that have ferrite, especially those welds with ferrite content on the higher end of the allowable range. The potential for synergistic effects between multiple degradation mechanisms operating simultaneously remains an area of active research. Extended plant operation beyond original design lifetimes may introduce aging effects not fully captured in current understanding. These uncertainties suggest that continued materials research and operational monitoring remain important elements of nuclear safety programs.

Importantly, the stable crack growth and LBB behavior demonstrated by ductile nuclear-piping materials implies that the maximum credible break size is substantially smaller than a full double-ended rupture. Materials science indicates that realistic failure modes involve gradual crack opening with significantly reduced effective flow areas compared to DEGB assumptions. This has direct implications for ECCS sizing: systems designed to handle instantaneous full-diameter rupture inherently possess substantial margin for any credible piping failure mode, suggesting that ECCS acceptance criteria could be met with smaller, more-realistic postulated break sizes.

3. INTERNATIONAL REGULATORY EXPERIENCE AND ALTERNATIVE APPROACHES

The international regulatory landscape provides valuable evidence that comprehensive alternatives to DEGB-based design requirements can maintain excellent safety performance while reducing economic burden and enhancing operational flexibility. Several major nuclear nations have successfully implemented alternative approaches over multiple decades, providing empirical validation of technically sound alternatives to non-credible event protection.

Yonggwang Units 3 and 4 in South Korea are the first nuclear power plants in the world that adopted the LBB concept in early stages of design. In Germany, for all advanced Convoi nuclear power plants, the break preclusion concept including LBB was introduced during the design phase of the plants and was successfully achieved during their construction. The reactor cooling system piping in Areva’s new US EPR design (currently being reviewed by the NRC) is designed using the LBB concept. This eliminates the need to design reactor cooling-system components and piping and supports accommodation of the dynamic effects of large or double-ended ruptures in these piping systems. Consequently, large pipe-whip restraints and jet-impingement shields are not required. To justify the LBB design, a monitoring system is required to detect leakage from this piping into the reactor building. Olkiluoto 3 (OL3), the first third-generation plant in the world, also designed by Areva and currently being built in Finland, and Flamaville 3, currently under construction in France, consider the LBB concept in the design stage. Both these plants use the EPR model as well.

3.1. German Break Preclusion: Comprehensive Alternative Implementation

Germany developed the most-systematic alternative to DEGB requirements through implementation of comprehensive break-preclusion concepts beginning in the late 1970s. The German Reactor Safety Commission (Reaktor-Sicherheitskommission [RSK]) Guidelines for PWRs, revised in 1981, formally eliminated DEGB postulation for structural-force calculations while establishing maximum allowable leak areas of 0.1A, representing 10% of pipe cross-sectional area (RSK 1981). The German approach represented a fundamental departure from non-credible event protection toward mechanistically justified design requirements based on realistic failure modes.

The German break-preclusion concept reflects what regulatory authorities described as “a logical consequence of the ‘concept of basic safety’” where “a guillotine type break or any other break type resulting in a large opening is not postulated any longer for the calculation of reaction and jet forces” (Schulz 1997). This systematic elimination of non-credible event assumptions has been extended from main piping to “an integrated concept for the whole pressure boundary within the containment” and applied in periodic safety reviews of operating plants throughout the German nuclear fleet.

The implementation of break-preclusion concepts has provided substantial benefits that validate the technical adequacy of alternatives to DEGB protection. These benefits include elimination of massive pipe whip-restraint systems throughout nuclear facilities, reduced containment-sizing requirements to enable more-efficient plant layouts, simplified structural requirements to reduce construction complexity and costs, and improved maintenance access to enhance operational efficiency while reducing radiation exposure to workers (Bartholomé 1995). German nuclear plants have operated under break-preclusion approaches for almost four decades without any documented reduction in safety performance, providing empirical validation of the approach. This approach could be reviewed for compliance with Part 53, where the methodology could significantly decrease design and operational costs.

3.2. French Regulatory Evolution and Advanced Reactor Applications

French regulatory policy has evolved substantially regarding alternatives to DEGB requirements, reflecting changing technical understanding and operational experience. While the French nuclear safety authority, ASN, historically maintained conservative approaches that required anti-whipping systems and comprehensive structural protection, recent policy development demonstrates significant evolution toward evidence-based regulation. ASN now accepts break-preclusion approaches for EPR and EPR-2 reactors through Technical Guidelines Sections B.1.2 and B.1.3, recognizing the technical justification for eliminating DEGB from design-basis scenarios for advanced-reactor designs (ASN 2021). This eliminates DEGB scenarios for main coolant pipes and main steam lines from the design basis (Chapuliot and Migné 2013). This integration from the design stage enables more-comprehensive optimization than retrofitting existing designs, demonstrating the potential benefits of systematic application of evidence-based approaches to new nuclear technologies.

The French regulatory evolution reflects broader international recognition that alternatives to non-credible event protection can maintain safety while enabling more-efficient reactor designs. The French approach demonstrates that even traditionally conservative regulatory authorities can adapt their requirements based on accumulated evidence and technical advancement, providing precedent for similar evolution in other regulatory jurisdictions.

3.3. United States Implementation Experience

The United States has accumulated substantial practical experience with alternatives to DEGB protection through widespread implementation of LBB methodology for PWRs. The NRC has approved LBB applications for almost all the PWR units covering primary-coolant-loop piping (U.S. NRC 2018). This extensive implementation demonstrates both regulatory confidence in the technical approach and practical validation of the methodology's effectiveness.

The U.S. implementation extends beyond primary coolant systems to auxiliary systems, including residual-heat removal, chemical and volume control, and safety-injection systems, providing experience with LBB applications across diverse nuclear plant systems. The technical-review process involves comprehensive NRC staff evaluation, including detailed fracture-mechanics analysis verification, material-property evaluation and acceptance, load definition and analysis-methodology assessment, and leak-detection system adequacy evaluation (NUREG-0800).

The most-significant U.S. implementation precedent comes from the AP1000 reactor design, which successfully incorporated comprehensive LBB methodology with full NRC approval for both the certified design and plant-specific applications at Vogtle Units 3 and 4 (U.S. NRC 2013).

The AP1000 design achieved substantial benefits using modernization, passive safety systems and LBB, including complete elimination of pipe whip restraints throughout the plant, 75% reduction in safety-piping length, 60% fewer valves, 35% fewer pumps, and 50% less seismic building volume compared to conventional reactors, optimized containment design that enables significant cost and construction-schedule benefits, and integrated passive safety systems that reduce dependence on active ECCSs (Schultz 2006). Operating experience from six AP1000 units—including Vogtle Units 3 and 4 and four Chinese units (Sanmen 1 and 2, Haiyang 1 and 2)—provides early validation that LBB-based designs perform as intended without DEGB protection systems. While operational history remains limited, these units demonstrate the practical implementation of comprehensive LBB concepts in modern reactor designs with zero documented failures in approved applications, which validates the technical adequacy and conservative nature of the regulatory acceptance criteria.

3.3.1. Boiling-Water Reactor Considerations and Intergranular Stress Corrosion Cracking Management

The NRC has not approved LBB applications for BWR primary systems due to concerns about intergranular stress corrosion cracking (IGSCC) in recirculation piping. IGSCC differs from the degradation mechanisms in PWR stainless steel systems in that it can occur in the sensitized heat-affected zones of welds under BWR water-chemistry conditions.

However, comprehensive IGSCC management programs implemented across the BWR fleet since the 1980s provide multiple layers of defense that collectively preclude DEGB scenarios:

1. Materials Solutions: Replacement of susceptible Type 304 stainless steel with resistant materials (Type 316L, 316NG nuclear grade) in recirculation piping (NUREG-0313)
2. Water Chemistry Control: Implementation of hydrogen water chemistry and noble-metal chemical addition to reduce electrochemical potential below the IGSCC threshold
3. Enhanced Inspection: Deployment of advanced ultrasonic-testing techniques capable of detecting and sizing IGSCC before through-wall penetration
4. In-Service Inspection Programs: Regular examinations under ASME, Section XI, with documented success in identifying and dispositioning IGSCC indications.

Operational experience demonstrates that properly implemented IGSCC management programs have been highly effective. While through-wall leaks have occurred in BWR systems, these have been detected by leak-detection systems and addressed through orderly shutdown and repair, validating LBB behavior even in the presence of IGSCC. No DEGB has occurred in BWR piping despite over 4,000 BWR reactor-years of operational experience globally.

The BWRX-300 design's acceptance by both the NRC and the Canadian Nuclear Safety Commission (CNSC) (World Nuclear News, 2020) indicates regulatory confidence that modern BWR designs incorporating passive safety systems, optimized water chemistry, and advanced materials can achieve adequate safety without DEGB-based design requirements.

3.4. International Coordination and Technology Transfer

International cooperation through the IAEA and OECD NEA has facilitated substantial technology-transfer and regulatory-harmonization activities that validate LBB technical approaches across diverse regulatory frameworks. The IAEA guidance for LBB implementation provides comprehensive technical frameworks that have been successfully applied in developing countries that implement nuclear programs, demonstrating the global applicability of alternatives to non-credible event protection (IAEA 1994).

The OECD NEA has coordinated extensive international benchmarking activities involving 14 organizations from 11 countries, demonstrating consistent technical approaches across participating nations and common recognition of LBB technical validity (OECD NEA 2021). These collaborative efforts have identified opportunities for further harmonization in implementation details while confirming the fundamental technical adequacy of alternatives to DEGB-based design requirements.

The international experience demonstrates that alternatives to DEGB requirements represent mature, technically validated approaches rather than experimental concepts. The successful implementation across multiple regulatory jurisdictions and reactor technologies provides empirical evidence that such alternatives can maintain safety performance while enabling more-efficient nuclear plant designs and operations.

Contemporary regulatory practice demonstrates convergence on alternatives to DEGB protection, specifically for PWR systems. The commissioning of four APR-1400 units at the Barakah Nuclear Power Plant in the United Arab Emirates between 2021 and 2024—employing designs derived from the U.S.-certified System 80+ technology and incorporating LBB methodology—demonstrates strong international confidence in modern, risk-informed reactor-design practices (NUREG-1462, Kim 2009, and IAEA 2024). This represents validation that current regulatory direction supports the systematic elimination of non-credible-event protection in favor of technically justified alternatives. However, the NRC has not issued LBB license amendments because of IGSCC in BWRs. Nonetheless, the GE Vernova Hitachi Nuclear Energy's BWRX-300, currently under licensing review by both the U.S. NRC and CNSC, leverages the Economic Simplified Boiling Water Reactor design certification that incorporates modern risk-informed approaches, rather than traditional non-credible event protection (U.S. NRC 2025b, World Nuclear News 2020, CNSC 2025, djysrv[iv] 2025).

[iv] "djysrv" is the actual name of the author of the article, which can be found in the References.

4. ECONOMIC IMPLICATIONS AND RESOURCE ALLOCATION

The economic implications of current DEGB protection requirements extend far beyond direct capital costs to encompass operational complexity, maintenance burden, and broader impacts on nuclear competitiveness in energy markets. A comprehensive assessment reveals that protection against non-credible events imposes substantial economic penalties while potentially creating operational complications that may reduce, rather than enhance, safety performance.

4.1. Direct Capital-Cost Assessment

DEGB protection systems impose substantial direct capital costs that represent significant portions of total nuclear plant-construction expenses. An analysis of reports and other literature provides specific DEGB-driven protection systems, subsystems, and components (NUREG-0609, NUREG-1061, IAEA 1993, U.S. NRC 1996). Reclassifying a DEGB as a beyond-design-basis event would impact design provisions related to dynamic rupture effects—such as pipe whip, jet impingement, and associated localized load combinations—and allow for reductions in ECCS performance margins and containment pressure-temperature load assumptions. These reductions are based on a mechanistic or break-size-limited LOCA approach wherein the maximum credible break area is demonstrated—through fracture-mechanics analysis and material-property evaluations—to be smaller than a complete guillotine rupture. The aggregate of these systems is estimated to cost hundreds of millions of dollars per plant (NUREG-0609), reflecting the massive scale and complexity required for non-credible event protection. These costs encompass multiple system categories including: pipe whip restraint systems and jet impingement shields requiring large capital costs per plant for high-strength steel construction capable of arresting severed piping motion and for protective barriers designed to prevent damage from high-energy fluid discharge; structural reinforcement necessitating an estimated tens of millions of dollars per plant for enhanced building design to accommodate these protection-system loads; and ECCS oversizing requiring an estimated from tens to hundreds of millions per plant for equipment scaled to handle massive instantaneous coolant-discharge scenarios. It also constrains fuel design, particularly those related to fuel fragmentation, relocation, and dispersal, which currently pose regulatory challenges for proposed higher-enriched, high-burnup fuel.

The ECCS oversizing creates cascading design impacts throughout nuclear facilities. The original purpose of the DEGB postulate was to provide a limiting basis for sizing reactor-containment systems, but the concept was subsequently extended to ECCS design (Ali 2010). The ECCS must be sized to provide adequate makeup water to compensate for the largest-diameter pipe breaks, requiring enormous water-storage tanks, multiple redundant pump trains, and extensive piping systems (Congressional Research Service 2012). The ECCS also drives diesel-generator sizing, emergency-bus sizing, and all the corresponding electrical components to handle the loads. This oversizing will also add to the already large containment volume to accommodate the additional equipment and associated support systems, driving containment construction costs substantially higher. Larger containment structures face increased structural challenges because design pressure requirements, combined with greater structural spans and surface areas, create more-complex loading conditions (Wang et al. 2024). The increased size requires thicker concrete walls, more-extensive reinforcement, and more-sophisticated structural analysis to ensure integrity under design-basis loads, including seismic events. Containment structures must remain elastic under seismic conditions, with concrete sections potentially experiencing biaxial tension not typically encountered in conventional construction (ACI-349). These factors compound the safety challenges while significantly increasing construction costs and complexity, creating a cascade of oversizing effects that could actually introduce new safety concerns (Bourga 2015).

The aggregate economic impact across the nuclear industry represents a massive allocation of resources toward non-credible event protection. The current U.S. nuclear fleet of 94 operating units carries an estimated DEGB-related costs that could measure in the tens of billions of dollars, representing substantial capital that could potentially have been deployed more effectively for genuine safety improvements or cost reductions that enhance nuclear competitiveness. Globally, the approximately 400 operating commercial nuclear units carry estimated DEGB-related costs in the tens of billions, suggesting potential worldwide benefits from systematic elimination of requirements to protect from non-credible events (IAEA 2000).

The NRC's own analysis provides the most-authoritative quantification of economic benefits available from alternatives to DEGB protection. NRC staff evaluations have concluded that application of the LBB concept has produced very large cost savings—on the order of hundreds of millions of dollars in aggregate—by avoiding extensive backfits at many operating nuclear power plants (NUREG-1061) while application of LBB to plants under construction has resulted in cost savings of tens of millions of dollars (Ali 2010). These savings demonstrate that alternatives to non-credible-event protection can provide substantial economic benefits without compromising safety performance.

4.2. Operational Complexity and Maintenance Burden

The ongoing operational costs imposed by DEGB-protection systems extend far beyond an initial capital investment that creates persistent economic burdens throughout plants' operational lifetimes. Maintenance-access restrictions created by massive pipe-whip restraints and jet-impingement shields increase maintenance duration and labor costs, often requiring temporary removal or modification of protective systems to access equipment for routine maintenance activities. These access complications increase worker radiation exposure because maintenance personnel must work longer periods in proximity to radioactive systems while navigating around extensive protective structures that also expose workers to industrial hazards in a challenging environment.

Inspection requirements for DEGB protection systems themselves create additional operational burdens requiring specialized expertise and documentation. Periodic structural-integrity assessments must verify that restraints and shields maintain their design capabilities under thermal cycling and potential seismic loading (ASME OM Code 2023). These inspection programs require sophisticated analytical techniques and comprehensive documentation that add to operational complexity without addressing credible threats to plant safety.

The operational inflexibility created by DEGB protection systems affects plant modification and upgrade activities throughout operational lifetimes. Future plant improvements must accommodate existing restraint and shield systems, often requiring costly modifications or engineering workarounds that would be unnecessary in plants designed without non-credible-event constraints. This inflexibility can increase costs for plant life-extension projects, power uprates, and equipment-replacement activities that are essential for continued competitive operation. This leads to a high safety cost for very little safety benefit.

4.3. Economic Opportunity Assessment for New Construction and Ongoing Costs for Existing Plants

The economic benefits available through elimination of DEGB requirements may prove most significant for new nuclear construction, where integrated design optimization can achieve benefits impossible through retrofitting existing plants. This approach could yield hundreds of millions of dollars in economic benefits per plant, primarily through reduced construction and maintenance costs, improved plant layout, and lower radiation exposure for workers. Current new plant projects have experienced substantial cost overruns and schedule delays partly attributable to complex design requirements including DEGB protection systems (Potter 2022). These measures encompassed extensive pipe-whip restraints and jet shields, over-engineered support structures, increased seismic design margins, and complex inspection and maintenance protocols. Although there were large overruns due to incomplete engineering on secondary systems coupled with poor project management, the Vogtle project's final cost of approximately $30 billion for two AP1000 units demonstrates the scale of nuclear construction costs, with DEGB-related requirements contributing significantly to total project expense (Johnson 2024).

As previously discussed, elimination of DEGB constraints could reduce new plant capital costs by hundreds of millions of dollars per unit while enabling simplified plant layouts that accelerate construction schedules. The construction-schedule benefits may prove even more valuable than direct cost savings because simplified designs without massive restraint and shield systems enable more-efficient construction sequences and reduced complexity of coordination. Adherence to DEGB requirements have both one-time (i.e., capital expenditure) and ongoing (i.e., operational expenditure) costs. However, it is important to note that cost benefits differ substantially between new construction and existing plants. New plants can realize full capital-cost avoidance through integrated design optimization while existing plants primarily benefit from reduced operational burden and enhanced flexibility for modifications and power uprates. The power-uprate potential is particularly significant for existing plants where DEGB currently limits achievable power increases.

Table 1 presents the one-time implementation costs, Table 2 outlines the recurring or operational costs, and Table 3 provides a consolidated summary of the total cost estimates (Swamy 1997, NUREG-0933, U.S. NRC2004, NUREG-1061, U.S. NRC2006). Cost estimates are order-of-magnitude approximations and were derived from limited publicly available data. Actual values can vary substantially depending on plant vintage (e.g., Generation II versus Generation III), reactor design (PWR versus BWR), and the scope of installed protection systems.

Table 1. One-time costs incurred during design, licensing, and construction phases.

Category	Estimated Costs (millions of U.S. dollars)	Notes
Pipe-whip restraints and jet shields	100s	Eliminated hardware and associated installation labor
Seismic and structural overdesign	10s	Reduced structural reinforcement needs
ECCS overdesign	10s	Reduced design, licensing, and operational needs
Licensing simplification	10s	Faster NRC reviews, fewer design basis events to justify
Design optimization	10s	More efficient layouts, reduced material use

Table 2. Ongoing costs that are realized over the 40–60-year operating life of the plant.

Category	Estimated Savings (millions of U.S. dollars)	Notes
Maintenance and ISI	10s	Fewer components to inspect, less downtime
Radiation exposure reduction (ALARA)	10s	Easier access, fewer man-hours in high-rad areas
Operational flexibility	Variable	Easier upgrades, fewer outage complications

Table 3. Summary of costs by plant category.

Category	New Construction (millions of U.S. dollars)	Existing Plants (millions of U.S. dollars)
One-Time Capital Savings	100s	N/A (sunk costs)
Ongoing Operational Savings	10s	10s
Licensing/Flexibility Benefits	10s	10s
Enhanced Power Uprate Potential	100s (NPV)	100s (NPV)

Note: Power uprate potential represents Net Present Value (NPV) of additional revenue over remaining plant life, discounted at 7%. This benefit is independent of physical system modifications and can be realized through analytical changes alone, making it particularly relevant for existing plants.

In addition, a construction-schedule acceleration of 6–24 months could provide financing-cost benefits that can exceed direct capital-cost reductions, particularly for projects with high capital costs and financing charges.[v]

As mentioned previously in Section 3, the AP1000 design provides demonstration of the benefits available through systematic elimination of non-credible event protection (Schultz 2006). The design achieved 80% reduction in safety-piping length compared to traditional PWR designs, complete elimination of pipe-whip restraints throughout the plant, 35% fewer pumps and associated support systems, and optimized containment design that enables significant construction and operational benefits. These achievements demonstrate that alternatives to DEGB protection can maintain safety while enabling substantially more-efficient nuclear plant designs.

4.3.1. Power Uprate Margin Recovery: Quantitative Impact for Existing Fleet

For the existing U.S. nuclear fleet, the most-significant economic benefit from DEGB elimination may be recovery of margin for power uprates. Current 10 CFR 50.46 analyses show that DEGB scenarios consume the majority of available margin to the 2200°F peak cladding temperature (PCT) limit, typically yielding calculated PCTs of 1850–2000°F and leaving only 200–350°F of remaining margin. Meeting these design limits often requires restrictions on fuel power peaking that result in higher fuel costs.

- Representative Case Analysis: A typical large PWR with DEGB PCT of 1870°F and small-break LOCA PCT of 1167°F demonstrates that DEGB consumes approximately 700°F of the available margin (from the 2200°F limit). This margin consumption directly constrains achievable power uprates because any power increase raises both initial stored energy and decay heat.
- Economic Significance: Industry experience indicates that power-uprate feasibility is often limited by ECCS/LOCA analyses rather than by plant equipment capacity. A modest 2% power uprate for a 1,000 MWe plant generates approximately $8 million in additional annual revenue ($50/MWh, 92% capacity factor), while a 5% uprate generates $20 million annually. Over a 20-year remaining license

[v] In some cases, the daily interest costs on under construction plants can be as high as $1 million per day.

period, discounted at 7%, a 5% uprate that would be precluded by DEGB margin constraints represents approximately $212 million in lost value per plant.

- Fleet-Wide Impact: Of the 92 currently operating U.S. reactors, approximately 40–50 units could potentially pursue measurement-uncertainty recapture or stretched uprates if DEGB margin consumption were eliminated. This represents potential cumulative fleet value of $4–10 billion in NPV terms, far exceeding the operational cost savings identified in previous analyses.

This economic benefit is realizable today for existing plants without requiring removal of physical protection systems, making it particularly relevant to regulatory decision-making for the current fleet.

4.4. Resource Allocation and Competitive Implications

The economic resources devoted to DEGB protection represent opportunity costs that could be deployed more effectively for genuine safety improvements or competitive enhancements that support continued nuclear operation. The billions of dollars spent on protection against non-credible events could instead support enhanced aging-management programs that address realistic degradation mechanisms, advanced monitoring and diagnostic systems that provide improved operational awareness, cybersecurity improvements that address evolving threats to nuclear facilities, research and development investments that support advanced-reactor technologies and improved operational efficiency, or value that is returned to shareholders to make nuclear more competitive (NEI 2021).

The preservation of existing nuclear capacity provides substantial environmental and energy-security benefits that justify regulatory-reform efforts even without considering new construction applications. Recent plant closures, including Indian Point, Pilgrim, and Duane Arnold, demonstrate the economic pressures facing nuclear plants in competitive energy markets although that trend is now turning towards the restart of shuttered or unfinished plants. Cost reductions enabled by elimination of non-credible-event protection could help ensure continued operation of existing nuclear plants that might otherwise face premature closure and the restarting of formerly closed plants, thereby preserving clean, baseload energy-generation capacity while maintaining employment and economic activity in nuclear communities.

The economic stakes are particularly significant for aging facilities, as demonstrated by the 2018 closure of Oyster Creek Nuclear Generating Station, which was the oldest operating U.S. reactor (commissioned in 1969) before its retirement made Nine Mile Point, Unit 1, the new record holder (U.S. NRC 2025c). These aging plants face intensified economic pressures; every additional cost impacts the viability of continued operation versus premature closure. For facilities like Nine Mile Point, Unit 1, which has already received license extension through 2029, the economic burden of maintaining DEGB protection systems designed for non-credible events becomes increasingly difficult to justify when weighed against realistic operational needs and competitive market pressures.

The broader implications for nuclear competitiveness extend beyond direct cost considerations to encompass innovation capability and technological advancement. The allocation of resources toward non-credible-event protection undermines the nuclear industry’s ability to invest in genuine improvements that could enhance both safety and competitiveness. International experience demonstrates that countries implementing more-efficient regulatory approaches achieve better cost and schedule performance in nuclear construction, suggesting that regulatory reform could position the United States as a more-competitive participant in global nuclear markets.

However, economic analysis must also acknowledge that transition costs and implementation challenges could offset some potential benefits from DEGB elimination. Existing plants face complex retrofit considerations that may limit achievable savings while new plant designs require integrated optimization from initial stages to realize full potential benefits. Regulatory-approval processes for alternatives to established requirements may involve additional licensing costs and schedule uncertainties that affect project economics.

5. LIMITATIONS, UNCERTAINTIES, AND AREAS FOR CONTINUED RESEARCH

While accumulated evidence provides substantial support for reassessing DEGB requirements, limitations and uncertainties merit careful consideration in policy development. These encompass potential technical gaps in current understanding, implementation challenges, and evolving operational conditions that could affect assessment validity.

As mentioned in Section 2, long-term aging effects represent the most-significant potential knowledge gap although operational experience from plants like Nine Mile Point, Unit 1, (operating since 1969, over 55 years) and the U.S. fleet's average age of 41.6 years provide considerable empirical data (U.S. NRC 2025a, U.S. NRC 2024). However, extended operation to beyond 80-year lifetimes may introduce degradation mechanisms not fully captured by current analysis (NUREG 1800, NUREG-1801, NUREG-2191, NUREG-2192). The potential for synergistic effects between multiple degradation mechanisms creates additional uncertainty because current fracture-mechanics models typically analyze SCC, fatigue cycling, and thermal aging in isolation (Chopra 2016). Research on simultaneous mechanism interactions remains limited for extended operating periods.

Site-specific factors create uncertainties affecting probability estimates and failure assessments. Seismic hazards (see Section 2 for a discussion on seismic hazards) vary dramatically across plant locations. Materials evolution presents both opportunities and uncertainties because advanced nuclear materials with enhanced properties could further reduce failure probabilities, but they may have different aging characteristics, requiring continued validation.

Implementation faces several practical challenges that affect achievable benefits and transition strategies. Current regulatory frameworks provide limited standardized pathways for DEGB alternatives, requiring case-by-case approvals. These create licensing burden (NUREG-0800). Existing plants face complex retrofit considerations that may limit economic benefits from DEGB elimination, particularly where removing protection systems would require extensive building modifications. However, it is important to recognize that analytical adjustments can be made to realize value without physical modifications, as noted in Section 4.3.1. Stakeholder acceptance represents another challenge; safety advocates may view requirement changes as reductions to safety margins despite technical evidence while public confidence depends partly on perceived regulatory conservatism. Acceptance requires careful communication and technical justification.

The current operational database, while extensive, represents primarily Generation II reactor experience under specific conditions (OECD NEA 2024). Advanced-reactor technologies may have different materials, operating conditions, or system designs that affect failure characteristics. Geographical and temporal distribution creates potential biases because documented experience comes largely from plants in stable condition with well-developed maintenance programs. Future deployment may occur in different environments affecting failure rates or detection capabilities. The evolution of inspection technologies means earlier experience may not reflect current capabilities to detect and mitigate problems (ASME BPVC V 4).

Research priorities should focus on understanding long-term aging mechanisms and synergistic degradation effects for extended plant operation and advanced-reactor technologies operating under conditions different from current plants (Chopra 2016). Probabilistic methodologies require development to incorporate evolving understanding of aging effects and external hazards, including climate-change impacts. Under GDC 30 in 10 CFR 50, Appendix A, nuclear power plants must ensure the reactor-coolant pressure boundary is designed and maintained to the highest practical quality standards. This includes provisions for leak-detection capability. Specifically, plants are required to have systems that can detect and identify leaks of reactor coolant at a rate per the technical specification limiting condition for operation requirements within a specified time frame. RG-1.145 provides NRC-accepted guidance on types of leakage-detection systems, redundancy and diversity, and expected sensitivity ranges. It historically supports detection capability of ~1 gal/minute unidentified leakage for PWRs. The intent is to identify promptly abnormal leakage before it challenges core cooling or containment integrity, thereby supporting fracture prevention and maintaining compliance with the operational limits of RG-1.147.

Enhanced monitoring technologies—including advanced leak detection, continuous structural-health monitoring, and predictive maintenance—could provide real-time integrity assessments while reducing dependence on conservative assumptions (EPRI 2015, Ferdinand 2014). Modern leak detection uses multiple, redundant methods, including high-precision reactor-coolant system mass balance, containment-sump level monitoring, humidity and temperature monitoring, radiation monitors, etc. These methods can detect leaks from ~0.05 to 0.5 gal/minute, which can be up to an order of magnitude smaller than the leak-detection requirement. These modern detection techniques further support LBB viability. Collection of operational experience should expand to encompass advanced designs and extended conditions, with standardized databases facilitating international cooperation (OECD NEA 2024).

6. POLICY FRAMEWORK AND IMPLEMENTATION CONSIDERATIONS

The accumulated evidence regarding DEGB credibility creates an opportunity for thoughtful regulatory evolution that maintains safety while optimizing resource allocation and enhancing nuclear competitiveness. Successful implementation requires careful consideration of policy-development approaches, stakeholder-engagement strategies, and transition mechanisms that advance evidence-based regulation while respecting legitimate stakeholder interests. Recently, the NRC issued draft Interim Staff Guidance (ISG), NRC-DSS-ISG-2025-XX, "Treatment of Certain Loss-of-Coolant Accident Locations as Beyond-Design-Basis Accidents," for public comment in the Federal Register, proposing the treatment of certain LOCAs—such as a DEGB—as beyond design-basis events (U.S. NRC 2025d). The draft ISG proposes a case-by-case exemption process for treating certain large-break LOCAs as beyond-design-basis. While this represents progress toward risk-informed regulation, we recommend that the NRC pursue generic rulemaking rather than plant-specific exemptions for the following reasons:

1. Technical Basis is Generic: The evidence supporting DEGB elimination (operational experience, materials science, probabilistic analysis) applies across the fleet, not to individual plants
2. Administrative Efficiency: Generic rulemaking avoids duplicative technical review for each exemption request
3. Regulatory Certainty: Clear generic requirements provide better guidance for new-reactor licensing and existing plant modifications
4. International Alignment: Generic rulemaking would align U.S. practice with international precedents (Germany, France, Korea, UAE)
5. Precedent Alignment: Generic rulemaking is consistent with how the NRC has historically addressed fleet-wide technical issues where the underlying basis applies broadly (e.g., digital instrumentation and control, fire protection, station blackout).

The compensatory measures proposed in the draft ISG—while well-intentioned—are not well-aligned with the actual risk profile demonstrated by five decades of operating experience. The evidence suggests that standard in-service inspection programs and leak-detection systems already provide adequate defense-in-depth without additional compensatory measures specifically targeted to DEGB scenarios.

The reassessment of DEGB should be conducted within established risk-informed regulatory frameworks that recognize both benefits and limitations of current understanding (Regulatory Guide 1.174). The NRC's risk-informed decision-making (RIDM) framework provides a systematic methodology to incorporate operational experience, probabilistic analysis, and deterministic engineering judgment into regulatory decisions (NUREG-2150). The RIDM process explicitly addresses scenarios where traditional conservative assumptions may not reflect realistic risk profiles, establishing precedent for evidence-based regulatory evolution while maintaining defense-in-depth principles.

Comprehensive alternatives to DEGB requirements necessitate coordinated revision across multiple regulatory domains to ensure consistent implementation. The ECCS requirements in 10 CFR 50.46 require revision to eliminate DEGB as the upper bound for system design while maintaining protection for credible accidents, establishing leak-area limitations based on mechanistic analysis, as does Germany's 10% pipe cross-sectional-area approach (10 CFR 50.46, RSK 1981). Structural protection requirements in GDC 4 should formalize LBB as a primary method through streamlined review procedures and standardized methodologies (10 CFR 50, APP A GDC4). Containment design and environmental qualification requirements merit updates to reflect realistic energy-release scenarios, rather than non-credible post-DEGB conditions (10 CFR 50, APP A GDC50, 10 CFR 50.49).

Implementation should employ graduated approaches that validate alternative methodologies while managing transition risks. Phase 1 could streamline existing LBB processes and develop standardized criteria building upon all the PWR units with approved applications (U.S. NRC 2018). Phase 2 could conduct risk-informed evaluation across regulatory domains with economic-impact assessment (NUREG-1061). Phase 3 could integrate findings into broader risk-informed frameworks that support advanced-reactor deployment (10 CFR 53). Throughout all phases, continuous monitoring should track safety performance, economic efficiency, and stakeholder confidence while ensuring responsiveness to evolving technical understanding.

Stakeholder engagement should address legitimate concerns while building confidence in evidence-based evolution. The engagement strategy should emphasize transparent communication that acknowledges uncertainties while leveraging substantial operational experience and international precedents. German break-preclusion implementation over four decades and successful Vogtle Units 3 and 4 LBB applications provide compelling precedents demonstrating that alternatives maintain public confidence while providing economic benefits (Schulz 1997, U.S. NRC 2013).

The reassessment offers particular opportunities for advanced-reactor technologies facing economic challenges under current regulatory frameworks (Mignacca and Locatelli 2020, Stewart and Shirvan 2023). The proposed 10 CFR 53 framework explicitly embraces risk-informed approaches that could naturally accommodate DEGB reassessment principles. Part 53’s emphasis on technology-inclusive design criteria, performance-based requirements, and integrated risk-assessment methodologies provides a regulatory template for eliminating non-credible event protection while maintaining safety margins through functional requirements rather than specific hardware (10 CFR 53). The Part 53 risk approach allows systematic application of break-preclusion concepts across diverse reactor technologies, including small modular reactors and gas-cooled, liquid-metal, and molten-salt systems (U. S. DOE 2021).

6.1. Maintaining Safety Culture Through Evidence-Based Regulation

Some stakeholders may be concerned that eliminating DEGB requirements could be perceived as reducing safety margins or compromising safety culture. This concern warrants a clear and focused response.

First, evidence-based regulation strengthens rather than weakens safety culture by demonstrating that regulatory requirements are grounded in technical reality rather than legacy-driven conservatism. A safety culture that cannot adapt to new evidence is fragile, not robust.

Second, the resources currently devoted to DEGB protection could be reallocated to address actual safety challenges—including aging management for extended operation, enhanced cybersecurity capabilities, and severe-accident mitigation—potentially yielding greater safety benefit per dollar invested. Alternatively, cost reductions could improve plant economics by supporting continued operation of at-risk units and enabling investment in fleet modernization.

Third, international experience demonstrates that countries eliminating DEGB requirements (Germany, France for advanced reactors) maintain exemplary safety records, suggesting that evidence-based regulation does not compromise safety performance.

Finally, the technical community's confidence in LBB methodology—validated through decades of successful application—reflects mature understanding of piping integrity, rather than an erosion of conservative principles. Sound engineering judgment distinguishes between appropriate conservatism (defense-in-depth, margin) and non-technical conservatism (protection against physically implausible scenarios).

7. CONCLUSIONS AND STRATEGIC RECOMMENDATIONS

The accumulated evidence from five decades of nuclear operation provides compelling justification to reassess DEGB regulatory requirements through evidence-based policy evolution. The complete absence of documented DEGB events across more than 20,000 reactor-years of global experience—combined with probabilistic assessments indicating extremely low direct-failure probabilities and material science that demonstrates stable crack-growth behavior incompatible with instantaneous severance—suggests that current requirements impose economic burdens disproportionate to realistic safety threats.

The operational experience and material science provide sufficient evidence that current regulatory assumptions do not reflect realistic piping-failure modes. International databases consistently document LBB behavior across all recorded failure mechanisms, with no instances of instantaneous circumferential severance despite comprehensive monitoring systems that would guarantee detection of such events. This empirical foundation is reinforced by successful international implementation of alternatives, particularly Germany's break-preclusion approach over four decades and the United States' LBB applications in almost all PWR units, demonstrating that sound alternatives can maintain safety performance while reducing economic burden. Although uncertainties remain relevant to policy development, the inclusion of DEGB as a design-basis event is no longer grounded in current technical understanding. For example, seismic pathways for DEGB occurrence show the potential for higher probabilities, but current data for plants in seismically active regions is trending towards lower probabilities. Probability calculations should be reevaluated using the current data and include both inspections and material behavior.

The economic implications are substantial, with current DEGB-protection systems imposing costs of hundreds of millions of dollars per plant and totaling billions globally for non-credible-event protection. The NRC's own documentation confirms "hundreds of millions in backfit costs saved" through LBB implementation. Eliminating non-credible event protection could:

- Reduce new plant costs, including ongoing costs, by hundreds of millions of dollars per unit
- Potentially accelerate construction schedules by 6–24 months
- Reallocate resources toward genuine safety improvements
- Support continued operation of economically pressured plants
- Enable modest but impactful power uprates for existing plants.

These benefits could redirect resources toward enhanced aging-management programs or advanced-reactor development that provides greater safety and competitive advantages.

Based on five decades of operating experience, an established technical-evidence base, and widespread regulatory acceptance of methodologies such as LBB and in-service inspection programs, the probability of a DEGB in reactor coolant-loop piping is extremely low under all plant conditions, including seismic events. This evidence demonstrates that the DEGB scenario is not a realistic or necessary basis for modern plant-safety design. We recommend that DEGB be removed from design-basis requirements and, where addressed at all, be treated in the same category as other extremely low-probability events (frequency $<10^{-6}$/year) that do not drive deterministic design requirements. This is consistent with the frequency-consequence framework adopted for advanced reactors in NEI 19-04/ RG 1.233, where events with frequencies below 10^{-4}/year are not considered design-basis events requiring deterministic compliance demonstration. Furthermore, this position aligns with the intent of DSS-ISG-2025-XX, but further recommends codifying the removal of the DEGB assumption for large-break LOCA requirements where technically justified, rather than relying solely on exemptions.

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