RULEMAKING ISSUE
NOTATION VOTE

July 23, 2001 SECY-01-0133

FOR: The Commissioners

FROM: William D. Travers
Executive Director for Operations

SUBJECT: STATUS REPORT ON STUDY OF RISK-INFORMED CHANGES TO THE TECHNICAL REQUIREMENTS OF 10 CFR PART 50 (OPTION 3) AND RECOMMENDATIONS ON RISK-INFORMED CHANGES TO 10 CFR 50.46 (ECCS ACCEPTANCE CRITERIA)

PURPOSE:

To provide the fourth status report on the staff's study of possible risk-informed changes to the technical requirements of 10 CFR Part 50, and to specifically provide the staff's recommendations for risk-informed changes to 10 CFR 50.46 ("Acceptance Criteria for Emergency Core Cooling Systems (ECCS) for Light-Water Nuclear Power Reactors").

SUMMARY:

The staff has developed recommendations for Commission consideration on risk-informed changes that can be made to 50.46. The staff recommends: (a) modification of the existing 50.46 to change the ECCS acceptance criteria and the Appendix K ECCS evaluation model; and (b) development of a voluntary risk-informed alternative to 50.46, Appendix K and General Design Criterion (GDC) 35 that will change the ECCS reliability requirements. Additional technical work, described in this paper, will be needed to support implementation of the recommendations.

The staff believes that additional changes to 50.46 may be merited. These changes which relate to the scope of pipe break sizes relevant to 50.46, require further technical evaluation and thus more time. The staff intends to continue its assessment of the feasibility of such changes. This feasibility study could require significant staff and industry resources, but could result in considerable reduction of unnecessary regulatory burden. If found feasible, a separate rulemaking would be required.

CONTACT:
Mary T. Drouin, RES
415-6675

BACKGROUND:

In a June 8, 1999, staff requirements memorandum (SRM) on SECY-98-300, the Commission approved proceeding with a study of risk-informing the technical requirements of 10 CFR Part 50. The staff provided its plan and schedule for this work in SECY-99-264, “Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50,” dated November 8, 1999. The plan describes two phases to the staff’s work. Phase 1 is an evaluation of the feasibility of risk-informed changes and results in recommendations to the Commission on proposed rulemaking. Phase 2 is an implementation phase, consisting of rulemaking (based on recommended changes resulting from Phase 1 and approved by the Commission), and performing needed technical analyses. The Commission approved proceeding with this plan in a February 3, 2000, SRM. Since that time, the staff has provided:

- First status report, SECY-00-0086 (“Status Report on Risk-Informing the Technical Requirements of 10 CFR Part 50 (Option 3)”), which provided the staff’s framework for risk-informing the technical requirements of Part 50. The framework document provides the guidelines that the staff is applying in reviewing, formulating, and recommending risk-informed alternatives to the technical requirements of Part 50.

- Second status report, SECY-00-0198 (“Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control)”), which provided recommendations on a risk-informed alternative to 10 CFR 50.44 (“Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors”) and policy issues. In a January 19, 2001 SRM, the Commission directed the staff to proceed expeditiously with rulemaking regarding a risk-informed alternative to 50.44 and to review the resource estimates associated with the overall Option 3 effort. Staff work has begun with respect to 50.44; this will be the subject of a separate paper. This fourth status paper discusses resources associated with Option 3.

- Third status report, memorandum to the Commission (“Third Status Report on Risk-Informing the Technical Requirements in 10 CFR Part 50 (Option 3)”) dated February 5, 2001, in which the staff indicated that preliminary recommendations and a detailed plan and schedule on the feasibility of risk-informed changes to 50.46 would be provided to the Commission in the next status report.

This fourth status paper includes recommendations for rulemaking on 50.46 and a plan and schedule for the technical work needed to support the rulemaking. The staff expects that a proposed rule can be provided to the Commission within 12 months of Commission approval of the staff’s recommendations discussed in this paper (i.e., receipt of the SRM in response to this paper).

DISCUSSION:

Since the third status report in February 2001, the staff’s Option 3 work has involved:

- completing a feasibility assessment with respect to specific changes to 50.46 and the development of recommendations for rulemaking,

- continuing a feasibility assessment of additional possible changes to 50.46,

- continuing with other Option 3 activities (e.g., assessing the feasibility of expanding the single failure criterion beyond ECCS), and

- meeting with stakeholders to obtain their input on these activities (including updating the staff's framework to reflect comments received in public meetings and from the Advisory Committee on Reactor Safeguards (ACRS)).

Feasibility Assessment of Changing 10 CFR 50.46 and Recommendations for Rulemaking

The staff's feasibility assessment of possible changes to 50.46 included an evaluation of the current requirements, their basis and evolution; a review of related regulations and implementing documents; a review of risk information relevant to 50.46 and related accidents; development and comparison of potential options for risk-informing current requirements; and development of recommendations for changes. The staff's feasibility assessment is provided in Attachment 1; a summary is provided below.

The technical requirements of 50.46 and the related regulations (i.e., GDC 35, "Emergency Core Cooling" and Appendix K, "ECCS Evaluation Models") call for an ECCS for postulated loss-of-coolant accidents (LOCAs). These requirements are grouped into four technical areas:

- <u>ECCS reliability.</u> The ECCS is designed to codes and standards applicable to safety-related systems, and is designed to be reliable by the application of the single failure criterion and specifications on offsite power availability. More specifically, the system is designed to meet specified functional requirements with an assumed single failure and an assumed loss of offsite power simultaneous with the LOCA.

- <u>ECCS acceptance criteria.</u> Calculated parameters such as peak cladding temperature, total cladding oxidation, and maximum hydrogen generation are used as metrics for comparison to the specified criteria.

- <u>ECCS evaluation model.</u> Appendix K of Part 50 describes the most commonly used method for evaluating ECCS performance.

- <u>ECCS spectrum of break sizes and locations.</u> This spectrum includes 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 staff believes that changes to the first three technical areas above may be justified and the last is potentially feasible. More specifically, the staff believes that the ECCS reliability resulting from the current technical requirements is not commensurate with the risk significance of the various LOCA sizes and that unnecessary conservatisms exist in the requirements.

Observations and conclusions that support this staff position for each technical area include:

- ECCS reliability. Current ECCS reliability requirements may be overly conservative for large-break LOCAs.

- ECCS acceptance criteria. Use of a performance-based requirement rather than the current prescriptive ECCS acceptance criteria would allow use of cladding materials other than zircaloy or ZIRLO without licensees having to submit an exemption request.

- ECCS evaluation model. Current evaluation models of ECCS performance may be overly conservative for all LOCAs.

- ECCS spectrum of break sizes and locations. Given current estimates of the frequency of large-break LOCAs (NUREG/CR-5750 indicate 95th percentile values of 10^{-5} per critical year for pressurized water reactors [PWRs] and 10^{-4} per critical year for boiling water reactors), the reliability of the ECCS (and containment functions) is generally sufficient to assure that large-break LOCAs (> 6 inches in diameter) are not significant contributors to risk. However, the current estimates of large-break LOCA frequencies are uncertain and are not low enough to allow elimination of all large-break LOCA sizes from the design bases. In addition, plant equipment that is designed, at least in part, to the requirements of design-basis LOCAs also provides defense against a spectrum of beyond-design-basis accidents.

Based on the above analysis, the staff recommends (A) changes to the technical requirements of the current 50.46 related to acceptance criteria and evaluation model, and (B) development of a voluntary risk-informed alternative to the reliability requirements in 50.46. In developing the proposed rule(s)[1] for these two recommendations, the staff will follow the guidelines in its Option 3 framework. The framework is designed to ensure that changes are risk-informed, and include consideration of defense-in-depth principles.

(A) Changes to the Current 10 CFR 50.46

The staff recommends that rulemaking should be undertaken to change the current 50.46.[2] These changes would include:

1. Replace the current prescriptive ECCS acceptance criteria in 50.46 with a performance-based requirement. This requirement would, one, demonstrate adequate post-quench cladding ductility and adequate core-coolant flow area to ensure that the core remains amenable to cooling, and, two, for the duration of the accident, maintain the calculated core temperature at an acceptably low value and remove decay heat. AND Use of a performance-based requirement rather than the current prescriptive criteria would allow use of cladding materials other than zircaloy or ZIRLO without licensees having to submit an exemption request.

---

[1]It is not yet clear if it would be more effective and efficient to make the recommended changes in one or two rulemakings. This will be clarified as staff performs technical work and prepares to begin rulemaking.

[2]The recommendations provided here are limited to changes in the ECCS requirements. There are additional changes, not yet studied, that could “spin-off” including, for example, changes to requirements for containment design or equipment qualification (EQ). The staff’s recommendations provided in this paper presume that other requirements (beyond those for the ECCS) remain unchanged until proper study has been performed.

2. Revise the requirements for the ECCS evaluation model to be based on more realistic analyses. In the near term, this revision would involve an update of Appendix K requirements based on more current and realistic information. Specifically this update could involve:
   a. replacing the current 1971 American Nuclear Society (ANS) decay heat curve with a model based on the 1994 ANS standard.
   b. replacing the current decay heat multiplier of 1.2 with an NRC-prescribed uncertainty treatment.
   c. deleting the limitation on PWR reflood steam cooling for small reflood rates.
   d. replacing the Baker-Just zirconium steam model with the Cathcart-Pawel zirconium steam oxidation model for heat generation.
   e. deleting the prohibition on return to nucleate boiling during blowdown.

   As part of this update, the staff will also consider the recognized nonconservatisms and model limitations to insure that proper safety focus is incorporated in any new rule.

The recommended rulemaking is based on a feasibility study, and additional technical work is required to support the actual rule changes. Attachment 2 provides a detailed discussion of the needed technical work; in summary, the staff will undertake work to:

- support removal of unnecessary conservatisms from Appendix K.

- develop guidelines for demonstrating adequate post-quench ductility as a replacement for the current prescriptive acceptance criteria, including specified peak cladding temperature and total cladding oxidation limits.

- support development of the regulatory guides needed for implementing the modifications to the existing rule.

This technical work is estimated to take approximately 12 months and up to 1.3 FTE and cost about $350k. The staff will continue to perform the technical work needed to support the rulemaking: However, the staff will begin developing the related proposed rulemaking upon Commission approval (i.e., receipt of the SRM), and expects to provide this proposed rulemaking within 12 months. This activity is estimated to take up to 2.3 FTE. The rulemaking will ensure that the approach taken considers backfit implications.

The staff believes that outcomes of this rulemaking will be that safety will be maintained, NRC activities and decision-making will be more effective, efficient and realistic, and unnecessary regulatory burden will be reduced, and that public confidence will also be maintained if we effectively communicate how safety will be maintained. With respect to the unnecessary burden reduction, industry estimates (letter from R. Bryan, Westinghouse Owner's Group (WOG), to T. King, NRC, dated October 17, 2000) indicate that this rulemaking could result in savings of $100K/year/unit to $3100K/year/unit, depending on the specific plant and scope of changes.

In addition, this rulemaking would address a petition for rulemaking (PRM-50-71) submitted by the Nuclear Energy Institute (NEI) on April 12, 2000. NEI has requested that NRC amend its regulations in 50.44 and 50.46 to allow nuclear power plant licensees to use zirconium-based cladding materials other than zircaloy or ZIRLO, provided the cladding materials meet the requirements for fuel cladding performance and receive approval by the NRC staff. This objective, with respect to 50.46, would be accomplished by the changes to the ECCS acceptance criteria recommended above.

(B) Development of a Risk-Informed Voluntary Alternative to 10 CFR 50.46

The staff recommends that rulemaking should be undertaken to develop a risk-informed alternative to the current 50.46.[3] This alternative would be voluntary on the part of licensees and would include technical requirements to ensure an ECCS reliability that is commensurate with the frequency of challenge to systems. This revision would replace the current approach for obtaining ECCS reliability with more risk-informed and realistic approaches. In place of the simultaneous loss of offsite power requirement and single failure criterion, two options would be offered to accomplish ECCS system reliability (further explanation of these options is provided in Attachment 2):

1. A deterministic system reliability requirement based on risk information (e.g., an ECCS design requirement that only one train of ECCS is required for LOCAs larger than a specified size). OR

2. An ECCS functional reliability requirement that is commensurate with the LOCA frequency (e.g., a requirement that ECCS design must be such that the core damage frequency [CDF] associated with a specified set of LOCAs is less than an NRC-specified CDF threshold, with due consideration of uncertainties).

This recommended rulemaking is also based on a feasibility study, and additional technical work is required to support the actual rule changes. Attachment 2 provides a detailed discussion of the needed technical work. In summary, the staff will undertake work to:

- determine acceptable methods and assumptions for performing LOCA CDF and ECCS reliability analyses for those alternatives requiring such analyses, including evaluation of uncertainties. In addition, appropriate reliability and CDF threshold values would have to be selected.

- further examine the likelihood of loss of offsite power following a LOCA, and to determine acceptable methods and assumptions for estimating plant-specific probability of loss of offsite power given a LOCA.

- support development of the regulatory guides needed for implementing the recommended risk-informed alternative rule.

This work is estimated to take approximately 9 months and up to 0.5 FTE and cost about $350K. The staff will continue to perform the technical work needed to support the rulemakings. The staff will begin developing the related proposed rulemaking upon Commission approval (i.e., receipt of the SRM), and expects to provide this proposed rulemaking within 12 months.

The staff believes that the outcomes of this rulemaking will be that safety will either be maintained or enhanced; NRC activities and decision-making will be more effective, efficient

[3]As noted before, the recommendations provided here are also limited to changes in the ECCS requirements. Additional changes, not yet studied, could also “spin-off (see Footnote 2).”

and realistic; and unnecessary regulatory burden will be reduced; and that public confidence will also be maintained if we effectively communicate how safety will be maintained. With respect to the unnecessary burden reduction, industry estimates (letter from R. Bryan, WOG), indicate that this risk-informed alternative could result in additional savings of $400K/year/unit up to $1200K/year/unit, depending on the specific plant and scope of changes.

Feasibility Assessment of Additional Changes to 10 CFR 50.46

The staff believes that additional changes to 50.46 may also have merit, and will continue to perform the technical work to determine its feasibility. More specifically, the extent of potential change to 50.46 is dependent on the state-of-knowledge of the frequency of LOCAs of various break sizes. For example, if a set of LOCAs can be demonstrated to have a collective mean frequency of occurrence of below 10^{-4} per year (/yr), some regulatory relief may be appropriate in terms of the level of conservatism and redundancy required in the design. If a set of LOCAs can be demonstrated to have a collective mean frequency of occurrence of below 10^{-5}/yr, it may be appropriate to remove these LOCAs from the plant design basis, as long as some mitigative capability remains in the plant, e.g., there is an expectation of success under accident management. Lastly, if a set of LOCAs can be demonstrated to have a collective mean frequency of occurrence of below 10^{-6}/yr, it may be appropriate to remove these LOCAs from the plant design basis. Attachment 2 provides a detailed discussion of the needed technical work.

With the current state-of-knowledge on the frequencies of large breaks, the staff believes that some changes can now be made. These changes are included in the rulemakings recommended above. The staff plans to continue to improve the state-of-knowledge of LOCA frequencies and to continue to assess the feasibility of further changes to 50.46. As part of this, the staff will continue to meet with representatives of the nuclear industry in public meetings to address a set of technical issues. These issues include, for example, initial flaw distributions, degradation mechanisms, material response and uncertainty analysis (see Appendix A of Attachment 2 for more detail). Resolution of the technical issues will be pursued as part of the staff's feasibility assessment. If found feasible, the staff will recommend additional changes, potentially including rulemaking to change the wording in 50.46 and Appendices A and K of Part 50 which would allow the licensee to use an alternate pipe size, subject to some level of NRC approval.

Resource requirements for this feasibility study, i.e., to support a rigorous analysis of LOCA frequencies, could be significant. It is estimated to take 2-3 years and up to 2.4 FTE and cost about $1.2 million. With respect to the potential benefit, industry estimates[4] indicate that this redefinition of a large-break LOCA could potentially result in a total savings to industry in the range $100K/year/unit up to $5500K/year/unit and a potential one-time savings up to $8300K (for baffle barrel bolt replacement), depending on the specific plant and scope of changes (and the degree to which previous changes were implemented).

---

[4]Letter from R. Bryan, WOG; and WOG letter from L. Liberatori, Jr., to T. Essig (NRC), "Westinghouse Owners Group: NRC Review of WCAP-14748 (Proprietary), Revision 0, and WCAP-14749, Revision 0 (Non-Proprietary), 'Justification of Increasing Postulated Break Opening Times in Westinghouse Pressurized Water Reactors'," December 10, 1988.

Other Option 3 Activities

GDC 35 requires that the ECCS safety function be accomplished assuming a single failure. As indicated above, the staff recommends replacing this single failure criterion in its recommended alternative rule, but only as it affects ECCS. However, the single failure criterion, as discussed in Appendix A of Part 50, is applied to more than just the ECCS. GDCs 17, 34, 38, 41 and 44 also contain the single failure criterion. In addition, the footnote to the definition also states that *"Single failures of passive components in electric systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development."*

The staff believes that a generic change to the Part 50 Appendix A single failure criterion definition may be warranted and intends to assess the feasibility of a single generic change under Option 3. This could be a very significant change to Part 50, so the feasibility study will include careful consideration of implications. The Option 3 framework will be used, as well as internal and external stakeholders meetings, to ensure that fundamental regulatory principles are not inadvertently compromised. Such a risk-informed definition would also address the Commission's guidance in the SRM of February 3, 2000 which stated *"the staff should also review safety issues noted in Part 50 as being 'under consideration' or 'under development,' e.g., ... failure of passive components..., as discussed in footnotes ... 2 to the Definitions and Explanations of Appendix A, and consider their resolution."* As discussed below, the staff will reassess the priority of this work late this year.

The staff, as part of Option 3, has also begun to investigate changes to the special treatment technical requirements of Part 50. The staff has deferred further work on this to better focus its resources on assessments of 50.44 and 50.46, but will reassess its priority late this year.

Stakeholder Communication:

The staff has held several meetings with stakeholders to communicate, receive feedback on, and enhance public confidence in the technical merit of the staff's work. In addition, the staff has had several discussions with the ACRS (both the sub- and the full committee) and plans to continue to meet with them on a regular basis. The staff has continued to maintain the interactive Web site[5]. As information is ready for stakeholder review, it is posted to this Web page (and placed in the public document room for those who do not have internet access).

The principal stakeholder feedback has included comments on the need to complete 50.44 and 50.46 rulemakings and various comments on the framework, which have been incorporated into the version used in the assessment of 50.46, and owner's groups' input with respect to redefining the large-break LOCA design basis accident. An updated version of the framework addressing stakeholder, including ACRS, comments will be issued in August 2001.

---

[5] The Web site is accessed via the NRC Web site under the Nuclear Reactors icon, and then selecting the "Risk-Informed Part 50 Initiatives" line item, followed by the "Risk-Informed Part 50, Changes to Technical Requirements (Option 3)" line item.

RESOURCES:

It is the staff's intent that the first priority for the Option 3 resources for FY 2002 and FY 2003 will be to complete the technical work and rulemaking support for 50.44 and the recommended 50.46 rulemaking(s).[6]

While the staff priority is to focus efforts on 50.44 and 50.46 (and 50.61), input from stakeholders will be solicited at a workshop towards the end of this calendar year on the priority and schedule for additional candidate regulations to be risk-informed. This discussion of priorities will include the application of the revised single failure criterion to all of Part 50 and the modification of special treatment requirements.

Staff resources for proceeding with rulemaking(s) on 50.46 and the associated technical work, and for completing the recommended longer-term feasibility study on additional changes to 50.46, are estimated as follows:

Staff Activity	Schedule	Budget
Perform rulemaking to change 50.46 to replace current prescriptive ECCS acceptance criteria and revise requirements for evaluation model		
• Develop proposed rule (NRR)	12 months from date of SRM or 2 months after completion of technical work (whichever is later)	2.3 FTE, $0K [FY01: 0.3 FTE FY02: 2.0 FTE]
• Perform supporting technical work (RES)	On or before July 2002	Up to 1.3 FTE, $350K [FY01: 0.3 FTE; $150K FY02: 1.0 FTE; $200K]
Perform rulemaking to develop voluntary alternative requirements to ensure ECCS reliability commensurate with frequency of challenge		
• Develop proposed rule (NRR)	12 months from date of SRM or 2 months after completion of technical work (whichever is later)	2.3 FTE, $0K [FY01: 0.3 FTE FY02: 2.0 FTE]
• Perform supporting technical work (RES)	On or before April 2002	Up to 0.5 FTE, $350K [FY01: 0.1 FTE; $200K FY02: 0.4 FTE; $150K]
Continue longer-term feasibility assessment on additional changes to 50.46, including rigorous analysis of LOCA frequencies (RES)	Up to 3 years	Up to 2.4 FTE and $1,200K [FY01: 0.4 FTE; $200K FY02: 1.0 FTE; $500K FY03: 1.0 FTE; $500K]

[6] In parallel, the staff is continuing its reevaluation of the technical basis of the Pressurized Thermal Shock rule (10 CFR 50.61) to reflect results of research on reactor vessels, new risk methods development, operational data, and Commission policies. This work, described in SECY-00-0140 and SECY-01-0045, will be completed as a high priority project in FY2002.

These resources are included in the current staff budgets for FY2001 and are included in the budget requests for FY2002 and FY2003.

COORDINATION:

The Office of the General Counsel has reviewed this paper and has no legal objections. The Office of the Chief Financial Officer has reviewed this paper for resource implications and has no objections.

The topic of risk-informed changes to 50.46 has been the subject of continuing interactions between the staff and ACRS. The staff briefed the ACRS on March 16 and June 6, 2001, and modified the report to reflect comments received. The staff will brief the ACRS on the final version of this paper on July 9 and 11, 2001. A letter from ACRS on the staff's paper is expected following the July briefing.

RECOMMENDATIONS:

The staff recommends that the Commission approve proceeding with rulemaking for:

- modification of the existing 10 CFR 50.46 and Appendix K, and
- development of a risk-informed alternative to 10 CFR 50.46, Appendix K and GDC 35.

In order to improve the timeliness of these rulemaking(s), the staff does not intend to prepare a rulemaking plan, but rather to proceed with the technical work supporting the rulemaking(s) and expects to deliver proposed rule(s) to the Commission within 12 months of Commission approval (i.e., receipt of Commission SRM).

Because of the potential benefits of the recommended changes and the high interest by the public, the staff recommends that the Office of the Secretary release this paper to the public 10 days from the date of the paper.

/RA by William F.* Kane *Acting for/

William D. Travers
Executive Director
for Operations

Attachments: 1. Feasibility study on 50.46
2. Technical work to support rulemaking

Attachment 1

FEASIBILITY STUDY OF A RISK-INFORMED ALTERNATIVE TO 10 CFR 50.46, APPENDIX K AND GDC 35

Revision 0

June 2001

TABLE OF CONTENTS

TABLE OF CONTENTS (cont'd)

LIST OF FIGURES

LIST OF TABLES

ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic Depressurization System
AEC	U.S. Atomic Energy Commission
AFW	Auxiliary Feedwater
ANS	American Nuclear Society
AOO	Abnormal Operating Occurrence
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
B&W	Babcock and Wilcox
BTP	Branch Technical Position
BWR	Boiling Water Reactor
CCDP	Conditional Core Damage Probability
CCW	Component Cooling Water
CDF	Core Damage Frequency
CE	Combustion Engineering
CHF	Critical Heat Flux
CLERP	Conditional Large Early Release Probability
CST	Condensate Storage Tank
DBA	Design Basis Accident
DEGB	Double-Ended Guillotine Break
DG	Diesel Generator
DHR	Decay Heat Removal
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EPG	Emergency Procedure Guideline
EPRI	Electric Power Research Institute
ESF	Emergency Safeguard Feature
EQ	Equipment Qualification
FPI	Fast Probability Integration
GDC	General Design Criterion
GL	Generic Letter
GSI	Generic Safety Issue
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
HPSI	High Pressure Safety Injection
HVAC	Heating, Ventilating and Air-Conditioning
IGSCC	Intergranular Stress Corrosion Cracking
IPE	Individual Plant Examination
ISI	In-Service Inspection
LBB	Leak-Before-Break
LBLOCA	Large-Break Loss-of-Coolant Accident
LER	Licensee Event Report
LERF	Large Early Release Frequency
LHS	Latin Hypercube Sampling
LLNL	Lawrence Livermore National Laboratory
LLRF	Large Late Release Frequency
LOCA	Loss-of-Coolant Accident
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection

ABBREVIATIONS (cont'd)

LPCS	Low Pressure Core Spray
LPSI	Low Pressure Safety Injection
LWR	Light-Water Reactor
MOV	Motor-Operated Valve
NPSH	Net Positive Suction Head
NRC	U.S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PCT	Peak Cladding Temperature
PFM	Probabilistic Fracture Mechanics
PORV	Power-Operated Relief Valve
POS	Plant Operational State
PRA	Probabilistic Risk Assessment
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
QHO	Quantitative Health Objective
RCIC	Reactor Core Isolation Cooling
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RES	Office of Nuclear Regulatory Research (NRC)
RG	Regulatory Guide
RSS	Reactor Safety Study
RWST	Refueling Water Storage Tank
SBLOCA	Small-Break Loss-of-Coolant Accident
SBO	Station Blackout
SCSS	Sequence Coding Search System
SRM	Staff Requirements Memorandum
SRP	Standard Review Plan
SRV	Safety-Relief Valve
SW	Service Water
TAF	Top of Active Fuel
TMI-2	Three Mile Island, Unit 2
USI	Unresolved Safety Issue
WOG	Westinghouse Owner's Group

1. INTRODUCTION

1.1 Background

In a June 8, 1999, staff requirements memorandum (SRM) on SECY-98-300, the Commission approved proceeding with a study of risk-informing the technical requirements of 10 CFR Part 50. The Commission specifically directed the staff to pursue the *"study on an aggressive timetable and provide, for Commission approval, a schedule for this activity. The staff should periodically inform the Commission on progress made in the study....if the staff identifies a regulatory requirement which warrants prompt revision..., the Commission should be...provided with a recommended course of action."*

The staff provided its plan and schedule for the study phase of its work to risk-inform the technical requirements of 10 CFR Part 50 in SECY-99-264, "Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50," dated November 8, 1999. The plan consists of two phases. Phase 1 is strictly an initial study in which only an evaluation is performed of the feasibility of risk-informed changes along with recommendations to the Commission on proposed changes. Phase 2 is an implementation phase of the rulemaking (based on recommended changes resulting from Phase 1 and approved by the Commission). Phase 2 also consists of performing the technical analysis and developing the regulatory guides needed to support the rulemaking. The Commission approved proceeding with the plan in a February 3, 2000, SRM.

The changes identified and evaluated as part of this effort can include adding provisions to Part 50 allowing for risk-informed alternatives to the present requirements, revising specific requirements to reflect risk-informed considerations, or deleting unnecessary or ineffective regulations.

10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," was promulgated with Appendix K to 10 CFR Part 50, "ECCS Evaluation Models," to implement General Design Criterion (GDC) 35, "Emergency Core Cooling," of Appendix A to 10 CFR Part 50. These requirements were, for the most part, promulgated based on knowledge available in the late 1960s and early 1970s to provide a conservative design basis for emergency core cooling systems. Specifically, 10 CFR 50.46 and Appendix K delineate acceptance criteria and modeling requirements for use in evaluating the adequacy of emergency core cooling systems for hypothetical design basis loss-of-coolant accidents (LOCAs). In this document, the current requirements in 10 CFR 50.46, Appendix K and GDC 35 are evaluated, potential options are identified for risk-informing these requirements, and a risk-informed alternative is proposed.

1.2 Objectives

In SECY-98-300, the U.S. Nuclear Regulatory Commission (NRC) staff delineated the following broad objectives for its work to risk-inform 10 CFR 50:

- Enhance safety by focusing NRC and licensee resources in areas commensurate with their importance to health and safety

- Provide NRC with the framework to use risk information to take action in reactor regulatory matters, and

- Allow use of risk information to provide flexibility in plant operation and design, which can result in burden reduction without compromising safety.

The objective of this document is to identify and describe potential risk-informed alternatives to 10 CFR 50.46, Appendix K, and GDC 35. The work documented herein is intended only to demonstrate the feasibility of risk-informed changes to the candidate regulations. If the Commission approves going to rulemaking, additional analyses will be required.

1.3 General Comment and Limitations

The work to risk inform the technical requirements of GDC 35, 10 CFR 50.46 and Appendix K was carried out in the following manner:

- The approach follows the framework for the risk-informing process described in reference [Ref. 3]. The staff expects that the changes to requirements resulting from this work would be consistent with the defense-in-depth approach delineated in the framework, maintain sufficient safety margins, be performance-based to the extent possible, and result in changes in risk that are reasonable compared to the Safety Goals. This approach would also ensure that adequate protection continues to be maintained.

- The study focuses on developing risk-informed alternatives to 10 CFR Part 50.46, Appendix K to 10 CFR Part 50, and GDC 35 of Appendix A to 10 CFR Part 50. Since the impact of these regulations may stem from the regulations themselves or from supporting regulatory guides, standard review plan sections, branch technical positions, or other implementing documents, all such documents are reviewed and, as necessary, considered for change.

- When implemented, compliance with a risk-informed alternative regulation would be voluntary. It is anticipated that licensees would have the option to comply with all of the requirements of the existing regulations or with all of the requirements of a risk-informed alternative.

- The risk-informed alternative regulations may modify or eliminate requirements contained in existing regulations or add new requirements. If changes are identified that have the potential to pass the backfit rule, they will be referred to the Generic Safety Issues program to assess the need for mandatory implementation for all licensees (not just those that choose a risk-informed alternative).

- Any criteria applied in this study for risk categorization will build upon and be consistent with those being used in the Option 2 work as described in SECY-99-256 [Ref. 1]. It will also build upon and be coordinated with the risk-informed plant oversight process.

- The criteria established in this study with respect to needed quality of a licensee's probabilistic risk assessment (PRA) will be consistent with those proposed in SECY-99-256 and RG 1.174 [Ref. 2]. PRA standards, either developed by standards-setting organizations (e.g., American Society of Mechanical Engineers [ASME] and American Nuclear Society [ANS]) and endorsed by NRC, or developed by NRC are intended to be important mechanisms for ensuring needed quality.

- The principal focus of this work is on the current set of licensed reactors. However, one factor in the staff's prioritization process will be the potential impact on future reactors, so that potential regulatory changes that impact both current and future plants will receive higher priority than those only affecting current reactors. Those changes affecting only future plants will be of lowest priority.

- Codes and standards referenced in 10 CFR Part 50 will not be addressed in this study.

- The study may identify requirements that, while important to safety, are not directly related to the concern being addressed by the rule under consideration. For these cases, the requirement will be retained (even though it is not directly related to the concern) rather than moving it to a more relevant rule. This is done to avoid the additional effort that would be associated with deleting the requirement and moving it to another rule.

1.4 Organization of Report

Chapter 2 of this report describes, at a high level, how the objective of risk-informing the regulations in 10 CFR Part 50 will be accomplished and how success will be measured. The approach is that set forth in the implementation section of the proposed framework for risk-informing the technical requirements of 10 CFR Part 50 [Ref. 3]. Subsequent chapters document the application of this approach to the current requirements of 10 CFR 50.46, Appendix K and GDC 35.

Chapter 3 provides the basis for selecting 10 CFR 50.46, Appendix K and GDC 35 as candidate regulations to be risk-informed, and includes a description of the regulatory concern being addressed. This chapter also includes relevant background information on the history of the design basis LOCA and emergency core cooling system (ECCS) acceptance criteria, and includes a detailed examination of the current technical requirements of 10 CFR 50.46, Appendix K and GDC 35. Relationships to other regulations and implementing documents are identified. This information is needed because changes to the regulations could potentially impact related regulations or implementing documents.

In Chapter 4, the risk significance of LOCAs and the ECCS is discussed. The purpose of this chapter is to identify the needed attributes for risk-informed alternative regulations.

Chapter 5 presents options for risk-informing various regulatory requirements or practices associated with 10 CFR 50.46, Appendix K and GDC 35. For each option, information is provided regarding existing regulatory requirements or practices, risk perspectives, safety considerations, potential for unnecessary burden reduction, implementation steps, time and resource requirements, applicability to advanced reactors, and any other relevant considerations. This chapter also includes a comparison of the different options, resulting in identification of a risk-informed alternative.

1.5 References

1. SECY 99-256, “Rulemaking Plan for Risk-Informing Special Treatment Requirements,” October 29, 1999.

2. USNRC, “Regulatory Guide 1.174: An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis,” July 1998.

2. APPROACH

This section describes at a high level how the objective of risk-informing the regulations in 10 CFR Part 50 will be accomplished. The approach is that described in detail in the Risk-Informing 10 CFR Part 50 Framework [Ref. 1] and summarized below. In developing the framework to risk-inform the technical requirements of 10 CFR Part 50, three concepts are integrated together in its formulation as shown below in Figure 2-1.

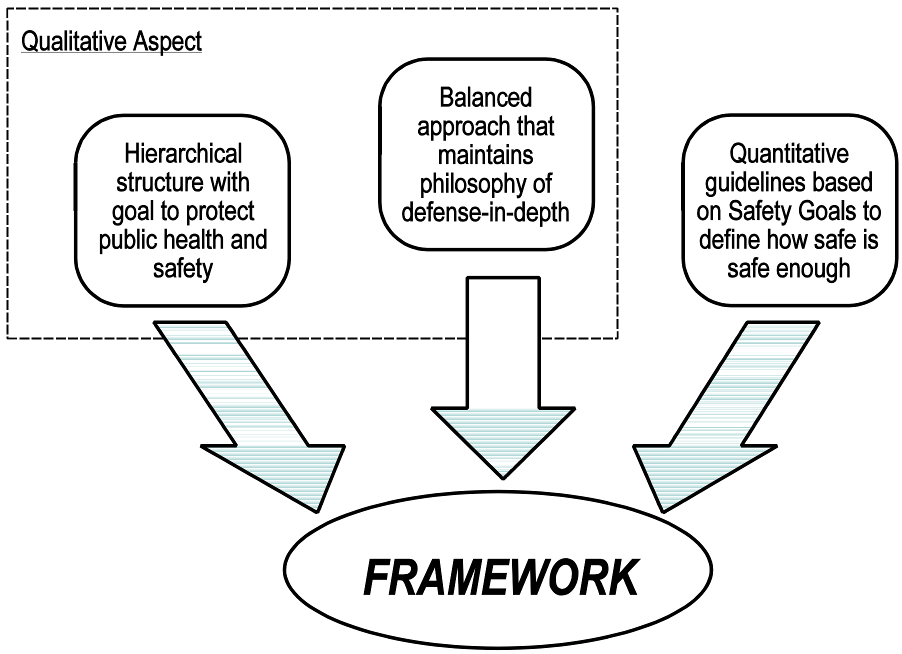


Figure 2-1 Framework Concepts

Hierarchical Structure —

One qualitative aspect of the framework is a hierarchal structure defining elements that relate the goal of protecting the public health and safety to the regulations. The framework is comprised of four major elements:

1. Element 1 is the goal of protecting the public health and safety

2. Element 2 are the NRC Reactor Inspection and Oversight Program cornerstones for safe nuclear power plant operation needed to meet the goal

3. Element 3 are the strategies for implementing the cornerstones, and

4. Element 4 are the tactics used in formulating and implementing the various regulations

The hierarchal structure of the framework is shown in Figure 2-2 below.

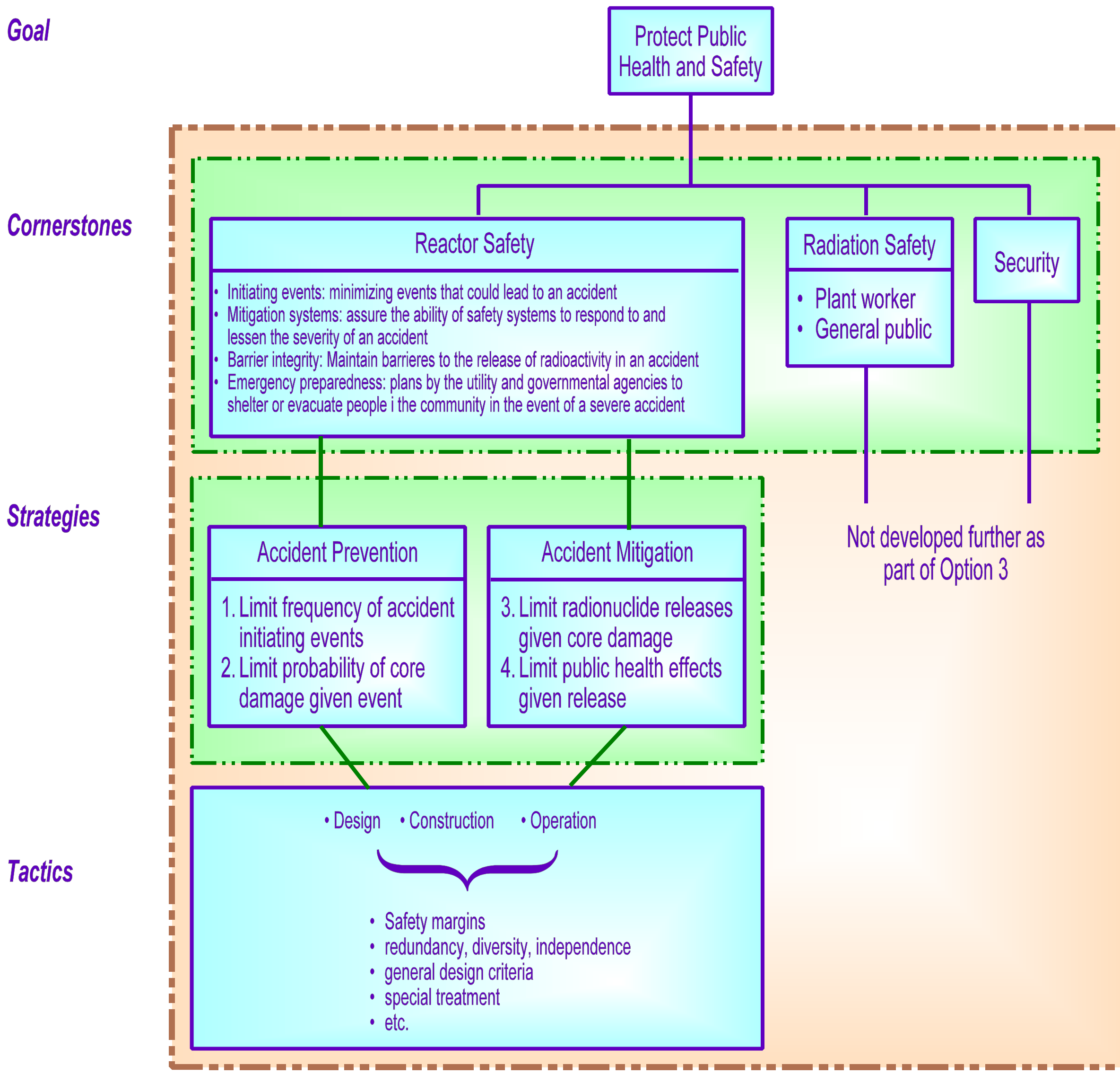


Figure 2-2 Framework Hierarchy.

Defense-in-Depth Approach —

A balanced approach that maintains defense-in-depth is used in developing the framework. At a high level, the cornerstones of safe nuclear power plant operation (in particular the four reactor safety cornerstones) are the bases for the strategies developed for risk-informing existing technical requirements. The strategies, therefore, incorporate the defense-in-depth philosophy and address preventing core damage and mitigating radionuclide releases should core damage occur. At a lower level, defense-in-depth is also incorporated into the framework: ***in formulating and implementing a regulation it must meet the defense-in-depth principles.***

The approach used in Option 3 is summarized in the following working definition:

Defense-in-depth is the approach taken to protect the public health and safety.

At a high level, risk-prevention and mitigation strategies are applied in formulating the regulation:

1. *limit the frequency of accident initiating events*

2. *limit the probability of core damage given accident initiation*

3. *limit radionuclide releases during core damage accidents*

4. *limit public health effects due to core damage accident*

At a lower level, the risk-informed regulation is formulated and implemented in such a way that the following defense-in-depth principles are met:

- *A reasonable balance is provided among the strategies.*

- *over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.*

- *independence of barriers is not degraded.*

- *safety function success probabilities commensurate with accident frequencies, consequences, and uncertainties are achieved via appropriate*
  - *redundancy, independence, and diversity,*
  - *defenses against common cause failure mechanisms,*
  - *defenses against human errors, and*
  - *safety margins*

- *the defense-in-depth objectives of the current GDCs in Appendix A to 10 CFR 50 are maintained.*

Quantitative Guidelines —

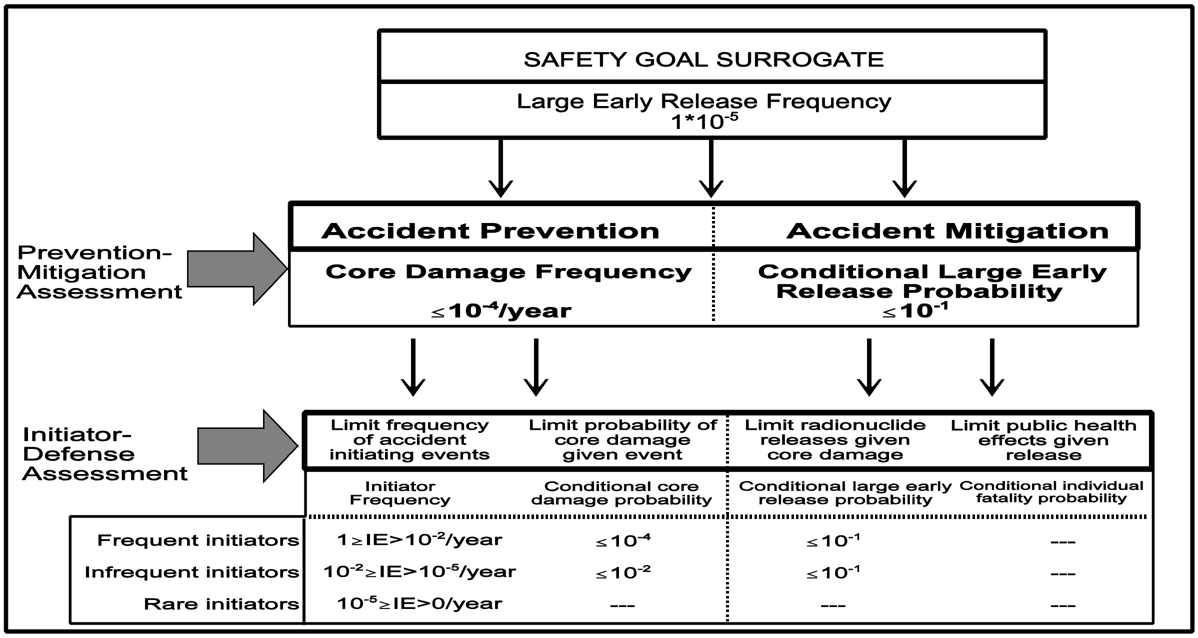
To bring risk insights into the process, the framework is extended to include quantitative guidelines associated with the strategies for risk informing existing technical requirements. These guidelines are for the staff use in identifying existing regulations that are candidates for risk-informed change, formulating and evaluating change options, and recommending the changes to be included in alternative, risk-informed regulations.

The quantitative guidelines are not proposed regulatory requirements and will not appear in risk-informed regulations; however, they may appear in implementing documents such as regulatory guides when probabilistic analyses are deemed appropriate.

The framework quantitative guidelines are also based on the Safety Goal Quantitative Health Objectives (QHOs). Consistent with the framework structure, quantitative guidelines are provided for the two high level strategies (i.e., accident prevention and accident mitigation) and for the four supporting strategies (e.g., limit frequency of accident initiating events). To be consistent with the defense-in-depth principles, guidelines were developed as a function of initiating event frequency. Three categories of initiators are defined: frequent, infrequent and rare.

The quantitative guidelines are summarized below in Figure 2-3. The guideline values presented in Figure 2-3 represent mean values, in order to account for uncertainty.

Figure 2-3 Framework Quantitative Guidelines



In assigning guidelines for the strategies, the product across each row gives a large early release frequency (LERF) of <10^{-5}/year. Therefore, a quantitative guideline of 0.1 is assigned for accident mitigation (i.e., conditional large early release probability). Though the product across each row individually gives a LERF of <10^{-5}/year, it is the intent of these guidelines that the combined LERF for all initiators should remain less than 10^{-5}/year. Finally, quantitative guidelines are derived for each strategy for different categories of initiators. Three categories are identified:

- ***Frequent initiators*** (or anticipated) are those events expected to occur or may well occur during the life of an individual plant. The sum of the frequency of frequency initiators is typically greater than 10^{-2} per year and less than 1.0.

- ***Infrequent initiators*** are those events not expected to occur over the life of any single plant, but may occur in the population of plants and could be risk significant. The sum of the frequency of infrequent initiators is typically greater than 10^{-5} per year and less than 10^{-2}.

- ***Rare initiators*** are those events that are extremely unlikely and not expected to occur in the population of plants. The sum of the frequency of rare initiators is typically greater than zero per year and less than 10^{-5}.

For frequent and infrequent event, when the first two strategies, prevent initiators and prevent core damage, are considered as a pair, the relevant quantitative guideline is a core damage frequency (CDF) less than 10^{-4} per year. When these strategies are considered individually, the products of the quantitative guidelines for the two strategies is the 10^{-4} per year CDF quantitative guideline. That is, meeting the risk-informed regulations should be consistent with achieving a CDF of less than 10^{-4} per year. To meet such a guideline, the regulations should assure a higher response reliability (perhaps more redundancy and diversity) for more frequent initiators.

A different approach has been taken for rare events. Some of these events, should they occur, have the potential to progress directly to offsite releases of radionuclides. Because the core damage prevention and containment strategies may be unavailable for rare initiators, the frequency quantitative guideline for rare initiators is set more stringently than 10^{-4} per year. Specifically, the quantitative guideline is less than 10^{-5} rare initiators per year with no single type of rare initiator being allowed to account for the entire guideline.

The fourth high-level defense-in-depth strategy involves emergency planning and response, which are essential for protecting the public health and safety. Although a quantitative guideline has not been set for this strategy, credit has been taken for its effectiveness in establishing subsidiary quantitative guidelines compatible with the QHOs for the first three strategies. In addition, pre-planned protective actions may be particularly important for accident scenarios in which one or more of the first three strategies are compromised. For example, for an ISLOCA, which bypasses containment, an early containment failure guideline cannot be used; therefore, the fourth strategy becomes necessary.

In deriving the guidelines for conditional core damage probability (CCDP) and conditional large early release probability (CLERP), the upper end of the range for the initiating event frequencies is used.

The use of a LERF guideline developed from the early-fatality QHO, does not imply that risks associated with late containment failures can or will be ignored. Measures to mitigate late large releases are also appropriate. A conditional probability of a late large release (up to 24 hours after the onset of core damage) of #10^{-1} is also proposed.

References

1. USNRC, "Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control)," SECY-00-0198, September 14, 2000.

3. OVERVIEW OF 10 CFR 50.46, Appendix K AND GDC 35

3.1 Historical Background - Evolution of Design Basis LOCA and 10 CFR 50.46

Design-basis LOCAs for early commercial reactors were assumed to potentially lead to substantial fuel melting given failure of the ECCS. Emphasis was, therefore, placed on the containment capability, low containment leak rate, heat transfer out of the containment to prevent unacceptable pressure buildup, and containment atmospheric cleanup systems. The earliest commercial reactor containments were designed to confine the fluid release from a double-ended guillotine break (DEGB) of the largest pipe in the reactor coolant system (RCS). Long-term core cooling capability was provided, but before 1966, high-capacity ECCSs were not required.

In 1966, during the review of applications for construction permits for large power reactors, early models of basemat meltthrough made it apparent to the Atomic Energy Commission (AEC) and the Advisory Committee on Reactor Safeguards (ACRS) that containment might not survive a core meltdown accident. An ECCS task force, which was appointed to study the problem, concluded in 1967 that more reliable, high-capacity ECCSs were needed to assure that larger plants could safely cope with a major LOCA. The GDCs in Appendix A to 10 CFR Part 50, which were being developed at the time, included requirements to this effect. The ECCS was to be designed to accommodate pipe breaks up to and including a double-ended guillotine break of the largest pipe in the reactor coolant system.

In the 1966-1967 time frame, research results indicated that zircaloy cladding exposed to LOCA-like conditions with peak temperatures in the vicinity of 1370°C (well below the zircaloy melting point of 1820°C) embrittled and ruptured, or even shattered upon cooldown. This threatened the integrity of the core geometry, which, in turn, was perceived to threaten core coolability. Therefore, instead of the criterion of no (or very little) clad melt, which was based in part on the concern over the autocatalytic effect on zirconium oxidation, and which had been proposed by the nuclear steam supply system (NSSS) vendors and accepted for some months, a much lower limit on the highest acceptable clad temperature during a LOCA was indicated, somewhere between 1204°C and 1370°C (2200°F to 2498°F). In 1971, the AEC issued a policy statement containing interim acceptance criteria for ECCS for light water reactors [Ref. 1].

Various intervenor groups challenged the interim acceptance criteria in individual licensing hearings. Consequently, the AEC scheduled public rulemaking hearings. The hearings began in January 1972 and took 125 days over 23 months. All of the documents utilized during these hearing were included in the hearing record, which contains more than 22,000 pages (Docket RM-50-1). The Regulatory Staff filed its concluding statement with the recommendation that the ECCS criteria be made more conservative in several aspects, especially by decreasing the acceptable temperature limit for the cladding from 1260°C to 1204°C (2300°F to 2200°F) and by increasing the conservatism of the methods used to calculate the temperature of the fuel cladding during the LOCA. The revised requirements appeared in the U.S. Federal Register on January 4, 1974 and published as 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling for Light Water Cooled Nuclear Power Reactors [Ref. 2]. Further details on the historical development of the ECCS acceptance criteria can be found in [Ref. 3] and [Ref. 4].

Appendix K to 10 CFR 50 was promulgated with 10 CFR 50.46 to specify required and acceptable features of ECCS evaluation models. Included are assumptions regarding initial and boundary conditions, acceptable models, and imposed conditions for the analysis. In developing Appendix K, conservative assumptions and models were imposed to cover areas where data were lacking or uncertainties were large or unquantifiable. Many required and acceptable features remain in their original form in Appendix K. The evolution of required and acceptable features in Appendix K is discussed in [Ref. 5].

In 1975, a second major review of the LOCA/ECCS issues and related research programs was carried out by the American Physical Society [Ref. 6]. The succeeding research program followed the recommendations of this review and the Commission Opinion following the ECCS rulemaking [Ref. 7].

The AEC staff supplemental testimony on the interim acceptance criteria emphasized that statistically based best estimate methods were then (1971-72 time frame) under development and should be applied to LOCA analyses: "there should be an effort to incorporate two fundamental concepts: (1) treatment of individual components of the analysis procedures as realistically as possible and (2) keeping the techniques as simple as possible. These concepts are complementary in purpose; that is, they promote understanding of the physical phenomena by avoiding unnecessary mathematical complexity and unwarranted simplifying assumptions." It wasn't until 1988 that 10 CFR 50.46 was revised to permit the use of best-estimate analyses in lieu of more conservative Appendix K calculations, provided that uncertainties in the best-estimate calculations are quantified. In effect a third major review of LOCA/ECCS issues was conducted to provide technical bases for this revision [Ref. 8]. Regulatory Guide 1.157 [Ref. 9] presents acceptable procedures and methods for realistic or best-estimate ECCS evaluation models.

3.2 Description of Regulations

Figure 3-1 graphically depicts the technical requirements associated with 10 CFR 50.46, and related regulations (i.e., Appendix K, “ECCS Evaluation Models” and GDC 35, “Emergency Core Cooling”). A more detailed description of the technical requirements associated with each of these regulations is provided below.

3.2.1 10 CFR 50.46

The technical requirements of 10 CFR 50.46 call for an ECCS whose performance following postulated LOCAs meets the following criteria:

- Peak cladding temperature (PCT) less than or equal to 2200°F,
- Local peak cladding oxidation less than or equal to 0.17 times the total cladding thickness before oxidation,
- Maximum hydrogen generation less than or equal to 0.01 times the hypothetical amount from complete reaction of the fuel cladding (excluding the cladding surrounding the fuel plenum volume),
- Core geometry remains amenable to cooling, and
- Core temperature maintained at acceptably low value for long-term decay heat removal.

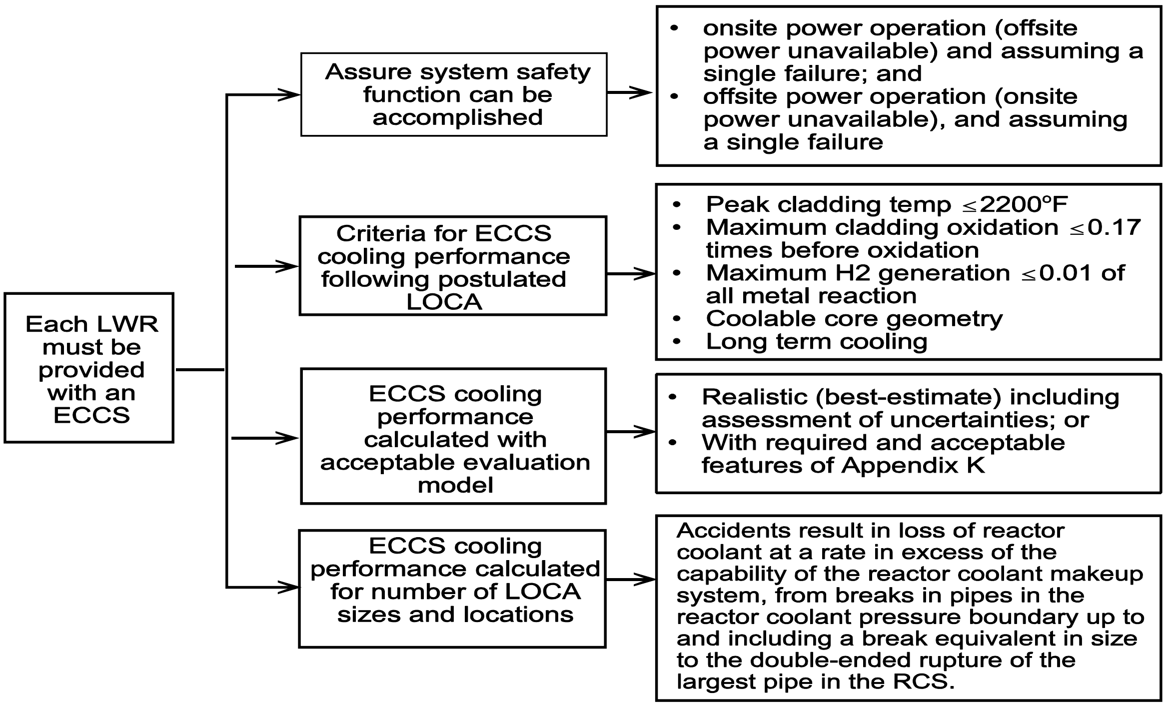


Figure 3-1 Overview of ECCS Performance Technical Requirements

The ECCS cooling performance must also be calculated for a number of LOCA sizes (up to and including double-ended rupture of the largest pipe in the RCS), locations and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated, using one of the following two types of acceptable evaluation models:

(1) An ECCS model with the required and acceptable features of 10 CFR Part 50, Appendix K, or
(2) A best-estimate ECCS evaluation model which realistically represents the behavior of the reactor system during a LOCA, and includes an assessment of uncertainties which demonstrates that there is a high level of probability that the above acceptance criteria are not exceeded.

The evaluation model documentation requirements are provided in Appendix K (see Section 3.2.2). Reportability requirements in 10 CFR 50.46 regarding model changes or errors are as follows:

(1) The significance of the change or error must be determined (a significant change or error is one which results in a change in calculated PCT of more than 50°F from that calculated for the limiting transient with the last acceptable model, or is a cumulation of errors and changes such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50°F),

(2) Significant errors or changes must be reported within 30 days; other changes or errors must be reported annually, and

(3) Any change or error correction that results in a calculated ECCS performance that does not meet the above acceptance criteria is a reportable event as described in 10 CFR 50.55(e), 10 CFR 50.72 and 10 CFR 50.73.

The requirements of 10 CFR 50.46 are in addition to the general ECCS cooling performance design requirements found elsewhere in 10 CFR Part 50, in particular the system safety function requirements in GDC 35.

3.2.2 Appendix K to 10 CFR Part 50

Appendix K to 10 CFR Part 50 provides the required and acceptable features of ECCS evaluation models, as well as the evaluation model documentation requirements. The required and acceptable features fall into the following four categories:

(1) Sources of heat during the LOCA
(2) Swelling and rupture of the cladding and fuel rod thermal parameters
(3) Blowdown phenomena
(4) Post-blowdown phenomena; Heat removal by the ECCS

Appendix K specifies the sources of heat during the LOCA that need to be considered as:

(1) The initial stored energy in the fuel
(2) Fission heat
(3) Decay of actinides
(4) Fission product decay
(5) Metal-water reaction rate
(6) Reactor internals heat transfer
(7) Pressurized water reactor (PWR) primary-to-secondary heat transfer

Appendix K specifies that “each evaluation model shall include a provision for predicting cladding swelling and rupture from consideration of the axial temperature distribution of the cladding and from the difference in pressure between the inside and outside of the cladding, both as functions of time.”

Appendix K specifies the blowdown phenomena that need to be considered as:

(1) Break characteristics and flow
(2) Frictional pressure drops
(3) Momentum equation
(4) Critical heat flux (CHF)
(5) Post-CHF heat transfer correlations
(6) Pump modeling
(7) Core flow distribution during blowdown (applies only to PWRs)

For post-blowdown phenomena and heat removal by the ECCS, Appendix K requires consideration of the following:

(1) Single failure criterion
(2) Containment pressure
(3) Calculation of reflood rate for PWRs
(4) Steam interaction with ECC water in PWRs
(5) Refill and reflood heat transfer for PWRs
(6) Convective heat transfer coefficients for boiling water reactor (BWR) fuel rods under spray cooling
(7) The BWR channel box under spray cooling

The evaluation model documentation requirements specified by Appendix K include the following:

(1) Description of each evaluation model
(2) Listing of each computer program
(3) Demonstration of solution convergence for each computer program
(4) Sensitivity studies for each evaluation model
(5) Comparison with experimental information
(6) Technical adequacy of calculational methods (must meet general standards for acceptability)

3.2.3 Appendix A to 10 CFR Part 50: GDC 35

Appendix A to 10 CFR Part 50, GDC 35, requires that a system must be designed to provide abundant emergency core cooling. The system safety function following any loss of reactor coolant is to remove heat from the reactor core in order to (1) prevent fuel and clad damage that could interfere with continued effective core cooling; and (2) limit clad metal-water reaction to negligible amounts. Accomplishment of the system safety function must be assured for:

(1) onsite power operation (with offsite power unavailable), and assuming a single failure; and
(2) offsite power operation (with onsite power unavailable), and assuming a single failure.

3.3 Related Regulations and Implementing Documents

Prior to developing a risk-informed alternative to a regulation, or set of regulations, it is necessary to perform a review to determine the relationship of each candidate regulatory requirement to other related regulations and implementing documents, such as regulatory guides (RGs), standard review plan (SRP) sections, branch technical positions (BTPs), generic letters (GLs), etc. The purpose of this review is to obtain a detailed understanding of the implications of revising any particular requirement in terms of its impact across the body of the regulations and implementing documents.

GDC 35 does not reference any other regulations. Appendix K to 10 CFR Part 50 references only 10 CFR 50.46. Regulations other than Appendix K and GDC 35 referenced from 10 CFR 50.46, and the specific cross-references, are provided in Table 3-1.

Appendix K and GDC 35 are only referenced by 10 CFR 50.46. Regulations other than Appendix K which reference 10 CFR 50.46, and the specific cross-references, are provided in Table 3-2.

Table 3-1 Regulations Referenced from 10 CFR 50.46

Referenced Regulation	Cross-Reference
10 CFR 50.4, *Written communications*	For each change to or error discovered in an acceptable evaluation model or in the application of such a model that affects the temperature calculation, the applicant or licensee shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission at least annually as specified in §50.4.
• 10 CFR 50.55, Conditions of construction permits • 10 CFR 50.72, *Immediate notification requirements for operating nuclear power reactors* • 10 CFR 50.73, *Licensee event report system*	Any change or error correction that results in a calculated ECCS performance that does not conform to the criteria set forth in paragraph (b) of this section is a reportable event as described in §§50.55(e), 50.72 and 50.73.
10 CFR 50.82, *Termination of license*	This section [50.46(a)(1)(i)] does not apply to a nuclear power reactor facility for which the certifications required under §50.82(a)(1) have been submitted.

Table 3-2 Regulations Referencing 10 CFR 50.46

Referencing Regulation	Cross-Reference
10 CFR 50.8, *Information collection requirements: OMB approval*	The approved information collection requirements contained in this part appear in §§50.30, 50.33, 50.33a, 50.34, 50.34a, 50.35, 50.36, 50.36a, 50.36b, 50.44, 50.46, 50.47, 50.48, 50.49, 50.54, 50.55, 50.55a, 50.59, 50.60, 50.61, 50.62, 50.63, 50.64, 50.65, 50.66, 50.71, 50.72, 50.74, 50.75, 50.80, 50.82, 50.90, 50.91, 50.120, and Appendices A, B, E, G, H, I, J, K, M, N, O, Q, R, and S to this part.
10 CFR 50.34, *Contents of applications; technical information*	Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §50.46 of this part for facilities for which construction permits may be issued after December 28, 1974.
10 CFR 50.44, *Standards for combustible gas control system in light-water-cooled power reactors*	<u>50.44(d)</u> (1) For facilities that are in compliance with §50.46(b), the amount of hydrogen contributed by core metal-water reaction (percentage of fuel cladding that reacts with water), as a result of degradation, but not total failure, of emergency core cooling functioning shall be assumed either to be five times the total amount of hydrogen calculated in demonstrating compliance with §50.46(b)(3), or to be the amount that would result from reaction of all the metal in the outside surfaces of the cladding cylinders surrounding the fuel (excluding the cladding surrounding the plenum volume) to a depth of 0.00023 inch (0.0058 mm), whichever amount is greater. A time period of 2 minutes shall be used as the interval after the postulated LOCA over which the metal-water reaction occurs. (2) For facilities as to which no evaluation of compliance in accordance with §50.46(b) has been submitted and evaluated, the amounts of hydrogen so contributed shall be assumed to be that amount resulting from the reaction of 5 percent of the mass of metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume.

Table 3-2 Regulations Referencing 10 CFR 50.46

Referencing Regulation	Cross-Reference
	50.44(h) As used in this section: (1) Degradation, but not total failure, of emergency core cooling functioning means that the performance of the emergency core cooling system is postulated, for purposes of design of the combustible gas control system, not to meet the acceptance criteria in §50.46 and that there could be localized clad melting and metal-water reaction to the extent postulated in paragraph (d) of this section. The degree of performance degradation is not postulated to be sufficient to cause core meltdown.

As indicated in Table 3-2, 10 CFR 50.44, “Standards for combustible gas control system in light-water-cooled power reactors,” references the total amount of hydrogen calculated in demonstrating compliance with §50.46(b)(3) as one of two means for determining the amount of hydrogen contributed by core metal-water reaction, for use in determining the necessary size (capacity) of the hydrogen recombiners. As will be seen in Chapter 5 of this attachment, §50.46(b)(3) is one of the prescriptive acceptance criteria in 10 CFR 50.46 that is recommended for elimination as part of changes to the existing 10 CFR 50.46 requirements. While this would appear to cause an incongruity in the existing regulations, it should be noted that the requirements of the existing 10 CFR 50.44, particularly with respect to the hydrogen recombiners, are currently in the process of being modified [Ref. 10]. The staff expects that the modifications to 10 CFR 50.44 will resolve this potential incongruity in the regulations.

Some other regulatory requirements related to the broader context of design-basis LOCAs are listed in Table 3-3, and some of the principal implementing documents associated with 10 CFR 50.46, Appendix K and GDC 35 are listed in Table 3-4.

Table 3-3 Partial List of LOCA-Related Regulatory Requirements

- Appendix A to 10 CFR Part 50, GDC 1, *Quality standards and records*
- Appendix A to 10 CFR Part 50, GDC 4, *Environmental and dynamic effects design bases*
- Appendix A to 10 CFR Part 50, GDC 5, *Sharing of structures, systems and components*
- Appendix A to 10 CFR Part 50, GDC 13, *Instrumentation and control*
- Appendix A to 10 CFR Part 50, GDC 14, *Reactor coolant pressure boundary*
- Appendix A to 10 CFR Part 50, GDC 15, *Reactor coolant system design*
- Appendix A to 10 CFR Part 50, GDC 16, *Containment design*
- Appendix A to 10 CFR Part 50, GDC 17, *Electric power systems*
- Appendix A to 10 CFR Part 50, GDC 19, *Control room*
- Appendix A to 10 CFR Part 50, GDC 27, *Combined reactivity control systems capability*
- Appendix A to 10 CFR Part 50, GDC 30, *Quality of reactor coolant pressure boundary*
- Appendix A to 10 CFR Part 50, GDC 31, *Fracture prevention of reactor coolant pressure boundary*
- Appendix A to 10 CFR Part 50, GDC 32, *Inspection of reactor coolant pressure boundary*
- Appendix A to 10 CFR Part 50, GDC 34, *Residual heat removal*
- Appendix A to 10 CFR Part 50, GDC 36, *Inspection of emergency core cooling system*
- Appendix A to 10 CFR Part 50, GDC 37, *Testing of emergency core cooling system*
- Appendix A to 10 CFR Part 50, GDC 38, *Containment heat removal*
- Appendix A to 10 CFR Part 50, GDC 41, *Containment atmosphere cleanup*
- Appendix A to 10 CFR Part 50, GDC 44, *Cooling water*
- Appendix A to 10 CFR Part 50, GDC 50, *Containment design basis*
- Appendix A to 10 CFR Part 50, GDC 54, *Systems penetrating containment*

Table 3-4 Partial List of Implementing Documents for 50.46, Appendix K and GDC 35

- SECY-83-472, *Emergency Core Cooling System Analysis Methods*
- Regulatory Guide (RG) 1.1, *Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps*
- RG 1.32, *Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants*
- RG 1.53, *Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems*
- RG 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)*
- RG 1.79, *Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors*
- RG 1.82, *Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident*
- RG 1.93, *Availability of Electric Power Sources*
- RG 1.97, *Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident*
- RG 1.157, *Best-Estimate Calculations of Emergency Core Cooling System Performance*
- Draft Regulatory Guide (DG) 1096, *Transient and Accident Analysis Methods*
- Standard Review Plan (SRP) Section 3.6.2, *Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping*
- SRP Section 6.3, *Emergency Core Cooling System*
- SRP Section 8.2, *Offsite Power System*
- SRP Section 8.3.1, *A-C Power Systems (Onsite)*
- SRP Section 9.2.1, *Station Service Water System*
- SRP Section 9.2.2, *Reactor Auxiliary Cooling Water Systems*
- SRP Section 9.2.5, *Ultimate Heat Sink*
- SRP Section 15.6.5, *Loss-Of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary*

3.4 References

1. "Criteria for Emergency Core Cooling Systems for Light Water Power Reactors - Interim Policy Statement," U.S. Federal Register, Vol. 36, No. 125, June 29, 1971 and No. 244, December 18, 1971.

2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," and Appendix K to 10 CFR 50, "ECCS Evaluation Models," U.S. Federal Register, Vol. 39, No. 3, January 4, 1974.

3. Wm. B. Cottrell, "The ECCS Rule-Making Hearing," *Nuclear Safety*, Vol. 15, No. 1, January 1974.

4. D. Okrent, "Nuclear Reactor Safety, On the History of the Regulatory Process," The University of Wisconsin Press, Madison, Wisconsin, 1981.

5. D. E. Bessette, "Initial and Boundary Conditions to LOCA Analysis, An Examination of the Requirements of Appendix K," *ICONE 8, 8th International Conference on Nuclear Engineering, Baltimore, MD, April 2-6, 2000*.

6. H. W. Lewis et al, "Report to the American Physical Society by the Study Group on Light Water Reactor Safety," American Physical Society, Reviews of Modern Physics, Vol. 47, No. 1, pp. S1-S123, June 1975.

7. USAEC Opinion of the Commission in the Matter of “Rulemaking Hearing, Acceptance Criteria for ECCS for Light Water Cooled Nuclear Power Reactors,” CLI-73-39, December 28, 1973.

8. Compendium of ECCS Research for Realistic LOCA Analysis," U.S. Nuclear Regulatory Commission, NUREG-1230, December 1988.

9. Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance."

10. Memo from A. L. Vietti-Cook to W. D. Travers, “Staff Requirements - SECY-00-0198 - Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control),” January 19, 2001.

4. RISK SIGNIFICANCE OF LARGE LOCAs AND ECCS

4.1 Introduction

LOCAs occur as the result of a breach in the reactor coolant pressure boundary. Since the RCS boundary includes pipes, the reactor pressure vessel, valves, pumps, and steam generators (in PWRs), ruptures in any of these components can result in a LOCA. The failures in the RCS boundary can be induced by random component failures (e.g., pipe ruptures), by external events (e.g., a seismically-induced pipe failure), and by human error (e.g., inadvertently opening a valve leading to a draindown event). In order to mitigate LOCAs, an ECCS is required to be included in the design of light water reactors. The ECCS is currently required to be designed to mitigate a LOCA from breaks in RCS pipes up to and including a break equivalent in size to a DEGB of the largest diameter RCS pipe. The ECCS is required to have sufficient redundancy that it can successfully perform its function with or without the availability of offsite power and with the occurrence of an additional single failure.

This chapter provides information on the risk-significance of LOCAs and the importance of the ECCS to preventing core damage from any accident initiator. This risk information was utilized to help formulate and determine the feasibility of risk-informed options to 10 CFR 50.46 and associated requirements. The main effort was to identify information on the frequency of pipe break LOCAs and their contribution to both the core damage and large early release frequencies. However, in order to risk-inform 50.46, additional initiator frequency information was obtained for other RCS boundary failures (i.e., for other than pipe failures). Additional information was reviewed in order to establish the risk-significance of specific requirements such as the need to evaluate the ECCS for loss of offsite power (LOOP) scenarios. Finally, some issues related to the design and operation of the ECCS were identified and the need for potential requirements in proposed risk-informed options was assessed.

4.2 Risk from Large Pipe-Break LOCAs

LOCAs as defined in 10 CFR 50.46 are "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." In PRAs, a large-break LOCA is generally a break greater than 5 or 6 inches in diameter and includes DEGBs in the largest RCS pipe. As discussed in this section, estimated frequencies of large-break LOCAs are small, especially the frequency of DEGBs. In addition, the results of the Individual Plant Examinations (IPEs) indicated that large-break LOCAs are not significant contributors to risk for most BWRs and many PWRs.

4.2.1 RCS Pipe Break Frequencies

Most PRAs use RCS pipe break frequencies that can be traced back to the Reactor Safety Study (RSS) [Ref. 1]. The RSS pipe break frequencies were based on pipe break data obtained from numerous sources that included both nuclear and non-nuclear data and U.S. and foreign information. The frequencies for small, medium, and large LOCAs were estimated and the same frequencies were utilized for both PWRS and BWRs. Large LOCAs are typically defined in PRAs as corresponding to break areas equivalent to greater than a 5 or 6 inch (inside) diameter pipe. Small LOCAs are typically less than an equivalent break in a 2 inch (inside) diameter pipe (medium LOCAs are between 2 and 5 or 6 inches in diameter). The NUREG-1150 study [Ref. 2] generated two sets of RCS pipe frequencies; one for BWRs and one for PWRs. The BWR frequencies utilized the WASH-1400 values while the PWR frequencies were selected to reflect the range of values that had been used in both industry and NRC-sponsored PRAs. Similarly, most of the IPEs utilized existing RCS pipe break frequencies from WASH-1400, NUREG-1150, and other PRAs. Thus, it is not surprising that the RCS pipe break frequencies used in WASH-1400, NUREG-1150, and the

IPEs are very similar. A comparison of the mean PWR LOCA frequencies used in these and other studies is shown in Figure 4-1 (mean estimates were generated from the median values provide in WASH-1400 by assuming a lognormal distribution and an error factor of 10) . The BWR LOCA frequencies from these sources are compared in Figure 4-2.

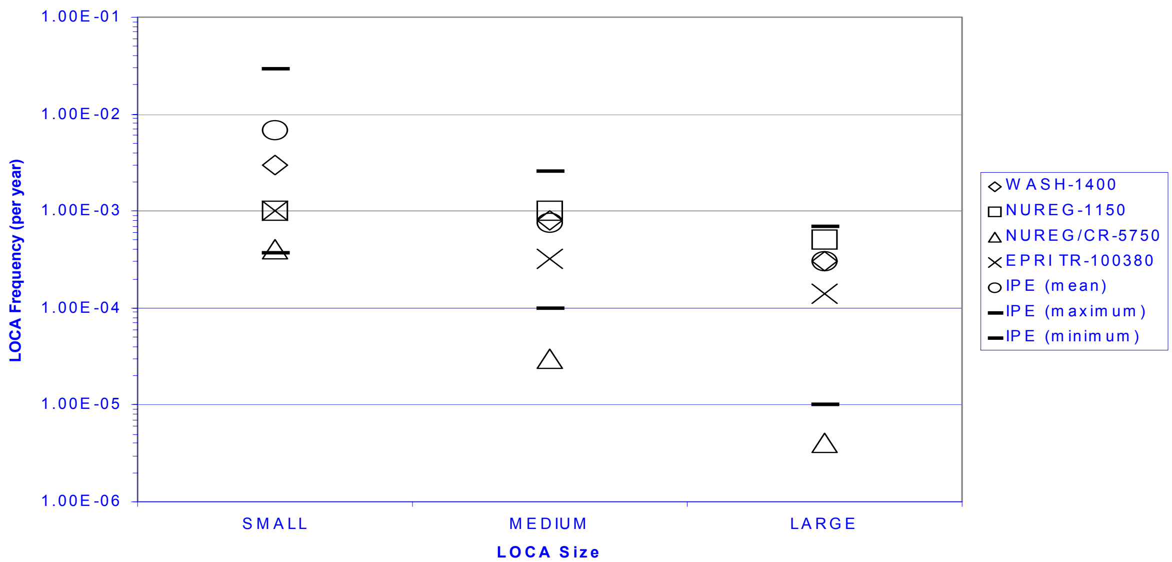


Figure 4-1 PWR pipe break LOCA frequencies used in PRAs and IPEs.

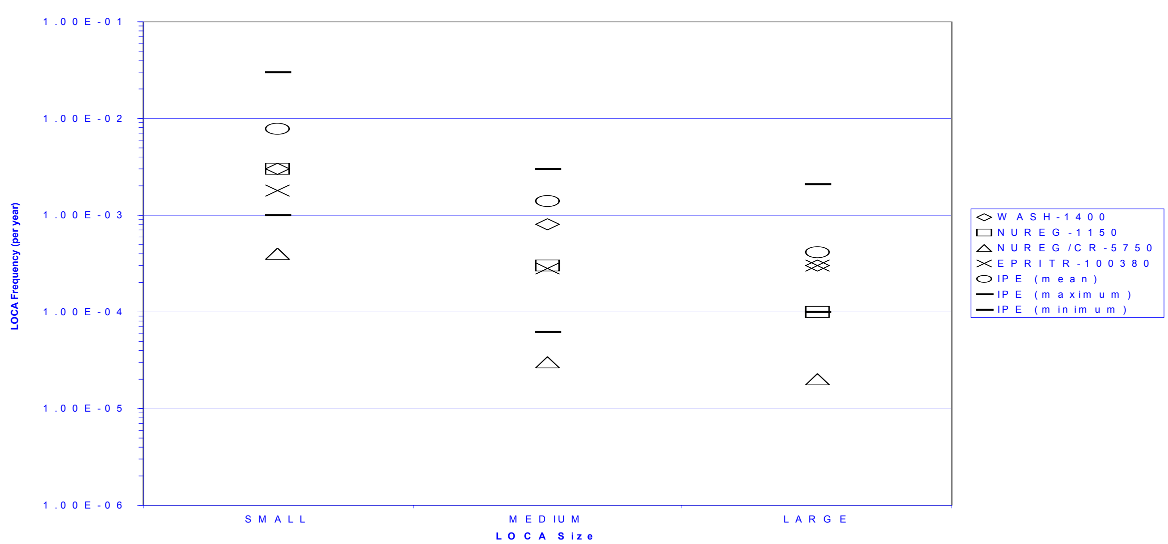


Figure 4-2 BWR pipe break LOCA frequencies used in PRAs and IPEs.

In commercial U.S. nuclear power plant history, there has never been a break in the RCS piping. With 2102 calender years of light-water reactor (LWR) operation in the U.S. through 1997 [Ref. 3], an estimate of the mean frequency of any size RCS pipe break is approximately 2×10^{-4}/calender year (calculated using a Bayesian update of a Jeffrey's noninformative prior). Estimates for BWRs and PWRs based on 710 and 1392 calender years of operation in commercial U.S. nuclear power plants through 1997 are 7×10^{-4} and 4×10^{-4}/calender year, respectively. The BWR and PWR pipe break frequency estimates are even lower when one considers that there has been 1748 and 3362 calender years of operation, respectively, in western-type reactors (results in pipe break estimates of 3×10^{-4}/calender year for BWRs and 1×10^{-4}/calender year for PWRs). These values are comparable to the RCS large pipe break frequencies but less than typical small break LOCA frequencies used in most PRAs.

A recent effort to generate new estimates of RCS pipe break frequencies is documented in NUREG/CR-5750 [3]. RCS pipe break initiator frequencies were estimated based on operating plant data (through 1997) and current knowledge of pipe break mechanisms. Frequencies of large and medium RCS pipe breaks were estimated by calculating the frequency of observed through-wall cracks and estimating the probability of rupture given a through-wall crack based on a technical review of information on fracture mechanics, data on high-energy pipe failures and cracks, and assessments of pipe-break frequencies estimated by others. The probability of rupture given a through-wall crack was modeled using a correlation developed by Beliczey and Schulz [Ref. 4]:

$$P_{R/TW} = 2.5/DN$$

where

$P_{R/TW}$ = mean probability of rupture given a through-wall crack

DN = nominal pipe diameter in mm

By this model, $P_{R/TW}$ is about 0.1 for a pipe with an inside diameter of 1 inch (25 mm) and 0.01 for pipes with inner diameters of 10 inches. In NUREG/CR-5750, as an added measure of conservatism, the probability of rupture of pipes larger than 10 inches in diameter was assumed to be 0.01. As indicated in Section 4.2.2, the Beliczey and Schulz correlation is supported by results from other studies.

The operating plant data used in the NUREG/CR-5750 study for estimating the frequency of through-wall cracks for large and medium sized pipes is summarized below:

<u>Relevant PWR throughwall cracks in large and medium RCS pipes</u>

- Dominant mechanism is thermal fatigue
- Data from 3362 calendar years of U.S. and foreign (Western-style) PWRs
- 1 through-wall crack in a large (8-inch) pipe
- 5 through-wall cracks in medium pipes
  - 1 in a 2.5-inch pipe
  - 4 in 6-inch pipes (including only U.S. event, Farley 2, 1987)

<u>Relevant BWR throughwall cracks in large and medium pipes</u>

- Dominant mechanism is intergranular stress corrosion cracking (IGSCC)
- Data from 710 calendar years
  - only U.S. plants considered

  - data from Big Rock Point, Dresden 1, La Crosse, and Humbolt Bay excluded because these plants are not representative of currently operating BWRs
- All cracks were caused by IGSCC in piping weldments
- 34 through-wall cracks in large pipes
  - all in pipes greater than 10-inches in diameter
  - only one event (in 1990) since IGSCC mitigation efforts started in mid-1980s
- 15 through-wall cracks in medium pipes
  - 13 in 4-inch pipes
  - 2 in 6-inch pipes
  - last event in 1984

Based on this data, the frequency of through-wall cracks (i.e., leaks) in large diameter pipes (>6 inches in diameter) was estimated in NUREG/CR-5750 to be 3.0×10^{-4}/calender year in PWRs and 2.4×10^{-3}/calender year in BWRs. For medium diameter pipes (2 inches < diameter < 6 inches), the frequency of through-wall cracks was estimated to be 1.5×10^{-3}/calender year in PWRs and 1.0×10^{-3}/calender year for BWRs. The BWR through-wall crack frequency estimates given above include an improvement factor of 20 to account for the effectiveness of IGSCC mitigation efforts. Use of this improvement factor is supported by experience data that shows the crack occurrence rate in BWRs has decreased significantly after IGSCC mitigation strategies were implemented.

The throughwall crack frequencies calculated from operational experience are significantly higher than those estimated in probabilistic fracture mechanics predictions. Probabilistic fracture mechanics calculations reported in NUREG-1061 [Ref. 5] resulted in estimates of the frequency of through-wall cracks in PWR RCS pipes ranging from 1.5×10^{-8}/yr to 3.8×10^{-7}/yr. A probabilistic fracture mechanics evaluation for an older BWR [Ref. 6] yielded a frequency of through-wall cracks in RCS pipes (with no IGSCC modeled) ranging from 6×10^{-8}/yr to 1×10^{-6}/yr.

The frequency of through-wall cracks obtained from operational experience were multiplied by the probability of rupture given a through-wall crack ($P_{R/TW}$) to obtain the NUREG/CR-5750 pipe break frequency estimates for large and medium sized pipes presented in Table 4.1. Since the publication of NUREG/CR-5750 several instances of through-wall cracking have been identified including cracks at the V.C. Summer plant. The estimates listed in Table 4.1 would, therefore, be slightly different (i.e., a factor of two higher) when the operating experience is updated. Error factors of 10 were assigned based on engineering judgement and are consistent with the values used in WASH-1400 and NUREG-1150.

Table 4-1 LOCA Frequency Estimates from NUREG/CR-5750

		Lower Bound Freq (per rx-cl-yr)[a,b]	**Mean Freq (per rx-cl-yr)[a]**	**Upper Bound Freq (per rx-cl-yr)[a,b]**
Large-Break LOCAs	PWR	1E-7	4E-6	1E-5
	BWR	9E-7	2E-5	9E-5
Medium Break LOCAs	PWR	1E-6	3E-5	1E-4
	BWR	9E-7	3E-5	9E-5
Small Break LOCAs	PWR & BWR	1E-4	4E-4	1E-3

[a] To convert from a per-calendar-year basis to a per-critical-year basis NUREG/CR-5750 divides the above values by an industry average criticality factor of 75%.

[b] Upper and lower bounds are estimated using engineering judgment and attempt to capture the uncertainty in the various estimation parameters.

NUREG/CR-5750 also generated a small LOCA frequency estimate but utilizing a different approach than used for the medium and large LOCAs. The small break LOCA frequency was estimated from available U.S. operating experience data in a simple Bayesian update of the small break LOCA frequency from WASH-1400. Because no difference could be discerned between PWR and BWR operating experience (no leakage events in U.S. operating experience), a single frequency was generated for use by both types of plants. The small LOCA frequency is also shown in Table 4.1.

As indicated in Figures 4.1 and 4.2, the mean NUREG/CR-5750 frequency estimates are one to two orders of magnitude less than the corresponding estimates from WASH-1400 and NUREG-1150, and near or below the lower bounds of the corresponding point-estimate frequency estimates from the IPEs. NUREG/CR-5750 characterizes its large and medium pipe break frequencies as "reasonable but conservative" noting that "a more complete (best-estimate) analysis using data, fracture mechanics analyses, results from pipe fracture experiments, and an expert elicitation process could likely produce more definitive estimates." The requirements for fracture mechanics analyses to generate pipe break frequencies are delineated in Attachment 2.

LOCA frequency estimates have also been generated by industry groups. Foremost among these efforts are those by the Electric Power Research Institute (EPRI). An evaluation of actual operating experience was performed and utilized to develop LOCA frequencies using a different methodology than has been used to generate the other frequency estimates. The methodology which is documented in EPRI TR-100380 [Ref. 7] involves use of historical data to generate LOCA frequencies on a per pipe segment basis. Events with greater than 50 gpm of leakage were considered ruptures. Use of pipe segments as the basis for the LOCA frequencies was selected because pipe failure rates are dependent on pipe length and the number of welds, with the number of welds being the most important factor. The results from this study were reported in NUREG/CR-5750 and are shown in Figures 4.1 and 4.2. As indicated in these figures, the reported LOCA frequencies are comparable to those from WASH-1400 and NUREG-1150.

The large pipe break frequencies discussed above are for break areas with equivalent diameters of 6 inches or greater. Information on the frequency of DEGBs in RCS pipes is sparse. Estimates of DEGB frequencies for Combustion Engineering (CE) and Westinghouse plants were obtained using probabilistic fracture mechanics techniques. Point estimates, reported in NUREG-1061, are plant-specific and range from 5.5×10^{-14} to 2.5×10^{-11}/calender year. Upper bound estimates (90^{th} percentiles) range from 1×10^{-12} to 1×10^{-9}/calender year. Fracture mechanics results for a BWR 4 plant without IGSCC effects are reported in NUREG/CR-4792 [6]. The point estimates for the DEGB frequency range from 2.5×10^{-13} to 3.8×10^{-11}/calender year. Upper bound estimates range from 1.4×10^{-10} to 1.2×10^{-9}/calender year. Estimates for the frequency of a DEGB due to IGSCC (with no mitigating actions) range from 8×10^{-5}/yr for resistant (316NG) material to 1×10^{-3}/yr in susceptible (304SS) material.

In conclusion, the values provided in NUREG/CR-5750 represent the best estimates of LOCA frequencies that are currently publicly available, but do not consider recent events involving primary stress corrosion cracking (PWSCC). The frequency information on different pipe break sizes suggest that, in a risk-informed environment, the evaluation of the ECCS could be based on pipe breaks significantly smaller than DEGBs but larger than 6 inches in diameter. However, the potential for other causes of RCS LOCAs in addition to random pipe failures must be considered before DEGB pipe breaks are eliminated as surrogates for these events (see Section 4.3.1 for a discussion of these other potential LOCA initiators).

The size of pipe break LOCAs to consider in a risk-informed environment should include the consideration of the expected frequency of the pipe break. Unfortunately, there are no known estimates of the pipe break frequency as a function of pipe diameter other than the coarsely binned information discussed previously. However, the information and approach used in NUREG/CR-5750 was used to generate rough estimates of the mean RCS pipe break frequencies as a function of pipe diameter. The method used to generate the results shown in Figure 4-3 has several limitations and thus the results should only be used as an indicator on whether it would be worthwhile to pursue more detailed efforts to determine RCS pipe break frequencies as a function of pipe diameter. These limitations include:

- the estimates are for ruptures in each size of pipe and do not reflect the frequency of a smaller size LOCA occurring in a large pipe (e.g., a 6 inch break in a 24 inch diameter pipe),
- the uncertainty in the data and methodology has not been quantified but would likely result in 95th confidence estimates higher than 10^{-6}/yr for the large pipe break frequencies shown in the figure,
- data on through-wall cracks in PWRs is sparse (estimates for western type PWRs were thus generated due to the sparsity of crack data in U.S. PWRs),
- the frequencies for pipe breaks in large diameter pipes in PWRs is based on no occurrences of through-wall cracks through 1997 (a Bayesian update was performed using a Jeffrey's non-informative prior),
- several occurrences of through wall cracks have occurred since 1997 (at V.C. Summer and Ringhals) which would double the frequencies for PWR large pipe breaks shown in Figure 4-3, and
- the through-wall cracks in U.S. BWRs occurred because of IGSCC and thus past through-wall frequency estimates may not be representative of the current BWR LOCA potential (consistent with NUREG/CR-5750, an improvement factor of 20 was used to reduce the BWR LOCA frequency estimates to account for IGSCC mitigation strategies).

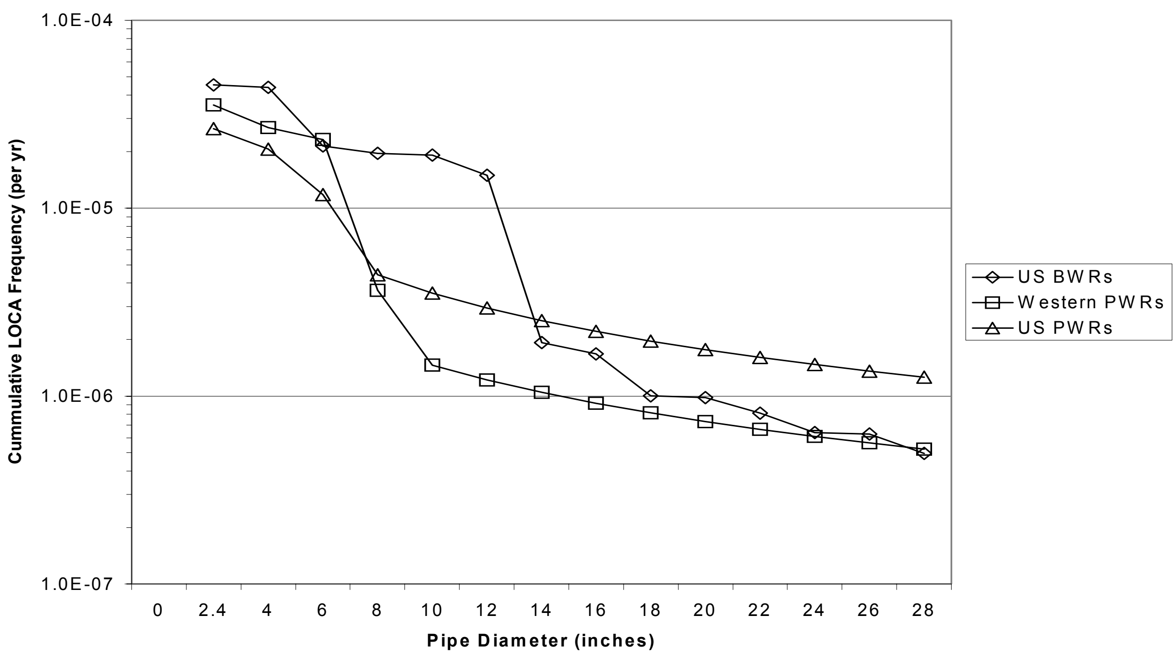


Figure 4-3 Mean pipe break frequency as a function of pipe diameter.

As part of the evaluation of the leak-before-break issue [Ref. 8][Ref. 9][Ref. 10], the frequency of seismic-induced DEGBs were estimated for a large fraction of operating PWRs using fracture mechanics. The frequency of both directly- (i.e., due to fatigue crack growth under the combined effects of thermal, pressure, seismic, and other cyclic loads) and indirectly-induced (i.e., due to other causes such as structural support failures) DEGBs were evaluated. The frequency of a DEGB due to direct seismic-induced failure of the piping was estimated to be one to three orders of magnitude lower than the random DEGB frequencies. However, the frequency of indirect seismic-induced DEGBs were significantly higher. As indicated in Table 4.2, the frequency of an indirectly-induced DEGB are generally low but was estimated to exceed 10^{-6}/yr at some plants. The major contributors were failure of primary equipment supports (i.e., reactor pressure vessel, steam generator, or reactor coolant pump) or the failure of an overhead crane. In addition, the frequency of directly-induced leaks (i.e., through-wall crack) were also estimated for Westinghouse and CE plants. The median seismically-induced through-wall crack frequencies for Westinghouse plants range from 6×10^{-8} to 4×10^{-7}/yr and best-estimates for the CE plant frequencies were all approximately 2×10^{-8}/yr.

The frequency estimates shown in Figure 4-3, the uncertainty associated with these estimates, and the consideration of recent pipe cracking events suggests that it is currently not possible based on existing crack data to conclude that random breaks in the largest diameter pipes are sufficiently rare events (i.e., have frequencies $<10^{-6}$/yr) to exclude them from the licensing bases. Furthermore, available frequency estimates obtained from fracture mechanics evaluations for indirect, seismically-induced DEGBs are also greater 10^{-6}/yr for some plants. Differences between the actual frequencies of through-wall cracks and estimates for pipe leaks reported in NUREG-1061 suggest that these seismic-induced frequencies and other existing pipe break frequencies obtained using probabilistic fracture mechanics may be optimistic. Requirements to improve analytical estimates of pipe break frequencies are delineated in Attachment 2.

Table 4-2 Seismic-Induced DEGB Frequencies for PWRs

Plants	Range in the Frequency of Seismic-Induced DEGBs (per plant-year)			
	Directly-Induced		Indirectly-Induced	
	Median	90th Percentile	Median	90th Percentile
CE PWRs	6E-14 to 5E-13	4E-12 to 7E-11	5E-17 to 6E-6	3E-14 to 5E-5
Westinghouse PWRs	2E-13 to 3E-11	8E-10 to 1E-9	5E-8 to 5E-6	1E-6 to 5E-5
B&W PWRs	Estimated to be <1E-10		6E-11 to 2E-7	8E-9 to 1E-5

4.2.2 Leak-Before-Break Ratio

The assessment that a large RCS pipe is more likely to leak and be detected by the plant's leakage monitoring systems long before cracks grow to unstable sizes is referred to as leak-before-break (LBB) and is a key factor used in complying with regulatory requirements related to the dynamic effects of a LOCA. The probabilistic fracture mechanics analyses documented in NUREG-1061 [5] were performed to help the NRC make decisions on the application of LBB. Probabilistic fracture mechanics calculations for CE and Westinghouse PWRs were performed to determine estimates of the frequency of leaks and DEGBs. The median frequency of leaks for Westinghouse plants ranged from 5.5×10^{-8} to 3.8×10^{-7}/calender year. The corresponding median frequency for DEGB in Westinghouse plants range from 2.2×10^{-13} to 2.5×10^{-11}/calender year. For CE plants, the calculated best estimate leak frequencies range from 1.5×10^{-8} to 2.3×10^{-8}/calender year and the best estimate DEGB frequencies range from 5.5×10^{-14} to 4.5×10^{-13}/calender year. Since not all pipe breaks will be DEGBs, the ratio of the probabilistic fracture mechanics results can only provide an upper bound for a LBB ratio for large RCS pipes. The leak/DEGB ratios range from approximately 10^4 to 5×10^5.

The Beliczey and Schulz correlation was used in NUREG/CR-5750 to determine the probability of a pipe rupture given the occurrence of a through-wall crack (see Section 4.2.1). Comparisons of this correlation with other data and analyses is also reported in NUREG/CR-5750. When compared to conditional break probabilities obtained from a Bayesian assessment of the actual operating experience, the Beliczey and Schulz correlation provides very similar results. However, when compared to the results of probabilistic fracture mechanics evaluations, the Beliczey and Schulz correlation provides generally conservative results. Assuming that all ruptures are preceded by through-wall cracks and may be less than a DEGB, the inverse of this value (i.e., 10^2) provides a reasonable estimate of the LBB ratio for a 10 inch diameter pipe. For larger diameter RCS pipes, a LBB ratio based on the Beliczey and Schulz correlation could approach 10^3. Note that these values are a factor of 50 to 100 lower than the leak/DEGB ratios obtained by fracture mechanics evaluations.

4.2.3 Probability of Coincident Loss of Offsite Power

GDC 35 requires that the ECCS provide sufficient core cooling during a LOCA when it is powered by either onsite or offsite power. To comply with this requirement, ECCS evaluations generally assume a pipe break with a coincident LOOP. This assumption generally results in requirements for fast diesel generator start times that provide significant stresses on the diesels that may reduce their reliability. This section evaluates the potential for a pipe break LOCA followed by a LOOP.

An evaluation of the potential for a pipe break followed by a LOOP was documented in NUREG/CR-6538 [Ref. 11]. The evaluation was performed as part of the resolution of GSI-171 "ESF Failure from LOOP Subsequent to LOCA." In this report, three reasons were identified why there may be an increase in the likelihood of a LOOP immediately following a LOCA:

1. A LOCA will cause a reactor trip and a generator trip. In addition, the emergency diesel generators (EDGs) will start automatically, but will not be connected to the safety buses unless an undervoltage occurs at the buses. The loss of the main generator disturbs the offsite power grid and therefore can possibly lead to a loss of offsite power to the plant.

2. The reactor trip also will cause a fast transfer of the power supply to those buses that normally receive their power from the main generator. This transfer is from the auxiliary transformer to the startup transformer (powered from the offsite grid). Problems in the fast transfer could lead to a loss of power to the safety buses and require that the EDGs be connected to the safety buses.

3. If the fast transfer is successful, those loads that were originally on the safety buses will continue to operate without interruption, and the ECCS loads will be added onto the safety buses. The addition of the ECCS loads can cause an undervoltage at the safety buses requiring that the EDGs be connected to the buses.

The first two causes can occur subsequent to any reactor trip (including that caused by a LOCA), and the last cause can occur anytime the ECCS is actuated (including a LOCA). For this reason, NUREG/CR-6538 used experience data for LOOPs following any reactor trip or ECCS actuation in estimating the potential for a LOOP coincident with a pipe break LOCA.

Reactor trip events contained in the licensee event report (LER) sequence coding search system (SCSS) were reviewed in NUREG/CR-6538 along with other reports to identify the occurrences of reactor trip events with an immediate LOOP. The LER events in the SCSS were also used to identify occurrences of a LOOP immediately following an ECCS actuation. The results of the NUREG/CR-6538 evaluation are summarized in Table 4.3. The estimates provided in this table are based on relatively sparse data. The estimates are averages over the population of plants and values for specific plants may vary significantly depending on the plant's vulnerability to undervoltages. Furthermore, since the data was obtained from events prior to 1993, any decrease in the grid stability resulting from electric power deregulation may result in higher probabilities since there could be a greater potential for the occurrence of undervoltage events following reactor trips and ECCS actuations leading to a LOOP.

The probabilities of a LOOP following a LOCA generated in NUREG/CR-6538 are two orders of magnitude higher than those used in NUREG-1150 and the IPEs. Typically, these studies modeled a LOOP as being independent of the LOCA. As such, the probability of a LOOP during a 24 hour mission time was typically evaluated by multiplying the LOOP initiating event frequency by 24 hours. In the NUREG-1150 studies, this resulted in a probability of a LOOP following any reactor trip of 2×10^{-4}.

Table 4-3 Conditional Probability of a LOCA Followed by a LOOP

Plant Type	Probability of a LOOP given a reactor trip	Probability of a LOOP given an ECCS actuation	Probability of a LOOP given a pipe break LOCA[1]		
			Point Estimate	5^{th} Percentile	95^{th} Percentile
BWR	3.7E-03	5.6E-02	6.0E-02	4.5E–03	2.5E-01
PWR	3.9E-03	1.0E-02	1.4E-02	2.7E-03	5.5E-02
Total	3.8E-03	1.7E-02	2.1E-02	5.7E-03	6.0E-02

[1] The point estimate is the sum of the probabilities of a LOOP given a reactor trip and a LOOP given an ECCS actuation.

Although the conditional probability of a LOOP subsequent to a random pipe break LOCA is higher than as been generally modeled in existing PRAs, the frequency of this combination of events is small. Combining the conditional probabilities in Table 4.3 with the frequency of a large LOCA (>6 inches in diameter) shown in Table 4.1 results in point estimates for a random large pipe break LOCA with a subsequent LOOP of 2.4×10^{-7} and 5.6×10^{-8}/calender year for BWRs and PWRs, respectively. These results suggest that modeling of all sizes of random large pipe break LOCAs with a coincident LOOP in the evaluation of ECCS performance is not necessary in a risk-informed environment.

NUREG/CR-6538 also addresses the potential for a simultaneous LOCA/LOOP event. Based on the design of the electrical systems at nuclear power plants, the study concluded that a delay ranging from several seconds to several minutes will occur between the occurrence of a LOCA and a consequential LOOP. The delay is a consequence of delays in tripping the main generator following a reactor trip and delay in the bus undervoltage relays in sensing the power drop and transferring power from the switchyard to the EDGs. Actual LOOP events were also reviewed to estimate the timing of the LOOP events following a LOCA. The timing could only be identified for 5 of 12 events and ranged from 34 seconds to 5 minutes. Estimates for the remaining events, generated based on electrical design characteristics, were used to obtain the distribution for the timing of LOOP events shown in Figure 4-4. Based on this information, if a consequential LOOP does occur following a LOCA, the probability that it occurs in less than 5 seconds is approximately 0.1. Therefore, if a consequential LOOP occurs following a LOCA, it is highly probable that it will be a delayed LOOP.

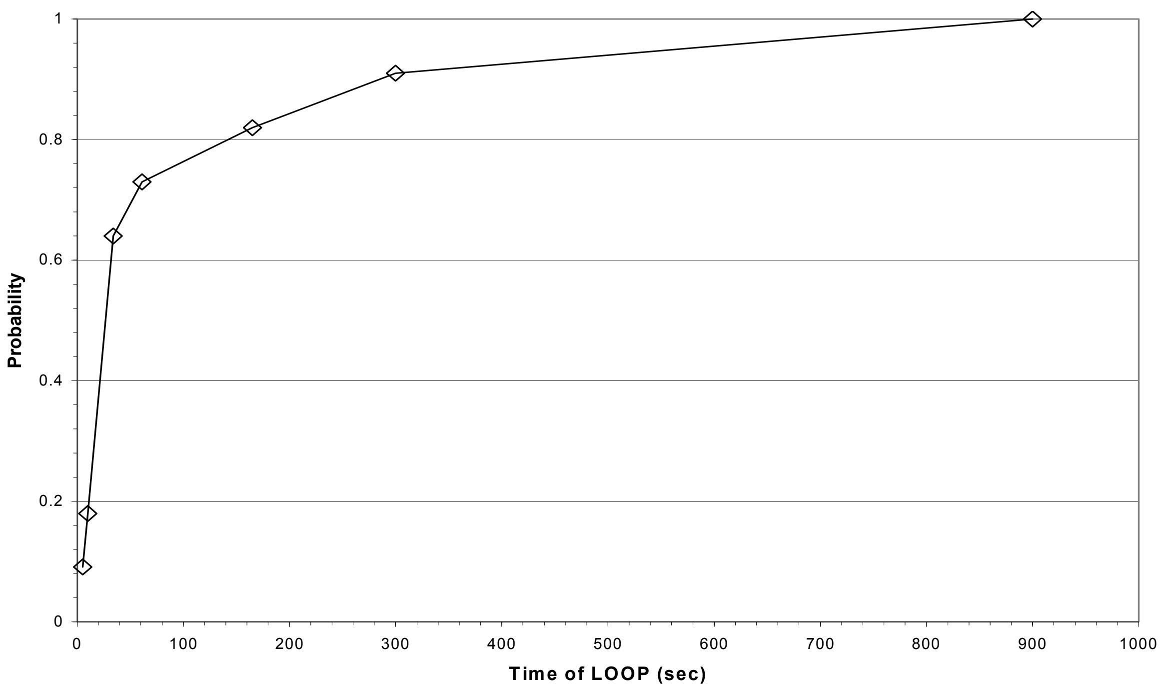


Figure 4-4 **Cumulative probability distribution for the timing of a LOOP event following a LOCA**

Currently, ECCS evaluations are performed assuming a simultaneous occurrence of a LOOP. This modeling may not represent the most limiting LOCA/LOOP situation. Specifically, the occurrence of a delayed LOOP will result in "double sequencing" of the ECCS and containment system equipment. The emergency equipment will initially be sequenced onto the offsite power system following a reactor trip. Following a delayed LOOP, the emergency systems will be separated from the offsite power system (the ECCS pumps will trip) and be sequenced on to a diesel generator before restarting. The impact of this double sequencing may influence the results of ECCS acceptance calculations.

A LOOP can also be followed by a LOCA. However, a LOOP would not increase the RCS pipe break frequency or introduce a new pipe break failure mechanism. Thus, the potential for an RCS pipe break following a LOOP remains mostly independent and small. One exception involves the potential for a LOOP to result in a LOCA involving one or more stuck-open safety-relief valves (SRVs) in a BWR or a stuck-open power-operated relief valve (PORV) or reactor coolant pump (RCP) seal LOCA in a PWR. These events can result in high CDFs particularly in PWRs where RCP seal LOCA contributions during station blackouts (SBOs) are high for many plants (see NUREG/CR-6538, Table 5-1).

The probability that a LOOP will occur during an earthquake is much higher than during non-seismic events. This is due to the fact that offsite transmission grids are not capable of withstanding significant size earthquakes. Therefore, it is not unreasonable to assume that offsite power will be lost in any earthquake sufficient in magnitude to challenge the integrity of RCS piping and the operability of the safety systems in nuclear power plants. Estimates of the frequency of seismic events leading to through-wall cracks and a LOOP are sufficiently low that the frequency of such an event appears to be a rare event. However, as previously indicated, through-wall crack frequency estimates generated by fracture mechanics techniques do not agree well with actual operating data. Furthermore, the largest estimated frequency of an indirectly-induced large pipe break (a LOOP would likely occur) during an earthquake is 6×10^{-6}/yr (see Table 4.2). These estimates suggest that the frequency of a seismically-induced LOCA/LOOP event may be greater than the frequency limit in the Option 3 framework for a rare event and thus may have to be considered in ECCS evaluations.

4.2.4 Single Failure Probabilities in Large-Break LOCAs

GDC 35 requires that the ECCS be capable of providing sufficient core cooling during a LOCA even when a single failure is assumed. SRP 6.3 interprets this as requiring the ECCS to perform it's function during the short-term injection mode in the event of the failure of a single active component and to perform it's long-term recirculation function in the event of a single active or passive failure. In practice, the single active failure of one of the trains of emergency AC power is the most limiting when offsite power is unavailable since it disables one train of ECCS. This section discusses the redundancy and reliability of the ECCS that results from implementation of the single failure criterion.

All reactors operating in the United States have multiple trains of ECCS capable of mitigating the full spectrum of LOCAs. PWRs have two or three trains of high-pressure safety injection (HPSI) and two trains of low-pressure safety injection (LPSI). Although BWRs only have one train of high-pressure core spray (HPCS) or high-pressure coolant injection (HPCI), they also have the capability to depressurize the reactor vessel so that low-pressure ECCS can be utilized. BWRs generally have one or two trains of low-pressure core spray (LPCS) and two or three trains of low-pressure coolant injection (LPCI). Redundant divisions of electrical power and trains of cooling water are also available to support ECCS operation and together, provide the redundancy necessary to meet the single failure criterion.

Figure 4-5 provides the CCDPs calculated for LOCAs in the IPEs. The CCDPs for large and medium LOCAs primarily reflect the unreliability of the ECCS while the small break CCDPs includes credit for non-ECCS systems (e.g., the reactor core isolation cooling [RCIC] system in BWRs).

The average CCDPS for BWRs are approximately an order of magnitude lower than the values for PWRs and reflect the greater redundancy for providing coolant injection that exists in BWRs. On average, the CCDPs for both PWRs and BWRs meet the quantitative goal of 0.01 set for infrequent events in the Option 3 framework. However, the results also show a significant variation in the CCDPs that is due to a combination of plant design characteristics and assumptions used in the IPE analyses. One important contributor to the higher CCDPs in PWRs without automatic switchover to recirculation is an operator error to manually perform the switchover.

4.2.5 Contribution of Large Pipe Break LOCAs to CDF

A comparison of the CDFs from different random pipe break LOCA sizes for BWRs and PWRs is provided in Figures 4.6 and 4.7, respectively. As indicated in the figures, the contribution of large pipe break LOCAs to CDF is generally small. Most CDF estimates for large-break LOCAs in PWRs are less than 10^{-5}/yr and are less than 10^{-6}/yr for BWRs. If the values presented in Table 4.1 are representative of LOCA frequencies, the CDF estimates would decrease by an order of magnitude. On average, the contribution for PWRs reflected in Figure 4-6 is approximately 5% of the total internal event CDF. For BWRs, the large LOCA CDF contribution is significantly smaller (~1%) due to the greater redundancy in coolant injection capability.

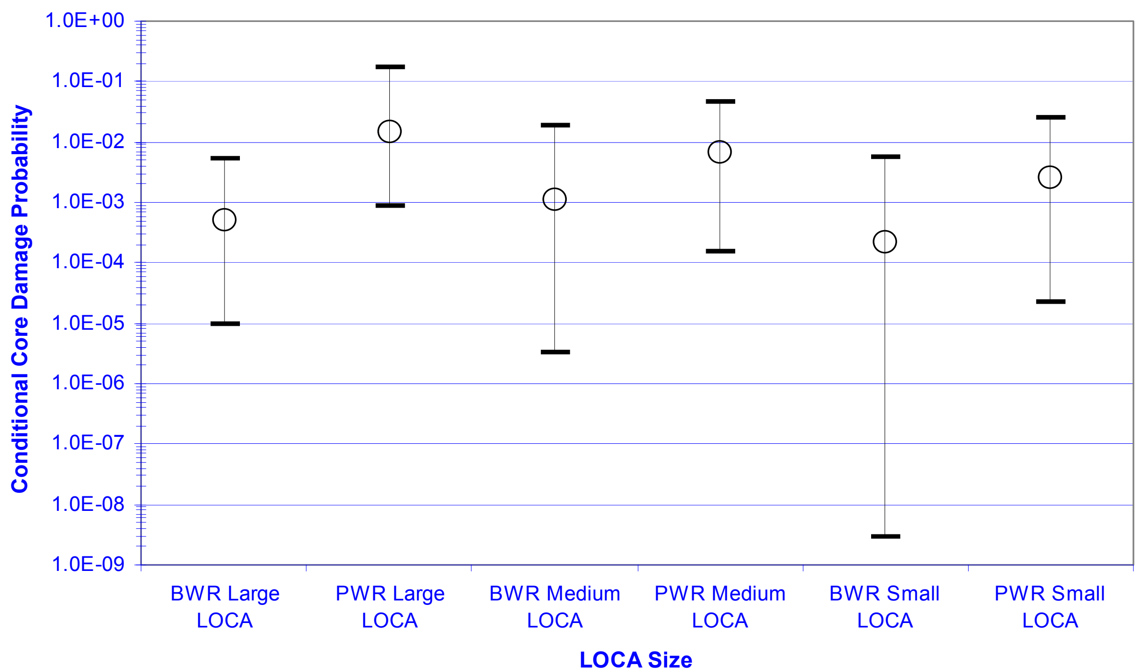


Figure 4-5 LOCA conditional probability of core damage (IPEs).

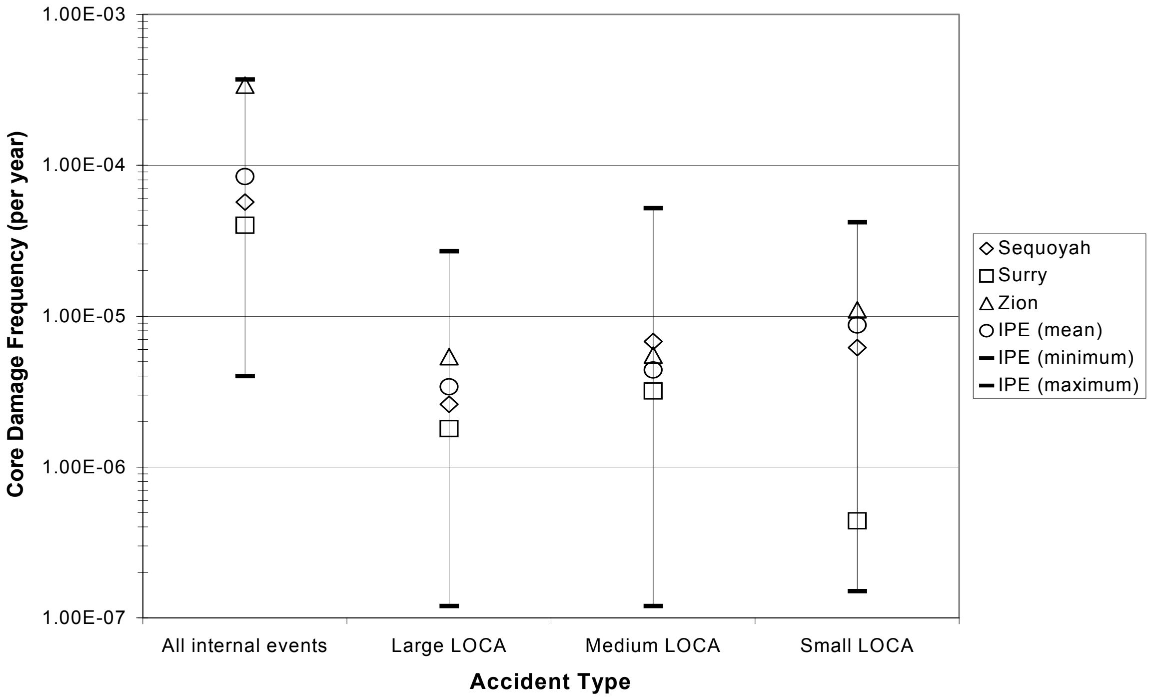


Figure 4-6 PWR pipe break LOCA CDFs.

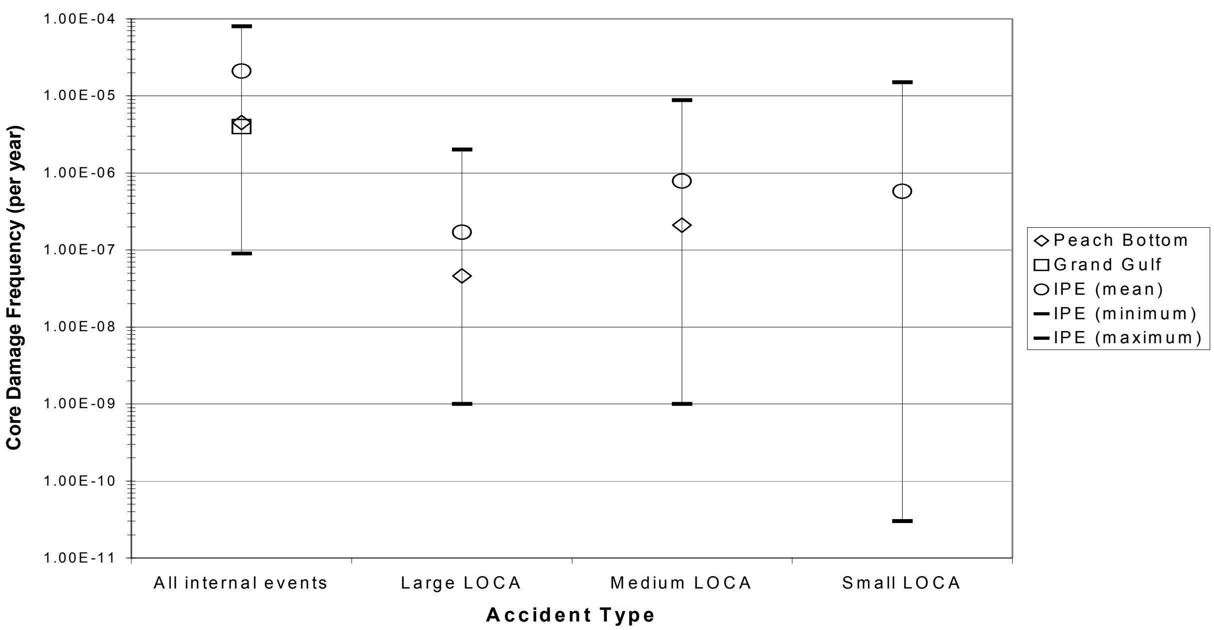


Figure 4-7 BWR pipe break LOCA CDFs

Although the contribution from large pipe break LOCAs has been calculated to be small in many PRAs, it should be realized that the modeling of these initiators has been generally incomplete. Specifically, the phenomenological impacts from a LOCA are generally ignored. These impacts can include pipe whip, jet impacts, asymmetrical blowdown effects, and ECCS pump suction plugging. The latter two are believed the most significant since pipe whip and jet impacts from RCS pipe breaks generally would only challenge the injection piping and valves for one ECCS train whereas asymmetrical blowdown loads can disrupt the core and ECCS sump plugging can fail all ECCS trains. Except for pump suction plugging(see Section 4.4.2.1 for further discussion), there are no known estimates of the contribution to the LOCA CDF from these effects.

An additional modeling area that was insufficient in past LOCA assessments is in the area of consequential LOOPs. As indicated in Section 4.2.3, the probability of a LOOP coincident with a LOCA could be two orders of magnitude higher than what has been typically assumed in past PRAs. LOCA/LOOP CDF estimates were generated in NUREG/CR-6538 for eight theoretical plant groups differentiated by electrical power design features. The CDFs from large LOCAs range from 7.3×10^{-7}/yr to 3.4×10^{-5}/yr for PWRs and 5.9×10^{-7}/yr to 2.9×10^{-5}/yr for BWRs. These values were generated using the NUREG-1150 LOCA frequencies and are higher than those typically reported in PRAs for LOCA/LOOP events. However, note that if the LOCA frequencies in Table 4.1 were used in the NUREG/CR-6538 evaluation, the CDF values would be lower by over an order of magnitude. Thus, the contribution from consequential LOCA/LOOP scenarios to CDF would be expected to be small.

There is very little information available concerning the risk during low power and shutdown conditions. Several studies sponsored by the NRC have been performed to evaluate the risk during low power and shutdown conditions. One BWR (Grand Gulf) and one PWR (Surry) have been the subject of these studies. Table 4.4 provides the LOCA CDF contributions for Grand Gulf during low power/shutdown conditions. The LOCA contributions during cold shutdown [Ref. 12] are higher than those during power operations. During refueling [Ref. 13], the LOCA contribution is due to drain down events.

In the Surry Low power/shutdown analysis [Ref. 14], no substantial CDF contribution from LOCAs was reported. However, in a subsequent analysis of Surry [Ref. 15], a large LOCA CDF contribution during cold shutdown and refueling operations of 2.5×10^{-6}/yr was reported. No contribution from drain down events was identified.

LOCA contributions reported in low power/shutdown PRAs of foreign reactors range from 1.4×10^{-8}/yr to 7×10^{-7}/yr for BWRs and 3.4×10^{-6}/yr to 3.7×10^{-7}/yr in PWRs. The contribution from drain down events in these foreign studies range from 2.9×10^{-8}/yr to 2.6×10^{-6}/yr for BWRs and 3.6×10^{-7}/yr to 1.5×10^{-6}/yr for PWRs.

Estimates of the CDF from seismically-induced LOCAs were calculated in the NUREG-1150 studies of the Peach Bottom [Ref. 16] and Surry plants [Ref. 17]. The contributions from seismically induced LOCAs of various sizes and from vessel rupture were estimated. The frequency of a vessel rupture was computed based on the failure of the reactor vessel supports (peach Bottom) or the supports of the steam generators and reactor coolant pumps (Surry). At both plants, the evaluation of large LOCAs only considered indirectly-induced pipe failures. The frequency of a seismically-induced large LOCAs was calculated based on the failure of the supports for the recirculation pumps at Peach Bottom and failure of the steam generator and reactor coolant at Surry. The small and medium LOCA frequencies were based on seismically-induced failures of the appropriate size pipes.

Table 4-4 LOCA Contributions During Low/Power Shutdown Operation at Grand Gulf

Accident Initiating Event	Initiating Event Frequency	Core Damage Frequency		
		Cold Shutdown (POS 5)[1]	Refueling (POS 6)[2]	Refueling (POS 7)[2]
Large LOCA	3.6E-05	4.8E-07	Screened[3]	Screened[3]
Large LOCA during hydro test	1.3E-04	2.1E-07	NA[4]	NA[4]
Medium LOCA	3.6E-05	2.5E-07	Screened[3]	Screened[3]
Medium LOCA during hydro test	1.3E-04	2.1E-07	NA[4]	NA[4]
Diversion to suppression pool via RHR	6.1E-02	1.3E-07	1.3E-08	7.6E-09
LOCA in RHR	1.6E-02	2.1E-08	4.2E-07	3.7E-07
Total		1.3E-06	4.3E-07	3.8E-08

[1] Core damage frequencies are from NUREG/CR-6143.
[2] Plant operational state (POS) 6 is during refueling when the water level is at the steam lines. POS 7 is during refueling when the vessel is flooded up to the upper containment pool and the refueling transfer tube is open. CDFs are from NUREG/CR-5593.
[3] LOCAs in the RCS were not considered credible during refueling due to the RCS being at atmospheric pressure.
[4] Not applicable during POS 6 or 7 since hydro testing is only performed during cold shutdown.

The seismic-induced LOCA contributions calculated in the NUREG-1150 studies are summarized in Table 4.5. Results for two different sets of seismic hazard curves (Lawrence Livermore National Laboratory [LLNL] and EPRI) are presented for each plant. For both plants, the CDFs for seismically-induced large LOCAs (using either hazard curve) is greater than the CDF from random large LOCAs. The important seismic-induced LOCA sequences for both plants involve a LOOP event. For Peach Bottom, all of the dominant LOCA scenarios include the additional failure of onsite power leading to LOCA/SBO scenarios.

Table 4-5 Seismic-Induced LOCA Core Damage Frequencies

Accident Type	Mean Core Damage Frequency	
	LLNL Hazard Curve	EPRI Hazard Curve
Large LOCA		
Peach Bottom (NUREG-1150)	1.9E-05	6.8E-07
Surry (NUREG-1150)	7.7E-06	1.3E-06
Medium LOCA		
Peach Bottom (NUREG-1150)	7.4E-06	2.1E-07
Surry (NUREG-1150)	1.5E-06	1.7E-07
Small LOCA		
Peach Bottom (NUREG-1150)	1.5E-06	5.5E-08
Surry (NUREG-1150)	6.8E-06	1.3E-06

Table 4-5 Seismic-Induced LOCA Core Damage Frequencies

Accident Type	Mean Core Damage Frequency	
	LLNL Hazard Curve	EPRI Hazard Curve
Vessel Rupture		
Peach Bottom (NUREG-1150)	8.9E-06	3.3E-07
Surry (NUREG-1150)	3.3E-06	5.5E-07

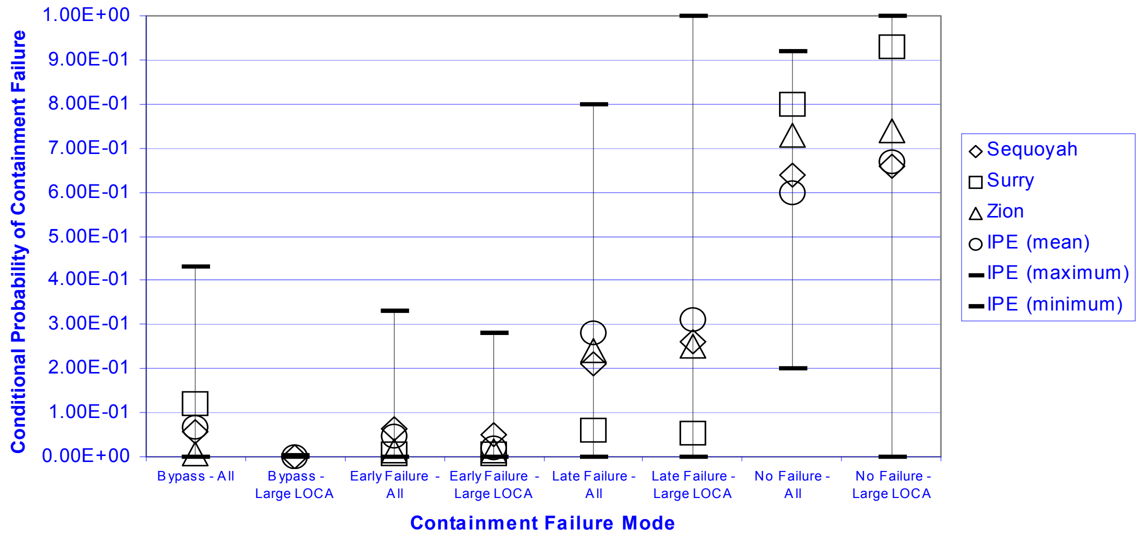
In summary, current estimates of the CDF contribution from random LOCAs during full power operation are generally less than 10^{-5}/yr for PWRs and less than 10^{-6}/yr for BWRs. Thus, the estimated large LOCA CDFs for most plants are less than 10% of the Option 3 framework CDF guideline. Limited information on the large LOCA contribution during low power and shutdown conditions were reviewed in this assessment. The limited information suggests that large LOCA contributions to CDF during low power and shutdown conditions is small but that seismically-induced LOCAs may result in contributions equivalent to the contribution from random pipe breaks.

4.2.6 Conditional Probability of Containment Failure Given Core Damage

All of the PWR and BWR containments are designed to withstand large-LOCA blowdown loads with considerable margin. Containment failure pressures are generally greater than two times the design pressure. Containments are also designed to have very low leakage rates when subjected to the internal pressures that would occur during a design-basis LOCA. However, if a LOCA proceeds to core damage, the challenge to containment is more severe.

The conditional probabilities of different containment failure modes given core damage following a LOCA event are shown in Figures 4.8 and 4.9. The probabilities of containment failure during LOCAs from the NUREG-1150 studies and IPEs are contrasted with the containment failure modes from all internal events. For both BWRs and PWRs, the conditional probabilities of the different containment failure modes following a LOCA are similar the values for all internal events.

The Option 3 framework includes quantitative guidelines for both LERF and the conditional large early release probability. Estimates of the average values for these parameters for large-break LOCAs obtained from the IPEs are provided in Table 4.6. As indicated in this table, the average LERF values for large LOCAs are significantly lower than the quantitative guideline of 10^{-5}/yr. In addition, the average conditional large early release probabilities during a large LOCA are also below the quantitative guideline of 0.1.



Fi

gure 4-8 **PWR conditional probability of containment failure.**

Figure 4-9 BWR conditional probability of containment failure.

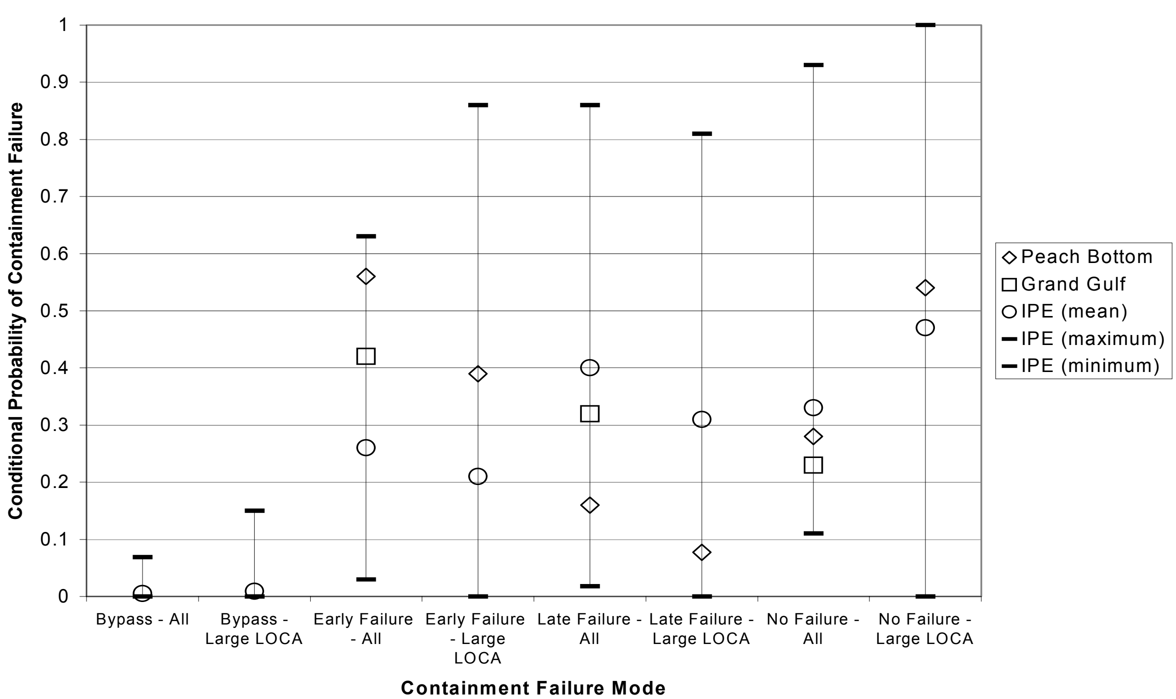


Table 4-6 Contribution to LERF from Large LOCAs

Plant Type	Average Large Early Release Frequency		Average Conditional Large Early Release Probability	
	All Internal Events	Large LOCAs	All Internal Events	Large LOCAs
PWRs	5E-6/yr	3E-8/yr	6E-2	1E-2
BWRs	2E-6/yr	4E-9/yr	9E-2	3E-2

4.3 Frequency of Other Large Internally-Initiated LOCAs

Loss-of-coolant accidents are defined in 10 CFR Part 50, Appendix A, "Definitions and Explanations," as "...breaks in the reactor 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." The definition of LOCA in 50.46 limits the breaks that have to be evaluated to breaks in RCS pipes. Currently, a DEGB in the largest RCS pipe is the upper bound that is used for evaluating the adequacy of the ECCS. In risk-informing 50.46, it is prudent to consider other RCS component failures that may result in large LOCAs. Table 4.7 presents estimates of the frequencies of both leaks and ruptures in other RCS components.

Table 4-7 Potential Causes and Frequencies of Loss of Primary Coolant

Source of Loss-of-Coolant	Frequency Estimates	References	Comments
Reactor Vessel			
Vessel rupture	3E-7/yr	WASH-1400	
CRDH housing failure	-		No frequency estimate available Too small for LBLOCA
Head closure seal leakage rupture	9E-5/yr 9E-7/yr	EGG-SSRE-9639 WSRC-TR-93-262	Reflects flange/gasket failure estimates. A rupture could be a LBLOCA.
Failure of instrumentation penetrations	-		No frequency estimate available Too small for LBLOCA
Inadvertently-open head vent valve	9E-4/yr	NUREG/CR-4550	Frequency of spuriously-open MOV, two valves may have too open Too small for LBLOCA
Pressurizer			

Table 4-7 Potential Causes and Frequencies of Loss of Primary Coolant

Source of Loss-of-Coolant	Frequency Estimates	References	Comments
Shell leakage rupture	9E-5/yr 9E-7/yr	EGG-SSRE-9639	Reflects data recommended for a pressurized tank. Estimated based on 12 leakage events (1 in the PCS) and 2 rupture events (0 in PCS). A rupture could be a LBLOCA.
Surge line rupture	-		Assumed included in other PCS piping failure frequency
Pressurizer spray line rupture	-		Assumed included in other PCS piping failure frequency
Inadvertently-open PORV	1E-3/yr	NUREG/CR-5750	Too small for LBLOCA
Stuck-open safety valves 1 SORV (BWR) 1 SORV (PWR) 2 or more SORVs	 5E-3/yr 4.6E-2/yr 3.2E-4/yr	 NUREG/CR-5750	Three SORVs would be a LBLOCA.
Failure of instrumentation penetrations	-		No frequency estimate available Too small for LBLOCA
Steam Generator			
Single tube rupture Multiple tube ruptures	7E-3/yr -	NUREG/CR-5750	Based on 3 reported events. Multiple SGTRs would result in a LBLOCA. No frequency estimate identified for multiple SGTRs.
Shell leakage rupture	9E-5/yr 9E-7/yr	EGG-SSRE-9639	Reflects estimates for heat exchangers. Estimates based on 2 leakage events and 0 rupture events in non-PCS HTXs. A rupture could be a LBLOCA.
Manway failure	-		No frequency estimate available
Failure of instrumentation penetrations	-		No frequency estimate available Too small for LBLOCA
Reactor Coolant Loops			
Pump casing leakage rupture	3E-4/yr 3E-6/yr	EGG-SSRE-9639	Based on 50 reported incidences of external leakage (4 in PCS) and 2 rupture events (0 in PCS). A rupture could be a LBLOCA.
RCS pump seal failure	2.5E-3/yr	NUREG/CR-5750	Too small for LBLOCA.

Table 4-7 Potential Causes and Frequencies of Loss of Primary Coolant

Source of Loss-of-Coolant	Frequency Estimates	References	Comments
Valve body leakage rupture	9E-5/yr 9E-7/yr	EGG-SSRE-9639	Estimated based on 170 incidences of reported leakage (29 in PCS) and 7 rupture events (none in the PCS). A rupture could be a LBLOCA.
Large pipe rupture BWR PWR	 2E-5/yr 4E-6/yr	 NUREG/CR-5750	Frequencies are for BWR pipe sizes >5 inches in diameter and PWR pipe sizes >6 inches in diameter.
Accumulator leakage rupture	9E-5/yr 9E-7/yr	EGG-SSRE-9639	Reflects data recommended for a pressurized tank. Estimated based on 12 leakage events (1 in the PCS) and 2 rupture events (0 in PCS). Valve failures would also have to occur to result in loss of primary coolant.
Failure of instrumentation penetrations	-		No frequency estimate available Too small for LBLOCA
Interfacing System LOCAs			
BWRs	9.6E-4/yr	NUREG/CR-5750	This is the mean CDF of the distribution of IPE point estimates. Not all ISLOCAs may be LBLOCAs.
PWRs	5.1E-5/yr	NUREG/CR-5750	This is the mean CDF of the distribution of IPE point estimates. Not all ISLOCAs may be LBLOCAs.
Low Power/Shutdown Events			
Draindown events	6.1E-2/yr	NUREG/CR-5593	Frequency is for a BWR. The frequency does not include the fraction of time the plant is shutdown.
Inadvertent overpressurization (makeup greater than letdown or spurious ECCS actuation)	1.6E-3/yr 1.4E-2/yr	NUREG/CR-5593	Frequency is for a BWR. Applicable for cold shutdown only. The frequency does not include the fraction of time the plant is in cold shutdown.
ISLOCA (RHR)	1.6E-2/yr	NUREG/CR-5593	Frequency is for a BWR. The frequency does not include the fraction of time the plant is shutdown.
Heavy load drop	3.4E-6/lift	[Ref. 18]	Estimate is for a non single-failure proof load handling system. For systems that are single-failure-proof, the estimate is 1E-7/lift.

The estimates provided in the table indicate that the frequency of ruptures in RCS components are generally less than the mean large pipe break frequencies calculated in NUREG/CR-5750, though no effort has currently been made to compare uncertainty bounds. Many of the estimated frequencies in Table 4.7 are less than 1E-6/yr. However, there are some component failures where estimated failure frequencies have not been identified (e.g., steam generator manway failures and multiple steam generator tube ruptures) or have relatively high frequencies (e.g., multiple stuck-open SRVs and drain down events during shutdown). In addition, the size of these component ruptures or events are generally smaller in area than a DEGB in the largest RCS pipe (potential exceptions include a vessel rupture and steam generator manway failure). Thus, the capability of the ECCS to mitigate a DEGB pipe break in the RCS ensures that the majority of the RCS failures identified in Table 4.7 can also be mitigated.

4.4 Safety Significance of ECCS

The risk from large-break LOCAs is relatively small primarily due to the fact that the ECCS has been included in the design of nuclear power plants to mitigate DEGBs. The availability of the ECCS is also beneficial in the mitigation of the other accidents. Section 4.4.1 discusses the importance of the ECCS to the overall plant risk. Any changes to the ECCS with regard to LOCA mitigation may have implications on the ability to mitigate other potential accidents and thus could impact the overall plant risk.

Several issues concerning the design of the ECCS have been identified over time. Some of these issues are in the process of being evaluated. In addition, the IPE process has identified significant insights on the capability of the ECCS to mitigate the full spectrum of possible accidents. These issues and insights and their potential risk-significance are discussed in Section 4.4.2.

4.4.1 Importance of ECCS to Risk

In addition to mitigating LOCAs, the ECCS can be used to mitigate transients. If the normal decay heat removal capability is lost in either a BWR or PWR, the high-pressure ECCS systems can be used in a feed and bleed mode of operation to provide coolant to the vessel and remove decay heat from the core (some PWRs do not have this capability). BWRs also have the capability to use low-pressure ECCS for coolant injection to mitigate a transient due to the ability to depressurize the vessel.

A review of the IPE insights reveals that transient scenarios involving the failure of the ECCS results in a significant fraction of the total risk at most nuclear power plants. In many cases, the dominant failure mode of the ECCS involves failure of required support systems. This is particularly true in some PWRs where the failure of cooling water systems can result in an RCP seal LOCA and fail the ECCS pumps. In BWRs, an important failure of the ECCS involves the failure to depressurize the vessel to allow injection from low-pressure systems. Failure to remove heat from the containment during transients (or LOCAs) was predicted to result in adverse environmental conditions that can fail the ECCS (many BWRs are more susceptible to these scenarios than are PWRs due to ECCS pump design differences). These issues are discussed further in Section 4.4.2.

Station blackout scenarios are important contributors to risk at most BWRs and PWRs. PWRs rely on steam-driven auxiliary feedwater (AFW) to remove decay heat and do not have any ECCS systems that are capable of functioning during an SBO. Thus, core damage can occur during SBO scenarios due to the occurrence of RCP seal LOCAs or due to the failure of AFW. BWRs have limited ECCS capability during an SBO. Failure of steam-driven ECCS pumps due to battery depletion or high-temperature in the suppression pool were identified as important contributors in the IPEs.

4.4.2 Potential ECCS Risk Issues

This section identifies some risk issues related to the ECCS in both BWRs and PWRs. The majority of these issues were gleaned from the IPEs but the discussion also addresses some Generic Safety Issues that are still being evaluated. Since the issues are sometimes dependent upon the accident type, the issues are identified along those lines. To help determine if these issues should be pursued further in the alternatives for risk-informing 50.46, the risk significance of these issues was identified, where possible, and compared to the Option 3 framework quantitative guidelines for CDF, CCDP, and LERF.

4.4.2.1 BWR ECCS Issues

LOCAs

The LOCA CDFs for BWRs (excluding Big Rock Point) reported in the IPEs range from negligible to 7.8×10^{-6}/yr and the mean value is 1×10^{-6}/yr. The CCDP for LOCAs reported in the IPEs range from negligible to 3E-3 (excluding Big Rock Point). The mean LERF for large-break LOCAs reported in the IPEs is 4×10^{-9}/yr and the mean conditional probability of a large early release during a large-break LOCA is 3×10^{-2}. These values are 10% or less of the Option 3 framework quantitative guidelines.

One issue was identified related to ECCS operation during a LOCA that could increase the LOCA risk contribution. This issue is related to the LPCI success criteria for a large-break in a recirculation line. Because of the design of the plant, the core may only be reflooded to two-thirds core height for large LOCAs with break flows equivalent to the capacity of one LPCI pump. If only one LPCI pump is available for operation and if flow is not controlled, the water in the vessel maybe subcooled and there will not be steam cooling of the top third of the core. Current ECCS evaluation calculations do not proceed this far in time. This issue may be more significant for low-power shutdown events where the potential for subcooling would be greater due to the lower decay heat. Procedural guidance to maintain steam cooling in this situation would be beneficial. Also, this may be less of a problem for LPCI systems that inject into the shroud area. No risk estimate exists for this issue. However, any risk increase would be expected to be minimal due to procedural guidance to monitor core exit temperatures. Thus, no actions are anticipated to address this issue in risk-informed options. However, PRA evaluations used in complying with risk-informed alternatives to 50.46 and any other regulation should address the need to control ECCS flow in such scenarios.

Transients with loss of injection

An important BWR issue identified in the IPEs with regard to the use of the ECCS to mitigate transients is the failure to depressurize the vessel for low-pressure ECCS injection. Despite having an automatic depressurization system (ADS), automatic depressurization of the vessel is inhibited at most BWRs by procedures. The operator error rates reported in the IPEs for failing to manually depressurizing the vessel, after inhibiting ADS, varied by three orders of magnitude (from 10^{-1} to 10^{-4}; this can be considered the range in the CCDPs for this concern). Transient scenarios involving this error were important at many BWR 3/4s and 5/6s. Commonly identified plant improvements to address this issue involved procedural improvements and increased emphasis on operator training on vessel depressurization.

Additional ways to address this issue could include (a) removing the guidance for inhibiting ADS except for anticipated transient without scram (ATWS) scenarios and/or (b) changing the ADS initiation logic (e.g., instead of LEVEL 1, use top of active fuel [TAF] as the vessel level for actuation).

The total transient contribution for BWRs reported in the IPEs range from 1.3×10^{-8}/yr to 5×10^{-5}/yr and the average value is 9×10^{-6}/yr. CCDP values are not currently available but would be expected to have the same range in values since the frequency of transient initiators is approximately 1/yr. LERF values are not readily available. The average IPE transient CDF value is 10% or less of the Option 3 framework quantitative guideline. Since the contribution from failing to depressurize the vessel is only a fraction of the above CDF value, any changes to the ECCS to address this issue or any other concern pertinent to transients would be difficult to justify.

Transients with loss of decay heat removal

A problem identified in some PRAs is lack of adequate net positive suction head (NPSH) problems with ECCS pumps taking suction from suppression pool for scenarios where suppression pool cooling is unavailable. Related to this issue is the capability of the ECCS pumps to pump saturated water. High temperatures in the suppression pool during loss of decay heat removal (DHR) scenarios can result in ECCS pumps experiencing vortex or inadequate NPSH problems. This problem is dependent upon pump design and the pump-suction elevation difference. Containment venting can also result in a loss of adequate NPSH. These problems were identified in the IPEs for BWR 1/2/3s and 3/4s and may have been an important contributor at some BWR 3/4s. Plant improvements identified in the IPEs include procedural guidance to replenish the condensate storage tank (CST), delay switchover from the CST to the suppression pool (applicable for RCIC, HPCI, and HPCS), and align LPCI pump to the CST (LPCI normally only takes suction from the suppression pool).

Additional ways to address this issue could include (a) pump changes or suction elevation changes, (b) changes to suction source switchover logic (e.g., change high suppression pool switchover logic), (c) increase the CST ECCS dedicated capacity, or (d) add LPCI and/or LPCS connections to the CST.

Another issue concerns tripping steam-driven pumps on high turbine exhaust pressure. RCIC in particular (HPCI has a relatively high trip set point) can trip off due to a high turbine exhaust pressure trip following a loss of containment heat removal or due to vessel depressurization to maintain heat-capacity temperature limits. One method to alleviate this concern could be to increase the turbine exhaust back pressure trip but such a change would have to be reviewed for negative impacts on the system operation. Since RCIC is not an ECCS system, this issue is not being pursued further as a potential risk-significant concern.

High pressures following a loss of containment heat removal in Mark I and II containments can result in closure of the ADS valves before containment failure occurs resulting in vessel repressurization and the inability to inject water using low-pressure ECCS. This would not occur in plants with Mark III containments since they would likely fail before containment pressure closed the ADS valves. One solution to this problem is to use SRVs that are not susceptible to closure due to lack of the proper differential pressure.

As mentioned previously, the average IPE transient CDF and LERF values are 10% or less of the Option 3 framework quantitative guidelines. Since, the contributions from any of the above issues are likely only a fraction of the above values, any changes to the ECCS to address these issues or any other concern pertinent to transients would be difficult to justify. However, PRA evaluations of transient scenarios used in complying with risk-informed alternatives to any regulation should address these phenomena.

SBO

Failure of RCIC (not an ECCS system) and HPCI can occur during a station blackout before battery depletion at some plants due to several phenomenon. One of these phenomenon is related to pump seal failure resulting from taking suction off of the suppression pool when there is no decay heat removal. Delaying the pump suction switching from the CST to the suppression pool can delay this problem. This can be accomplished by making the switchover manual rather than automatic or by changing the automatic switchover set points. Another failure mode of these systems occurs due to steam-leak detection logic. Loss of heating, ventilating, and air-conditioning (HVAC) systems during an SBO can result in room temperature rises sufficient to trip the steam leak detection logic. Common methods to address this issue include bypassing the trip logic and opening doors where the steam leak detectors are located (not possible in the main steam tunnel). Other potential improvements include increasing the steam leak detection trip set point or changing the isolation valve to an AC motor-operated valve (MOV) so that it can’t close during an SBO.

The SBO contributions to CDF reported in the IPEs range from negligible to 3.4×10^{-5}/yr with an average of 8×10^{-6}/yr. CCDP and LERF values are not readily available. The mean CDF value is 10% or less of the Option 3 framework quantitative guideline. Since the contribution from the above issues is only a fraction of the CDF value, any changes to the ECCS to address concerns pertinent to SBOs would be difficult to justify. However, PRA evaluations of SBO scenarios used in complying with risk-informed alternatives to regulations should address these interactions.

ATWS

Two issues were identified related to the performance of the ECCS during ATWS scenarios. The failure to inhibit ADS and control low-pressure ECCS coolant injection flow can result in flushing boron from the core, large power oscillations, or core instabilities. The failure to perform these actions was assumed to lead to core damage in some IPEs and was a major contributor to the ATWS-related risk (ATWS was not a significant contributor to the BWR CDFs). Plant improvements were not identified for this action since it was not a major contributor to CDF. However, an automatic ADS inhibit based on ATWS symptoms and the use of flow control valves instead of full-open injection valves could be beneficial to safety.

The second issue concerns ECCS injection inside of the shroud. Spraying or injecting cold water directly into the core during an ATWS can result in positive reactivity insertions. It is desirable to mix the cold ECC water with hotter water in the downcomer before it is injected into the vessel. Some plants that have LPCI systems that inject into the shroud have procedural guidance for ATWS scenarios that directs the operators to align the LPCI injection through the shutdown cooling paths into the recirculation loops. Requiring all plants with this type of LPCI to follow this procedure could be considered. Note that the BWR emergency procedure guidelines (EPGs) also direct the operators to use HPCS and LPCS as the last resorts because of this concern. Most IPEs assumed LPCS would be successful during an ATWS but not all BWR 5/6s assumed HPCS could be used. One way to address this concern for these systems would be to arrange for alternate injection paths to the recirculation loops.

ATWS is not an important contributor to BWR CDF. The CDFs reported in the IPEs range from negligible to 4.7×10^{-6}/yr and the mean is 9.8×10^{-7}/yr. This is approximately 1% of the Option 3 framework CDF guideline. Thus, any changes to the ECCS to address ATWS concerns would be difficult to justify. However, PRA evaluations of ATWS scenarios used in complying with risk-informed alternatives to any regulation should address these phenomena.

4.4.2.2 PWR ECCS Issues

LOCAs

The contribution from LOCAs reported in the IPEs range from 1.8×10^{-6}/yr to 7.5×10^{-5}/yr with an average of 1.8×10^{-5}/yr. The CCDPs for LOCAs range from 2.3×10^{-5} to 0.17 (the average values range from 2.6×10^{-3} for small LOCAs to 1.6×10^{-2} for large LOCAs). An estimate of the average LERF value for large LOCAs is 3×10^{-8}/yr and the average conditional probability of an early release resulting from a large LOCA is 1×10^{-2}. Based on the CDF and CCDP values, one of the concerns listed below (ECCS pump suction plugging) could contribute 10% or more of the Option 3 framework guidelines for CCDP but technical resolution of the issue would likely reduce that contribution.

As mentioned previously, ECCS pump suction plugging due to debris collection in the containment sump was evaluated in Unresolved Safety Issue (USI) A-43. The assessment of debris accumulation on PWR sump performance is the subject of a generic safety issue (GI-191) that is currently being evaluated. As with the BWR issue discussed previously, the specific concern involves degradation of piping insulation in the containment from exposure to LOCA conditions resulting in particles with near-neutral buoyancy that can quickly plug the ECCS pump strainers resulting in loss of adequate NPSH for the pumps. A technical resolution to this issue is scheduled for 2001. Potential plant modifications to deal with this issue include the installation of backwash systems or the replacement of existing strainers with larger or self-cleaning strainers. Thus, there is no need to address this issue in a risk-informed alternative to 50.46 at this time.

Most licensees did not model ECCS pump suction plugging due to debris resulting from a LOCA or from other sources (e.g., corrosion products) in their IPEs. ECCS pump suction plugging should be included in PRA evaluations used in complying with risk-informed alternatives to 50.46 (and any other regulation) and the assigned plugging probabilities should reflect the method selected to resolve this issue.

Manual switchover of the ECCS pumps from the injection mode to the recirculation mode is required at many plants. For some plants, automatic switchover occurs for low-pressure recirculation but require manual actions to switchover to high-pressure recirculation. Failure to perform this action was an important contributor to the LOCA-related CDF at many plants. There was also a large variability in the human error probabilities for this event (several orders of magnitude). The human error probabilities in the IPEs for large LOCAs were generally higher than for small LOCAs since less time was available for performing the switchover. Important factors in determining the timing include the refueling water storage tank (RWST) size and the initiation set point of the containment spray which diverts water in the RWST from ECCS usage (low set points, 10 psig versus 30 psig for example, deplete the RWST inventory faster). The use of piggybacked systems for high pressure recirculation also tends to increase the complexity of the required human actions (high-pressure ECCS at most plants can not take suction from the containment sump). Common plant improvements identified in the IPEs to address this issue were focused on increased operator training and improved procedural guidance. Alternate methods for addressing this concern could include changing to automatic switchover, increasing the RWST size, improving the ability to refill the RWST, and/or increasing the containment spray actuation pressure.

The potential for backfitting PWRs with manual ECCS switchover to automatic switchover was addressed as GI-24. A cost-benefit evaluation of the issue, published in NUREG/CR-6432 [Ref. 19], led to a “no action” resolution of the issue. The evaluation estimated that a changeover from manual to semiautomatic ECCS switchover (automatic switchover of the LPSI pumps and manual switchover of the LPSI pumps) would reduce the CDF from all LOCAs by a mean value of 1.7×10^{-5}/yr (the 95th percentile is 7×10^{-5}/yr). For large LOCAs, the CDF would be reduced by 2.3×10^{-6}/yr. These values were generated using the NUREG-1150 LOCA frequencies. Estimates of the contribution of manual switchover failure during all LOCAs from several PRAs are also listed in NUREG/CR-6432 and range from 9×10^{-6} to 4.5×10^{-5}/yr. These values would decrease by an order of magnitude if the NUREG/CR-5750 LOCA frequencies were used and would be substantially below the CDF guideline in the Option 3 framework. Based on this fact and the fact that changing to a semiautomatic switchover can not be justified on a cost/benefit basis, action to consider this issue further in risk-informed alternatives to 50.46 would be difficult to justify. However, PRA evaluations of scenarios used in complying with risk-informed alternatives to any regulation should address these interactions.

SBO

RCS heat removal in PWRs during SBO events is provided by steam-driven auxiliary feedwater pumps. There is no ECCS capability during SBOs in PWRs.

Transients

The total transient contribution for PWRs reported in the IPEs range from 5.3×10^{-7}/yr to 3×10^{-4}/yr and the average value is 4.3×10^{-5}/yr. CCDP values are not currently available but would be expected to have the same range in values since the frequency of transient initiators is approximately 1/yr. An estimated average LERF value for PWRs from all internal initiators is 5×10^{-6}/yr and the average conditional probability of a large early release is 6×10^{-2}. Based on the CDF and CCDP values, some of the concerns listed below could contribute 10% or more of the Option 3 framework quantitative guidelines and need to be reviewed further for potential impacts on risk-informed alternatives to regulations.

Transient sequences involving failure of AFW require the operators to utilize a feed and bleed operation using HPSI and the PORVs. Some PWRs (e.g., most Babcock and Wilcox [B&W] plants) use the charging pumps as the HPSI system and thus can inject water at PORV set point pressures. Thus, opening the PORVs is not required for feed and bleed at these plants. However, other PWRs have separate HPSI pumps which can not inject at these pressures and require opening of the PORVs for feed and bleed. Also, the high-pressure pumps at some plants can take suction directly off the sump avoiding the need for low-pressure system operation during feed and bleed in the recirculation mode. The high-pressure pumps at other PWRs can’t take suction off of the sump and must be piggybacked off of the low-pressure pumps which are connected to the sump.

A limited review of the IPEs indicate that none of the factors listed above that can influence the potential for feed and bleed were significant contributors to CDF. Thus it does not appear that any action to address this issue in risk-informed alternative regulations is warranted at this time. However, further examination of IPE results should be performed to verify this conclusion.

The systems used to provide cooling to the ECCS pumps are the same systems that provide RCP seal cooling and cool the charging pumps which provide RCP seal injection. Thus, failure of a single cooling water system can result in an RCP seal LOCA and failure of the ECCS required to mitigate it. RCP seal LOCAs initiated by loss of these cooling water systems is a significant contributor to the CDF primarily at Westinghouse PWRs (CDFs range from $<5\times10^{-8}$/yr to 3.5×10^{-4}/yr). Many licensees have identified plant improvements to reduce the potential for RCP seal LOCAs. These include the use of new high-temperature O-rings, alternate seal cooling systems, alternate cooling systems for the charging pump, and better procedural guidance to trip the pumps on loss of cooling. Alternate methods for addressing this concern could include use of air-cooled charging pumps and depressurizing the reactor coolant system to reduce RCP seal leakage.

RCP seal LOCAs are also the subject of a generic safety issue (GI-23). The NRC closed GI-23 partially on the basis that industry was pursuing voluntary initiatives to implement corrective measures related to RCP seal failure. These measures include the use of improved O-ring polymer material in Westinghouse pump seals. The NRC will, however, continue to pursue plant-specific risk analysis of the loss of component cooling water (CCW)/service water (SW) systems to assess this contributor to RCP seal failure risk. In a similar vein, the issue of cooling water failure impacts on plant systems should be addressed in risk-informed alternatives for regulations dealing with such systems rather than in a risk-informed alternative of 50.46.

4.5 References

1. USNRC, “Reactor Safety Study,” WASH-1400, 1975.

2. USNRC, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants - Final Summary Report,” NUREG-1150, Vol. 1, 1990.

3. J. P. Poloski, D. G. Marksberry, C. L. Atwood, and W. J. Galyean, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995," NUREG/CR-5750, Idaho National Engineering and Environmental Laboratory, February 1999.

4. Beliczey, S. and H. Schulz, “Comments on Probabilities of Leaks and Breaks of Safety-Related Piping of PWR Plants,” *International Journal of Pressure Vessels & Piping*, Vol. 43, pp. 219-227, 1990.

5. USNRC, “Report of the U.S. Nuclear Regulatory Commission Piping Review Committee,” NUREG-1061, Vol. 3, November 1984.

6. G.S. Holman and C.K. Chou, “Probability of Failure in BWR Reactor Coolant Piping,” NUREG/CR-4792, Vol. 1, Lawrence Livermore National Laboratory, March 1989.

7. EPRI, “Pipe Failures in U.S. Commercial Nuclear Power Plants,” EPRI TR-100380, Interim Report, July 1992.

8. G.S. Holman and C.K. Chou, “Proability of Pipe Failure in the Reactor Coolant Loop of Westinghouse PWR Plants Volume 1: Summary Report,” NUREG/CR-3660, Vol. 1, Lawrence Livermore National Laboratory, July 1985.

9. G.S. Holman, T. Lo, and C.K. Chou, “Probability of Pipe Failure in the Reactor Coolant Loops of Combustion Engineering PWR Plants Vol.1: Summary Report,” NUREG/CR-3663, Vol. 1, Lawrence Livermore National Laboratory, January 1985.

10. G.S. Holman and C.K. Chou, "Probability of Pipe Failure in the Reactor Coolant Loops of Babcock and Wilcox PWR Plants Vol.1: Summary Report," NUREG/CR-4290, Vol. 1, Lawrence Livermore National Laboratory, May 1986.

11. G. Martinez-Guridi, P.K. Samanta, T.L. Chou, and J.W. Chang, "Evaluation of LOCA With Delayed Loop and Loop With Delayed LOCA Accident Scenarios," NUREG/CR-6538, Brookhaven National Laboratory, July 1997.

12. D.W. Whitehead, et al., "Evaluation of Potential Severe Accidents During Low Power and Shutdown Accident Operations at Grand Gulf, Unit 1, Analysis of Core Damage Frequency From Internal Events for Plant Operational State 5 During a Refueling Outage," NUREG/CR-6143, Vol.1, Sandia National Laboratories, 1994.

13. B.D. Staple, T.D Brown, J.J. Gregory, J.L. LaChance, "Risk Comparisons of Scheduling Preventive Maintenance for Boiling Water Reactors During Shutdown and Power Operations," NUREG/CR-5593, Sandia National Laboratories, July 1999.

14. T.L. Chu, et al., "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operation at Surry, Unit 1," NUREG/CR-6144, Brookhaven National Laboratory, June 1994.

15. J.W. Yang, et al., "Risk Comparison of Scheduling Preventive Maintenance During Shutdown vs. During Power Operation for PWRs," NUREG/CR-6616, Brookhaven National Laboratory, December 1998.

16. J.A. Lambright, et al., "Analysis of Core Damage Frequency: Peach Bottom, Unit 2 External Events," NUREG/CR-4550, Vol. 4, Rev. 1, Part 3, Sandia National Laboratories, December 1990.

17. M.P. Bohn, et al., "Analysis of Core Damage Frequency: Surry Power Station, Unit 1 External Events," NUREG/CR-4550, Vol. 3, Rev. 1, Part 3, Sandia National Laboratories, December 1990.

18. NRC, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants." October 2000.

19. B. Walsh, et al., "Estimated Net Value and Uncertainty for Automating ECCS Switchover at PWRs." NUREG/CR-6432, Science and Engineering Associates, Inc., February 1996.

5. POTENTIAL RISK-INFORMED OPTIONS

5.1 Introduction

As the discussion in Chapter 4 indicates, core-cooling related research and risk studies suggest that any risk-informed alternative to the current 10 CFR 50.46 and related regulations should account for the following:

- Current estimates of the frequency of large-break LOCAs (LBLOCAs) (i.e., LOCAs large enough to require low pressure injection) are uncertain, and not low enough to permit all LBLOCAs to be excluded from the design basis.

- Because large-breaks in the reactor coolant system pressure boundary are very unlikely, the reliability of core cooling and containment functions is generally sufficient to assure that LBLOCAs are not significant contributors to risk.

- Plant equipment that is designed, at least in part, to the requirements of design-basis LOCAs also provide defenses against a spectrum of beyond-design-basis accidents.

- Current evaluation models of ECCS performance may be overly conservative for LBLOCAs.

Section 5.2 presents eight options for risk-informing 10 CFR 50.46 and related requirements and practices. The regulatory requirements and practices associated with 10 CFR 50.46, Appendix K and GDC 35 can be divided into four categories: (1) spectrum of breaks to be considered, (2) ECCS functional reliability, (3) ECCS evaluation models, and (4) ECCS acceptance criteria. Each of the options discussed in this chapter strives to risk-inform a specific type of regulatory requirement or practice associated with one of the above categories. Option 1 deals with postulated breaks and postulated break characteristics. Options 2 and 3 deal with other failure events postulated in design-basis LOCAs, which are used to establish the reliability of the ECCS function. Option 4 deals with conservative requirements for ECCS evaluation models set forth in Appendix K, and Options 5, 6, and 7 deal with the uncertainty analyses required when realistic ECCS evaluation models are used. Option 8 deals with the ECCS acceptance criteria. Table 5-1 summarizes each option.

Section 5.2 includes a subsection for each of the four categories of regulatory requirements and practices associated with 10 CFR 50.46, Appendix K and GDC 35. Each of these subsections, in turn, contains a separate subsection for each associated option. Table 5-2 summarizes by subheading the topics discussed for each option. These include existing regulatory requirements or practices, risk perspectives, safety considerations, potential for unnecessary burden reduction, implementation steps, time and resource requirements, and other considerations relevant to the option.

Section 5.3 compares the eight options. Based on the comparison, a risk-informed alternative is identified in Section 5.4, and discussed in more detail in Attachment 2.

As discussed in Section 1.2, this work is intended to demonstrate the feasibility of risk-informed changes to 10 CFR 50.46 and related requirements and practices. If the Commission approves going to rulemaking, additional analyses will be required.

Table 5-1 Summary of Potential Risk-Informed Options

	Option	Description
Spectrum of Breaks:		
1	LBLOCA Redefinition	Permit each plant to define a maximum design-basis LOCA size based on LBB and probabilistic fracture mechanics (PFM) analyses performed in accordance with NRC-approved methods and assumptions.
ECCS Functional Reliability:		
2	Modify Design-Basis LOCA-LOOP Assumptions	Drop the requirement that LOOP be postulated in larger, more unlikely design-basis LOCAs.
3	Use Reliability Analyses in Lieu of Single Failure Criterion for ECCS	Permit the use of ECCS reliability and LOCA frequency information to establish ECCS reliability requirements in lieu of the single failure criterion.
ECCS Evaluation Models:		
4	Relax Excessive Appendix K Conservatisms	Revise Appendix K to 10 CFR Part 50 to permit excessively conservative features (e.g., decay heat and cladding oxidation models) to be replaced by more realistic ones.
5	Make Best-Estimate ECCS Performance Analyses Less Burdensome	Apply advanced methods to accelerate uncertainty analyses (and potentially model reviews) for best-estimate evaluations of ECCS performance.
6	Propagate Uncertainty in LBLOCA Size & Location	Permit uncertainties in large-break size and location to be addressed along with and in the same manner as uncertainties in other inputs to best-estimate ECCS evaluation models.
7	Enable Best-Estimate Analyses with Approved Uncertainty Increments	Enable licensees to evaluate ECCS performance using best-estimate code predictions with NRC-approved allowances added to account for uncertainties.
ECCS Acceptance Criteria:		
8	Modify ECCS Acceptance Criteria	Replace the current prescriptive ECCS acceptance criteria in 10 CFR 50.46 with a performance-based requirement to demonstrate adequate post-quench cladding ductility and adequate core-coolant flow area to ensure that the core remains amenable to cooling, and for the duration of the accident, maintain the calculated core temperature at an acceptably low value and remove decay heat. Permit demonstration of adequate post-quench ductility through testing as a performance-based alternative to the current acceptance criteria for peak cladding temperature and maximum oxidation.

Table 5-2 Topics Covered for Each Option

Subheading	Description
Requirement	Current regulatory requirements or practices that would be risk-informed by the option
Risk Significance	Risk significance of the current regulatory requirements or practices or the option proposed to risk-inform them. Consistent with the Option 3 framework, the focus is on CDF and LERF; however, decisions on risk-informed alternatives are not based solely on these considerations.

Table 5-2 Topics Covered for Each Option

Subheading	Description
Description	The proposed changes that constitute the option
Safety Perspectives	Potential safety benefits of the option Safety concerns related to the option and approaches that could be taken to address these concerns
Potential for Unnecessary Burden Reduction	Changes to plant equipment, technical specifications, procedures, training, licensee analyses, regulatory practices, etc. that could result from the option and reduce unnecessary burden
Implementation	Activities that would be undertaken to prepare for rulemaking and to implement the option
Needed Technical Analyses	Technical analyses that would be required to demonstrate the feasibility of the option Other technical analyses that would be required to prepare for rulemaking or for licensee applications
Time and Resources Required	Time and resources required to develop and demonstrate the option Time and resources required for licensee applications and NRC reviews
Other Considerations	Other considerations including the types of plants that could potentially apply the option, variations of the option, and potential synergism with other options

5.2 Options for Risk Informing Current Requirements

5.2.1 Spectrum of Breaks

5.2.1.1 <u>OPTION 1: LBLOCA Redefinition</u>

Requirement

Appendix A to 10 CFR Part 50 defines 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*." Paragraph (a)(1)(i) of 10 CFR 50.46 requires that ECCS cooling performance "*be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated*." The spectrum of breaks to be considered is made more specific in 10 CFR 50 Appendix K (C)(1) which says "*In analyses of hypothetical loss-of-coolant accidents, a spectrum of possible pipe breaks shall be considered. This spectrum shall include instantaneous double-ended breaks ranging in cross-sectional area up to and including that of the largest pipe in the reactor coolant system. The analysis shall also include the effects of longitudinal splits in the largest pipes, with the split area equal to the cross-sectional area of the pipe*."

Risk Significance

As indicated in Chapter 4, especially for PWRs, the frequency of occurrence of the larger pipe ruptures currently postulated as design-basis initiators is estimated to be very low. On the other hand, plant characteristics (e.g., ECCS capacity, containment design pressure, equipment qualification envelopes, etc.) established, in part, to cope with large design-basis LOCAs provide defenses for other accidents as well.

Description

LBLOCA Redefinition would permit each plant to define a maximum design-basis LOCA size based on LBB, PFM, and risk analyses performed in accordance with NRC-approved methods and assumptions. The regulatory changes required to enable this option would be relatively straightforward. For example, the definition of design-basis LOCAs in 10 CFR 50.46, and Appendices A and K to 10 CFR Part 50, could be modified to allow licensees to specify an alternate maximum pipe size. At the same time, references to "*any loss-of-coolant accident*" in the General Design Criteria could be changed to "*any design-basis loss-of-coolant accident*." Additional regulatory changes might be needed to preserve key elements of defense-in-depth (e.g., containment design pressure) as discussed below under safety concerns.

Safety Perspectives:

Potential Safety Benefits: The potential for safety benefits from LBLOCA redefinition is high relative to Options 2 through 8. Because a simultaneous LOOP must be postulated, a LBLOCA is generally believed to be the design basis accident (DBA) that establishes the diesel generator (DG) start time (typically at 10 seconds). While this may not actually be the case (an analysis performed for a typical Westinghouse four-loop plant using a best-estimate code with the required features of Appendix K found that current ECCS acceptance criteria could be met with a 30-second DG start time not only for LOCAs but also for other design basis accidents [Ref. 1]), relaxing the simultaneous LOOP requirement for LBLOCAs may make it easier for licensees to extend the DG start time. Rapid start tests could degrade bearings, gears, the governor, and power transmission such that the reliability of emergency diesel generators might be diminished. Speed overshoots during fast starts could increase the potential for an overspeed trip. If the required DG starting time could be lengthened, challenges to emergency AC power system reliability and availability associated with emergency operation and tests of rapid DG starts and load sequencing could then be avoided. Similar reliability benefits could be obtained for fast-operating valves whose operating times are derived from LBLOCA analyses. By eliminating larger pipe breaks as design basis initiators, flow orifices provided to prevent high-pressure injection pump runout during LBLOCAs could be sized to permit greater flow for more-likely, smaller LOCAs. Both licensee and NRC resources could be focused on smaller, more likely LOCAs. Finally, the use of LBB analyses to justify the exclusion of certain pipe breaks as design-basis events for global as well as dynamic effects would result in improved regulatory consistency.

Safety Concerns: Design-basis RCS pipe breaks may serve as a surrogate for other non-pipe break LOCA initiators such as steam-generator manway failure. Such initiators would have to be identified and demonstrated to pose low risks. Estimates of large-break frequencies would have to adequately account for such non-pipe-break initiators, for all potential failure mechanisms, and for uncertainties in models and model parameters. Safety concerns regarding plant changes would have to be addressed via predetermined change control processes, where necessary, using state-of-the-art PRAs performed in accordance with NRC-approved standards. The impact of plant changes proposed based on LBLOCA redefinition should be demonstrated to have acceptable impacts on risks posed by other accidents. For example, longer diesel generator start time should not lead to steam-generator dryout in the event of a simple loss of offsite power. Plant changes that would compromise the key elements of defense-in-depth and, thereby, the ability of plants to cope with severe accidents would not be permitted. For example, the containment design pressure for some plants is derived from LBLOCA analyses. A reduction in containment capability would, generally, not be risk-informed.

Potential for Unnecessary Burden Reduction

The potential for unnecessary burden reduction is high relative to other options. Table 5-3 summarizes industry estimates of a number of potential cost savings based primarily on input from the Westinghouse Owner's Group (WOG). All of the listed cost savings would not be realized at every plant, and the cost savings actually realized would be plant specific. For example, the balance of plant systems for some plants would not permit power uprates. The extent of additional analytic margin obtained in LBLOCA analyses of ECCS and containment heat removal systems would be highly plant specific, and power uprates could consume much of the increase in analytic margin that would otherwise be available from LBLOCA redefinition. It should also be noted that some of the potential benefits could only be realized for a very small maximum design-basis break size. For example, without changing the current assumption of a 1 millisecond break opening time, elimination of the need for baffle barrel bolt replacement could, for some plants, require a maximum break size less than that currently approved based on LBB analyses for eliminating dynamic effects as design-basis events.

Table 5-3 Potential Cost Savings Resulting from LBLOCA Redefinition

Item	Potential Cost Savings
Relaxation of technical specification requirements related to 10-second diesel-generator start times.	WOG estimates savings of $400,000 to $1,200,000 per plant year depending on whether DG teardowns are on the critical path for refueling outages.
Increases in peaking factors for many plants.	WOG estimates $100,000 to $300,000 per plant year.
Power uprates from 1% to 3% for plants whose power conversion systems permit such upgrades	WOG estimates $1,700,000 to $2,800,000 per plant year.
Relaxation of technical specification requirements related to accumulators	WOG estimates $17,000 per plant year.
Reductions in LBLOCA analysis costs	WOG estimates $50,000 to $300,000 per plant year depending on evaluation model (Appendix K or best estimate) and plant activities requiring LOCA evaluation..
Reductions in licensee response costs associated with the potential elimination or simplification of generic issues and letters related to 50.46 and design-basis LOCAs.	WOG estimates $75,000 per plant year.
Reduction in costs of 50.46 reporting requirements.	Commonwealth Edison estimates $20,000 per plant year

Table 5-3 Potential Cost Savings Resulting from LBLOCA Redefinition

Item	Potential Cost Savings
Avoidance of one-time cost associated with baffle barrel bolt replacement for some plants if a maximum design-basis break size is justified to be less than that currently approved based on LBB analyses and used to exclude dynamic effects including asymmetric loads on reactor vessel internals.	WOG estimates $3,600,000 to $8,300,000.

Implementation:

LBB and PFM methods and assumptions suitable for LBLOCA redefinition would have to be developed. Either regulatory guidance or NRC approval of industry topical reports would be required to define LBB and PFM methods and assumptions acceptable to NRC. It is anticipated that Owner's groups (for larger break categories) and licensees (for smaller, plant-specific break categories) would develop justifications for maximum LOCA sizes using NRC-approved methods and assumptions.

Significant plant changes that could potentially be proposed based on LBLOCA redefinition would have to be identified and either allowed, prohibited by regulation, or subjected to an appropriate pre-determined approval process. It may be possible to justify some changes generically or for specific plant types. Other changes would require licensees to develop and submit plant-specific change requests following appropriate change control processes. Change controls similar to those set forth in Regulatory Guide 1.174 [Ref. 2] could require state-of-the-art risk assessments performed in accordance with NRC approved standards. Additions or revisions to current regulatory guides, standard review plans, and other NRC implementing documents would be necessary to reflect LBLOCA redefinition. Specific regulatory guidance could be developed for widely anticipated changes such as lengthening diesel-generator start times.

Needed Technical Analyses:

LBLOCA Redefinition would involve two stages of analysis. The first stage would justify the exclusion of breaks larger than a selected size as design basis initiators by demonstrating that accidents initiated by the excluded breaks pose sufficiently low risks. The second stage would justify that plant changes proposed based on LBLOCA redefinition would have acceptable risk implications for other accidents.

Consider the analyses to justify the exclusion of larger breaks as design-basis initiators. The upper bounds of the uncertainty ranges for recent NRC estimates of LBLOCA frequencies (10^{-5} per critical year for PWRs and 10^{-4} per critical year for BWRs per NUREG/CR-5750 [Ref. 3]) do not permit all LBLOCAs to be excluded based on initiator frequency alone. Rupture frequency decreases with pipe size, so it might be possible to exclude very large-breaks (e.g., double-ended ruptures of PWR hot legs and cold legs) as design-basis initiators based on frequency estimates. However, to justify the exclusion of smaller breaks would require a demonstration that LOCAs initiated by these breaks are not significant contributors to CDF, LERF, or large late release frequency (LLRF). Specifically the contributions of excluded LOCAs to CDF, LERF, and LLRF should not be substantial fractions (e.g., 0.1 or greater) of the corresponding quantitative guidelines from the Option 3 framework (10^{-4} per calendar year for CDF and 10^{-5} per calendar year for LERF and LLRF) [Ref. 4]. The feasibility of Option 1 is low relative to other options because substantial advances in LBB/PFM methods would be required in order to confidently demonstrate sufficiently low contributions of excluded LOCAs to CDF, LERF, and LLRF. The characteristics of the LBB and PFM analyses that would be required are discussed in Appendix A of Attachment 2.

In addition to the analyses used to redefine the maximum design-basis break size, analyses might also be required to demonstrate acceptable risk impacts associated with plant changes proposed based on LBLOCA redefinition. State-of-the-art PRAs conforming to NRC approved standards could be needed in order to justify significant changes to ECCS, emergency power, and other safety significant systems, structures, or components.

Time and Resources Required

Time and resources required for Option 1 would be high relative to other options. This is because advanced LBB/PFM methods would have to be developed, and proposed plant-specific change proposals might need to be subjected to appropriate NRC reviews and approvals. Where necessary, change proposals would be based on state-of-the-art PRAs performed in accordance with NRC-approved standards. Industry owner's groups have budgeted several million dollars over 3 to 4 years for methods development and approval of LBLOCA redefinition. Significant NRC resources would also be required for confirmatory research and for methods review and approval. Plant-specific changes could also involve significant licensee and NRC costs although some changes could probably be approved generically or by plant type.

Other Considerations

In the past, LBB has not been applied to BWR piping because of intergranular stress corrosion cracking concerns. LBB and PFM methods are, therefore, less developed for BWRs than for PWRs. On the other hand, BWRs typically have more systems that can deliver water to the core in LOCAs than do PWRs. The extent to which LBB and PFM methods would have to be developed in order to apply LBLOCA redefinition to BWRs has not yet been determined.

NEI and the owner's groups for Babcock and Wilcox, Combustion Engineering, General Electric, and Westinghouse plants have all expressed support for the LBLOCA redefinition option in public meetings.

Performance monitoring (e.g., in-service inspection [ISI] and leak detection) aimed at LOCA prevention would be expected to continue under LBLOCA redefinition or any of the other risk-informed options.

Industry has indicated that some plant mitigative capability would be retained even for those size LOCAs ultimately excluded from the design basis. One possibility that has been identified is to maintain expectation of success under accident management.

5.2.2 ECCS Functional Reliability

5.2.2.1 <u>OPTION 2: Modify Design-Basis LOCA-LOOP Assumptions</u>

Requirement

GDC 17, *Electric Power Systems*, requires an onsite electric power system and an offsite electric power system be provided to assure "*the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents*." GDC 34, 35, 38, 41, and 44 stipulate that systems required to respond to or mitigate the effects of LBLOCAs be designed 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*." In design-basis LOCA analyses LOOP is postulated to occur immediately. Under this assumption, it is generally believed that rapid (typically 10 s) emergency diesel generator start times are required in order to meet ECCS acceptance criteria.

Risk Significance

The probability of LOOP following a pipe break LOCA was evaluated (based on data that predominately preceded electric power deregulation) to resolve Generic Safety Issue (GSI)-171 (NUREG/CR-6538 [Ref. 5]). As indicated in Table 4-3, the point estimate values were 0.06 for BWRs and 0.014 for PWRs. For PWRs, combining the upper bound LBLOCA frequency estimate of 10^{-5} per reactor calendar year from Table 4-1 (obtained from NUREG/CR-5750) with the point estimate probability of LOOP given a LOCA gives a LBLOCA-LOOP frequency of 1.4×10^{-7} per reactor calendar year. This is considerably less than the quantitative guideline of 10^{-6} per reactor calendar year set in the Option 3 framework document for excluding a class of accident initiating events from the design-basis. In contrast, even combining the mean small-break LOCA (SBLOCA) frequency estimate of 4×10^{-4} per reactor calendar year with the point estimate probability of LOOP given a LOCA gives a SBLOCA-LOOP frequency of 5.6×10^{-6} per reactor calendar year, which exceeds the Option 3 framework guideline of 10^{-6} per reactor calendar year.

Plant-specific design features and grid instabilities should be considered in estimating the probability of LOOP given a LOCA. Nevertheless, it appears that some plants would be able to demonstrate LOCA-LOOP frequencies demonstrably below 10^{-6} per reactor calendar year, at least for less likely larger break sizes.

Recent studies indicate that LOOP following a LOCA initiator is more likely to be delayed (from a few seconds to about one minute) than immediate. Existing PRAs and IPEs do not model vulnerabilities associated with delayed LOOP following LOCAs (e.g., double sequencing of safety equipment). Determining the risk implications of delayed LOOP, particularly in combination with plant operation during periods of degraded grid voltage, may require further study with or without the risk-informed option described below.

Description

Option 2 would permit combinations of LBLOCA initiators and simultaneous LOOP events that are highly unlikely to be excluded as design basis accidents. To enable Option 2, wording changes would be required to 10 CFR 50.46 and/or GDC 35, as well as to GDCs 34, 38, 41, and 44 if this option were to be extended beyond just the ECCS.

Safety Perspectives:

Potential Safety Benefits

Because a simultaneous LOOP must be postulated, a LBLOCA is generally believed to be the DBA that establishes the DG start time (typically at 10 seconds). While this may not actually be the case (see discussion of Potential Safety Benefits in Section 5.2.1.1), relaxing the simultaneous LOOP requirement for LBLOCAs may make it easier for licensees to extend the DG start time. Rapid start tests could degrade bearings, gears, the governor, and power transmission such that the reliability of emergency diesel generators might be diminished. Speed overshoots during fast starts could increase the potential for an overspeed trip. If the required DG starting time could be lengthened, challenges to emergency AC power system reliability and availability associated with emergency operation and tests of rapid DG starts and load sequencing could then be avoided. The design-basis LBLOCA also requires rapid operation times for certain engineered safety feature (ESF) valves. Lengthening ESF valve operation times could have a positive impact on the reliability of the affected components.

For design-basis LOCA-LOOP sequences under existing regulations or Option 2, if emergency AC power systems were designed for delayed versus instantaneous LOOP following LBLOCAs, additional options for electrical equipment protection could be provided (e.g., to avoid degraded voltage and double sequencing of safety equipment).

Safety Concerns:

The estimates of large-break frequencies that would be necessary for Option 2 should account for non-pipe breaks and all potential failure mechanisms.

The impact of proposed plant changes (e.g., DG start times or load sequencing times) might have to be analyzed and approved based on licensee-originated plant-specific analyses performed using methods and assumptions acceptable to NRC.

Evidence of electrical grid stability may be a necessary part of the justification for eliminating larger LOCA-LOOP sequences as DBAs. Monitoring may be required to preclude unacceptable long term deterioration of grid stability. As envisioned, no new technical specification requirement on operability of offsite power would be required.

With proper treatment of the concerns discussed above, Option 2 would not impact the key elements of defense-in-depth listed in the Option 3 framework [Ref. 4]. The diversity of electric power systems would not be compromised.

Potential for Unnecessary Burden Reduction

The relaxation of technical specification requirements related to DG start times would result in cost savings (WOG estimates $400,000 to $1,200,000 per plant year depending on whether or not diesel generator teardown is on the critical path for refueling outages). Lengthening the diesel generator start time could delay actuation of containment heat removal systems at some plants. This could reduce analytic margin in design-basis containment analyses. The bulk of the potential cost reductions identified in Table 5-3 would not be achieved by Option 2.

Implementation:

To permit LOOP to be excluded as a design-basis event for LOCA initiators of sufficiently low frequency would require wording changes to 10 CFR 50.46 and/or GDC 35, as well as to GDCs 34, 38, 41, and 44 if this option were to be extended beyond just the ECCS.

Analyses that would be required to justify the elimination of the LOOP assumption for larger LOCAs are described under Needed Technical Analyses below.

Regulatory guidance describing acceptable justification, methods, and analyses for implementing Option 2 would have to be developed. A demonstration plant could be useful for developing such guidance.

Changes reflecting Option 2 in affected SRPs would ultimately be required.

Needed Technical Analyses:

Analyses of LBLOCA frequencies and conditional LOOP probabilities would be needed to justify the exclusion of LBLOCA-LOOP sequences with sufficiently large-break areas as DBAs. The rigor of break frequency estimation would be less than required for Option 1 in which breaks with sufficiently low frequencies would be completely eliminated as design-basis events. One important frequency value could be associated with the break size that represents the transition point to LBLOCAs often used in probabilistic risk assessments (e.g., >6 inches for PWRs). One possibility for obtaining the frequency of this transition point and the frequency of other LOCAs might involve a review of current LOCA frequencies used in PRAs, and consideration of pipe break frequency data and insights, to determine an appropriate frequency based on engineering judgement. Another possibility might involve an update of the NUREG/CR-5750 analysis of pipe-break LOCA frequencies, to account for more recent operating experience. For this possibility, a less subjective analysis of uncertainties in the frequency estimates for LOCAs may be required to assure that suitably conservative estimates are used in implementing those alternatives requiring analyses of LOCA initiator frequencies or LOCA CDF contributions.

The plant-specific features that tend to decrease the probability of a LOOP given a LOCA would need to be identified. Also, acceptable methods and assumptions would have to be identified for quantifying the probability (and potentially the timing) of LOOP given a LBLOCA. Such probability estimates might have to account for plant-specific electrical design differences, grid instabilities arising from electric-power deregulation, and plant operation during periods of degraded grid voltage.

Plant-specific analyses might be required to demonstrate that plant changes (e.g., lengthening the required diesel-generator start time) proposed based on the exclusion of larger LOCA-LOOP sequences as DBAs would have acceptable risk implications. Appropriate methods and assumptions for performing such analyses, if required, would have to be identified. For some plants, it might suffice to demonstrate compliance with acceptance criteria for other design basis accidents (e.g., for PWRs, DBAs requiring emergency power for auxiliary feedwater pumps). Alternatively, it might be necessary to quantify changes in risk measures associated with lengthening diesel-generator start time. If required, analyses of plant risks before and after proposed changes to DG start times or load sequencing times might have to account for LBLOCA vulnerabilities associated with delayed LOOP and plant operation during periods of degraded grid voltage. Detailed system studies (including thermal-hydraulic analyses) could be required to investigate vulnerabilities associated with delayed LOOP and degraded grid voltage during LBLOCAs and methods for coping with such vulnerabilities.

Time and Resources Required

Guidance regarding estimates of LOCA frequencies might have to be developed and should account for non-pipe breaks and all potential failure mechanisms.

Research and development efforts in support of Option 2 could be significantly impacted if it is determined that LBLOCA vulnerabilities to delayed LOOP have to be investigated.

With appropriate regulatory guidance, licensee implementation as well as NRC review costs should be reasonable; however, plant-specific applications and reviews might be required.

Other Considerations

Option 2 would be applicable to BWRs and PWRs.

Option 2 might reasonably be coupled with Option 3 (Relax single failure criterion for sufficiently large LOCAs)

As noted previously, an analysis performed for a typical Westinghouse four-loop plant using a best-estimate code with the required features of Appendix K found that current ECCS acceptance criteria could be met with a 30-second DG start time not only for LOCAs but also for other design basis accidents [Ref. 1]. The option to demonstrate the adequacy of a longer diesel-generator start time using an approved ECCS performance model exists under current regulations.

5.2.2.2 <u>OPTION 3: Use Reliability Analyses in Lieu of Single Failure Criterion for ECCS</u>

Requirement

GDC 35 stipulates that the ECCS be designed to assure that the system safety function can be accomplished assuming a single failure.

Risk Significance

Because large-breaks in the RCS pressure boundary are very unlikely, the reliability of the core cooling function is generally sufficient to assure that LBLOCAs are not dominant risk contributors. To illustrate, consider the upper bound LBLOCA frequency estimates from NUREG/CR-5750 [Ref. 3], which are 10^{-5} per reactor calendar year for PWRs and 9×10^{-5} per reactor calendar year for BWRs. These frequencies imply LBLOCA contributions to CDF of less than or equal to 10^{-6} per reactor calendar year for core cooling failure probabilities less than or equal to 0.1 for PWRs or 0.01 for BWRs. At these achievable levels of reliability, risks posed by LBLOCAs are small compared to the quantitative guidelines presented in the Option 3 framework (CDF # 10^{-4} per plant year and LERF # 10^{-5} per plant year).

Description

Option 3 would permit the use of ECCS reliability information in conjunction with LOCA frequency information to establish ECCS reliability requirements commensurate with the LOCA core damage frequency. This information could include plant-specific PFM calculations to establish LOCA frequencies and ECCS reliability evaluations that reflect unique plant features and plant-specific operational data. Under this option, the NRC could set an acceptable threshold value for the frequency of core damage associated with a specified set of LOCA initiators. The licensee would be required to meet these core damage thresholds values using plant-specific LOCA frequencies and ECCS reliability values. Alternatively, the NRC could establish system reliability requirements for different LOCA frequency intervals based on generic information for the industry as a whole, or by plant type, and once implemented, licensees would be free to choose this alternative without any NRC review or approval.

Any changes to the ECCS requirements based on reliability arguments would have to be based on the reliability of the ECCS for all accidents, not just LOCAs, unless there is an ECCS component that is only required for a LOCA (e.g., the accumulators).

In lieu of the single failure criterion, functional reliability (or failure) analyses could also be used to assure suitably low failure probabilities for containment heat removal and containment atmospheric cleanup in design-basis LOCAs (by meeting NRC-specified CDF or reliability thresholds).

Safety Perspectives:

Potential Safety Benefits

Option 3 could confirm, quantify, and potentially enhance the element of defense-in-depth related to safety function success probabilities. Appropriate redundancy, independence, diversity, defenses against common cause failure mechanisms, defenses against human errors, and safety margins would have to be maintained in order to meet the quantitative reliability objectives.

Safety Concerns:

As for Option 2, estimates of LOCA frequencies would be necessary for Option 3 and such estimates should account for non-pipe breaks and all potential failure mechanisms. Assuming this can be accomplished, Option 3 would not adversely impact the key elements of defense-in-depth listed in the Option 3 framework [Ref. 4]. The reliability analyses and reanalyses of non-LOCA DBAs should suffice to justify plant changes based on Option 3.

Potential for Unnecessary Burden Reduction

Option 3 should permit some relaxation of technical specification requirements (e.g., component outage limitations); but, the potential for unnecessary burden reduction for Option 3 is low relative to other options.

Implementation:

This option would require wording changes to 10 CFR 50.46 and/or GDC 35, as well as to GDCs 34, 38, 41, and 44 if this option were to be extended beyond just the ECCS.

Regulatory guidance describing acceptable methods and assumptions for implementing Option 3 would have to be developed. A demonstration application might be useful for this purpose.

Changes reflecting Option 3 in affected SRPs would ultimately be required.

Plant-specific reliability and DBA analyses might have to be performed by licenses (and reviewed by NRC) to justify plant changes proposed based on Option 3.

Needed Technical Analyses:

Analyses of LOCA frequencies and conditional probabilities of core damage given a LOCA would be needed to establish appropriate ECCS reliability requirements. As for Option 2, analysis of LOCA frequencies should consider non-pipe breaks and all potential failure mechanisms, but would not need to be as rigorous as for Option 1, in which breaks with sufficiently low frequencies would be completely eliminated as design-basis accident initiators. Technical analysis will be required to support development of regulatory guidance that would specify acceptable methods and assumptions for performing LOCA CDF and ECCS reliability analyses. In addition, appropriate reliability and CDF threshold values would have to be selected.

The feasibility of Option 3 is high relative to other options because established reliability (or failure) analysis methods would be utilized.

Time and Resources Required

Time and resources required for development and demonstration would be low relative to other options. Regulatory guidance regarding LOCA frequencies would be the same as for Option 2. Regulatory guidance regarding the use of established reliability analysis methods should be relatively straightforward to develop.

With appropriate regulatory guidance, licensee implementation as well as NRC review costs should be reasonable; however, plant-specific applications and reviews might be required.

Other Considerations

Option 3 could help to resolve passive single failure footnote in 10 CFR 50, Appendix A; however, regulatory analysis and resulting changes to regulations and implementing documents might be complicated by the fact that applications of the single failure criterion are not limited to just the ECCS or LBLOCAs.

Option 3 would be applicable to BWRs and PWRs

Option 3 could reasonably be coupled with Option 2 (Modify simultaneous LOOP assumption for LBLOCAs).

5.2.3 ECCS Evaluation Models

5.2.3.1 <u>OPTION 4: Relax Excessive Appendix K Conservatisms</u>

Requirement

10 CFR 50.46(a)(1)(i) requires ECCS performance to be calculated in accordance with an acceptable evaluation model. Best-estimate models can be used provided uncertainties in calculated results are quantified. Alternatively, models with the required features of Appendix K to 10 CFR Part 50 can be used. Analyses using evaluation models based on Appendix K are generally less costly but more conservative than analyses using realistic ECCS evaluation models with uncertainty quantification. At this time, less than half of U.S. plants have used or plan to use best-estimate ECCS evaluation models. The majority of plants use ECCS evaluation models based on Appendix K.

Risk Significance

The ECCS acceptance criteria permit some cladding failures but preserve the basic core geometry and hence core coolability. ECCS evaluations either use conservative models and assumptions as set forth in Appendix K or compare best-estimate peak cladding temperatures at the 95^{th} percentile to the 2200°F ECCS acceptance criteria. Option 4 could enable excessive conservatism to be removed from Appendix K models. Therefore, it is important that licensees and owners of Appendix K evaluation models remain vigilant in assuring that nonconservatisms do not become significant relative to the remaining conservatism.

Description

Option 4 would revise Appendix K to 10 CFR Part 50 to permit the use of more realistic alternatives to excessively conservative models and assumptions. As indicated above, such changes to Appendix K would not have significant risk implications; however, other options might reduce the need for changes to Appendix K (e.g., if LBLOCAs were eliminated as DBAs under Option 1). As a result, Option 4 is being considered in the larger context of the initiative to risk-inform 50.46 and related regulatory requirements.

Safety Perspectives:

Potential Safety Benefits

Plants utilizing ECCS evaluation models based on Appendix K would obtain additional analytic margin under Option 4. The increase in analytic margin would be plant specific as would the nature of changes proposed to take advantage of the increase. It is conceivable that some changes could lead to safety benefits (e.g., fuel loading changes that significantly reduced neutron fluxes to the reactor pressure vessel could reduce or delay threats of pressurized thermal shock); however, it is assumed that additional margin would be utilized to maximize economic benefits to the licensees. Safety benefits would not necessarily be realized.

Safety Concerns:

Option 4 could enable excessive conservatism to be removed from Appendix K models. Therefore, it is important that licensees and owners of Appendix K evaluation models remain vigilant in assuring that nonconservatisms do not become significant relative to the remaining conservatism. As long as known nonconservatisms do not significantly impact the revised ECCS evaluation model, Option 4 would not impact the key elements of defense-in-depth listed in the Option 3 framework. The ECCS acceptance criteria would still have to be met for all design-basis LOCAs based on demonstrably conservative Appendix K models.

Potential for Unnecessary Burden Reduction

The increase in analytic margin obtained under Option 4 would be plant-specific as would changes proposed to take advantage of the increase. The extent of unnecessary burden reduction would be similarly plant specific. Of the potential cost reduction items listed in Table 5-3, additional analytic margin achieved via Option 4 could conceivably be useful for lengthening diesel generator start times, increasing peaking factors, uprating power, or reducing LOCA analysis costs (fewer iterations required in design-basis LOCA analyses). The relative potential for unnecessary burden reduction is low to medium for Option 4 because the additional analytic margin is greater for options utilizing best-estimate ECCS evaluation models and because the other potential cost reductions listed in Table 5-3 could not be achieved based solely on Option 4.

Implementation:

The changes to Appendix K would be expected to be straightforward.

A process may be needed to address known nonconservatisms in ECCS evaluation models based on Appendix K, and to assess the remaining overall conservatism of such evaluation models when less conservative, alternative features are applied.

A revised evaluation model based on Appendix K would be applied in the same manner as its predecessor.

Needed Technical Analyses:

One of the significant conservatisms in Appendix K is the requirement to use the 1971 ANS-5 standard for decay heat with a multiplier of 1.2. When Appendix K was developed (1974), the 1.2

multiplier was thought to approximately represent a suitably conservative fit to available data. A 1979 decay heat standard [Ref. 6] utilized a more extensive analysis of an improved data base to reduce the uncertainty in decay heat.

Another well-known conservatism in Appendix K is the required use of the Baker-Just metal-water reaction model, especially at temperatures above ~2000°F. Regulatory Guide 1.157 [Ref. 7] accepts the less-conservative Cathcart-Pawel metal-water reaction model for best-estimate analyses.

A preliminary evaluation performed by the NRC Office of Research (RES) examined the effects of using more realistic decay heat and metal-water reaction models in Appendix K calculations for Westinghouse and Combustion Engineering plants. (BWRs and B&W plants are generally not limited by ECCS performance analyses.) Replacing the 1971 decay heat model and multiplier of 1.2 with the 1979 decay heat standard and a multiplier of 1.036 (approximately two standard deviations) caused a decrease in predicted PCT ranging from 250°F to 500°F for large LOCAs and 500 to 1000°F for small LOCAs. The change to the Cathcart-Pawel metal-water reaction model resulted in at most a 75°F reduction in PCT and only a 2°F decrease when coupled with the 1979 decay heat model.

It is generally believed that a large degree of conservatism exists in Appendix K evaluation models and that this conservatism more than compensates for known and unknown nonconservatisms. The margin between Appendix K and best-estimate (at the 95th percentile) peak cladding temperatures for large LOCAs was, however, found to be as little as 200°F. Appendix K calculations based on the 1979 decay heat standard with a 1.036 multiplier can actually predict lower PCTs than best-estimate models at the 95th percentile. This indicates that nonconservatisms in Appendix K models for PWRs may be more significant than generally believed. As a specific example, Westinghouse has shown with their best estimate model that, for some plants, downcomer boiling after accumulator injection stops can add as much as 400°F to the LBLOCA PCT. Most, if not all, Appendix K models do not include the capability to properly calculate downcomer boiling.

Preliminary analyses support the feasibility of alternatives to current Appendix K decay-heat requirements. Additional NRC analyses would be required to examine more recent ANS decay heat standards and associated uncertainty estimates, and to examine the potential for relaxing any other Appendix K conservatisms. As the excessive conservatism in Appendix K is reduced, it is important that licensees and owners of Appendix K evaluation models remain vigilant in assuring that nonconservatisms do not become significant relative to the remaining conservatism.

Time and Resources Required

Time and resources required for development and demonstration under Option 4 would be low to medium compared to other options. NRC analyses required to evaluate particular features of Appendix K for rulemaking would be relatively straightforward, but time and resources would increase with the number of features analyzed.

Once revised Appendix K evaluation models were approved, time and resources required for routine applications and NRC reviews would be the same as for current applications of Appendix K evaluation models. However, significant time and resources (for licensees and NRC) could be required in order demonstrate that Appendix K models with known nonconservatisms remain conservative with respect to best-estimate results at the 95th percentile.

Other Considerations

Most BWRs and some PWRs (e.g., B&W plants) are not significantly limited by results obtained from ECCS evaluation models based on Appendix K and would not significantly benefit from Option 4.

5.2.3.2 OPTION 5: Make Use of Best-Estimate ECCS Performance Analyses Less Burdensome

Requirement

10 CFR 50.46(a)(1)(i) permits licensees to use realistic (best-estimate) ECCS performance models provided that the uncertainty in the calculated results can be estimated. Regulatory Guide 1.157 describes acceptable methods and assumptions for best-estimate ECCS performance calculations with uncertainty propagation. Both model/method approval and computational costs associated with realistic thermal-hydraulic analyses and uncertainty propagation are currently high (~3 years for method approval and ~$2M per analysis). As stated earlier, at this time, less than half of U.S. plants have used or plan to use the realistic model option in 10 CFR 50.46.

Risk Significance

The ECCS acceptance criteria permit some cladding failures but preserve the basic core geometry and hence core coolability. Risk dominant accidents involve much more core damage, specifically, partial to full core meltdown. ECCS evaluations either use conservative models and assumptions as set forth in Appendix K or conservatively compare best-estimate peak cladding temperatures at the 95th percentile to the 2200^{o}F ECCS acceptance criteria. Option 5 would make current best-estimate analyses more efficient but would not alter the underlying best-estimate models. As a result, Option 5 would not have significant risk implications.

Description

Available methods would be applied to accelerate the uncertainty analyses (and potentially the model reviews) required for realistic ECCS performance calculations. Accelerated uncertainty analysis methods that could be applied include:

- Latin hypercube sampling (LHS) and fast probability integration (FPI) methods[Ref. 8], or
- Adaptation of best-estimate codes to perform simultaneous adjoint calculations.

Parallel computing could be applied to further accelerate either of the preceding methods. No regulatory change would be required. Current regulations do not preclude Option 5.

Safety Perspectives

Potential Safety Benefits

As noted in Regulatory Guide 1.157 [Ref. 7], "Safety is best served when decisions concerning the limits within which nuclear reactors are permitted to operate are based upon realistic calculations." Option 5 would encourage the development and use of best-estimate methods. This is consistent with the philosophy of risk-informed regulation.

For plants currently limited by Appendix K calculations, transition to a best-estimate analysis method would result in additional analytic margin (with or without Option 5). The increase in analytic margin would be plant specific as would the nature of changes proposed to take advantage of the increase. Some changes could result in safety benefits (e.g., a lengthened DG start time that resulted in improved DG availability and reliability or fuel loading changes that significantly reduced neutron fluxes to the reactor pressure vessel thereby reducing or delaying the threat of pressurized thermal shock). Without further analysis, it cannot be determined whether the additional analytic margin obtained under Option 5 would permit any specific safety benefit to be realized for any particular group of plants.

The increase in analytic margin provided by shifting from Appendix K to a best-estimate approach should be greater than that obtained under Option 4 (Relax Excessive Appendix K Conservatisms).

Safety Concerns:

No safety concerns have been identified for Option 5. Option 5 would not impact the key elements of defense-in-depth listed in the Option 3 framework. ECCS acceptance criteria would still have to be met (e.g., 95^{th} percentile peak cladding temperature less than 2200°F) for the limiting case in a complete spectrum of pipe breaks.

Potential for Unnecessary Burden Reduction

NRC would likely develop independent capabilities for accelerated uncertainty analyses in order to assist in review/approval of methods, models, and software proposed by industry. Development costs for implementing improved uncertainty analysis methods would be substantial, particularly if existing best-estimate codes were adapted to perform simultaneous adjoint calculations. On the other hand, the turn-around time of realistic ECCS performance analyses with uncertainty propagation could be significantly reduced, especially by parallel computing. The net cost impact of Option 5 would depend on the development versus application cost tradeoff.

Even if more-efficient best-estimate analysis capabilities were developed under Option 5, it is unclear if all licensees who could take advantage of less costly realistic calculations would choose to do so. Even under Option 5, best-estimate analysis costs could still be high compared to Appendix K analysis costs.

For plants that did choose to shift from an Appendix K approach to a best estimate approach, excess conservatism associated with Appendix K calculations of peak cladding temperatures would be reduced providing additional analytic margin. Resulting changes to fuel design, technical specifications, and plant equipment would depend on the amount of additional analytic margin, which could be highly plant-specific. The extent of unnecessary burden reduction would be similarly plant specific. Of the potential cost reductions listed in Table 5-3, additional analytic margin achieved via Option 5 could conceivably be useful for lengthening diesel generator start times, increasing peaking factors, or uprating power. Cost reductions that require changes to design-basis break requirements (e.g., avoiding baffle barrel bolt replacement) could not be achieved via Option 5. LBLOCA analysis costs would only be reduced for plants already using best-estimate analysis methods.

Implementation:

No rulemaking would be required. Current regulations do not preclude Option 5.

Implementing guidance in the form of a NUREG or regulatory guide (e.g., a revised Regulatory Guide 1.157) would be required.

As discussed under Needed Technical Analyses below, software development and demonstration efforts required for implementation would be substantial.

Needed Technical Analyses:

The technical feasibility of Option 5 is medium relative to other options. Accelerated uncertainty analysis methods have been used in a variety of applications including analyses of some aspects of ECCS performance. Full applicability to an ECCS performance analysis would have to be demonstrated.

Considerable software development would be required to adapt, implement, demonstrate, and facilitate the application of efficient uncertainty analysis methods for routine applications in ECCS performance evaluations. Software development would be required either to augment existing best-estimate codes to perform simultaneous adjoint calculations or to automate/facilitate the application of LHS and FPI methods for ECCS performance uncertainty analysis. Software development would also be required to enable either adjoint or LHS/FPI analyses to be performed in a parallel computing environment. Additional development would be required to apply efficient sampling methods (e.g., LHS) to select or automate audit calculations and analyses.

Time and Resources Required

Time and resources required for development and demonstration efforts would be medium to high relative to other options. Adaptation, demonstration, and acceptance of improved methods would be time-consuming and resource intensive.

Time and resources required for realistic ECCS performance analyses and reviews would decrease under Option 5 but would still exceed those for Appendix K analyses and reviews. Time and resources required for licensee applications and NRC reviews would be greater than for Option 4, which assumes continued use of Appendix K models, but less than for Option 6, which assumes the use of existing, less efficient uncertainty analysis methods.

Other Considerations

No other considerations have been identified for Option 5.

5.2.3.3 OPTION 6: Propagate Uncertainty in LBLOCA Size and Location

Requirement

Unless an ECCS evaluation model is developed in conformance with Appendix K to 10 CFR Part 50, a best-estimate analyses of ECCS performance must be conducted in accordance with the following requirements from 10 CFR 50.46(a)(1)(i). "*ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated. Except as provided in paragraph (a)(1)(ii) of this section, the evaluation model must include sufficient supporting justification to show that the analytic technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability* *that the criteria would not be exceeded.*"   Regulatory Guide 1.157 [Ref. 7] describes acceptable methods and assumptions for best-estimate ECCS performance calculations with uncertainty propagation.

Risk Significance

Under the current best-estimate approach, uncertainties in model parameters and certain initial conditions (e.g., power level and linear heat rate) are propagated to obtain a distribution of peak cladding temperatures. Because such analyses are computationally intensive, uncertainties are generally only quantified for the limiting break, that is, for the break size, break location, and LOOP/single-failure assumptions that should result in the highest peak cladding temperature based on scoping calculations and engineering insights. The 95^{th} percentile peak cladding temperature for the limiting break is required to be below the ECCS acceptance criterion of $2200^{o}F$.

From a risk perspective it is conservative to require that ECCS acceptance criteria be met at the 95^{th} percentile confidence level for design basis LBLOCAs for several reasons:

- The limiting break for which acceptance criteria must be met under current regulations may not be the most likely in size or location and may be very unlikely if it also involves LOOP and/or another single failure.

- The level of core damage permitted by ECCS acceptance criteria involves some cladding failures but preserves the basic core geometry. Exceeding the $2200^{o}F$ ECCS acceptance criteria for peak cladding temperature, even by several hundred degrees, does not necessarily imply risk significant core damage. The risk dominant in-vessel release of fission products is initiated when the cladding melt temperature (typically about $3200^{o}F$) is reached and fuel liquefaction is initiated. If adequate coolant flow is established in time to prevent significant fuel liquefaction, a large release would not occur.

- As discussed in Chapter 4, the conditional probability of core damage given a LBLOCA due to failure events not postulated in design-basis accidents can exceed 0.1 for PWRs and 0.01 for BWRs. The same additional failure events would take any design basis LBLOCA to core damage.

In essence, current practice permits only a 5% chance of the most limiting design-basis LBLOCA exceeding ECCS acceptance criteria due to uncertainties associated with best-estimate thermal-hydraulic analyses, but the chance that a design-basis LBLOCA may proceed all the way to core meltdown due to additional equipment failures or operator errors can be as great as ~10% for some plants.

This does not imply that extensive resources should be directed at reducing the conditional probability of core damage given a LBLOCA. As discussed in Chapter 4, because large-breaks in the RCS pressure boundary are very unlikely, the reliability of core cooling and containment functions is generally sufficient to assure that LBLOCAs are not significant contributors to risk.

Description

Option 6 would permit the uncertainty in break size, location, and other characteristics (e.g., double-ended versus slot break) to be propagated with other uncertainties when using realistic (best-estimate) ECCS evaluation models. No regulatory change would be required. ECCS performance analyses would still "*provide assurance that the most severe loss-of-coolant accidents are* *calculated*" and demonstrate "*a high level of probability that the criteria would not be exceeded*" as required in 50.46(a)(1)(i).

The distinction between Option 6 and current practice is in the application and interpretation of comparisons at the 95th percentile. In current practice, the 95th percentile peak cladding temperature for the limiting break within the spectrum of breaks covered by an approved best-estimate ECCS performance model must be less than 2200°F. Under Option 6, the 95th percentile peak cladding temperature integrated over breaks within the spectrum would be required to be less than 2200°F. Limiting conditions with respect to LOOP and another single failure would still be postulated.

Heuristically, Option 6 would permit a 5% chance of exceeding ECCS acceptance criteria given a LBLOCA. Current practice allows a 5% chance of exceeding ECCS acceptance criteria given the limiting LBLOCA.

Consider the simple example depicted in Figure 5-1. Differences in break location are ignored. The breaks considered range from 1 ft^2 to 16 ft^2. This is typical of the size range covered by approved best-estimate ECCS performance models for PWRs. The probability that a break in this range occurs in dA about area A is denoted by p(A)dA. p(A) is assumed to be inversely proportional to A and the integral of p(A)dA from 1 to 16 is normalized to one.

The middle graph in Figure 5-1 is a plot of the 2200°F nonexceedance percentile e(A), which is assumed to decrease with break area from 99.7 for a 1 ft^2 break to 95 for a 16 ft^2 break. Clearly, the limiting break in the example is the 16 ft^2 break. Such a plot would not be a required product of an Option 6 analysis, but it is useful for examining the relationship between Option 6 and the current approach. The level of confidence that ECCS acceptance criteria would be met given a design-basis break within the 1-16 ft^2 range is given by the average 2200°F nonexceedance percentile, which is 0.990.

$$\bar{E} = \int_{1}^{16} e(A)\; p(A)\; dA = 0.990$$

A uniform distribution, p(A)=1/15, would have given $\bar{E}$=0.984; that is, for the example, the average 2200°F nonexceedance potential is not sensitive to the postulated break size distribution.

The lower graph in Figure 5-1 is a plot of the 95th percentile peak cladding temperature as a function of break size. Again, such a plot would not be a required product of an Option 6 analysis, but it serves to illustrate another sensitivity. The plot assumes the 95th percentile peak cladding temperature increases linearly with break area from 1800°F at 1 ft^2 to 2200°F at 16 ft^2, the limiting break size. The 95th percentile peak cladding temperature averaged over the 1 to 16 ft^2 spectrum is 1918°F.

$$\bar{T}_{95} = \int_{1}^{16} T_{95}(A)\; p(A)\; dA = 1918^{o}F$$

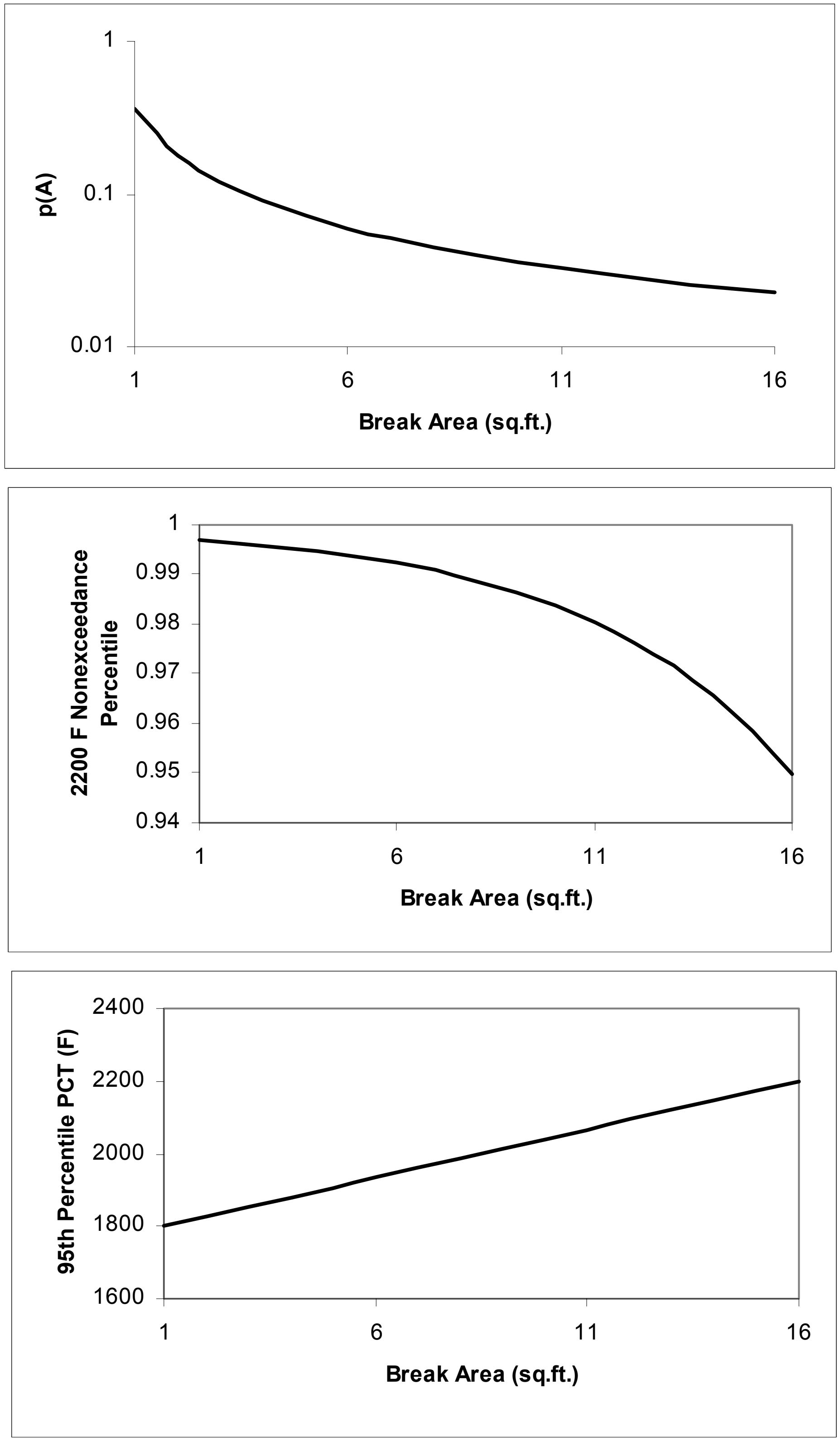


Figure 5-1 Example to Illustrate Propagation of Break Size Uncertainty

A uniform distribution, p(A)=1/15, would give $\overline{T}_{95}$ = 2000°F, so T_{95} is somewhat sensitive to the break size distribution. But even with the use of a conservative uniform distribution, in this simple example, the 95th percentile peak cladding temperature decreases by 200°F from the 2200°F obtained for the limiting break under current practice.

In practice, the uncertainty in break area would simply be included with the other uncertainties that are propagated. The 95th percentile peak cladding temperature for the entire spectrum of breaks would be compared to the 2200°F acceptance criterion. Nonexceedance percentiles and 95th percentile PCTs would not have to be calculated for individual break sizes.

For (the majority of) plants currently limited by more unlikely larger breaks, as in the simple example, Option 6 would result in a significant increase in analytic margin compared to a best-estimate analysis under current practice. But, for plants limited by more likely smaller breaks Option 6 would only result in a slight increase in analytic margin.

At present, best-estimate ECCS evaluation models have only been approved for larger breaks within the spectrum of breaks classified as large in PRAs. Typically the currently approved ECCS evaluation models cover break areas of ~1 ft^2 or more; whereas, the lower-bound for LBLOCAs in PRAs is ~0.4 ft^2 (double-ended break of 6-inch pipe) for PWRs. The simplest implementation of Option 6 would only utilize currently approved models. A full implementation would require the development and approval of ECCS evaluation models covering the additional breaks from ~0.4 to ~1 ft^2.

Safety Perspectives:

Potential Safety Benefits

As noted in Regulatory Guide 1.157, "Safety is best served when decisions concerning the limits within which nuclear reactors are permitted to operate are based upon realistic calculations." Option 6 would encourage the development and use of best-estimate methods. This is consistent with the philosophy of risk-informed regulation.

Plants utilizing best-estimate ECCS evaluation models would obtain additional analytic margin under Option 6. The increase in analytic margin would be plant specific as would the nature of changes proposed to take advantage of the increase. It is conceivable that some changes could lead to safety benefits (e.g., fuel loading changes that significantly reduced neutron fluxes to the reactor pressure vessel could reduce or delay threats of pressurized thermal shock). Without further analysis, it cannot be determined whether the additional analytic margin obtained under Option 6 would permit any specific safety benefit to be realized for any particular group of plants. Option 6 should provide significantly greater increase in analytic margin than Option 4 (Relax Excessive Appendix K Conservatisms), but the increase relative to current best-estimate analyses would require further study.

Safety Concerns:

The relative frequencies of different break sizes, locations, and types (double-ended versus slot) would have to account for non-pipe breaks (e.g., steam-generator manway failure) and for all potential failure mechanisms.

Subject to the preceding caveat, Option 6 would not impact the key elements of defense-in-depth listed in the Option 3 framework. ECCS acceptance criteria would still have to be met with a high level of probability for a complete spectrum of pipe breaks, but the breaks would be weighted according to their relative likelihoods.

Potential for Unnecessary Burden Reduction

Excessive conservatism associated with failing to account for the much lower probabilities of very large-breaks would be eliminated.

For plants currently limited by results of ECCS performance calculations for larger pipe breaks (e.g., cold-leg or hot-leg breaks), additional analytic margin could be used to justify changes that would provide cost savings in a plant-specific manner. Of the potential cost reductions listed in Table 5-3, additional analytic margin achieved via Option 6 could conceivably be useful for lengthening diesel generator start times, increasing peaking factors, or uprating power. Other potential cost reductions listed in Table 5-3 (e.g., reducing LOCA analysis costs and avoiding baffle barrel bolt replacement) could not be achieved via Option 6.

Option 6 would increase computational costs associated with best-estimate ECCS performance analysis. A complete spectrum of breaks would have to be analyzed. Significant time and resources could be required to adapt advanced uncertainty analysis methods to facilitate routine applications under Option 6.

Implementation:

No regulatory change would be required. Existing regulations do not preclude Option 6.

Analyses required to support rulemaking would be moderately complex. Methods for specifying distributions that appropriately account for the relative likelihoods of different break sizes and locations would have to be developed and demonstrated. Regulatory guidance describing acceptable analysis methods and assumptions for implementing Option 6 would have to be developed. The types of analyses that would be required to develop appropriate guidance are described below under Needed Technical Analyses. One or more demonstration analyses could be helpful in developing appropriate guidance.

Best estimate analyses of ECCS performance would increase in complexity. Extensive thermal hydraulic calculations analyses would be required for licenses to apply Option 6. Relative to current best-estimate uncertainty analyses, the number or computations for Option 6 would increase with the range of break sizes covered (two versus one thermal-hydraulic codes to cover entire break spectrum) and with the number of break locations and geometries examined.

Needed Technical Analyses

PFM analyses would be used to develop methods for characterizing the uncertainty in LBLOCA break size and location for various plant types. Relative (not absolute) frequencies of different break sizes and locations would be required. LOCA initiators for which large pipe breaks currently serve as surrogates would have to be identified and their relative frequencies would have to be estimated.

Phenomenological transitions can complicate uncertainty analyses because code outputs cannot be fit by a single regression model when two or more significantly different behavior regimes are spanned. One or more additional uncertainty analyses might be required for significantly different break locations (e.g., pump suction, pump-discharge, and hot leg for PWRs). Similarly, if the limiting failure events changed with break size or location, if both slot and circumferential breaks were probable, or if two or more approved best-estimate models were used to span the entire range of LBLOCA sizes, additional analyses might be required.

Option 6 would clearly benefit from the more efficient uncertainty analysis methods discussed under Option 5. However, significant software development would be required to adapt, implement, demonstrate, and facilitate the use of either adjoint or LHS/FPI uncertainty analysis methods in routine applications. As currently envisioned, Option 6 would not require the calculation of extreme (e.g., $99th) nonexceedance percentiles; however, LHS/FPI methods [Ref. 8] could be applied should $99th percentiles be needed.

Option 6 is considered more feasible than Option 1 (LBLOCA Redefinition). Unlike Option 1, Option 6 would not seek to eliminate any design-basis LBLOCAs, Option 6 would rely on relative not absolute estimates of large-break frequencies, and Option 6 would not require risk assessments of proposed plant changes.

Time and Resources Required

The rigor of LBLOCA frequency analyses required for Option 6 would be less than required for Option 1 (LBLOCA Redefinition) and possibly less than required for Option 2 (Modify Design-Basis LOCA-LOOP Assumptions) or for Option 3 (Use Reliability Analyses in Lieu of Single Failure Criterion for LBLOCAs). This is because a) relative rather than absolute break frequencies are required, and b) the results of Option 6 may not be extremely sensitive to reasonable variations in relative break frequencies.

Time and resources required for developing and demonstrating acceptable methods and assumptions for Option 6 would be medium compared to Option 1.

Resources required for realistic ECCS performance analyses would exceed those required for current best-estimate analyses and far exceed those required for Appendix K analyses.

Relative to other options time and resources required would be high for demonstration and rulemaking and also medium to high for applications and approvals.

Other Considerations

Computational costs for best-estimate ECCS performance uncertainty analyses (with or without Option 6) would benefit substantially from Option 5 (Make Best-Estimate ECCS Performance Analyses Less Burdensome).

Most BWRs and Babcock & Wilcox PWRs are not limited by results of ECCS performance analyses. Option 6 would not be useful for such plants.

The arguments presented above under Risk Significance and Needed Technical Analyses suggest at least two other, less computationally intensive options: 1) make best-estimate comparisons for the limiting break at less than the 95^{th} percentile or 2) develop appropriate input values to permit the application of best-estimate codes to ECCS performance analyses of limiting breaks in a slightly conservative point estimate mode.

5.2.3.4 OPTION 7: Enable Best-Estimate Analyses with NRC-Approved Uncertainty Increments

Requirement

10 CFR 50.46(a)(1)(i) permits licensees to use realistic (best-estimate) ECCS performance models provided that the uncertainty in the calculated results can be estimated. Regulatory Guide 1.157 describes acceptable methods and assumptions for best-estimate ECCS performance calculations with uncertainty propagation. Both model/method approval and computational costs associated with realistic thermal-hydraulic analyses and uncertainty propagation are currently high (~3 years for method approval and ~$2M per analysis). At this time, less than half of U.S. plants have used or plan to use best-estimate ECCS evaluation models.

Risk Significance

As discussed in Chapter 4, large LOCAs are generally not dominant contributors because of the low frequency of large-break initiators and because plants are specifically designed to cope with the most probable failures that could follow a large-break initiator including loss of offsite power and postulated failure events. ECCS acceptance criteria permit some cladding failures but preserve the basic core geometry and hence core coolability. Risk dominant accidents involve much more core damage, specifically, partial to full core meltdown. Current ECCS evaluations either use conservative models and assumptions as set forth in Appendix K or conservatively compare best-estimate peak cladding temperatures at the 95th percent probability level to the 2200^{o}F acceptance criteria. Option 7 would enable the use of best-estimate analyses with NRC-approved uncertainty increments thereby making best-estimate analyses comparable in scope to Appendix K analyses. The fixed uncertainty increments would have to be demonstrably conservative. As a result, Option 7 would not have significant risk implications.

Description

Option 7 would enable licensees to perform best-estimate ECCS performance analyses adding NRC-approved allowances to account for uncertainties. No regulatory change would be required. Current regulations do not preclude Option 7.

Safety Perspectives

Potential Safety Benefits

As noted in Regulatory Guide 1.157, "Safety is best served when decisions concerning the limits within which nuclear reactors are permitted to operate are based upon realistic calculations." Option 7 would encourage the development and use of best-estimate methods. This is consistent with the philosophy of risk-informed regulation.

Nevertheless, the potential for safety benefits is low relative to other options. It is not clear that improved analytic margin would be achieved to enable plant changes that would provide safety benefits.

Safety Concerns:

No safety concerns have been identified for Option 7. Option 7 would not impact the key elements of defense-in-depth listed in the Option 3 framework. ECCS acceptance criteria would still have to be met (e.g., calculated percentile peak cladding temperature plus pre-approved uncertainty allowance less than 2200^{o}F) for a complete spectrum of pipe breaks.

Potential for Unnecessary Burden Reduction

The potential for unnecessary burden reduction is low compared to other options. Computational costs associated with realistic ECCS performance analyses would be substantially reduced to resemble those of Appendix K analyses. It is not clear, however, that fixed allowances for uncertainty could be developed that would result in less conservatism than Appendix K analyses. Most of the cost reductions sought under Option 1 could not be achieved with Option 7.

Implementation:

No regulatory change. Current regulations do not preclude Option 7.

Regulatory guidance would have to be developed describing acceptable uncertainty allowances and other computational assumptions for implementing Option 7. Resources required for these development efforts could well be comparable to those required for Option 6; therefore, before proceeding, an assessment by thermal-hydraulic experts of the potential for success of Option 7 would be appropriate.

Licensee best-estimate codes would have to be demonstrated to remain conservative when used in conjunction with pre-approved uncertainty allowances.

Needed Technical Analyses:

The feasibility of Option 7 is low compared to other options. Extensive best-estimate ECCS performance analyses would have to be conducted to develop fixed allowances for uncertainty in peak cladding temperature (and possibly local oxidation and global hydrogen production) by plant type, break size, break location, and possibly fuel loading characteristics. Reasonable yet bounding uncertainty increments could depend on many factors: plant type, break size, break location, and core design parameters. As a result, computational analyses required to develop fixed uncertainty increments would be complex. Should future core designs be proposed with unanticipated characteristics, analyses like those described above would have to be repeated.

Plant applications and approvals would be similar to current Appendix K analyses.

Time and Resources Required

Time and resources required for development and demonstration would be high relative to other options. Considerable computational resources (hundreds of best-estimate code runs) would be needed to demonstrate the feasibility of Option 7 for a selected plant type. Either NRC, a licensee, a NSSS vendor, or an owner's group could initiate the required analyses for a given plant type, but NRC review and approval would be required in all cases.

Time and resources required for plant applications and approvals would be low relative to other options. Time and resources required to apply best-estimate codes with fixed allowances for uncertainties would be comparable to those for current Appendix K analyses.

Other Considerations

No other considerations have been identified for Option 7.

5.2.4 ECCS Acceptance Criteria

5.2.4.1 OPTION 8: Modify ECCS Acceptance Criteria

Requirement

10 CFR 50.46(b) sets forth five acceptance criteria for ECCS performance in design-basis LOCAs:

1. *The calculated maximum fuel element cladding temperature shall not exceed 2200^{o}F.*
2. *The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.*
3. *The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.*
4. *Calculated changes in core geometry shall be such that the core remains amenable to cooling.*
5. *After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.*

Risk Significance

The ECCS acceptance criteria were established in the early 1970s. Prior to the ECCS rulemaking ECCS acceptance criteria were plant specific, and peak cladding temperature criteria extended up to 2700^{o}F.

The intent of the first two criteria (PCT < 2200^{o}F and local oxidation < 17%) was to assure, based on information available at the time, that cladding ductility would remain sufficient to preserve the basic core geometry of vertical spaced fuel rods. Preserving this basic geometry is necessary for core coolability.

Exceeding the current 2200^{o}F PCT or 17% local oxidation criteria does not necessarily imply a large release of radionuclides from the core. The risk-dominant in-vessel release of fission products is initiated when the cladding melt temperature (typically about 3200^{o}F) is reached and fuel liquefaction is initiated. If sufficient coolant flow is reestablished before the onset of fuel liquefaction, the hot core can still be rapidly quenched; however, significant uncertainty exists regarding the geometry and coolability of post-quench core debris. The uncertainty is aggravated by the rapid, highly-exothermic cladding oxidation rates that would exist with cladding temperatures approaching or exceeding 2200^{o}F. Although the upper debris bed in the damaged Three Mile Island, Unit 2 (TMI-2) core region provides some evidence that rubble beds arising due to extensive fracturing of brittle cladding may be coolable, the configuration, characteristics, and coolability of debris that would result from delayed quenching in a large-break LOCA cannot be predicted with confidence. Current ECCS evaluation models do not attempt to predict debris formation or coolability, but simply model to the point where the basic core geometry would begin to deteriorate. Accordingly, modifying ECCS acceptance criteria to permit a more risk-significant level of core damage is not practical because of modeling complexities and uncertainties.

Degradation to a noncoolable core geometry can only occur if the ductility criteria are exceeded, or if there is excessive ballooning of the cladding resulting in inadequate core-coolant flow area.

The current ECCS acceptance criterion related to global hydrogen production is generally not controlling.

Description

Replace the current prescriptive ECCS acceptance criteria in 10 CFR 50.46 with a performance-based requirement to demonstrate adequate post-quench cladding ductility and adequate core-coolant flow area to ensure that the core remains amenable to cooling, and for the duration of the accident, maintain the calculated core temperature at an acceptably low value and remove decay heat. Adequate post-quench ductility could be demonstrated by three methods: (1) The present 17% oxidation limit and 2200EF temperature limit would be retained in a regulatory guide (as opposed to the regulation) as embrittlement criteria for zircaloy and ZIRLO cladding. (2) For zircaloy, ZIRLO, or any other cladding alloy, post-quench ductility could be demonstrated by testing under conditions that would be described in the regulatory guide. (3) Embrittlement criteria that had been previously approved as part of an exemption from 50.46(b) would continue to be accepted. Option 8 would, therefore, remove the limitation in 10 CFR 50.46 to applications with zircaloy and ZIRLO cladding.

Safety Perspectives

Potential Safety Benefits: Plant changes proposed to take advantage of any increased analytic margin could result in safety benefits (e.g., fuel loading changes that significantly reduced neutron fluxes to the reactor pressure vessel would reduce or delay threats of pressurized thermal shock [PTS]). However, the extent of additional analytic margin would be cladding dependent, and the utilization of such margin would be plant specific. At this time, neither the extent of additional analytic margin nor the nature or magnitude of safety benefits that might result can be estimated.

Safety Concerns: No significant safety concerns have been identified for Option 8. The implications of plant changes proposed based on alternative ECCS acceptance criteria would have to be assessed and either accepted based on generic analyses or subjected to NRC review and approval.

Potential for Unnecessary Burden Reduction

If a cladding material proved resistant to embrittlement beyond the current peak cladding temperature and oxidation limits, additional analytic margin would be obtained in ECCS performance analyses. This additional analytic margin could conceivably be useful for lengthening diesel generator start times, increasing peaking factors, or uprating power. However, if this additional analytic margin were to be utilized, additional steam or hydrogen blowdown to containment would also be predicted. This could render design-basis LOCA calculations for containment more limiting than ECCS performance calculations. At present, no unnecessary burden reduction is anticipated to be achieved via Option 8. The primary purpose of this option is to allow use of cladding materials other than zircaloy or ZIRLO without licensees having to submit an exemption request.

Implementation

The ECCS acceptance criteria of 50.46(b) would be revised.

Needed Technical Analyses

Regulatory guidance would need to be developed to specify suitable means for demonstrating post-quench ductility through experiments or test. The types of experiments envisioned and research results that support the feasibility of Option 8 are discussed in Attachment 2.

Time and Resources Required

Time and resources required for development, demonstration, and rulemaking would be medium to high compared to other options.

Time required for routine applications to advanced cladding materials and NRC reviews and approvals would be low to medium relative to other options.

Other Considerations

No other considerations have been identified for Option 8.

5.3 Comparison of Options

In this section, comparisons are made between the eight options described in Section 5.2. The comparisons are made with respect to six different categories as shown in Table 5-4.

Tables 5-5 through 5-10 provide a brief summary of the relevant information for each option, for each of the comparison categories. More detail is contained in Section 5.2, which discusses each of the options.

Table 5-4 Option Comparison Categories

1. Technical Feasibility	Based on the discussions presented under ***Technical Analyses*** for each option in Section 5.2
2. Potential for Safety Benefits	Based on the corresponding discussions presented under ***Safety Perspectives*** for each option in Section 5.2.
3. Potential for Unnecessary Burden Reduction	Based on the discussions presented under ***Potential for Unnecessary Burden Reduction*** in Section 5.2.
4. Significance of Safety Concerns	Based on the corresponding discussions under ***Safety Perspectives*** for each option in Section 5.2
5. Time and Resources Required for Development and Demonstration:	Based on the discussions under ***Time and Resources Required*** for each option in Section 5.2
6. Time and Resources Required for Applications and Reviews	Based on the discussions under ***Time and Resources Required*** for each option in Section 5.2

Table 5-5 Comparison of the Options by Technical Feasibility

Option		Technical Feasibility
1	LBLOCA Redefinition	Rigorous code analyses of large-break frequencies accounting for non-pipe breaks, all potential failure mechanisms, and uncertainties would require substantial advances in probabilistic fracture mechanics data and methods.
2	Modify Design-Basis LOCA-LOOP Assumptions	Suitable methods and assumptions for estimating large-break frequencies could have to be developed by NRC. Methods for estimating the plant-specific probability of LOOP given a LBLOCA could also have to be developed.
3	Use Reliability Analyses in Lieu of Single Failure Criterion for ECCS	Suitable methods and assumptions for estimating large-break frequencies could have to be developed by NRC. Existing reliability analysis methods would be applied.
4	Relax Excessive Appendix K Conservatisms	Preliminary analyses support the feasibility of alternatives to current Appendix K decay-heat requirements.
5	Make Best-Estimate ECCS Performance Analyses Less Burdensome	Advanced uncertainty analysis methods exist, but full applicability to ECCS performance analyses has not been demonstrated.
6	Propagate Uncertainty in LBLOCA Size & Location	Relative frequencies of possible break sizes and locations accounting for non-pipe-breaks and all potential failure mechanisms would be required. Considering different break locations could make uncertainty propagation costly.
7	Enable Best-Estimate Analyses with Approved Uncertainty Increments	Reasonable yet bounding uncertainty increments could depend on too many factors: plant type, break size, break location, and core design parameters.
8	Modify ECCS Acceptance Criteria	Suitable means for demonstrating adequate post-quench ductility through experiments or tests would have to be developed.

Table 5-6 Comparison of the Options by Potential for Safety Benefits

Option		Potential for Safety Benefits
1	LBLOCA Redefinition	NRC and licensee analyses, personnel, and ECCSs could be focused toward more risk-significant accidents. Plants could lengthen DG start times and operation times of some ESF valves reducing challenges to the reliability and availability of these components. Fast neutron flux to reactor pressure vessel might be reduced at some plants.
2	Modify Design-Basis LOCA-LOOP Assumptions	Some plants could be able to lengthen the DG start time reducing challenges to DG reliability and availability.
3	Use Reliability Analyses in Lieu of Single Failure Criterion for ECCS	Assurance of ECCS reliability appropriate for LOCA threats would be obtained, but other safety benefits have not been identified.

Table 5-6 Comparison of the Options by Potential for Safety Benefits

Option		Potential for Safety Benefits
4	Relax Excessive Appendix K Conservatisms	Plants using Appendix K evaluation models could obtain additional analytic margin, but the extent of additional margin, its utilization, and safety benefits that accrued would be plant-specific.
5	Make Best-Estimate ECCS Performance Analyses Less Burdensome	Additional analytic margin could be obtained for plants shifting from Appendix K to best-estimate evaluation models, and the impact should be greater than for Option 4, but the extent of additional margin, its utilization, and safety benefits that accrued would be plant-specific.
6	Propagate Uncertainty in LBLOCA Size & Location	Analytic margin should be greater than for current best-estimate analyses, although further study would be needed to characterize the difference. The extent of additional margin, its utilization, and safety benefits that accrued would be plant-specific.
7	Enable Best-Estimate Analyses with Approved Uncertainty Increments	It is not clear that a significant increase in current analytic margins would be achieved because large increments to best-estimate code results could be required to account for uncertainties.
8	Modify ECCS Acceptance Criteria	The impact on current analytic margin and the potential for safety benefits would depend on the specific criteria adopted.

Table 5-7 Comparison of Options by Potential for Unnecessary Burden Reduction

Option		Potential for Unnecessary Burden Reduction
1	LBLOCA Redefinition	Potentially, all LBLOCAs could be eliminated as DBAs, but, even if only larger breaks were eliminated from the design basis, increased analytic margin could be sufficient to permit many PWRs to lengthen DG start times, increase peaking factors, relax technical specification requirements, reduce LBLOCA analysis and licensee response & reporting costs, and/or uprate power by 1 to 3%.
2	Modify Design-Basis LOCA-LOOP Assumptions	Lengthening current 10-second diesel-generator start time requirements could result in significant cost savings.
3	Use Reliability Analyses in Lieu of Single Failure Criterion for ECCS	Burden reduction could be limited to relaxing technical specification requirements related to component outage times on a plant-specific basis.
4	Relax Excessive Appendix K Conservatisms	For plants using Appendix K evaluation models, increased analytic margins and associated burden reductions would be very plant specific.
5	Make Best-Estimate ECCS Performance Analyses Less Burdensome	For plants shifting from Appendix K to best-estimate evaluation models, increased analytic margin and associated burden reduction would be very plant specific. Analysis costs would still exceed those for Appendix K analyses.
6	Propagate Uncertainty in LBLOCA Size & Location	Analytic margin would increase relative to current best-estimate analyses, although further study would be needed to characterize the difference. Associated burden reduction would be plant specific, but analyses clearly would be more costly.

Table 5-7 Comparison of Options by Potential for Unnecessary Burden Reduction

Option		Potential for Unnecessary Burden Reduction
7	Enable Best-Estimate Analyses with Approved Uncertainty Increments	It is not clear that a significant increase in current analytic margins would be achieved because large increments to PCTs obtained from an acceptable best-estimate code calculation could be required to account for uncertainties.
8	Modify ECCS Acceptance Criteria	At present, no unnecessary burden reduction is anticipated to be achieved via this option. The primary purpose of this option is to allow use of cladding materials other than zircaloy or ZIRLO without licensees having to submit an exemption request.

Table 5-8 Comparison of Options by Significance of Safety Concerns

Option		Significance of Safety Concerns
1	LBLOCA Redefinition	Estimates of large-break frequencies should account for non-design-basis breaks, all potential failure mechanisms, and uncertainties in models and model parameters. Design-basis LBLOCAs influence ECCS capacity and reliability, containment capacity, and equipment qualification envelopes; therefore, acceptable risk impacts should be demonstrated for plant changes proposed under Option 1.
2	Modify Design-Basis LOCA-LOOP Assumptions	Data-based estimates of large-break frequencies should account for non-design-basis breaks and all potential failure mechanisms. Acceptable risk impacts should be demonstrated for plant changes proposed based on Option 2. Plant electrical grid stability should be monitored.
3	Use Reliability Analyses in Lieu of Single Failure Criterion for ECCS	Data-based estimates of large-break frequencies should account for non-design-basis breaks and all potential failure mechanisms. The reliability analyses and analyses of other DBAs should justify plant changes.
4	Relax Excessive Appendix K Conservatisms	Known nonconservatisms should not jeopardize the overall conservatism of modified Appendix K evaluation models.
5	Make Best-Estimate ECCS Performance Analyses Less Burdensome	No safety concerns have been identified for Option 5.
6	Propagate Uncertainty in LBLOCA Size & Location	The relative probabilities of different break sizes and locations should account for non-design-basis breaks and all potential failure mechanisms.
7	Enable Best-Estimate Analyses with Approved Uncertainty Increments	No safety concerns have been identified for Option 7.
8	Modify ECCS Acceptance Criteria	No safety concerns have been identified for Option 8.

Table 5-9 Comparison of Options by Time & Resources Required for Development and Demonstration

Option		Time & Resources Required for Development and Demonstration
1	LBLOCA Redefinition	Substantial advances in probabilistic fracture mechanics data and methodologies would be necessary.
2	Modify Design-Basis LOCA-LOOP Assumptions	Methods for estimating the plant-specific probability of LOOP given a LBLOCA could have to be developed. Guidance regarding estimates of LOCA frequencies might also have to be developed.
3	Use Reliability Analyses in Lieu of Single Failure Criterion for ECCS	Regulatory guidance regarding the use of existing reliability analysis methods should be straightforward to develop. Guidance regarding estimates of LOCA frequencies might also have to be developed.
4	Relax Excessive Appendix K Conservatisms	The overall conservatism of revised Appendix K codes that contain known nonconservatisms could be difficult to demonstrate for some plant types.
5	Make Best-Estimate ECCS Performance Analyses Less Burdensome	Advanced uncertainty analysis methods exist, but full applicability to ECCS performance analyses has not been demonstrated and software development requirements would be significant.
6	Propagate Uncertainty in LBLOCA Size & Location	Methods for characterizing uncertainty in break size and location and methods for propagating such uncertainties would have to be developed and appropriate regulatory guidance promulgated.
7	Enable Best-Estimate Analyses with Approved Uncertainty Increments	Extensive best-estimate analyses would be required to develop reasonable yet bounding uncertainties, which could depend on plant type, break size, break location, and core design parameters.
8	Modify ECCS Acceptance Criteria	Suitable means for demonstrating adequate post-quench ductility through experiments or tests would have to be developed.

Table 5-10 Comparison of Options by Time & Resources Required for Applications and Reviews

Option		Time & Resources Required for Applications and Reviews
1	LBLOCA Redefinition	Leak-before break, probabilistic fracture mechanics, and risk analyses would be plant specific and highly detailed. ECCS performance analyses would still be required for LOCAs not eliminated as DBAs.
2	Modify Design-Basis LOCA-LOOP Assumptions	Given demonstration of feasibility and appropriate regulatory guidance, analyses of plant-specific LBLOCA frequencies and plant-specific probabilities of LOOP given a LBLOCA, if required, should be relatively straightforward.
3	Use Reliability Analyses in Lieu of Single Failure Criterion for ECCS	With appropriate regulatory guidance, standard reliability analysis of functions provided to cope with design-basis LOCAs should be straightforward to conduct and review.
4	Relax Excessive Appendix K Conservatisms	Once revised Appendix K codes were approved, analysis costs would be the same as for current Appendix K analyses. However, depending on how this option is implemented, confirmatory analyses might be necessary for some plant types to assure that revised Appendix K models retain sufficient conservatism in light of known nonconservatisms.

Table 5-10 Comparison of Options by Time & Resources Required for Applications and Reviews

Option		Time & Resources Required for Applications and Reviews
5	Make Best-Estimate ECCS Performance Analyses Less Burdensome	Once advanced uncertainty analyses methods were implemented and approved best-estimate analyses of ECCS performance would be significantly less costly; however, best-estimate analysis costs would still exceed Appendix K analysis costs.
6	Propagate Uncertainty in LBLOCA Size & Location	Characterizing uncertainty in break size and location and propagating such uncertainties would be more computationally intensive than current best-estimate analyses of ECCS performance.
7	Enable Best-Estimate Analyses with Approved Uncertainty Increments	Assuming pre-approved uncertainty increments could be developed, analysis costs would be comparable to current Appendix K analyses.
8	Modify ECCS Acceptance Criteria	Comparable to current ECCS performance analyses unless Appendix K cannot be extended to treat greater cladding oxidation. Analyses comparable to current best-estimate analyses might then be required.

Option 1 (LBLOCA Redefinition) requires substantial methodological advances to justify the elimination of some or all LBLOCAs as DBAs. On the other hand, there is significant potential for safety benefits and unnecessary burden for Option 1 because considerable improvement in analytic margin could accrue from the elimination of some or all LBLOCAs as DBAs. If LBLOCA redefinition were to facilitate lengthened diesel-generator start times, this could lessen challenges to DG reliability and availability and, at the same time, provide significant cost savings. Safety benefits could also accrue from lengthening valve operation times, increasing peaking factors in a way that reduced the fast neutron flux to the reactor pressure vessel, and by permitting licensees and NRC to shift resources from LBLOCA analyses toward more risk significant accidents. Because of uncertainties regarding LBLOCA frequencies and influences of LBLOCAs on plant design, there are a relatively large number of safety concerns that could have to be addressed under Option 1. Similarly, time and resources required for licensee analyses and NRC reviews could be substantial for Option 1 due to the complexity of the analyses that would be required, some of which could be plant-specific.

At the other end of the spectrum, Option 3 (Use Reliability Analyses in Lieu of Single Failure Criterion for ECCS) involves the application of well-established methods of reliability analysis and much of the required analysis could be extracted from existing PRAs. While Option 3 would assure functional reliabilities appropriate for LOCA threats, there could be limited potential for safety benefit and unnecessary burden reduction.

Option 7 (Enable Best-Estimate Analyses with NRC-Approve Uncertainty Increments) was rejected because its technical feasibility is uncertain, it would entail significant development and demonstration costs, and there is little potential for safety benefit and unnecessary burden reduction.

Option 5 (Make Use of Best-Estimate ECCS Performance Analyses Less Burdensome) would apply advanced uncertainty analysis methods. This option would not require a rulemaking; however, some interest on the part of industry would be required in order to justify significant NRC expenditures on Option 5, because development and demonstration costs could be relatively high. This is also true for Option 6 (Propagate Uncertainty in LBLOCA Size and Location). Although NRC is open to industry initiatives related to Option 5 or Option 6, because no regulatory change is required to enable these options and because no current industry interest in these options has been expressed, NRC staff has no current plans to investigate these options further. However, arguments presented under Option 6 suggest the possibility of allowing best-estimate comparisons for larger, more unlikely breaks at less than the 95^{th} percentile.

At present, no unnecessary burden reduction is anticipated to be achieved via Option 8. The primary purpose of this option is to allow use of cladding materials other than zircaloy or ZIRLO without licensees having to submit an exemption request.

Option 2 (Modify Design-Basis LOCA-LOOP Assumptions), Option 3 (Use Reliability Analyses in Lieu of Single Failure Criterion for ECCS), and the decay-heat changes discussed under Option 4 (Relax Excessive Appendix K Conservatisms) are technically feasible and collectively offer the potential for safety benefits and unnecessary burden reduction. Proposed changes to 10 CFR 50.46 that would implement aspects of these three options, as well as Option 8, are summarized in the next section, and examined in more detail in Attachment 2.

5.4 Proposed Changes to 50.46

The preceding sections approached the risk-informing of 10 CFR 50.46, Appendix K and GDC 35 by identifying potential options to specific regulatory requirements and practices. Regulatory concerns and observations identified in the previous sections are summarized below. Conclusions regarding the suitability of each option are presented in Section 5.4.2. The proposed changes to 10 CFR 50.46 are presented in Section 5.4.3, and are described in more detail in Attachment 2.

5.4.1 Regulatory Concerns and Observations

The following regulatory concerns and observations were identified in the process of delineating, considering, and comparing risk-informed options in the previous sections.

Regulatory Concerns

- ECCS capacity and reliability should be sufficient to limit plant risk associated with LOCAs including (but not limited to) ruptures in the reactor coolant system pressure boundary, stuck open relief valves, transient-induced LOCAs, and draindown events during shutdown.

- Current design-basis LOCA requirements influence key elements of defense-in-depth including ECCS capacity and reliability, containment capacity, and equipment qualification envelopes. Changing current design-basis LOCA requirements should not be permitted to adversely impact key elements of defense-in-depth in a way that could increase severe accident risks.

- Current design-basis LOCA requirements may preclude some plant changes (e.g., lengthening DG start times) that appear to have safety benefits (see Options 1 & 2).

- Current ECCS acceptance criteria and evaluations of ECCS performance may be overly conservative given that LBLOCAs are not risk-dominant. Resources currently devoted to LBLOCA analyses might be better utilized on more risk-significant accidents.

Observations

- Based on available data and analysis methods, it would be difficult to demonstrate that LBLOCAs or even a significant subset of LBLOCAs have a frequency low enough (e.g., 10^{-6} per reactor calendar year) to permit them to be eliminated as design-basis accidents (see Chapter 4, Section 5.2, and Attachment 4).

- As discussed in the Option 3 framework [Ref. 4], it is desirable to have some tie between the frequency of a design-basis initiator and requirements imposed; that is, more stringent criteria should be applied to more frequent initiators (e.g., prevent fuel damage for anticipated operational occurrences (AOOs) but limit the extent of fuel damage for DBAs).

- Larger design basis LOCAs in which both LOOP and an additional single failure are postulated can have frequencies less than 10^{-6} per reactor calendar year. Consistent with the Option 3 framework, sequences of events with mean frequencies less than 10^{-6} per reactor calendar year could be excluded from the design-basis (see Options 2 & 3).

- Because a simultaneous LOOP must be postulated, a LBLOCA is generally believed to be the DBA that establishes the DG start time (typically at 10 seconds, see Option 2). While this may not actually be the case (NSAC-130 [Ref. 1]), relaxing the simultaneous LOOP requirement for LBLOCAs may make it easier for licensees to extend the DG start time. The design-basis LBLOCA also requires rapid operation times for certain ESF valves.

- Lengthening DG start time and ESF valve operation times could have a positive impact on the reliability of the affected components (see Options 1 and 2).

- Getting ESFs on early in a LOCA can reduce the peak containment pressure but drive the PCT higher due to lower containment backpressure. For some plants, the limiting design-basis LOCA for ECCS performance is one in which no failures of electric power or of ECCS components are postulated. For such plants, eliminating the requirement to postulate LOOP or a single failure would provide no reduction in peak cladding temperature and, therefore, no additional analytic margin in ECCS performance analyses. However, eliminating the requirement to postulate LOOP or a single failure may allow these plants to relax some technical specifications.

- If a newer decay heat standard is used, the reduction in overall conservatism could be significant. For the more probable small breaks, the Appendix K decay heat requirement is one of the most significant sources of conservatism. Therefore, it is important that licensees and owners of Appendix K evaluation models remain vigilant in assuring that nonconservatisms do not become significant relative to the remaining conservatism (see Option 4).

- It is preferable to have licensee's accident evaluations based on best-estimate code predictions rather than conservative (e.g., Appendix K) codes. Licensees and NRC can then better predict the occurrence and timing of key events and be better prepared to deal with accidents should they occur.

- Current best-estimate analyses are conservative in comparing LBLOCA peak cladding temperature at the 95th percentile to the 2200^{o}F ECCS acceptance criterion (see Option 6).

- Basing ECCS acceptance criteria on residual cladding ductility (e.g., PCT < 2200^{o}F and local oxidation < 17% of cladding thickness) is conservative with respect to a risk-significant level of core damage. A large release would generally require melting of cladding and associated fuel liquefaction, which would be initiated when the peak cladding temperature reached the cladding melt temperature (~3200°F for Zircaloy 4). However, modifying ECCS acceptance criteria to permit a more risk-significant level of core damage is not practical because of modeling complexities and uncertainties. (See Options 6 and 8.)

- The impact of potential regulatory changes that could result in additional analytic margin with respect to ECCS performance may be limited by containment considerations. An increase in predicted steam and hydrogen blowdown to containment could cause the peak containment pressure to approach or exceed the containment design pressure.

- The preceding observation may not be risk significant. It is known from PRAs and IPEs that the containment ultimate failure pressure exceeds (typically by a factor of 2 or more) the containment design pressure. For some plants, PreOp testing takes containment to 1.15 times design pressure.

- The main impact of design-basis LOCAs on equipment qualification envelopes is sustained exposure to a steam-saturated containment. This is true for both large and smaller LOCAs. At many plants, other DBAs (e.g., main steam line breaks) determine the peak containment temperature and/or the peak containment pressure.

- It is preferable that a risk-informed alternative to 10 CFR 50.46 and related regulations be simple and easy to implement. Otherwise, time and resources spent in licensee analyses and NRC reviews can make an alternative undesirable. There is little point in providing a risk-informed alternative that no licensee would volunteer to implement.

5.4.2 Contribution of Options to Proposed Changes

It is clear that none of the options, by itself, addresses all of the identified regulatory concerns. Table 5-11 provides conclusions regarding the appropriate role for each of the options with respect to the changes to 10 CFR 50.46, Appendix K and GDC 35. The proposed changes include elements of several of the options, including Option 2 (Modify LOCA-LOOP Assumptions), Option 3 (Use Reliability Analyses in Lieu of Single Failure Criterion for ECCS), Option 4 (Relax Excessive Appendix K Conservatisms), and Option 8 (Modify ECCS Acceptance Criteria).

Table 5-11 Contribution of Options to Proposed Changes

<table>
<tr><th colspan="2">Option</th><th>Description</th></tr>
<tr><td>1</td><td>LBLOCA Redefinition</td><td>NRC should continue to work with industry to ascertain the technical feasibility of LBLOCA redefinition.</td></tr>
<tr><td>2</td><td>Modify Design-Basis LOCA-LOOP Assumptions</td><td rowspan="2">Replace the simultaneous LOCA/LOOP requirement and the single failure criterion now used to accomplish system reliability with either deterministic system reliability requirements based on risk information, or with an ECCS functional reliability commensurate with the LOCA core damage frequency.</td></tr>
<tr><td>3</td><td>Use Reliability Analyses in Lieu of Single Failure Criterion for ECCS</td></tr>
<tr><td>4</td><td>Relax Excessive Appendix K Conservatisms</td><td>Appendix K to 10 CFR Part 50 should be revised to permit the use of less conservative decay heat models. The possibility of relaxing other unnecessary conservatisms in Appendix K should be examined.</td></tr>
</table>

Table 5-11 Contribution of Options to Proposed Changes

Option		Description
5	Make Best-Estimate ECCS Performance Analyses Less Burdensome	Current regulations permit Option 5, 6, and 7; however, Option 7 does not appear to be practical. Further, in the absence of industry interest, no further work by NRC is planned to develop Option 5 or Option 6.
6	Propagate Uncertainty in LBLOCA Size & Location	
7	Enable Best-Estimate Analyses with Approved Uncertainty Increments	
8	Modify ECCS Acceptance Criteria	Existing ECCS acceptance criteria can be simplified. The criteria should assure adequate post-quench ductility and core-coolant flow area, and for the duration of the accident, should assure that calculated core temperature is maintained at an acceptably low value and decay heat is removed. It would not be practical to set or demonstrate compliance with acceptance criteria based on more risk-significant levels of core damage (e.g., fuel liquefaction).

5.4.3 Proposed Changes

The proposed changes to 10 CFR 50.46, Appendix K and GDC 35, which are a composite of various aspects of different options as discussed in the previous section, include both (A) changes to the technical requirements of the current regulations, and (B) a risk-informed alternative to the current regulations. These changes are summarized below, and are discussed in more detail in Attachment 2. Note, the changes described here are limited to the ECCS requirements. There are additional changes, not yet studied, that could "spin-off" including, for example, changes to requirements for containment design or equipment qualification (EQ). The changes included here presume that other requirements (beyond those for the ECCS) remain unchanged until proper study has been performed.

(A) <u>Changes to the Current 10 CFR 50.46</u>:

1. Replace the current prescriptive ECCS acceptance criteria in 50.46 with a performance-based requirement. This requirement would, one, demonstrate adequate post-quench cladding ductility and adequate core-coolant flow area to ensure that the core remains amenable to cooling, and, two, for the duration of the accident, maintain the calculated core temperature at an acceptably low value and remove decay heat. Use of a performance-based requirement rather than the current prescriptive criteria would allow use of cladding materials other than zircaloy or ZIRLO without licensees having to submit an exemption request. AND

2. Revise the requirements for the ECCS evaluation model to be based on more realistic analyses. In the near term, this revision would involve an update of Appendix K requirements based on more current and realistic information. Specifically this update could involve:
   a. replacing the current 1971 ANS decay heat curve with a model based on the 1994 ANS standard.
   b. replacing the current decay heat multiplier of 1.2 with an NRC-prescribed uncertainty treatment.

c. deleting the limitation on PWR reflood steam cooling for small reflood rates.
d. replacing Baker-Just zirconium steam model with the Cathcart-Pawel zirconium steam oxidation model for heat generation.
e. deleting the prohibition on return to nucleate boiling during blowdown.

As part of this update, the staff will also consider the recognized nonconservatisms and model limitations to insure that proper safety focus is incorporated in any new rule.

(B) Development of a Voluntary Risk-Informed Alternative to 10 CFR 50.46:

Replace the current approach for obtaining ECCS reliability with more risk-informed and realistic approaches. This voluntary alternative would include technical requirements to ensure an ECCS reliability that is commensurate with the frequency of challenge to systems. In place of the simultaneous LOCA/LOOP requirements and single failure criterion, two options would be offered to accomplish ECCS system reliability:

1. A deterministic system reliability requirement based on risk information (e.g., an ECCS design requirement that only one train of ECCS is required for LOCAs larger than a specified size). OR

2. An ECCS functional reliability requirement that is commensurate with the LOCA frequency (e.g., a requirement that ECCS design must be such that the CDF associated with a specified set of LOCAs is less than an NRC-specified CDF threshold, with due consideration of uncertainties).

5.5 References

1. W. R. Schwarz and B. Chexal, "The Effect of Diesel Start Time Delay on Westinghouse PWRs," NSAC-130, Electric Power Research Institute.

2. USNRC, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Regulatory Guide 1.174, July 1998.

3. J. P. Poloski, D. G. Marksberry, C. L. Atwood, W. J. Galyean, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1985," NUREG/CR-5750, INEEL/EXT-98-00401, Idaho National Engineering and Environmental Laboratory, February, 1999.

4. USNRC, “Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control),” SECY-00-0198, September 14, 2000.

5. G. Martinez-Guridi, et al., “Evaluation of LOCA With Delayed Loop and Loop With Delayed LOCA Accident Scenarios,” NUREG/CR-6538, BNL-NUREG-52528, Brookhaven National Laboratory, July 1997.

6. American Nuclear Society Standards Committee Working Group ANS-5.1, “American National Standard for Decay Heat Power in Light Water Reactors, ANSI/ANS-5.1-1979, X 1.036, August 1979.

7. USNRC, "Regulatory Guide 1.157: Best-estimate Calculations of Emergency Core Cooling System Performance," May 1989.

8. F. Eric Haskin, Bevan D. Staple, and Chuanyi Ding, "Efficient uncertainty analyses using fast probability integration, *Nucl.Eng.&Design*, **166** (1996) 225-248.

Attachment 2

TECHNICAL WORK TO SUPPORT RULEMAKING FOR A RISK-INFORMED ALTERNATIVE TO 10 CFR 50.46, APPENDIX K AND GDC 35

June 2001

TABLE OF CONTENTS

LIST OF TABLES

1. INTRODUCTION

In Attachment 1, a feasibility assessment was performed regarding risk-informing the technical requirements of 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors,” and related requirements in Appendix K to 10 CFR Part 50, “ECCS Evaluation Models,” and General Design Criterion (GDC) 35, “Emergency core cooling.” The assessment was performed to existing requirements in four major areas:

1. Acceptable evaluation models for calculating emergency core cooling system (ECCS) performance
2. Acceptance criteria for ECCS performance following postulated loss-of-coolant accidents (LOCAs)
3. Assurance that the ECCS function can be accomplished reliably
4. The spectrum of breaks for which ECCS cooling performance must be calculated

The staff concluded that changes to the first three parts above of 50.46 are justified and the last is potentially feasible. Consequently, the staff:

- recommends changes to the technical requirements of the current 10 CFR 50.46,
- recommends development of a risk-informed alternative to 50.46, and
- plans on continuing the feasibility assessment regarding the spectrum of breaks.

Each of these activities, however, requires further technical work to either support a proposed rulemaking or determine the feasibility. This additional technical work, for each of the above, is discussed in detail in this attachment.

2. Description of Changes to Current 50.46 and Associated Technical Work

This section provides a description of the various proposed changes to the current10 CFR 50.46, Appendix K and GDC 35. Each section provides a description of the proposed changes and a summary of the technical work that is needed to support the rulemaking effort.

2.1 ECCS Evaluation Models

2.1.1 Proposed Change

Two options for ECCS evaluation models are proposed: the first option involves an update to the Appendix K model, in which unnecessary conservatisms are removed; the second option involves, unchanged, the existing realistic (i.e., best-estimate) model with propagation of uncertainties.

In the first option, the changes to Appendix K would permit the use of a more realistic alternative to existing decay heat requirements of Appendix K to 10 CFR Part 50. Specifically, the changes would permit use of the 1994 decay-heat standard with an NRC-approved method to account for uncertainty. Also, new information will be evaluated to determine if the following changes can be made:

- Delete the requirement involving pressurized water reactor (PWR) reflood steam cooling limitations,
- Replace the existing Baker-Just zirconium steam model with the Cathcart-Pawel zirconium steam oxidation model for heat generation, and
- Modify the requirement which prohibits return to nucleate boiling during blowdown, to allow return to nucleate boiling if justified.

In the first option, changes to Appendix K, licensees would be expected to remain vigilant in assuring that the required features of Appendix K continue to compensate for known nonconservatisms of Appendix K evaluation models.

Either of the options could be applied to a particular class of design-basis LOCAs. For example, the second option, best-estimate analyses with uncertainty propagation could, as in current practice, be applied to analyze the largest LOCAs while the first option, Appendix K models with unnecessary conservatisms removed, could be applied to all other LOCAs.

2.1.2 Needed Technical Work

The previous section described possible changes to the Appendix K ECCS evaluation model to remove unnecessary conservatisms. Many of the “required and acceptable features” of Appendix K models are specified to ensure that certain phenomena are considered in the analysis. Other features require sensitivity analysis to assure that a conservative, or at least a realistic selection is made for the subject parameter. Requirements specifying analyzed power level and peaking factors (paragraph I.A) are the most important boundary conditions in the ECCS analysis. It is not proposed these requirements be changed.

Only a few requirements actually contain specific conservatisms that could be considered for change. The U.S. Nuclear Regulatory Commission (NRC) staff has identified four requirements in Appendix K that warrant consideration for change. They are:

1. Decay Heat (paragraph I.A.3 & 4)
2. Heat Removal by ECCS (paragraph I.D.3-6)
3. Metal Water Reaction (paragraph I.A.5)
4. Nucleate Boiling Return (paragraph I.C.4.e)

Decay Heat (I.A.3 & 4) - Appendix K currently requires decay heat be modeled using the draft 1971 American Nuclear Society (ANS) standard for decay heat with a multiplier of 1.2. Information derived over the last 25 years would permit the use of a newer decay heat standard, which involves more sophisticated uncertainty methods and a greater number of model parameters and options. Sensitivity studies have indicated that a newer standard could reduce predicted peak cladding temperature (PCT) by as much as several hundred degrees. The 1994 ANS standard considers much more recent available data and methods. The 2s uncertainty as evaluated in the new standard is significantly less than the 0.2 value described in the 1971 standard. For any significant change in Appendix K evaluation models, such as decay heat, it should be determined that sufficient conservatism remains in the resulting analysis with respect to performance variables, in particular peak cladding temperature.

The uncertainty methods in the 1994 ANS standard will be assessed to assure that they can be reliably applied to Appendix K ECCS calculations. It is expected that the result would be a single multiplier, a time dependent multiplier or a simple uncertainty method. It should also be determined if separate uncertainties for actinides and recoverable fission energy should be included.

For Appendix K, additional analysis will be required to identify acceptable and practical model options and parameter values. This will include guidance regarding:

1. Whether the reactor operating history should be represented by a histogram of multiple irradiation intervals or must be modeled as a single interval,
2. Appropriate durations (T_a) of the irradiation interval(s),
3. Values of the recoverable energy per fission (Q_i) for U-235, Pu-239, and U-238, and Pu-241 (the contribution of each of these nuclides to decay heat is inversely proportional to its recoverable energy per fission),
4. The correction factor G(t) for neutron capture in fission products, and
5. The value of the atoms of U-239 produced per second per fission R, which is used in the model of U-239 and Np-239 decay heat power.

Heat Removal by ECCS (I.D.3-6) - Heat removal by ECCS includes several phenomena that are part of the heat removal process (coolant void generation, entrainment, reflood heat transfer, and certain system effects). The PWR reflood steam cooling limitations (I.D.5.b) leads to non-physical, computationally difficult and often unnecessarily conservative modeling requirements. Data developed since promulgation of 50.46 in 1974, could support deletion of this requirement. Evaluation of this data would be required to determine its applicability to the spectrum of PWR large break reflood analyses.

In small-break LOCAs (SBLOCAs) heat removal is dominated by system effects (e.g., natural circulation, steam generator heat transfer, ECCS pump capacity, operator actions). It should be noted that none of the specific requirements of Section I.D. of Appendix K, “Post-Blowdown Phenomena; Heat Removal by ECCS” are directly related to small breaks.

Metal-Water Reaction (I.A.5) - Metal water reaction can be an important heat source during periods of extended core uncovery, when the cladding reaches elevated temperatures. Appendix K currently requires the use of the Baker-Just zirconium steam model, which is known to be conservative compared to more recent data and models (e.g., the Cathcart-Pawel zirconium steam oxidation model). Previous studies showed that the effect in terms of peak cladding temperature is less than 80°F compared to less conservative models, when the peak cladding temperature remains below 2200°F. This is obviously less than the several hundred degrees for decay heat. If a revised acceptance criteria allowed peak cladding temperatures to exceed 2200°F, the effect could be very large depending on temperature. This is due to the exponential temperature dependence for the rate of this exothermic reaction. Also, metal water reaction rate is currently tied to the 2200°F PCT and 17% equivalent cladding reacted (ECR) embrittlement criteria discussed in Section 2.3, which discusses post-quench ductility as an alternate criterion. Changing the metal water reaction model would only apply to the heat release associated with oxidation. Models affecting embrittlement are described in Section 2.3.

Nucleate Boiling Return (I.C.4.e) - The Appendix K prohibition on return to nucleate boiling during blowdown is most important for plants that are predicted to have PCT early in the LOCA. It is not important for SBLOCA, since core design and experimentation have demonstrated that only core coverage is required for SBLOCA heat removal. To provide large-break LOCA (LBLOCA) relief from this requirement would involve a review of blowdown heat transfer data with appropriate "minimum film boiling temperature" data. The subject has not been extensively studied, and large uncertainties still exist. A review will be necessary to determine if current information is sufficient to allow relaxation of this requirement.

In addition to the technical work required to support removal of unnecessary conservatisms from the existing Appendix K ECCS model, there is a small amount of work that would be performed with respect to the existing best-estimate model. Paragraph (a)(1)(i) of 50.46 and Regulatory Guide 1.157 describe the features of a realistic (best-estimate) ECCS evaluation model including evaluation and propagation of uncertainties. There would be no fundamental change to that regulatory guidance for this alternative. However, some citation of the 1994 ANS decay heat standard could be added to Regulatory Guide 1.157 and other reference to more recent information could be added if that is deemed necessary.

A summary of the needed technical work to support rulemaking for the proposed changes associated with the ECCS evaluation models is provided in Table 2-1. This table also includes estimates of the time and resources required.

Table 2-1 Summary of Needed Technical Work (ECCS Evaluation Models)

Technical Work	Calendar Time to Perform Work	Resources Needed
• Assess uncertainty methods in 1994 ANS decay heat standard • Identify acceptable and practical decay heat model options and parameter values • Evaluate current (i.e., post-1974) reflood heat transfer data • Assure new metal water reaction models can be implemented properly • Evaluate current data on blowdown heat transfer and minimum film boiling temperature • Support modification of regulatory guide (RG) 1.157 to include 1994 ANS decay heat standard and other recent information	6-12 months	1.1 FTE $350K

2.2 ECCS Acceptance Criteria

2.2.1 Proposed Change

Two of the acceptance criteria in 10 CFR 50.46(b) were developed specifically for fuel rods with zircaloy cladding, and paragraph (a) restricts the application of 10 CFR 50.46 to zircaloy and a similar ZIRLO cladding alloy. These two criteria are the 2200°F peak cladding temperature limit and the 17% total cladding oxidation limit. It would be possible to remove the restriction on the application of 10 CFR 50.46 to zircaloy and ZIRLO by replacing the numerical values in the regulation with the underlying principle on which they were based. That principle is described in the Opinion of the Commission in the matter of the ECCS rulemaking hearing [Ref. 1].

During a LOCA, fuel rod cladding would be subjected to a high temperature transient in steam and it would experience rapid oxidation. Oxidation at high temperatures leads to changes in the metallurgy of the cladding and can result in complete embrittlement. To ensure that the cladding would remain intact during a LOCA, the Commissioners concluded that ductility should be retained after recovery from the high temperature transient. The presence or absence of ductility was determined from a mechanical test on rings of zircaloy cladding material that had been subjected to a LOCA-like high temperature transient and then quenched. Using data from Oak Ridge National Laboratory, the Commission concluded that ductility would be retained if the percent of the cladding thickness that was oxidized did not exceed 17% and the temperature of oxidation did not exceed 2200°F.

The ECCS acceptance criteria in the current 10 CFR 50.46 would be replaced by a performance-based requirement to demonstrate adequate post-quench cladding ductility and adequate core-coolant flow area to ensure that the core remains amenable to cooling, and for the duration of the accident, maintain the calculated core temperature at an acceptably low value and remove decay heat. Degradation to a noncoolable core geometry can only occur if the ductility criteria are exceeded, or if there is excessive ballooning of the cladding resulting in inadequate core-coolant flow area. As noted in Chapter 5 of Attachment 1, cores with more damage may be coolable, but phenomenological uncertainties and computational capabilities, as a practical matter, preclude the use of less conservative criteria. The current ECCS acceptance criterion related to global hydrogen generation is generally not controlling, and is not included in the risk-informed alternative. Global hydrogen generation is adequately dealt with by the hydrogen rule, 10 CFR 50.44.

It is anticipated that three options for assuring adequate ductility would be described in a regulatory guide. The first option would retain prescriptive acceptance criteria, though the specific criteria would now be contained in the regulatory guide, not in the actual regulation. This allows changes to be made to the prescriptive criteria, based on new information or for application to new cladding types, without necessitating another rulemaking. As a practical matter, for the time-being, the regulatory guide would specify the existing ECCS acceptance criteria for peak cladding temperature (≤ 2200°F) and local oxidation (≤ 17% of the cladding thickness), which would only be applicable for zircaloy and ZIRLO cladding. The second option would be a performance-based option, where the peak cladding temperature and local oxidation criteria are replaced by a more general requirement that adequate post-quench ductility be demonstrated. Specifically, the staff would adopt a criterion of non-zero post-quench ductility for fuel rod cladding. This performance-based approach, which would be applicable to zircaloy, ZIRLO, or any other cladding alloy, would provide flexibility for current and future fuel designs by eliminating reference to cladding types and numerical values. Procedures for determining post-quench ductility could be described in the regulatory guide.

The third option would apply to embrittlement criteria that had previously been approved as part of an exemption from 50.46(b), which would continue to be accepted.

2.2.2 Needed Technical Work

Current NRC research at Argonne National Laboratory is exploring post-quench ductility of high-burnup zircaloy cladding, and this research could provide information to support development of a regulatory guide. Funding for the research at Argonne National Laboratory is already in the budget so additional funding would not be needed. Interpretation of test results and formulation of the technical basis for a regulatory guide would be performed by the NRC Technical Advisory Group on Fuel (TAG-F).

A summary of the needed technical work to support rulemaking for the proposed changes associated with the ECCS acceptance criteria is provided in Table 2-2. This table also includes estimates of the time and resources required.

Table 2-2 Summary of Needed Technical Work (ECCS Acceptance Criteria)

Technical Work	Calendar Time to Perform Work	Resources Needed
• Research to explore post-quench ductility of high-burnup zircaloy cladding • Interpretation of test results and formulation of the technical basis for a regulatory guide	4 months[1]	0.2 FTE $0K[2]

[1] Due to the need for input from ongoing research at Argonne National Laboratory, technical work to support these modifications can not be completed before the end of calendar year 2001.

[2] Funding for the research at Argonne National Laboratory is already in the budget, so additional funding would not be needed.

3. Description of Risk-Informed Voluntary Alternative to 50.46 and Associated Technical Work

This section provides a description of the proposed voluntary alternative to 10 CFR 50.46, Appendix K and GDC 35 which addresses the ECCS functional reliability. Each section provides a description of the proposed technical requirement and a summary of the technical work that is needed to support the rulemaking effort.

3.1 Proposed Change

Current requirements postulate loss of offsite power (LOOP) and single failures in design-basis LOCA analyses in order to assure that the ECCS function can be achieved reliably. Risk analysis techniques provide a more realistic approach for assessing and ensuring reliability.

Two options to the existing LOOP/single-failure assumptions are proposed:

- The first option would replace the simultaneous LOCA/LOOP requirement and the single failure criterion now used to accomplish system reliability with deterministic system reliability requirements based on risk information. As such, it is anticipated that LOCAs in different frequency intervals would have different system reliability requirements applied to them. The different LOCA frequency intervals and associated system reliability requirements would be established by the NRC based on generic information for the industry as a whole, or by plant type, and once implemented, licensees would be free to choose this alternative without any NRC review or approval. The ECCS reliability information could be used to establish generic ECCS requirements in several different areas. First, for specific LOCA sizes where the LOOP contribution to ECCS unreliability is below a specified value, the thermal-hydraulic analyses required in 50.46 for these LOCA sizes would not require the assumption that offsite power is available. Second, the ECCS reliability information could be used to potentially specify the minimum features of the ECCS design required for different LOCA sizes (e.g., an ECCS design requirement that only one train of ECCS is required for LOCAs larger than a specified size). Based on these requirements, licensees could pursue modifications to operational limits (e.g., technical specifications) or, under some conditions, modifications to system design. Any generic changes to the ECCS requirements based on reliability arguments, however, would have to be based on the reliability of the ECCS for all accidents, not just LOCAs, unless there is an ECCS component that is only required for LOCAs (e.g., the accumulators). Third, the ECCS reliability information could also resolve the single passive failure issue by allowing any identified single passive failures in the ECCS design as long as the cumulative contribution of these failures to ECCS reliability is small.

- The second option would allow each licensee to use plant-specific information to establish their own LOCA frequency intervals and ECCS reliability requirements, commensurate with the LOCA frequency. This information could include plant-specific PFM calculations to establish LOCA frequencies and ECCS reliability evaluations that reflect unique plant features and plant-specific operational data. In this alternative, the NRC would set an acceptable threshold value for the frequency of core damage associated with a specified set of LOCA initiators. The licensee would be required to meet these core damage thresholds values using plant-specific LOCA frequencies and ECCS reliability values, and due consideration of uncertainties. Suppose, for example, that the threshold for all pipe-break-initiated LOCAs was set at 10^{-5} per reactor year. Let CDF_{LOCA} denote the corresponding

LOCA-initiated core damage frequency (CDF) estimate for a particular plant. To exercise the alternative would require that CDF_{LOCA} be less than the NRC-specified threshold value. If this were true, licensees could then use reliability analysis techniques to reduce the LOOP requirement for some LOCA sizes in the thermal-hydraulic calculations required by 50.46. In addition, the plant-specific analysis could be used to modify operational limits (e.g., technical specifications) or, under some conditions, to modify system design. Any plant-specific changes to the ECCS requirements based on reliability arguments, however, would have to be based on the reliability of the ECCS for all accidents, not just LOCAs, unless there is an ECCS component that is only required for LOCAs (e.g., the accumulators).

3.2 Needed Technical Work

The needed technical work to support rulemaking for the proposed changes associated with ECCS functional reliability involves the following three principal areas:

- LOCA frequencies
- Conditional probability of LOOP given a LOCA
- ECCS risk/reliability analyses

The required technical work associated with each of these areas is briefly described in the following sections. Section 3.2.4 summarizes this information and provides estimates of the time and resources required.

3.2.1 LOCA Frequencies

Technical analysis will be required to identify frequencies of LOCA break sizes. One important frequency value could be associated with the break size that represents the transition point to LBLOCAs often used in probabilistic risk assessments (PRAs) (e.g., >6 inches for PWRs). One possibility for obtaining the frequency of this transition point and the frequency of other LOCAs might involve a review of current LOCA frequencies used in PRAs, and consideration of pipe break frequency data and insights, to determine an appropriate frequency based on engineering judgement. Another possibility might involve an update of the NUREG/CR-5750 analysis of pipe-break LOCA frequencies, to account for more recent operating experience. For this possibility, a less subjective analysis of uncertainties in the frequency estimates for LOCAs may be required to assure that suitably conservative estimates are used in implementing those alternatives requiring analyses of LOCA initiator frequencies or LOCA CDF contributions.

Alternatively, probabilistic fracture mechanics (PFM) and a review of service history data could be used to identify the pipe diameter corresponding to a desired break frequency; however, this would be both costly and time-consuming. The technical work to support this option would be similar in scope to the work required for redefinition of the LBLOCA (see Section 4.2).

3.2.2 Conditional Probability of LOOP Given a LOCA

Technical analysis will be required to support the development of regulatory guidance that would specify acceptable methods and assumptions for quantifying the conditional probability of LOOP given a LOCA, and for identifying those plant-specific features that tend to decrease the likelihood of LOOP given an LOCA (e.g., plant capability to communicate with transmission system operators). All potential causes of LOOP given a LOCA would be considered. Both consequential and random LOOP events would be considered. Both prompt and delayed LOOP events would be considered. Earlier analyses of LOCA-LOOP frequencies [Ref. 2], which were performed as part of the resolution of Generic Safety Issue (GSI)-171, would be updated and upgraded to account for subsequent insights, operating experience, and the impact of electric power deregulation on grid stability. Prior analyses of automatic bus transfers would also be reexamined [Ref. 3]. These analyses would identify and/or confirm features that tend to decrease the likelihood of LOOP given a LOCA and compile relevant data. In particular, plant capabilities to communicate with transmission system operators, as mentioned above, and the availability of data on electric grid stability would be evaluated. The regulatory guidance may also specify methods for monitoring (e.g., electric grid stability) to assure the preceding analyses remain valid.

3.2.3 ECCS Risk/Reliability Analyses

Technical analysis will be required to support development of regulatory guidance that would specify acceptable methods and assumptions for performing LOCA CDF and ECCS reliability analyses for those alternatives requiring such analyses. In addition, appropriate reliability and CDF threshold values would have to be selected. Some specific issues that would need to be addressed include:

a. The ECCS success criteria in the CDF calculation are based on a CDF definition whereas the ECCS performance acceptance criteria are based on peak cladding temperature (PCT) and cladding oxidation (i.e., cladding ductility). Consideration will have to be given as to whether the definition of CDF should be the same as the ECCS acceptance criteria when utilizing this risk-informed alternative.

b. The proposed risk-informed alternative only considers pipe break LOCA initiators in this evaluation consistent with the scope of the current 10 CFR 50.46. However, to be completely risk-informed, all LOCAs (including stuck-open relief valves and seal LOCAs) should be considered when evaluating ECCS acceptability.

c. A determination will need to be made as to what are appropriate LOCA frequencies to use in these evaluations. Specifically, it will have to be decided whether operational data estimates such as in NUREG/CR-5750 are sufficient, or whether PFM analyses are required.

d. When evaluating the ECCS reliability, some issues not currently modeled in PRAs may need to be addressed. These include dynamic effects such as asymmetrical loads, pipe whip, and jet impingement; delayed LOOP and degraded voltage issues; and sump plugging. Guidance for treating these issues probabilistically could be required. In addition, guidance might be required for addressing some potential ECCS risk issues that have been inconsistently treated in past PRAs, as discussed in Section 4.4.2 of Attachment 1.

e. Since the use of mean CDFs could be allowed in the alternative, guidance with respect to the scope of needed uncertainty analyses might need to be developed.

f. The thermal-hydraulic (T-H) calculations cover a limited period of time during blowdown and recovery. The ECCS for PWRs would be in the injection mode. Thus, if the reliability evaluations are to be used for making decisions on the T-H conditions, the reliability calculations should be limited to evaluating the injection mode of ECCS and not the recirculation mode.

g. If the ECCS recirculation mode reliability is to be assessed, that information could be used in some fashion to address the long-term cooling requirement in 10 CFR 50.46. Currently, no calculations (PRA or T-H) are used to evaluate compliance with this requirement. Sump plugging is clearly a recirculation mode issue that could be addressed in a risk-informed 10 CFR 50.46 based on a reliability calculation. An appropriate mission time would need to be identified for use in this recirculation evaluation (e.g., 30 days).

h. When evaluating the LOCA CDF, it is currently envisioned that only ECCS systems would be credited. However, further consideration may need to be given to expanding the reliability calculations to include non-ECCS systems. The implications of this would need to be identified (e.g., would a non-ECCS system credited in the CDF evaluation have to be subjected to an Appendix K T-H calculation).

Demonstration analyses for selected plants may be helpful in developing the needed regulatory guidance.

3.2.4 Summary

A summary of the needed technical work to support rulemaking for the proposed changes associated with ECCS functional reliability is provided in Table 3-1. This table also includes estimates of the time and resources required.

Table 3-1 Summary of Needed Technical Work (ECCS Functional Reliability)

Technical Work	Calendar Time to Perform Work	Resources Needed
• Determine appropriate LOCA frequencies for use in LOCA CDF and ECCS reliability analyses • Identify features that tend to decrease the likelihood of LOOP following a LOCA, and develop guidance for estimating plant-specific probability of LOOP given a LOCA • Develop guidance for performing LOCA CDF and ECCS reliability analyses • Determine appropriate reliability and CDF threshold values	6-9 months	0.5 FTE $350K

4. Description of Additional Changes to 50.46 (Spectrum of Breaks) and Associated Technical Work

This section provides a description of the proposed additional changes to 10 CFR 50.46, Appendix K and GDC 35 which addresses the spectrum of breaks. Each section provides a description of the proposed technical requirement and a summary of the technical work that is needed to support the feasibility study.

4.1 Proposed Change

In 10 CFR Part 50, there are two places where the rules define loss-of-coolant accidents as hypothetical or postulated *"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 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"* (10 CFR 50.46(c) and 10 CFR Part 50, Appendix A, "Definitions and Explanations" -- note, in the definition in Appendix A, the words "in pipes" are not included). In addition, 10 CFR Part 50, Appendix K, paragraph (I)(C)(1), states that *"In analyses of hypothetical loss-of-coolant accidents, a spectrum of possible pipe breaks shall be considered. This spectrum shall include instantaneous double-ended breaks ranging in cross-sectional area up to and including that of the largest pipe in the primary coolant system."* The industry believes that the top priority in risk-informing 50.46 is redefining the maximum break size for the LBLOCA; i.e., defining a new maximum LOCA break size above which breaks would no longer need to be included in the plant design basis. The feasibility assessment for this effort has not yet been completed and recommendations are not proposed by the staff in the near term. However, if in the future, the industry, or an applicant, can successfully justify that a given set of LOCAs has a collective frequency below a specified threshold value, for example, 10^{-5}/yr, the staff would consider reduction of regulatory requirements to be commensurate with this frequency (e.g., allow best-estimate calculations without the need for uncertainty propagation, relax reporting requirements, or relax technical specifications). The staff has agreed to engage with the public in a series of meetings to further define and refine the feasibility effort for redefining the LBLOCA. The underlying concerns of degradation mechanisms, materials properties, and uncertainty analyses will be addressed, as well as issues of resources required to accomplish this work.

4.2 Needed Technical Work

In the longer term, the staff would continue to work with industry to establish the feasibility of LBLOCA redefinition, and develop regulatory guidance for plant-specific submittals (or, possibly, submittals by plant or reactor type). Technical work and resources to support redefinition of the LBLOCA are summarized in Table 4-1. Included in this work is development of an analytical approach, as described in Appendix A to this attachment. The results of this work will be conveyed to the public and industry through a series of meetings. Then, existing computer codes must be reviewed, adapted, undergo quality assurance checks, and be benchmarked in order to be applicable to redefinition of the LBLOCA. Existing codes are deficient for this purpose. The benchmarking would involve the use of sample plant evaluations or sample pipe system evaluations. These results would be compared to the results of other codes to ensure that any discrepancies are fully understood. A key result of this effort will be the identification of break frequency to a corresponding pipe diameter. This function cannot be extrapolated based upon a few data points, rather frequencies must be determined for each pipe diameter, since pipes of varying diameter will be exposed to different stresses and degradation mechanisms and thus result in different break frequencies.

Table 4-1 Summary of Needed Technical Work (Spectrum of Breaks)

Technical Work	Calendar Time to Perform Work	Resources Needed
Establish Feasibility and Support Development of Regulatory Guidance: • Develop analytic approach (including series of public meetings to convey technical requirements) • Review of existing computer codes • Updating/developing computer codes • Sample plant or pipe system evaluations • Code comparisons	2-3 years	2.4 FTE $1.2M

5. REFERENCES

1. USAEC Opinion of the Commission in the Matter of “Rulemaking Hearing, Acceptance Criteria for ECCS for Light Water Cooled Nuclear Power Reactors,” CLI-73-39, December 28, 1973.

2. G. Martinez-Guridi, P. K. Samanta, T. L. Chou, and J. W. Chang, "Evaluation of LOCA With Delayed LOOP and LOOP with Delayed LOCA Accident Scenarios," NUREG/CR-6538, July 1997.

3. S. Mazumdar, “Engineering Evaluation Report: Operational Experience on Bus Transfer,” AEOD/E90-05, June 1990.

APPENDIX A

TECHNICAL CONSIDERATIONS FOR REDEFINING THE LARGE-BREAK LOSS-OF-COOLANT ACCIDENT

APPENDIX A

Introduction

Leak-before-break (LBB) has been historically applied in many industries. In the nuclear industry, it has been applied to nuclear piping, primarily PWR Class 1 or primary piping systems. These analyses have been conducted since the draft standard review plan (SRP 3.6.3) for LBB was issued in 1983. The NRC's LBB procedure is a deterministic two-step approach with certain screening criterion. The screening criterion essentially requires that no active degradation mechanisms that can cause long surface flaws exist in the pipe system of interest, and that there are not any unquantifiable high stresses (i.e., water hammer) in the pipe system. Hence, LBB has traditionally been applied with some experienced-based risk knowledge. NRC LBB Draft SRP 3.6.3 requires computational procedures for all pipe systems, including those with no detectable flaws and low stresses. The LBB computational procedure is meant to demonstrate sufficient flaw tolerance for the material being evaluated using existing leakage detection requirements from NRC Regulatory Guide 1.45.

In several ways the current NRC Draft SRP 3.6.3 LBB procedure is simpler than if a leakage calculation and fracture assessment was to be done for an actual service crack. If risk-based analyses are used to generically reduce large-break LOCA requirements for 10CFR50.46, then the LBB methodology needs to include all the complications of making an actual leakage assessment as well as consideration of unknown future degradation mechanisms. The following general methodology is a broad summary of the procedures that are believed necessary to make these calculations with a greater degree of confidence than is currently used for LBB.

The following inputs are necessary to make a generic risk-based assessment:

- Initial flaw distributions,
- Pipe-system boundary conditions and postulated flaw locations,
- Degradation mechanisms (flaw growth evaluations, as well as crack morphology parameters for leak-rate calculations),
- Normal operating and transient loads,
- Material response, i.e., strength, toughness, including aging effects or loading-rate effects on material properties,
- Leak-rate estimations,
- In-service inspection, and
- Uncertainty analyses including all the above variables.

Some preliminary recommendations of these different input variables are given below.

Initial Flaw Distribution

An initial flaw distribution is used for life calculation in structures. This has been done for reactor pressure vessels for pressurized thermal shock. Initial flaw distributions could be determined from inspection results and validation of inspection results, i.e., cutting apart pipe weld, and expert processes and computer codes, i.e. PRODIGAL. Work in the PISC program (an international Program for the Inspection of Steel Components) may be helpful. Additionally there are requirements for pre-service flaw sizes in ASME Section XI Article IWB-3500. Records of acceptable flaw sizes from pre-service inspections would be helpful if they are available. The probability of multiple flaws in a weld should also be included.

Degradation Mechanisms

Degradation mechanisms are important from several considerations. First, will there be some degradation mechanism that will cause the initial flaws to grow. Secondly, and perhaps more important, is whether the mechanism of interest will result in long surface flaws as occurred in intergranular stress corrosion cracking (IGSCC) flaws in boiling water reactor (BWR) plants in the late 1960 and 1970's. Thirdly, the degradation mechanism will also affect the surface roughness and straightness of the flow path for leakage considerations. LBB analyses assuming smooth fatigue cracks would give shorter crack lengths for a given leak rate than IGSCC or primary water stress corrosion cracking (PWSCC) cracks with rougher surfaces that might actually occur in service.

In regards to flaw length development, the degradation mechanism could force either short or long surface flaws to develop. For instance, intergranular stress-corrosion crack growth is dominated by girth weld residual stress fields in the heat-affected zone of the base metal adjacent to the weld. These stress fields can be high and relatively uniform around the circumference. The service stresses may not be non-uniform enough around the circumference to make the surface flaws grow only over a short length. Thermal fatigue loads or thermal striping can cause cyclic loads over either large areas or very short lengths, respectively, and hence have different significance on LBB behavior.

As previously described, PWSCC cracking has occurred in small-diameter instrumentation lines, steam generator tubes, control-rod head-penetration tubes, and more recently in bimetallic lines of the large-diameter primary piping in PWRs. The cracking in the control-rod head tubes was first axial with little structural concern, but more recently some circumferential cracks have occurred that are of concern. The bimetallic weld metal cracks in PWR large-diameter piping were mainly axial cracks, but the potential for circumferential cracks forming needs to be assessed for continued LBB acceptability.

Generic long-term acceptance of LBB behavior for reducing LBLOCA requirements needs to consider these types of mechanisms, as well as the possibility of long-surface flaw degradation producing mechanisms.

Normal Operating and Transient Loads

In typical LBB analyses, the normal operating loads need to be considered for leakage detection capabilities. Normal operating loads are also needed for flaw growth analyses in probabilistic evaluations. The normal operating loads need to consider:

- pressure,
- dead-weight loads,
- thermal expansion,
- residual stresses,
- cyclic thermal loads, and
- fabrication stresses.

The first of these three stress contributors are included in typical design reports. Residual stresses are important for crack initiation and growth considerations, and their effect on the crack opening for leakage rates should also be addressed. Cyclic thermal loads such as thermal striping are not design-basis loads and need to be considered in probabilistic flaw growth analyses. Fabrication stresses are difficult to quantify, but should be evaluated. It is well known that in pipe system repairs, the pipes may spring apart when cut. Some determination of the equivalent stresses from such realistic behavior should be included. Another non-design stress is the effect of weld repairs (i.e., weld overlays) on stresses in other locations of the pipe system. These stresses should also be included in the analysis.

Hoop stresses are the controlling stress for axial flaws. For circumferential flaws, leakage and fracture are controlled by longitudinal membrane (from pressure) and bending stresses, but torsional stresses should also be included. Methods to determine equivalent bending stresses for combined bending and torsional stresses to determine leakage and fracture behavior exist.

Transient loads are considered to typically be the safe-shutdown earthquake (SSE) loads. This is a design load that occurs with some probability of occurrence. However, other greater and lesser earthquake loads can occur with different probabilities and should be included in a complete probabilistic analysis. In some pipe systems, the thermal transients during start-up or shutdown can result in higher stresses than the SSE loads. Such thermal transient loads should be included in the crack growth and fracture analyses. It is highly unlikely that an SSE event would occur during a transient thermal load, but if the thermal loads occur for a longer time period, then the combined load probability should be included.

Pipe-System Boundary Conditions and Postulated Flaw Locations

The pipe dimensions, pipe supports, and other pipe boundary conditions need to be specified. The location of a postulated crack in the pipe system determines the stresses, but an often overlooked aspect is that the pipe-system boundary conditions can also affect the leakage and fracture calculations from the viewpoint of how the pipe rotation is restricted by the rest of the pipe system. Typical LBB leakage analyses assume the ends of the pipe are free to rotate like an end-capped vessel. Restraint of this rotation will make the leakage size flaw larger and the same transient loads will be more detrimental to the structural integrity.

In LBB analyses, the location with the worst material properties and highest stresses is typically considered. Probabilistically, there could be a combination of material properties, lower normal operating stresses and slightly lower SSE stresses that might be more detrimental to fracture behavior than the location with the highest SSE stresses.

Flaw orientation is also an important consideration for actual flaw assessments. In hypothetical flaw analyses for LBB, the flaws are typically circumferential flaws in straight-pipe girth welds. Axial flaws in nuclear piping are rare, except for erosion-corrosion flaws in Class 2 or 3 piping which are not of interest here. For circumferential flaws, existing LBB analyses assume that the flaw is centered on the bending plane for normal operating stresses and the bending plane under the transient stresses. Using the location of the flaw being centered on the bending plane for transient stresses is conservative for the fracture analyses, but using the center of the bending plane for the leakage calculations is the least conservative flaw location around the pipe circumference. Realistically, the normal operating stress bending plane and the transient bending planes are probably different. More importantly, fabrication flaws can occur anywhere around the pipe circumference and may not be located on either bending plane. Some analyses exist for consideration of the effects of off-centered crack for leakage and fracture analyses, and should be incorporated into a generalized probabilistic approach.

The flaws could be located in the weld metal, base metal, or heat-affected zones (HAZ). Typical LBB analyses use the lower of the weld metal or base metal properties. However, cracks in stainless steel weld tests frequently grow and follow the fusion line of the weld. Consideration of the toughness in the HAZ or fusion lines should be included in a probabilistic analysis.

The occurrence of flaws in elbows and other pipe fittings than just straight pipe should also be considered.

Material Response

Some of the key material response factors that should be considered in any risk-based analyses are:
- the strength of the material,
- fatigue/environmental crack initiation properties,
- fatigue crack growth rate including environmental effects, and
- fracture toughness.

The strength of the material changes with temperature, so that the operating temperature needs to be considered. Code strength or actual strength values could be used in deterministic analyses, but statistical variability of the material strength (yield, ultimate, and strain-hardening exponent) could be employed in a probabilistic analysis.

Fatigue initiation life with environmental effects (i.e., ?K_{th} or K_{Iscc}) could use lower-bound values or a statistical distribution if enough data exists. These values are for the temperate, water chemistry of interest and vary by material. Even within a class of materials, the effect of certain elements, i.e., sulphur in ferritic steels, can change the material sensitivity to environmental effects. Loading rates in service and hold times can also effect the environmental crack growth rates.

The material fracture toughness values can differ significantly between the base metal and weld metal. Wrought stainless steels usually have a very high toughness, but that toughness may be lower if there is a higher amount in impurities such as sulphur and phosphorous. The type of weld procedure can greatly effect the toughness and needs to be considered. For stainless steels, SMAW and SAW welds statistically have the same toughness which is significantly lower than TIG welds. There is also some evidence that the toughness in the fusion line of austenitic welds can be lower than the weld metal, so that fusion line toughness and heat-affected zone (HAZ) toughness values should also be included in a probabilistic distribution of properties for the degradation mechanism of interest.

There are many factors that can also affect the fracture toughness, such as
- thermal aging of duplex stainless steel materials,
- carbon depletion in some bimetallic welds,
- dynamic strain aging in ferritic steels and weld, and

- cyclic loading.

Thermal aging of cast stainless steels (particularly those with higher ferrite content) has been well documented and can cause the base metal toughness of pipe or elbows to be much lower than wrought stainless steel or even lower than the weld metal toughness. Stainless steel welds can have duplex structures. Thermal aging effect on lowering the toughness of stainless steel welds is somewhat known, but has not been used in LBB evaluations to date.

Bimetallic welds can have a carbon depletion problem in the ferritic material heat-affected zone that will lower the toughness in that region. This would occur in older nuclear plants where the bimetallic weld were entirely made of stainless steel weld metal with no buttering of the ferritic steels using Inconel weld metal.

Since seismic loads are both dynamic and cyclic, effects from these loading conditions should be considered. Dynamic strain aging is an effect that can raise the ultimate strength, reduce the percent elongation in tensile tests, and lower the fracture toughness of carbon steels under certain conditions. Dynamic strain aging is a dislocation pinning phenomena which is sensitive to temperature and loading rate. The dislocation pinning can be by free atomic nitrogen or carbon atoms and occurs at temperatures between 300F and 650F for ferritic steels and their welds. This temperature range is in the operating temperature range of primary nuclear piping systems. In more recent years the effects of loading rate on the toughness of ferritic steels has been determined to be important for ferritic steels due to this dynamic strain aging effect. The loading rate to be used in testing should correspond to reaching crack initiation in the fracture test in the time that corresponds to one-quarter of the period of the first natural frequency of the pipe system of interest.

There has also been some results showing that cyclic loads during ductile tearing can also lower the fracture resistance. This is much harder to quantify, but generally if the ratio of the minimum to maximum stresses (in the presence of a crack) is more negative than -0.5 then cyclic loads could be detrimental to the material fracture resistance.

Leak-Rate Estimation

Leak-rate evaluations are made for several purposes. First, in LBB analyses, the leak-rate is used as a crack detection method. Secondly, the residual crack opening after a transient event can determine if the crack resulted in a small-break or large-break opening. Finally the rate of crack opening during a large-break determines the thermal-hydraulic loads on the reactor core supports, as well as pipe whip and jet impingement forces.

The NRC LBB methodology for nuclear piping involves a two-step analysis procedure. The first is to determine the crack size that corresponds to the detectable leak rate at normal operating conditions with some safety factor on leak-detection capability. That crack must be stable under transient loads with an additional safety factor on the crack length.

There are a number of leak-rate related factors to consider in making such an assessment that may be pertinent in a risk-based analysis. These factors include:

- crack morphology parameters (i.e., roughness, number of turns in the crack path, general straightness of the crack path through the thickness),

- effect of pipe-system boundary conditions (see earlier discussion) on restraining the crack-opening displacement for a cracked pipe under internal pressure,
- effect of weld residual stresses on the crack-opening displacements,
- effect of crack-face pressure,
- effect of off-center cracks, and
- potential of particulate plugging with tight cracks.

The choice of the crack morphology parameters can have a major role in estimating the detectable leakage crack size, and thus the probability of pipe break occurring. Analysis have shown that an LBB applicant's choice for crack morphology parameters resulted in a predicted leakage size flaw about half the crack size as when the statistical values of surface roughness and number of turns from service cracks were used. These crack morphology parameters are dependant on the degradation mechanism, and care should be undertaken in selecting the proper values as well as validated leak-rate computer codes. Statistical variability of the crack morphology parameters has been defined in NUREG/CR-6004 for several degradation mechanisms.

The restraint of pressure-induced bending has been determined to be an important aspect in some LBB analyses to date. This comes about from the crack-opening analyses in typical LBB applications assuming that the ends of the pipe are free to rotate when there is a circumferential crack in a pipe under internal pressure. The pipe-system geometry and boundary conditions (from pipe supports, nozzles, elbows, etc.) will restrain that pipe rotation and hence have a lower crack-opening displacement. This means that for the same leak rate, the crack length would be longer when considering the pipe-system boundary conditions on the crack-opening displacement. Since the effect of restraint of pressure-induced bending is strongly a function of crack size, this effect will have a bigger impact on small-diameter pipe than large diameter pipe, except perhaps for steam lines where larger cracks are required to get the same leakage as in subcooled-water lines.

From an LBB perspective, weld residual stresses are more important for "thin-wall" pipe than they are for "thick wall" pipe, and since thin-wall pipes are more closely associated with smaller diameter pipes, this effect will also be more important in smaller-diameter pipes. Since the effect of weld residual stresses in thin-wall pipe is to rotate the crack faces such that flow through the crack is choked off, ignoring this effect could cause an under prediction of the pipe break frequency for the smaller-diameter pipes.

The effect of crack-face pressures is to open the crack more so that leakage detection is easier for shorter cracks. This effect is seldom used in LBB analysis.

As noted in the section on crack location, having a circumferential crack off-centered from bending plane will reduce the crack opening and result in having a longer crack for the same leak rate. Some analyses exist to account for this behavior.

Leak-rate codes that give similar accuracies are the PICEP and SQUIRT codes. The uncertainty in these models have been established by comparisons to experimental data. Accuracies to plus or minus a factor of 2 is typical. Both of these leak-rate codes can have problems with the friction factor partes of their analyses, that is accounted for by using global roughness values verus local roughness parameters as explained in NUREG/CR-6004. Plugging by particulates with tight cracks may also be a concern.

Leak rate analyses can be used to quantify thrust loads imparted. Obviously, this sort of analysis is totally different than that invoked in a LBB assessment. This type of analysis will involve single-phase flow through a large break where such factors as the crack-morphology parameters are of little concern. Furthermore, before using any such analyses, it would be necessary to provide some level of validation. Further note that some leak-rate codes, such as SQUIRT, are not appropriate in this regime since they were written for two-phase flow cases only.

In-Service Inspection

In-service inspection (ISI) capabilities have increased significantly since the building of the initial nuclear power plants. Nevertheless, there are certain material/flaw combinations that are difficult to detect or size. The probability of detection needs to be incorporated in the risk-based analyses.

Typical detection of leaking cracks has frequently been determined by visual observations, not by sophisticated leak-detection equipment. Water puddles on the floor, or piles of boric acid powder in areas have been detection sources in many early cases of a particular degradation mechanism.

Risk-based inspection techniques have told operators to look at historical cracking locations. This is a necessary but not complete approach. Such methods may have told operators to minimize or eliminate inspections from other areas where cracks have not been seen in the past, but may be the source of new cracking mechanisms. The occurrence of PWSCC in hot legs is one such example.

Nevertheless, the ability to nondestructively size flaws is a highly desirable capability. Even if flaw depths cannot be accurately determined, the determination of flaw lengths is essential in augmenting LBB behavior for a degradation mechanism can produce long-surface flaws.

The assistance of ISI in improving LBB behavior should be considered in a risk-based LBB analysis, but the detrimental aspect of focusing resources away for locations where future flaw degradation mechanisms may develop needs to also be included.

Uncertainty Analyses

Several probabilistic models exist for determining the risk and uncertainties in the risk evaluations. One of the earlier ones was the PRAISE code, which was modified for the Westinghouse SRRA (Structural Reliability and Risk Assessment) code. These codes have made improvement since the early days of the PRAISE code from the days of the resolution of Generic Issue A2 on asymmetric blow down loads on older nuclear power plants. However, they are lacking in many of the deterministic leak-rate and fracture considerations in the NUREG/CR-6004 procedure as well as many of the above considerations.

A more complete leak-rate and fracture analysis is included in NUREG/CR-6004 for probabilistic LBB analyses, but even since the NUREG/CR-6004 report in 1995, many additional deterministic improvements have been made. The NUREG/CR-6004 probabilistic analysis does not include calculations of the probability of crack initiation and crack growth, but determines the conditional probability of a crack occurring with a given leak rate and no imposed safety factor for determination of the failure probability of an SSE event. Although useful in showing that the leakage size flaw and normal operating stresses are more important than the SSE loads in determining failure probabilities, the NUREG/CR-6004 study was only a sensitivity analysis that was not carried out far enough.

The PRAISE, SRRA, and NUREG/CR-6004 analyses are lacking in determining the failure probabilities with the deterministic analyses that currently exist as well as many of the considerations discussed above.

Other Failures

One aspect about the double-ended pipe fracture assumptions for ECCS sizing, which is not directly addressed, is that this loss-of-coolant sizing approach not only accounts for hypothetical cracks occurring in the piping systems, but also accounts for many other potential failure causes that are not related to cracks occurring in pipes. The LBLOCA pipe break analysis is a surrogate for these mechanisms, as well as the more frequently analyzed hypothetical pipe breaks. Some examples of these other failure modes that are covered by the LBLOCA analysis include:

- failure of the bolting for steam generator manways,
- failure of bolts on valve bonnets for LOOP stop valves,
- multiple failure sources, i.e., control rod drive mechanisms (CRDMs), and
- indirect sources.

During the initial nuclear piping LBB work at Lawrence Livermore for the evaluation of Generic Issue A2 (on asymmetric blow-down loads on older nuclear plants without pipe restraints at the vessel nozzles), some consideration was mode for indirect causes leading to pipe breaks. One such cause was the dropping of a large object from an overhead crane. The movement of heavy equipment in the containment building while the plant is under operation should have careful consideration. Similarly, bolt degradation criteria and protection from new bolting failure mechanisms should also be considered.

As noted earlier in this Attachment, the current risk-based models are good for near-term extrapolation of failure frequencies. However, if some new degradation mechanism reaches its incubation time, then the probabilities calculated from those risk analyses are invalid. Hence, these future degradation mechanisms may be considered to be precursors to future failure modes. By reducing the LBLOCA requirements, the protection against these other failure modes is reduced.