TECHNICAL WORK TO SUPPORT POSSIBLE RULEMAKING FOR A RISK-INFORMED ALTERNATIVE TO 10 CFR 50.46/GDC 35

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1. INTRODUCTION

1.1 Background

The Nuclear Regulatory Commission's (NRC) policy statement on probabilistic risk analysis (PRA) encourages greater use of PRA to improve safety decisionmaking and regulatory efficiency. Consistent with this policy, the staff presented options to the Commission for modifying the regulations in Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50) to make them risk-informed. One of the options (Option 3) involved making changes to specific requirements in the body of the regulations. 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 (U.S. NRC 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, which involves developing the technical bases for rule changes resulting from Phase 1 approved by the Commission (RES lead) and performing rulemakings to implement changes (NRR lead). The Commission approved proceeding with this plan in a February 3, 2000, SRM.

Based on input from stakeholders, the requirements for the analysis of design basis loss-of-coolant accidents (LOCAs) contained in 10 CFR 50.46 and Appendix K to Part 50, and the emergency core cooling systems (ECCS) requirements in General Design Criteria (GDC) 35 were chosen as high-priority candidates for risk-informed regulation. These requirements specify assumptions, methods, and acceptance criteria for use in evaluating the adequacy of the ECCS for design basis LOCAs.

An assessment of the feasibility of changes to existing ECCS-related regulatory requirements was conducted and provided to the Commission with SECY-01-0133 [Ref. 1]. This assessment included an evaluation of the current requirements, their basis and evolution; a review of related regulations and implementing documents; a review of relevant risk information and related accidents; development and comparison of potential options for risk-informing current requirements; and development of recommendations for changes.

In SECY-01-0133, one of the recommendations that the staff made to the Commission was that rulemaking should be undertaken to develop a risk-informed alternative to the current 10 CFR 50.46. 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 challenges to systems. In SECY-02-0057 [Ref. 2], which provided an update to SECY-01-0133, the staff stated that it continued to recommend the risk-informed alternative to the ECCS reliability requirements.

1.2 Objective

This document presents the technical work performed by NRC staff in support of possible rulemaking for a risk-informed alternative to 10 CFR 50.46, specifically in regard to the ECCS reliability requirements contained in GDC 35 of Appendix A to 10 CFR Part 50.

1.3 Scope, Limitations and General Comments

It is anticipated that the current wording of GDC 35 of Appendix A to 10 CFR Part 50 would be changed as part of possible rulemaking. The revised wording would permit the ECCS to be designed, operated or evaluated based on modern, quantitative reliability considerations instead of the currently stipulated loss of electric power and single failure assumptions. It should be noted that the only changes proposed here are to the ECCS requirements, i.e., GDC 35, and as stated in SECY-01-0133, no changes are currently being proposed to the requirements for containment design or equipment qualification (EQ), or to extend changes to the single failure criterion generically to other systems (e.g., GDCs 17, 34, 38, 41 and 44).

Changes to the ECCS acceptance criteria of 10 CFR 50.46(b) and changes to the requirements of Appendix K to 10 CFR Part 50 are also being studied, but are also beyond the scope of this document.

As discussed in SECY-01-0133, Attachment 1, "Feasibility Study of a Risk-Informed Alternative to 10 CFR 50.46, Appendix K and GDC 35," 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), 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. The results of this review for 10 CFR 50.46, Appendix K and GDC 35 are documented in Appendix A to this report.

Performance monitoring and corrective action strategies may need to be developed for various risk-informed ECCS applications. In such cases, the implications regarding risk estimates should be evaluated, and it should be determined whether the justification for ECCS-related changes remains valid as performance monitoring data are gathered.

The PRA models used to demonstrate ECCS safety function reliability in a risk-informed ECCS application would typically need to meet the Capability II requirements of the ASME PRA Standard (as expected to be endorsed [with exceptions, if necessary] by the NRC in a forthcoming regulatory guide). Some additional modeling issues that need to be addressed for this specific application, since they may have a significant impact on the results, are briefly described below (a more detailed description for some of these items is provided in Appendix B):

- The potential for the dynamic effects of LOCAs (i.e., pipe whip, jet impingement, and asymmetric loads on reactor coolant system components) to cause additional failures that could significantly affect LOCA progression and ECCS response should be examined and included in the PRA as appropriate.

- The potential for phenomena associated with consequential steam generator tube rupture given a LOCA or main steamline break may need to be included in the PRA as appropriate, depending on the extent of tube degradation at a particular plant.

- If a loss of offsite power (LOOP) occurs as the result of a LOCA, the LOOP is likely to be delayed from a few seconds up to about a minute after the onset of the LOCA. Issues

associated with a delayed LOOP (e.g., overloading of emergency diesel generators [EDGs], non-recoverable damage to EDGs and ECCS pump motors, lockout energization of breakers, or lockup of the load sequencer) should be addressed in an ECCS-reliability application, and included in the PRA as appropriate.

- In demonstrating that the ECCS functional reliability is commensurate with the frequency of accidents in which ECCS success would prevent core damage, it is recognized that current PRAs credit both ECCS and non-ECCS systems for providing coolant to the core. If reliance on non-ECCS equipment in the PRA is used to justify relaxation of ECCS capability, some increased level of regulatory treatment may be necessary for the non-ECCS equipment credited in the PRA.

As a final note, the ASME PRA Standard specifies the use of best estimate computer code calculations for use in determining system success criteria during accidents. However, regardless of the model chosen to determine the ECCS success criteria used in the PRA, evaluation model calculations that meet 10 CFR 50.46 requirements would still be required for any risk-informed change to the ECCS configuration, operational requirements or design basis. The one deviation would be that changes to eliminate scenarios from the ECCS design basis analyses would not require a 10 CFR 50.46 evaluation of the eliminated scenarios since that is the goal of the application.

1.4 Approach

The staff has developed a framework for risk-informing 10 CFR Part 50 [Ref. 3] that describes the approach, process and guidelines the staff applies in reviewing, formulating, and recommending risk-informed alternatives to 10 CFR Part 50 technical requirements. This framework was used to arrive at the current recommendations for the risk-informed alternative to GDC 35. The framework reflects experience from its use in risk-informing 10 CFR 50.44 [Ref. 3] and 10 CFR 50.46, and comments received from stakeholders. The structure and elements of the framework are consistent with established regulatory philosophy and have as a high level goal the protection of the public health and safety. A balanced high-level defense-in-depth approach (based on prevention and mitigation) is included in the framework to help achieve this goal.

1.5 Organization and Content

Section 2 of this report documents a detailed examination of the ECCS reliability requirements contained in GDC 35 of Appendix A to 10 CFR Part 50. The technical requirements imposed by GDC 35 are identified and described, and any relationship of these requirements to other regulations and implementing documents is identified. This information is needed because changes to the ECCS reliability requirements in GDC 35 could potentially impact some of the related regulations or implementing documents. Section 2 also documents the risk significance of the ECCS, and describes the proposed risk-informed alternative to GDC 35. Two approaches for accomplishing the alternative are identified, a plant-specific approach and a generic approach. Details of the plant-specific and generic approaches are provided in Sections 3 and 4, respectively. Additional technical work to support the generic approach is currently ongoing. Section 4 provides interim results which can be used to help evaluate the practicality of further pursuing the generic approach. A number of significant technical and policy issues have been identified during the performance of the technical work for this project, and these are discussed in Section 5.

2. RISK-INFORMED ECCS RELIABILITY REQUIREMENTS

2.1 Examination of ECCS Reliability Requirements

Paragraph (d) of 10 CFR 50.46 states:

> The requirements of this section are in addition to any other requirements applicable to ECCS set forth in this part. The criteria set forth in paragraph (b), with cooling performance calculated in accordance with an acceptable evaluation model, are in implementation of the general requirements with respect to ECCS cooling performance design set forth in this part, including in particular Criterion 35 of appendix A.

GDC 35 (Emergency core cooling) of Appendix A to 10 CFR Part 50 states:

> A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

The assumptions specified in GDC 35 regarding offsite power unavailability and a single additional failure are essentially deterministic surrogates for establishing a minimum ECCS reliability. The loss of offsite power assumption requires the ECCS to be designed to be able to perform its safety function while being powered solely by onsite emergency power, and the single failure assumption requires that the system be designed to have fully redundant capability.

2.2 Risk Significance of ECCS

All reactors operating in the United States have multiple trains of ECCS capable of mitigating the full spectrum of LOCAs. 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 required by GDC 35. As a result, PRAs have generally shown that the risk from large-break LOCAs is generally small.

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 boiling water reactor (BWR) or pressurized water reactor (PWR), the ECCS can be used 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 Individual Plant Examination (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 a reactor coolant pump (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).

Station blackout (SBO) 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.

These insights suggest that risk-information as generated in PRAs can be used to determine if the ECCS reliability is commensurate with the frequency of all challenges, including LOCAs and transients. More details on the risk significance of the ECCS and LOCAs in general are provided in Attachment 1 to SECY-01-0133.

2.3 Risk-Informed Alternative to GDC 35

The risk-informed alternative to GDC 35 would ensure an ECCS safety function reliability that is commensurate with the frequency of challenge to the ECCS safety function. This revision would permit use of more risk-informed and realistic approaches for demonstrating ECCS safety function reliability. An example of suggested wording for the risk-informed alternative would be to modify GDC 35 by adding an optional requirement at the end of the second paragraph as shown below:

> **Criterion 35** -- Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure***, OR functionality shall be demonstrated by assuring an ECCS reliability commensurate with the frequency of the challenge to the ECCS such that the risk to the public health and safety is not significant***.

Conformance with the risk-informed alternative requirement highlighted above requires that it be demonstrated that the ECCS functional reliability is commensurate with the frequency of accidents in which ECCS success would prevent core damage, or large early release. For the risk-informed alternative to GDC 35, in place of the assumptions that offsite power is not available and there is

a single additional failure, it is envisioned that two options would be offered in a regulatory guide to ensure ECCS safety function reliability:

1. A plant-specific approach where licensees, with appropriate consideration of uncertainties, demonstrate compliance with NRC-established acceptance guidelines, OR

2. A generic approach where a minimal set of ECCS equipment required to meet the NRC-established acceptance guidelines would be specified by the NRC, by generic plant group.

The plant-specific and generic approaches are discussed in Sections 3 and 4, respectively. As stated in SECY-02-0057, technical work is required to determine appropriate acceptance guidelines (threshold values) to be used with ECCS risk/reliability analyses in support of risk-informed ECCS-reliability applications. The RES technical work in this area is described in Section 3.1. SECY-02-0057 also identified the need for technical analysis pertaining to LOCA frequencies and the conditional probability of a LOOP given a LOCA. The RES technical work pertaining to LOCA frequencies is described in Sections 3.2 and 4.2 for the plant-specific and generic approaches, respectively. The RES technical work pertaining to the conditional probability of a LOOP given a LOCA is described in Sections 3.3 and 4.3 for the plant-specific and generic approaches, respectively.

As noted in SECY-02-0057, a fourth principal area where technical analysis is required to support possible rulemaking for the risk-informed alternative to GDC 35 is the formulating of generic plant groups, based on ECCS and support system configuration, to support the generic approach. The technical work associated with formulating generic plant groups is discussed in Section 4.4. Section 4.5 provides some insights from the initial generic calculations, and Section 4.6 identifies future work that needs to be accomplished to support the generic approach.

3. PLANT-SPECIFIC APPROACH

The plant-specific approach for the risk-informed alternative to GDC 35 would require licensees to demonstrate, with appropriate consideration of uncertainties, that the ECCS safety function reliability is commensurate with the frequency of the challenge to the system, i.e., is in compliance with acceptance guidelines established by the NRC staff for both core damage frequency (CDF) and large early release frequency (LERF). The licensee would be required to meet the CDF and LERF acceptance guidelines using plant-specific PRA models and data, and with appropriate consideration of uncertainties. The use of a plant-specific reliability design requirement in GDC 35 improves the coherence between the design requirements and the operational requirements (e.g., recent work on technical specifications and the Maintenance Rule).

Consistent with the framework for risk-informing 10 CFR Part 50, the quantitative guidelines are only one part of a risk-informed defense-in-depth approach for accomplishing the goal of protecting the public health and safety. This defense-in-depth approach includes both elements that are dependent upon risk insights (e.g., limiting CDF and LERF), and elements that are employed independent of risk insights (e.g., providing reasonable balance among prevention, containment and consequence mitigation and avoiding over-reliance on programmatic activities to compensate for weaknesses in plant design).

As discussed in Section 2.3, technical work is necessary in three principal areas to support possible rulemaking to risk-inform GDC 35 (acceptance guidelines, LOCA frequencies, and conditional LOOP probability). The CDF and LERF acceptance guidelines established by the NRC would apply equally to both the plant-specific and generic approaches. The LOCA frequency distributions obtained as a result of this work would be made available for use by licensees for demonstrating compliance with the acceptance guidelines. Guidance also would need to be provided to licensees with respect to calculation of a plant-specific conditional probability of LOOP given a LOCA or other reactor trip, since PRAs typically treat these consequential LOOPs as random, independent events. The results of the technical work associated with each of these areas are described in the following sections.

3.1 ECCS Risk/Reliability Analyses (Acceptance Guidelines)

It is envisioned that two types of ECCS-related changes may be proposed by licensees: changes in ECCS design or operation (e.g., removal of a piece of equipment or relaxation of technical specifications), or changes in the ECCS design basis (e.g., removal of an accident from the ECCS design basis analyses). The establishment and use of acceptance guidelines for each type of potential change is discussed separately below.

3.1.1 ECCS-Related Design and Operational Changes

Conformance with the alternate risk-informed GDC 35 would require that it be demonstrated that the ECCS functional reliability is commensurate with the frequency of accidents in which ECCS success would prevent core damage or a large early release. For proposed ECCS-related design and operational changes, this can be accomplished by demonstrating that the following acceptance guidelines are met:

1. The baseline total plant CDF and LERF (i.e., considering all initiating events and accident sequences) meet the quantitative guidelines specified in the framework for risk-informing 10 CFR Part 50 [Ref. 3], ***AND***
2. The resulting change in risk from a proposed ECCS-related design or operational change does not represent a significant risk increase.

The framework for risk-informing 10 CFR Part 50 specifies a CDF threshold of 10^{-4}/yr and a LERF threshold of 10^{-5}/yr. Since the framework threshold values apply to a full-scope PRA, the total plant CDF and LERF need to be determined (i.e., not just the CDF and LERF associated with LOCAs or ECCS failures). Appendix C to this report, which is a copy of an appendix to be included with a future revision of the framework document, demonstrates how these values are consistent with the quantitative health objectives (QHOs) specified in the Commission's Safety Goal Policy Statement. In demonstrating that the quantitative guidelines are met, it is necessary to address uncertainties, as discussed later in this section and in the framework document.

The framework for risk-informing 10 CFR Part 50 only specifies absolute risk guidelines, as opposed to limits on changes to risk. However, the CDF and LERF acceptance criteria in Section 2.2.4 of RG 1.174 can be used to demonstrate that the resulting change in risk from a proposed ECCS-related design or operational change does not represent a significant risk increase. In addition, consistent with RG 1.174, proposed ECCS-related design or operational changes may be allowable even for plants that fail to meet the framework quantitative guidelines if the change in risk can be demonstrated to be small.

With the plant-specific approach, licensees may propose changes to ECCS design or operation. As an example, if the generic approach described in Section 4 does not prove to be practical, some BWR 3 or 4 plants may propose to eliminate or reduce the redundancy of the core spray system, if they can demonstrate on a plant-specific basis that the proposed change meets the above acceptance guidelines.

As noted previously, the quantitative guidelines are only one part of the risk-informed defense-in-depth approach described in the framework for risk-informing 10 CFR Part 50. This approach maintains the philosophy of defense-in-depth at two levels. At a high level, the cornerstones of safe nuclear power plant operation (in particular, the four reactor safety cornerstones) provide the basis for the strategies that are applied in risk-informing existing regulations. The four reactor safety cornerstones and corresponding strategies constitute an application of the defense-in-depth philosophy that seeks not only to prevent core damage but also to mitigate radionuclide releases should core damage occur. At a low level, defense-in-depth is also applied (e.g., through multiple trains or systems to accomplish a given safety function). In formulating and implementing a risk-informed alternative to an existing regulatory requirement, important elements of defense-in-depth are preserved regardless of the risk information.

3.1.2 ECCS-Related Design Basis Changes

For applications that would seek to change the design basis for the ECCS (e.g., to remove an accident from the ECCS design basis analyses), the resulting change in CDF and LERF must also meet the criteria from the framework for risk-informing 10 CFR Part 50 and Regulatory Guide 1.174 as described in Section 3.1.1. The resulting change in CDF and LERF are determined by assuming that the plant can no longer respond to this particular accident (i.e., the subject accident

is assumed to lead directly to core damage). As an example, a large LOCA with a coincident LOOP could be removed from the design basis if the frequency of that combination of events from all possible contributors (random pipe breaks from all known mechanisms, seismic events, heavy load drops, etc.) were to be assumed to lead directly to core damage, but still meet the framework and RG 1.174 criteria. This could allow, for example, an increase in the diesel generator start time. The full extent of plant changes to be allowed, with respect to both design basis changes and design or operational changes, would need to be established by the staff.

3.1.3 PRA Scope and Uncertainty Analysis

As mentioned previously, the acceptance guidelines are intended for comparison with the results from a full scope PRA, consistent with RG 1.174. However, it is recognized that many PRAs are not full scope and thus the risk metrics associated with a proposed ECCS-related change may be incomplete. When the PRA is not full scope, it is necessary to address the significance of the out-of-scope items. For ECCS-related changes, this is difficult due to the fact that the ECCS is required for mitigating potential accidents during different modes of operation. The importance of assessing the contribution of the out-of-scope portions of the PRA to the base case estimates of CDF and LERF is related to the margin between the as-calculated values and the acceptance guidelines. When the contributions from the modeled contributors are close to the guidelines, the argument that the contribution from the missing items is not significant must be convincing, and will likely require additional PRA analyses. When the margin is significant, a qualitative argument may be sufficient. The contribution of the out-of-scope portions of the model to the change in metric may be addressed by bounding analyses, detailed analyses, or by a demonstration that the change has no impact on the unmodeled contributors to risk. In addition, it should also be demonstrated that changes based on a partial PRA do not disproportionally change the risk associated with those accident sequences that arise from the modes of operation not included in the PRA.

Because of the way the acceptance guidelines were developed, the appropriate numerical measures to use in the initial comparison of the PRA results to the acceptance guidelines are mean values, consistent with RG 1.174. The mean values referred to are the means of the probability distributions that result from the propagation of the uncertainties on the input parameters and those model uncertainties explicitly represented in the model. A formal propagation of the uncertainty is the best way to correctly account for state-of-knowledge uncertainties that arise from the use of the same parameter values for several basic event probability models.

While the analysis of parametric uncertainty in PRAs is fairly mature, the analysis of the model and completeness uncertainties cannot be handled in such a formal manner. Whether the PRA is full scope or only partial scope, it will be necessary to demonstrate that the choice of models or methods do not significantly change the assessment of the ECCS-related change. This demonstration can take the form of well formulated sensitivity studies. Model uncertainties that would drive the result toward unacceptableness should be identified and sensitivity studies performed or reasons given as to why they are not appropriate for the current ECCS application or for the particular plant. In general, the results of the sensitivity studies should confirm that the guidelines are still met even under the alternative model or assumptions.

3.2 LOCA Frequencies

Loss-of-coolant accidents are defined in 10 CFR 50.46(c) 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"* . 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."* However, the definition of LOCA in 10 CFR Part 50, Appendix A, "Definitions and Explanations" includes all "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."

Since, as discussed previously, the ECCS is used to mitigate all LOCAs as well as many transients, and transient scenarios involving the failure of the ECCS results in a significant fraction of the total risk at most nuclear power plants, the "challenges" to the system referred to in the risk-informed alternative to GDC 35 need to include all losses of reactor coolant, consistent with the LOCA definition provided in 10 CFR Part 50, Appendix A.

Any LOCA frequency estimates that would be used to justify an ECCS-related change under the proposed risk-informed alternative to GDC 35 should include not only pipe breaks but also other failures in the reactor coolant system (RCS) that would require similar ECCS response, such as:

- ruptures in other RCS components such as valves, pumps, penetrations (e.g., control rod drive housings), the pressurizer (PWRs), and the reactor vessel
- failures in reactor head, reactor coolant pump and other seals
- inadvertent valve openings, including power-operated relief valves (PORVs) and head vent valves in PWRs and safety relief valves (SRVs) in BWRs
- bolt failures leading to failures of the RCS boundary at locations such as steam generator manways or valve bonnets
- indirect causes of LOCAs such as from heavy load drops

Regulatory Guide 1.176 provides guidance on assessing the potential for pipe failures in in-service inspection (ISI) applications. Three methods for assessing pipe failure probabilities are identified: data analysis, fracture mechanics computer codes, and expert elicitation. These three methods are also applicable for determining LOCA frequencies (both pipe breaks and other RCS failures), but for applications involving a change to the ECCS design bases, more rigor in these methods is required. Details are provided below.

Large and intermediate LOCAs are rare events and as such, data estimates based on actual occurrences do not exist. The ASME PRA Standard guidance for such events is to use generic estimates and to account for plant-specific features. Methods, both classical and Bayesian, for calculating initiating event frequencies are provided in the Draft PRA Data Analysis Handbook [Ref. 4]. Some existing LOCA frequency estimates and methods to generate frequencies from historical data are discussed below.

Many PRAs use RCS pipe break frequencies that can be traced back to the Reactor Safety Study (RSS) [Ref. 5] as the frequencies of LOCAs. The RSS pipe break frequencies were based on pipe break data obtained from numerous sources that included both nuclear and non-nuclear facilities and U.S. and foreign information. Frequencies for small, medium, and large RCS pipe breaks were estimated and the same frequencies were used for both PWRs and BWRs. The median RSS frequencies are 10^{-4} per reactor calendar year for large breaks, 10^{-4} per reactor calendar year for medium breaks, and 10^{-3} per reactor calender year for small breaks. A lognormal distribution with an error factor of 10 was used to characterize the uncertainty in these estimates. The distribution means are 3×10^{-4} per reactor calendar year for large and intermediate pipe breaks and 3×10^{-3} per reactor calendar year for small pipe breaks.

A recent effort to generate new estimates of RCS pipe break frequencies is documented in NUREG/CR-5750 [Ref. 6]. In NUREG/CR-5750, 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. 7]:

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

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.

For large and intermediate pipe breaks, the NUREG/CR-5750 frequency percentiles are one to two orders of magnitude less than the corresponding RSS percentiles. NUREG/CR-5750 estimates have been adopted in a number of PRAs and in most of the NRC Standardized Plant Analysis Risk (SPAR) models. However, following the primary water stress corrosion cracking incident at the V.C. Summer plant, concerns over the NUREG/CR-5750 estimates have been expressed. The thrust of these concerns is to question whether the methodology utilized in NUREG/CR-5750 adequately accounts for known and potential unknown failure mechanisms. Efforts to resolve these concerns and update the NUREG/CR-5750 estimates by means of an expert elicitation process are scheduled to be completed in the time frame projected for the possible rulemaking. This expert elicitation process will consist of a panel solicited from largely non-NRC staff from industry, academia, contracting agencies, other government agencies, and international agencies. The panel members will represent the full range of technical expertise necessary to develop comprehensive LOCA frequencies. The process will be facilitated and elicited by NRC staff. It is

planned that the most relevant PRA studies documenting pipe break failures and failure rates of other LOCA-initiating sources will be utilized to form baseline distributions for elicitation. The elicitation will evaluate expected changes in these baseline distributions up to the end of the license renewal period. The elicitation results will be contrasted with probabilistic fracture mechanics (PFM) studies using standardized inputs also developed as part of the elicitation.

In the interim, a quick, informal, in-house expert elicitation to obtain “place-holder” LOCA frequencies for use in performing calculations to demonstrate the feasibility of the generic approach to a risk-informed GDC 35 has been implemented, as described in Section 4.2 (and Appendix F).

3.3 Conditional Probability of Consequential LOOP

In typical PRAs, the occurrence of a LOOP following a LOCA (or any other reactor trip) is assumed to be a random, independent event. However, more recent analysis (NUREG/CR-6538 [Ref. 8]) concludes that these events are not completely independent and that the conditional probability of a LOOP following a LOCA is greater than the random probability of a LOOP. Accordingly, there is a need to estimate the conditional LOOP probability following a LOCA.

NUREG/CR-6538 also demonstrated that the conditional probability of a LOOP following any reactor trip is greater than the random probability of a LOOP (though not as great as the conditional probability of a LOOP following a LOCA). For all potential ECCS-reliability applications, there would be a need to estimate the conditional LOOP probability following a reactor trip, in order to demonstrate compliance with the risk-informed alternative requirement proposed for GDC 35.

NUREG/CR-6538 provides the results of an analysis of the data for consequential LOOP following a reactor trip and following a LOCA, covering the period 1984-1993. The initial focus of the technical work documented here was to update the analyses in NUREG/CR-6538 to include data since 1993. During this effort, it was determined that recategorization of some of the data provided in NUREG/CR-6538 was necessary. In addition, while NUREG/CR-6538 analyzed data separately for BWRs and PWRs, a decision was made for this effort to treat all U.S. nuclear power plant data jointly, since no rationale could be identified for differentiating conditional LOOP probability based on BWR and PWR design differences.

The results of the update and revision to the consequential LOOP data from NUREG/CR-6538 are summarized in Section 4.3, since these data were used to obtain conditional LOOP probabilities that were used in the generic approach to risk-informing GDC 35. The generic probability of consequential LOOP following a reactor trip, as provided in Section 4.3, may be used for the plant-specific analysis. However, for consequential LOOP following a LOCA there are no data, and even for a consequential LOOP following a major ECCS actuation (which is used in the generic approach as a surrogate for a LOCA) data are extremely limited. Therefore, the results derived from the generic approach are not necessarily applicable to all plants (conditions and limitations are identified in Section 4.3). For plants for which the generic probability of consequential LOOP following a LOCA do not apply, or for plants which believe their specific probability of consequential LOOP following a LOCA is significantly lower than the generic value presented in Section 4.3, other methods for calculating a plant-specific conditional LOOP probability following a LOCA may be more appropriate.

Appendix D to this report presents one method for assessing plant-specific probability of a consequential LOOP due to a LOCA. A consequential LOOP can be caused by transient and/or plant-centered factors. In the study documented in Appendix D, the two transient factors are the electrical disturbance triggered by the LOCA and the conditions of the offsite transmission-system grid. Plant-centered factors are the failures of plant electrical equipment that contribute to a consequential LOOP (e.g., bus transfer schemes, undervoltage protection schemes, transformers and circuit breakers). Strictly speaking, the electrical transient triggered by a LOCA also is "plant-centered," but for convenience it was classified in Appendix D as a "transient factor."

The method for assessing the plant-specific probability of a consequential LOOP due to transient factors involves the following steps:

1. Gather historical data on the voltage conditions at the plant's switchyard.

2. Perform voltage analyses of the plant's electrical system.

3. Assess the average time per year that the plant is vulnerable to a LOOP condition.

4. Estimate the plant-specific probability of a LOOP due to transient factors.

5. Estimate the predicted performance of the offsite grid and conduct sensitivity studies.

The method for assessing the plant-specific probability of a consequential LOOP due to transient factors and the individual steps above are described in detail in Appendix D.

One method for assessing the plant-specific probability of a consequential LOOP due to plant-centered factors is to develop a PRA model (fault tree) of relevant components. The "top event" of the fault tree is LOOP occurs due to plant-centered factors after a LOCA. The fault tree must contain all the relevant electrical components involved in powering the Class 1E buses, including the transfer of power sources, if applicable. Some components may be grouped and replaced by modules for which probabilistic data are available, as long as the dependencies between components or modules are preserved. The fault tree also must contain the important contributors to risk, such as human errors and common cause failures.

To assess the overall plant-specific probability of a consequential LOOP, the plant-specific probabilities due to transient and plant-specific factors must be combined. Because a LOCA can cause an electrical disturbance in the offsite and onsite electrical systems, the probability of a consequential LOOP due to transient factors is not independent of the probability of consequential LOOP due to plant-centered factors. However, since either of the two factors (transient and plant-centered) can cause a consequential LOOP, an approximation of the plant-specific probability of a consequential LOOP given a LOCA can be obtained by adding the probability of LOOP due to transient factors and the probability of a LOOP due to plant-centered factors (as long as the individual probabilities are each less than 0.1).

Since the conditions of the offsite grid and the plant's electrical systems evolve over time, it is recommended that the evaluation of the plant-specific probability of a consequential LOOP be revised periodically. The objective of the revisions is to confirm that changes in the offsite grid and in the plant's electrical systems do not adversely affect this probability. The revisions would require

updating the evaluations discussed in Appendix D, which in turn would require gathering or estimating data reflecting the performance of the offsite grid, given that a reactor trip has occurred.

4. GENERIC APPROACH

4.1 Acceptance Guidelines

Section 3.1 provides recommended acceptance guidelines for the plant-specific approach for demonstrating that the ECCS functional reliability is commensurate with the frequency of accidents in which ECCS success would prevent core damage or a large early release. For the generic approach the same acceptance guidelines are considered to be applicable. That is, demonstrating that the ECCS functional reliability is commensurate with the frequency of the challenges to the system can be accomplished by demonstrating that the following acceptance guidelines are met:

1. The baseline total plant CDF and LERF (i.e., considering all initiating events and accident sequences) meet the quantitative guidelines specified in the framework for risk-informing 10 CFR Part 50 [Ref. 3], ***AND***
2. The resulting change in risk from a proposed ECCS-related design or operational change does not represent a significant risk increase.

As discussed in Section 3.1, consistent with the framework for risk-informing 10 CFR Part 50 [Ref. 3], it is recommended that (1) the total plant (i.e., full-scope PRA) CDF and LERF threshold values be established at 10^{-4}/yr and 10^{-5}/yr, respectively, and (2) the CDF and LERF acceptance criteria in RG 1.174 be used to limit any risk increase. For the generic approach, the acceptance guidelines would need to be met by all of the plants within a particular generic plant group.

As noted previously, the quantitative guidelines are only one part of the risk-informed defense-in-depth approach described in the framework for risk-informing 10 CFR Part 50. This approach maintains the philosophy of defense-in-depth at two levels. At a high level, the cornerstones of safe nuclear power plant operation (in particular, the four reactor safety cornerstones) provide the basis for the strategies that are applied in risk-informing existing regulations. The four reactor safety cornerstones and corresponding strategies constitute an application of the defense-in-depth philosophy that seeks not only to prevent core damage but also to mitigate radionuclide releases should core damage occur. At a low level, defense-in-depth is also applied (e.g., through multiple trains or systems to accomplish a given safety function). In formulating and implementing a risk-informed alternative to an existing regulatory requirement, important elements of defense-in-depth are preserved regardless of the risk information.

Consistent with the guidance for the plant-specific approach provided in Section 3.1, if an attempt is made to generically change the design basis for the ECCS for a group of plants (e.g., to remove an accident from the ECCS design basis analyses), the resulting change in CDF and LERF must also meet the criteria specified above from the framework for risk-informing 10 CFR Part 50 and Regulatory Guide 1.174. The resulting change in CDF and LERF are determined by assuming that the plants in the group can no longer respond to this particular accident (i.e., the subject accident is assumed to lead directly to core damage). As an example, a large LOCA with a coincident LOOP could be removed from the design basis if the frequency of that combination of events from all possible contributors (random pipe breaks from all known mechanisms, seismic events, heavy load drops, etc.) were to be assumed to lead directly to core damage, but still meet the framework and RG 1.174 criteria. This could allow, for example, an increase in the diesel generator start time. The full extent of plant changes to be allowed for a particular plant group, with respect to both

design basis changes and design or operational changes, would need to be established by the staff.

As discussed in Section 3.1.3, when a PRA is not full scope, it is necessary to address the significance of the out-of-scope items. For ECCS-related changes, this is difficult due to the fact that the ECCS is required for mitigating potential accidents during different modes of operation. The importance of assessing the contribution of the out-of-scope portions of the PRA to the base case estimates of CDF and LERF is related to the margin between the as-calculated values and the acceptance guidelines. When the contributions from the modeled contributors are close to the guidelines, the argument that the contribution from the missing items is not significant must be convincing, and will likely require additional PRA analyses. When the margin is significant, a qualitative argument may be sufficient. The contribution of the out-of-scope portions of the model to the change in metric may be addressed by bounding analyses, detailed analyses, or by a demonstration that the change has no impact on the unmodeled contributors to risk. In addition, it should also be demonstrated that changes based on a partial PRA do not disproportionally change the risk associated with those accident sequences that arise from the modes of operation not included in the PRA.

The risk contribution from internal fire and flood and external events is dependent not only on the system configurations but also the plant layout and location. Thus, the risk from these events is very plant-specific and it would be very difficult to generically establish the ECCS importance to these events. This implies that for the generic option, the minimum ECCS requirements can only be quantitatively established based on a PRA for internal events. This fact raises a policy issue on how to account for external events in establishing generic ECCS requirements (see Section 5).

Another issue for the generic option is the unavailability of low power/shutdown (LPSD) PRA models. Since the conditional risk during cold shutdown can be similar to that at power and the dominant sequences during LPSD include LOCAs and drain down events, the ECCS is important during these modes. However, the absolute risk from these modes of operation is generally small due primarily to the relatively short time the plant is in these modes. Generation of LPSD models would be required to quantitatively evaluate the minimum ECCS requirements for these modes (generic SPAR templates exist for creating these models, and an effort to use these templates to produce LPSD models for the various PWR and BWR classes is now underway). Alternatively, stricter configuration control requirements could be imposed to manage the risk during these modes. These requirements could take the form of technical specifications or risk limits for shutdown configurations.

Because of the way the acceptance guidelines were developed, the appropriate numerical measures to use in the initial comparison of the PRA results to the acceptance guidelines are mean values. The mean values referred to are the means of the probability distributions that result from the propagation of the uncertainties on the input parameters and those model uncertainties explicitly represented in the model. In the generic approach, parameter uncertainty was propagated through the models, and sensitivity studies were performed to test the robustness of the ECCS evaluations to uncertainties in the modeling.[(1)] The model uncertainties evaluated

[1]At the time of this report, propagation of parameter uncertainty and sensitivity studies have only been completed for the base case.

include credit for alternative coolant injection systems and the LPCI success criteria for large-break LOCAs. Treatment of uncertainty is discussed further in Section 5.5.

4.2 LOCA Frequencies

A near-term staff elicitation, or expert judgment process, was utilized to determine interim updated LOCA frequency distributions for use in demonstrating the feasibility of risk-informing GDC 35. The elicitation panel consisted of eleven staff from RES and NRR. Participants were chosen to provide expertise in the following relevant technical areas: probabilistic risk assessment; the ASME code; structural mechanics; thermo-hydraulics; piping systems; seismic, thermal and vibrational loading; environmentally assisted cracking; thermal aging; and alternative LOCA mechanisms.

The elicitation was structured in the following manner. A kick-off meeting was held to define the objective, distribute background documentation, discuss the underlying LOCA frequencies from Appendix J of NUREG/CR-5750 [Ref. 6], and identify key technical issues to consider in updating these frequencies. This kick-off meeting was followed with an issue development meeting to determine the issues that would be considered for elicitation, the decomposition of the technical issues which influence the initiating event frequencies, and the definition of the baseline frequencies. All elicitation questions were designed to solicit changes with respect to the baseline case.

The staff panel determined that piping systems should be grouped for BWR and PWR reactors in terms of functionality, size, and material. Then, the effect of each piping system on the LOCA frequencies should be evaluated. The effect of non-piping failures on the LOCA frequencies were considered separately for each component that the group deemed important. The group also determined during this meeting that the NUREG/CR-5750, Appendix J LOCA frequencies would serve as the baseline and that updates should be with respect to these values.

An elicitation questionnaire was developed based on the results of this meeting (see Appendix F). The questionnaire was submitted to each panel member and the results were supplied to RES for analysis. Whenever possible, separate LOCA frequencies were determined for each panel member. Details on the procedure utilized to calculate the individual results are provided in Appendix F. Other questionnaire responses were utilized to perform sensitivity analyses and cross-check LOCA frequency calculations to assess the consistency of the quantified results. The rationale supplied by each panel member supporting their quantitative estimates was also compiled and will be utilized in the forthcoming formal expert elicitation.

The median updated LOCA frequencies developed from the individual panel results serves as the mean of the updated distribution. Results are provided for each LOCA size and plant type in Table 4.1. Also shown in Table 4.1 is the ratio, or adjustment factor, between each updated and original NUREG/CR-5750, Appendix J result. Upper and lower bounds were developed by assuming a lognormal distribution with error factors of 3 for the small-break (SB) LOCA frequencies and 10 for the medium-break (MB) and large-break (LB) LOCA frequencies. This approach is consistent with the NUREG/CR-5750, Appendix J methodology. Results are presented in Table 4.1 to two significant digits to allow direct comparison with NUREG/CR-5750, Appendix J results. More details on the entire elicitation process are available in Appendix F.

It should be emphasized that the frequency distributions in Table 4.1 represent best estimate values which are applicable through the end of the license renewal period. These values have been quantified using individual panel responses to the elicitation questionnaire. Future estimates of the LOCA frequencies will be a function of the make-up of the expert elicitation panel, the baseline frequency assumptions, the elicitation questions, the current time-frame of the elicitation, and any subsequent information and events which occur in the interim.

RES is now embarking on the formal elicitation which will provide final updated LOCA frequency distributions. The expert panel will consist of participants from industry, academia, NRC and other government agencies, national laboratories, and international agencies. This effort is expected to provide results within 8 to 12 months after initiation.

Table 4.1 Updated Mean LOCA Frequencies and Bounds

Reactor Type	LOCA Size	Lower 5% Bound	Mean	Upper 95% Bound	NUREG/CR-5750 Adjustment Factor
BWR	SB	4.0×10^{-4}	1.5×10^{-3}	3.6×10^{-3}	3.7
	MB	3.4×10^{-6}	9.1×10^{-5}	3.4×10^{-4}	3.5
	LB	2.0×10^{-6}	5.2×10^{-5}	2.0×10^{-4}	2.2
PWR	SB	4.0×10^{-4}	1.5×10^{-3}	4.0×10^{-3}	3.7
	MB	2.3×10^{-6}	6.1×10^{-5}	2.3×10^{-4}	2.0
	LB	2.7×10^{-7}	7.2×10^{-6}	2.7×10^{-5}	2.0

4.3 Conditional Probability of Consequential LOOP

As discussed in Section 3.3, NUREG/CR-6538 [Ref. 8] demonstrated that the conditional probability of a LOOP following a reactor trip is greater than the random probability of a LOOP, and the conditional probability of LOOP following a LOCA is even greater than that following a non-LOCA reactor trip. NUREG/CR-6538 provides the results of an analysis of the data for consequential LOOP following a reactor trip and following a LOCA, covering the period 1984-1993. As part of the technical work documented here, the analyses in NUREG/CR-6538 were updated to include data from 1994-2001. During this effort, it was determined that recategorization of some of the data provided in NUREG/CR-6538 was necessary. In addition, while NUREG/CR-6538 analyzed data separately for BWRs and PWRs, a decision was made for this effort to treat all U.S. nuclear power plant data jointly, since no rationale could be identified for differentiating conditional LOOP probability based on BWR and PWR design differences.

A data-driven approach was initially used to assess the generic probability of a LOOP given any reactor trip, and the generic probability of a LOOP given a LOCA. Generic means an assessment

that would apply to the entire population of nuclear plants in the United States, or to a large group of them.

An approach consisting of examining the operational experience for the entire population of nuclear plants was used to obtain the generic probability of consequential LOOP. The source of the data was the LERs contained in the Sequence Coding and Search System (SCSS). Since no operational events of a LOCA with a consequential LOOP have occurred, it was necessary to identify operational events that match, as closely as possible, the occurrence of a LOCA. For this identification, two types of events that cause an impact similar to a LOCA were considered: reactor trips and major actuations of ECCS loads. The first surrogate can cause a consequential LOOP because the loss of the main generator disturbs the offsite power grid, and plant-centered failures of electrical equipment may occur after the trip of the main generator. The second surrogate, major actuation of the ECCS loads, causes a reactor trip and, in addition, the loading of the ECCS loads to the safety buses; the combined effect of these two events could cause an undervoltage at these buses, ultimately causing a LOOP.

The second surrogate, major actuation of the ECCS loads, is expected to more closely resemble a LOCA because it encompasses both the reactor trip and the loading of the ECCS loads to the safety buses. Accordingly, the generic probability of consequential LOOP following a LOCA was assessed using this surrogate.

The following subsections (4.3.1 and 4.3.2) summarize the derivation of generic probabilities of consequential LOOP after a reactor trip and after a major ECCS actuation, respectively. The details regarding the derivation of these probabilities are provided in Appendix G. It should be noted that the consequential LOOPs identified in the operational experience are due to “internal” causes, such as the electrical disturbance triggered by the generator trip. Therefore, in the following evaluation of the generic probability of consequential LOOP given a LOCA (or reactor trip), the estimate of this probability represents the probability due to “internal” events. An analysis of the impact of “external” events on the probability of consequential LOOP was not within the scope of this work. The term “external” events is used here to have the meaning in the typical PRA jargon, that is, those events such as seismic and fire events.

4.3.1 Generic Probability of Consequential LOOP After a Reactor Trip

Using the SCSS, the LERs from January 1, 1984 to October 31, 2001 were searched to count the number of reactor trips. The number of consequential LOOPs after a reactor trip that occurred from January 1, 1984 to December 31, 1992 was obtained from NUREG/CR-6538, and the number that occurred from January 1, 1993 to October 31, 2001 was obtained through a search and review of the LERs from the SCSS. Using this process, the total number of reactor trips was determined to be 3415, and the total number of consequential LOOPs after a reactor trip was determined to be 8.

The point estimate and uncertainty due to variability in the data were assessed using a binomial distribution. This distribution is used because the data, the number of failures in a given number of demands, and the consideration that the probability is constant across these demands, correspond to this distribution. Bayes’ theorem was then applied using a binomial model with a non-informative prior (i.e., a uniform prior), to obtain a posterior beta distribution. This yielded the

following parameters for the conditional probability of a LOOP following a reactor trip (details are provided in Appendix G):

5th percentile:	1×10^{-3}
mean:	$\mathbf{2\times10^{-3}}$
95th percentile:	4×10^{-3}

4.3.2 Generic Probability of Consequential LOOP After a Major ECCS Actuation

Using the SCSS, a search and review of the LERs from January 1, 1986 to November 30, 2001 was conducted in order to identify major ECCS actuations and LOOPs after such actuations. This review resulted in the identification of 14 major ECCS actuations and 1 consequential LOOP after a major ECCS actuation. Using the same statistical process as described in the previous section, the following parameters for the conditional probability of a LOOP following a major ECCS actuation were obtained:

5th percentile:	4×10^{-3}
mean:	$\mathbf{7\times10^{-2}}$
95th percentile:	3×10^{-1}

The number of demands used to assess the point estimate of the generic probability of LOOP given a major ECCS actuation (7×10^{-2}) is 14. This is a relatively small number of demands and, hence, the resulting point estimate may be inaccurate. In fact, the interval between the 95% and 5% confidence limits (3×10^{-1} and 4×10^{-3}) spans almost two orders of magnitude. To attempt to obtain an estimate of this probability that is more accurate, an approach that uses a model, insights gained during the data-driven evaluation, and engineering judgment was used, as described in Appendix G. An overview of this hybrid approach, as well as the results obtained, are provided below.

As discussed in Section 3.3, a consequential LOOP can be caused by transient and/or plant-centered factors. For the purpose of this study, the two transient factors are the electrical disturbance triggered by the LOCA and the conditions of the offsite transmission-system grid. Plant-centered factors are the failures of plant electrical equipment that contribute to a consequential LOOP (e.g., bus transfer schemes, undervoltage protection schemes, transformers and circuit breakers). Strictly speaking, the electrical transient triggered by a LOCA also is “plant-centered,” but for convenience it is classified in this study as a “transient factor.”

Generic Probability of Consequential LOOP due to Plant-Centered Factors

To assess the generic probability of a consequential LOOP due to plant-centered factors a model (fault tree) of typical designs of electrical equipment in nuclear power plants was developed and evaluated. A discussion of the hardware failures and human errors included in the model, as well as the fault tree and the input data, are provided in Appendix G.

The power sources for the safety buses during normal plant operation can be classified in two categories: 1) the safety buses are supplied by the main generator through a unit transformer and, 2) the safety buses are supplied by a preferred offsite power source. For plants in the first category, after the main generator has tripped, the power supply to the safety buses has to be

transferred to an offsite power source. Accordingly, “flag” or “conditional” events were included in the fault tree to allow it to be quantified both with and without consideration of the failure modes associated with the automatic transfer from the unit auxiliary transformer to the offsite power source.

The computer code SAPHIRE Version 7.17 was used to implement and evaluate the fault tree model. The probabilistic model yields the results shown in Table 4.2. The uncertainty calculations were conducted using the Latin Hypercube Sampling method with 10,000 samples.

Table 4.2 Generic Conditional Probability of LOOP Due to Plant-Centered Factors

Power supply during normal operation	Probability		
	5%	Mean	95%
Main generator through an auxiliary transformer	4.4E-3	1.2E-2	2.8E-2
Offsite power through a start-up transformer	2.3E-3	9.3E-3	2.5E-2

The mean probability of LOOP due to plant-centered factors for the design that is powered during normal operation by the main generator is about 30% higher than for the design powered by offsite power through a start-up transformer because of failures in the first design when the power source is transferred from the main generator to an offsite power source. Examples of failures affecting this transfer include human errors while manipulating the components involved in the transfer, failures of the relays that send a signal for the transfer, and failures of the circuit breakers that have to open and close to power the safety buses from offsite power.

The means of the generic probabilities of LOOP given a LOCA due to plant-centered factors (1.2×10^{-2} and 9.3×10^{-3}) are larger than the point estimate of generic probability of LOOP given a reactor trip (2×10^{-3}) that was obtained using the data-driven approach. The reason for this difference is that the events triggered by a LOCA are expected to impose a larger stress on the plant-centered electrical equipment than a reactor trip.

Generic Probability of Consequential LOOP due to Transient Factors

The generic probability of consequential LOOP due to transient factors is considered to be less than the generic probability of consequential LOOP due to plant-centered factors. This conclusion is based on the following observations:

1. There have been more consequential LOOPs due to plant-centered factors than due to grid-related factors. A review of the main causes of all the consequential LOOPs indicates that eight consequential LOOPs have occurred due to plant-centered factors, and only one due to grid-related factors. This argument, however, is not completely conclusive because most of the data on consequential LOOPs are for those following a reactor trip. As discussed earlier in this report, the impact of a reactor trip on the mechanisms related to a consequential LOOP is somewhat less severe than that resulting from a LOCA.

2. The data on LOOP events in NUREG/CR-5496 [Ref. 9] indicates that there are more LOOP events due to plant-centered failures than due to grid-related failures. NUREG/CR-5496 evaluates LOOP events at nuclear power plants during the period 1980 to 1996, and classifies the events in three categories:
"Plant-centered events are those in which the design and operational characteristics of the plant itself play the major role in the cause and duration of the loss of offsite power.
Grid-related events are those in which problems in the offsite power grid cause the [LOOP] and impact its duration."
Severe-weather events are those resulting from "weather with forceful and non-localized effects."
The data on LOOP events gathered by NUREG/CR-5496 indicate that there are 65 LOOP events due to plant-centered failures and 6 events due to grid-related failures. In other words, the number of LOOP events due to plant-centered failures is about one order of magnitude larger than the number of events due to grid-related failures. The data on LOOP events gathered by NUREG/CR-5496 includes events with short duration. In addition, Atwood, et al., state "...Based on this experience, grid instability has not been an important contributor to [LOOP] frequency."

Since the generic probability of consequential LOOP due to transient factors (grid-related failures) is considered to be less than the generic probability of consequential LOOP due to plant-centered factors, and since there is a lack of sufficient data or a model to accurately determine the probability of consequential LOOP due to transient factors, the generic probability of consequential LOOP due to plant-centered factors is used as a bounding value for the probability of consequential LOOP due to transient factors. This approach results in a conservative estimate of the generic probability of consequential LOOP due to transient factors.

Total Generic Probability of Consequential LOOP Given a LOCA

Since a LOCA can cause an electrical disturbance in the offsite and onsite electrical systems, the transient factors and the plant-centered factors are not independent of each other. However, since either of the two factors can cause a delayed LOOP, an approximation of the generic probability of a consequential LOOP given a LOCA can be obtained by adding the probability of LOOP due to plant-centered factors and the probability of a LOOP due to transient factors, as follows:

$$P(\text{LOOP given a LOCA}) \sim P(\text{LOOP due to plant-centered factors}) + P(\text{LOOP due to transient factors}) \quad (5)$$

This approximation only should be used when the individual probabilities on the right-hand side of the expression are less than 0.1.

Since P(LOOP due to plant-centered factors) is a function of the design of the plant, that is, whether the safety buses are supplied by the main generator through a unit transformer, or they are supplied by a preferred offsite power source, two evaluations of P(LOOP given a LOCA) were conducted. Table 4.3 presents the resulting generic conditional probabilities of LOOP given a LOCA.

Table 4.3 Generic Probability of LOOP Given a LOCA

Power supply during normal operation	Probability		
	5%	Mean	95%
Main generator through an auxiliary transformer	1.1E-2	2.4E-2	4.6E-2
Offsite power through a start-up transformer	8.6E-3	2.1E-2	4.3E-2

The uncertainty bounds of the generic probability of LOOP given a LOCA presented in Table 4.3 represent the uncertainty in the estimate of this probability mainly due to data variability. It is considered that the estimate of the generic probability of LOOP given a LOCA presented in Table 4.3 also has uncertainty due to incomplete knowledge about all the factors impacting this probability. This kind of uncertainty is called here epistemic uncertainty and is not included in the confidence limits presented in Table 4.3.

Due to incomplete knowledge (epistemic uncertainty), several assumptions were made in evaluating the generic probability of consequential LOOP due to plant-centered factors and the generic probability of consequential LOOP due to transient factors. In addition, and with regard to the last probability, as discussed in Appendix G (Section G.2), there is a concern that inadequate voltage in the offsite system can be caused by deregulation of the electrical industry. Deregulation is a relatively recent factor, and it is expected to keep growing.

Due to the epistemic uncertainty, it is suggested that the generic probabilities of LOOP given a LOCA presented in Table 4.3 apply only to plants that have certain "good" characteristics that make them less susceptible to a LOOP given a LOCA. A set of candidate characteristics was developed using the insights gained during the following activities: (1) review of consequential LOOPs after a reactor trip or a major ECCS actuation, (2) discussions with NRC staff, and (3) work carried out to produce NUREG/CR-6538. The characteristics were classified into two groups. Those characteristics that are related to onsite equipment are classified as "plant-centered," and those that are related to the interaction of the plant with the offsite grid are classified as "external."

Plant-centered characteristics

1. Onsite voltage correcting means such as automatic tap changer transformer (recent vintage, high reliability, fast responding type) or static volt amperes reactive (VAR) compensator.

2. A plant with electrical, instrumentation, and control equipment that is well-designed, analyzed, operated, and maintained. This applies to equipment located both onsite and offsite (up to the switchyard).

3. Multiple-unit sites maintain their generators in automatic voltage regulate.

4. Loading of ECCS loads to offsite power by sequencing, as opposed to block-loading.

5. On a plant trip, zero or minimal balance of plant loads are transferred to the same transformer winding that feeds AC Class 1E buses.

Items 4 and 5 are not required if item 2 is fulfilled. However, item 2 cannot be verified directly.

As seen from the analysis presented here, and as evidenced by operational experience, plants that power their Class 1E buses directly from the offsite source(s) have better reliability of offsite power to the plant than plants that power their Class 1E buses from the main generator (through an auxiliary transformer). For this latter class of plants, the reliability of offsite power to the plant can be improved by providing an automatic transfer to the delayed source of offsite power. With this change, if the automatic transfer to the preferred source of power fails, there would be an automatic transfer to an alternate source of offsite power, unless there is a common failure which affects both automatic transfers.

External characteristics

1. Voltage at the switchyard is kept close to nominal during normal operation of the plant.

2. Plant operators maintain protocols with their grid system operators that alert them to low switchyard voltages on a contingency, such as a plant trip or a LOCA.

A positive answer to the following questions related to deregulation [Ref. 10]:

3. Has the nuclear plant voltage requirement been determined through rigorous evaluation of all relevant conditions including evaluating the effect of a generator trip?

4. Does the nuclear plant voltage remain constant or rise when the nuclear plant trips under all allowed operating and credible contingency conditions?

5. Has the timing of the plant voltage requirements been coordinated with the power system's ability to provide that voltage post-contingency?

6. Is the geographic scope of real-time data collection, control, and system operations sufficient to address all nuclear plant offsite power reliability concerns?

7. Are real-time data tools, including state estimation and on-line contingency analysis, used by the control area operator, adjacent control areas (if their operation can influence the nuclear plants voltage), and the regional security coordinator?

8. Are system operators well trained in nuclear plant offsite power requirements and methods/procedures to assure adequacy of offsite supply?

4.4 Plant Grouping

The generic approach for identifying alternative ECCS configurations that meet established risk guidelines requires that the operating nuclear power plants be placed in groups that have similar configurations. Although plant grouping by nuclear steam supply vendor (NSSS) is possible, the plants within each vendor group are not standardized. Furthermore, the various architect and engineering (A&E) firms that constructed the plants generated different balance-of-plant and support system configurations. In many cases, the same A&E firm generated different plant

configurations for plants with the same NSSS design. The identification of generic plant groups must take these variables into consideration.

The process of placing plants into generic groups for the purpose of identifying alternative ECCS configurations that meet established risk guidelines considers several factors in the following order:

- the NSSS vendor,
- the vintage of the plant,
- differences in the frontline accident mitigating system configurations and success criteria,
- the availability of alternative systems for accident mitigation, and
- differences in the support system configurations and success criteria.

All of the operating nuclear power plants were first categorized by either the plant vintage or the nuclear steam supply system vendor. Hence, all of the plants were first categorized as either BWRs or PWRs. The BWRs were then further grouped according to the General Electric (GE) "product lines" to account for differences in plant design, especially in the ECCS. Specifically, these groupings include BWR 1, 2, or 3 designs with isolation condensers (ICs) as a group; BWR 3 and 4 designs with reactor core isolation cooling (RCIC) and high pressure coolant injection (HPCI) systems as a group; and BWR 5 or 6 designs with high pressure core spray (HPCS) and RCIC systems as the last group. The PWRs were put into three major groups by nuclear steam supply system vendor (i.e., Westinghouse (W), Combustion Engineering (CE), or Babcock & Wilcox (B&W)). The Westinghouse plants were further categorized on the basis of the number of primary coolant loops in the design (i.e., 2-, 3-, or 4-loop plants), since the plants in each of these groups have similar high-pressure ECCS configurations. The preliminary plant groups are summarized in Table 4.4. Table 4.4 also contains some descriptive information about the systems in each pant group, which was partially obtained from NUREG-1560 [Ref. 11].

Table 4.4 Summary of Nuclear Power Plants Sorted by NSSS Vendor and Vintage

Class	Plants
BWR 1/2/3	• Big Rock Point • Dresden 2&3 • Millstone 1 • Nine Mile Point 1 • Oyster Creek
	These plants generally have separate shutdown cooling and containment spray systems and a multi-loop core spray system. An isolation condenser is utilized for these plants with the exception of Big Rock Point.
BWR 3/4	• Browns Ferry 2&3 • Brunswick 1&2 • Cooper • Duane Arnold • Fermi 2 • FitzPatrick • Hatch 1 • Hatch 2 • Hope Creek • Limerick 1&2 • Monticello • Peach Bottom 2&3 • Pilgrim 1 • Quad Cities 1&2 • Susquehanna 1&2 • Vermont Yankee
	These plant are designed with two independent high pressure injection systems (RCIC and HPCI). The associated pumps are each powered by a steam driven turbine. These plants also have a have multi-loop core spray system and a multi-mode residual heat removal system that can be aligned for low-pressure coolant injection, shutdown cooling, suppression pool cooling and containment spray function.

Table 4.4 Summary of Nuclear Power Plants Sorted by NSSS Vendor and Vintage

Class	Plants
BWR 5/6	• Clinton • Grand Gulf 1 • LaSalle 1&2 • Nine Mile Point 2 • Perry 1 • River Bend • WNP 2
	These plants utilize a HPCS system that replaced the HPCI system. The HPCS system consists of a single motor-driven pump train powered by it's own electrical division complete with a designated diesel generator. These plants have a single train low-pressure core spray system and also have a multi-mode residual heat removal (RHR) system similar to the system design in the BWR 3/4 group.
Babcock & Wilcox	• ANO 1 • Crystal River 3 • Davis Besse • Oconee 1, 2 & 3 • TMI 1
	The B&W plants utilize once through steam generators. Primary system feed and bleed cooling can be established through the pressurizer power relief valves utilizing the high-pressure injection system (HPSI). The HPSI uses the charging system pumps (3) that have a pump shutoff head greater than the pressurizer safety relief valves setpoint. The low-pressure safety injection (LPSI) is a mode of the RHR system. Emergency core cooling recirculation requires manual alignment of the LPSI pumps to the containment sumps. High-pressure recirculation is accomplished by the HPSI pumps taking suction from the output of the LPSI pumps.
Combustion Engineering	• ANO 2 • Calvert Cliffs 1&2 • Fort Calhoun 1 • St. Lucie 1 • St. Lucie 2 • Maine Yankee • Millstone 2 • Palisades • Palo Verde 1,2&3 • San Onofre 2&3 • Waterford 3
	The CE plants utilize U-tube steam generators. The capability to establish feed and bleed cooling in this group is mixed. Several CE plants are not designed with pressurizer power operated valves. Separate HPSI pumps are available and are used for coolant recirculation from the containment sump (piggy-backing with the LPSI pumps is not required). The LPSI is a mode of the RHR system.
Westinghouse 2-loop	• Ginna • Kewaunee • Point Beach 1&2 • Prairie Island 1&2
	These plants utilize U-tube steam generators and are designed with air operated pressurizer relief valves. Decay heat can be removed from the primary system using feed and bleed cooling. Separate HPSI pumps are available and must be piggy-backed with the LPSI pumps in the recirculation mode. The LPSI is a mode of the RHR system. Emergency core cooling recirculation requires manual switchover to the containment sumps.
Westinghouse 3-loop	• Beaver Valley 1 • Beaver Valley 2 • Farley 1&2 • North Anna 1&2 • Robinson 2 • Shearon Harris 1 • Summer • Surry 1&2 • Turkey Point 3&4

Table 4.4 Summary of Nuclear Power Plants Sorted by NSSS Vendor and Vintage

Class	Plants
Westinghouse 3-loop (cont.)	This group is similar in design to the Westinghouse 2 loop group. One exception is that most of these plants use the charging pumps for HPSI. Another exception is that LPSI is a separate system (i.e., not a mode of RHR) at some plants.
Westinghouse 4-loop	• Braidwood 1&2 • Byron 1&2 • Callaway • Catawba 1&2 • Comanche Peak 1&2 • DC Cook 1&2 • Diablo Canyon 1&2 • Haddam Neck • Indian Point 2 • Indian Point 3 • McGuire 1&2 • Millstone 3 • Salem 1 • Salem 2 • Seabrook • Sequoyah 1&2 • South Texas 1&2 • Vogtle 1&2 • Watts Bar 1 • Wolf Creek • Zion 1&2
	The Westinghouse 4 loop group includes nine plants housed within ice condenser containments. These plants have both HPSI and charging pumps capable of mitigating small loss of coolant accidents (LOCAs). The LPSI is a mode of the RHR system at most of the plants. Piggy-backing of the HPSI and LPSI pumps is required for high-pressure recirculation. Many of these plants have large refueling water storage tanks such that switchover to emergency core cooling recirculation is either not needed or significantly delayed.

A summary of the available systems in the operating BWRs and PWRs was obtained from NUREG-1560 (see Appendix E, Tables E.2 and E.3). Review of this information showed that there can substantial differences in the types and number of accident mitigating systems even for plants of the same class. Consideration of the differences in the accident mitigating system configurations and success criteria can result in the need to subdivide the vendor/vintage groups. This is particularly true when considering the support system configurations. Also, the definition of the break size ranges for different LOCA size categories can vary substantially from plant-to-plant within a vendor/vintage group. This can have a significant impact on the assumed success criteria. Thus, in order to limit the number of plant groups to a manageable level, limiting configurations (with regard to the impact on CDF and LERF) have been selected in some cases. Where possible, the selected configurations are representative of the majority of the plants in the group.

For alternative coolant injection systems, only those credited by the majority of the plants in a group are credited. Sensitivity studies are performed to evaluate the effect of crediting those systems on the ability to identify alternative ECCS configurations for the plant group. The results of these sensitivity studies are used to establish the generic ECCS requirements and whether or not to modify those requirements for specific plants in the group.

At this time, only the BWR 3/4 group of plants has been analyzed in detail for this study. This plant group was selected for the initial analyses because of the high degree of redundancy present in low-pressure ECC systems. The analysis of the BWR 3/4 group of plants serves as a pilot study to help determine the practicality of identifying alternative ECCS configurations on a generic basis. Details on the differences in the BWR 3/4 class of plants are provided in Appendix E. Additional groups are being formulated for the remaining classes of BWRs and PWRs.

Plants placed in the BWR 3/4 group have two turbine-driven high-pressure injection systems, namely the RCIC system and the HPCI system. These plants also have a multi-loop low-pressure core spray (CS) system and a multi-mode RHR system that can be aligned for low-pressure coolant injection (LPCI), shutdown cooling, suppression pool cooling, and containment spray functions.

To create a composite SPAR model that represented the most limiting configuration for each system, a survey was conducted across the plants within the BWR 3/4 group. This survey addressed the following systems:

- BWR 3/4 primary and power conversion systems
- BWR 3/4 reactivity control functions
- BWR 3/4 coolant injection
  - reactor core isolation cooling
  - high-pressure coolant injection
  - core spray
  - low-pressure coolant injection
  - automatic depressurization system
  - alternative injection systems
  - influence of containment on coolant injection
- BWR 3/4 decay heat removal
- BWR 3/4 support systems
  - electric power systems
  - cooling water systems
  - other support systems

The NRC system notebooks and Revision 3i SPAR models were utilized initially for the survey. If the information provided in the SPAR models or system notebooks was found to be inconsistent with information provided in the Individual Plant Examinations (IPEs), the IPE was considered to supersede the SPAR model or system notebook information for the purpose of this survey. The NRC has underway activities which address differences between sources of plant information; one example is the SPAR model quality assurance (QA) process explained below. The results of the survey are provided in Appendix E.

The Revision 3 SPAR models are a set of 72 plant-specific, PRA-based models developed for use by staff analysts in their regulatory activities. The SPAR models should predict the response of the as-built, as-operated plant to a spectrum of accident initiators. To this end, the SPAR Model Development Program includes a systematic QA process, which was approved by the interoffice SPAR Model Users Group. Each of the Revision 3i ("i" denotes interim - the model has not completed the QA process) SPAR models is being subjected to an onsite QA review, which includes benchmarking the SPAR model against the licensee's updated plant PRA model.

4.5 Insights from Initial Generic Calculations

A modified version of the Revision 3i ("i" denotes interim - this model had not yet received an onsite QA review against the licensee's PRA model) SPAR model was used to reflect the representative configuration for the BWR 3/4 plant group. The Peach Bottom units have been examined extensively using PRA -- the Reactor Safety Study, NUREG-1150, the Peach Bottom IPE, and the Peach Bottom SPAR model. The Peach Bottom units are representative of plants with 4 core spray

pumps, 4 AC divisions per unit and above average emergency power reliability. Pending the outcome of this initial evaluation, units such as Brunswick 1&2 may need to be evaluated as representative of plants with 2 core spray pumps and 2 AC power divisions per unit.

For this application, the Revision 3i SPAR model for the Peach Bottom units was first changed to incorporate the best available pipe break frequencies, to specifically address the conditional probability of LOOP given a LOCA (or other reactor trip), to make the model more consistent with the plant's design features, and to eliminate nonrepresentative plant-specific features (e.g., back-up AC power from the Conowingo dam). The modified base case model obtained in this manner yielded a point-estimate CDF of 5×10^{-6} per reactor critical year. The modified base case model was examined to test the quantitative significance of alternative injection via non-safety systems. Studies were then performed to examine the feasibility of changing ECCS technical specifications, reducing ECCS redundancy, eliminating core spray pumps, or eliminating LOOP as a design basis event for large-break LOCAs and medium-break LOCAs. The analyses performed for Peach Bottom 2&3 are discussed in detail in Appendix H. The principal insights from these calculations are summarized below.

Injection Using Non-ECC Systems

An examination was performed of the significance of credit given for injection by non-ECC systems. When no credit is given for injection by non-ECC injection systems, the resulting CDF was ~5×10^{-4} per reactor critical year. A baseline CDF of this magnitude would preclude generic regulatory changes involving risk increases according to the current framework for risk-informing 10 CFR Part 50 [Ref. 3]. That is, credit for at least some non-ECC systems is needed to justify risk-informed changes to current regulatory requirements for ECC systems. An increased level of regulatory treatment for non-ECC systems may, therefore, have to be an integral part of any proposed change. In the following sensitivity studies, credit is given for all of the non-ECC systems included in the Peach Bottom SPAR model (see Appendix H, Table H.10, for a detailed list).

Analysis of ECCS Technical Specifications Requirements

For the plants in the BWR 3/4 group, low-pressure ECCS includes 4 RHR pumps in low-pressure coolant injection (LPCI) mode and four low-pressure core spray pumps. From a risk perspective, the RHR pumps are more important because they are also used to perform residual heat removal and containment spray functions. The initial analyses indicate that some relaxation to existing technical specifications regarding allowed outage times for low-pressure ECC pumps (particularly for core spray pumps) appears justifiable for the BWR 3/4 plant group. A summary of point-estimate single outage risks for outages of all possible pump pairs within the CS and LPCI systems is provided in Appendix H. The method employed follows that described in NUREG/CR-6141 [Ref. 12] and Regulatory Guide 1.177 [Ref. 13]. The results in Appendix H (Table H.11) show that, for the same outage duration, the maximum single outage risk associated with having any pair of RHR pumps inoperable exceeds the maximum single outage risk associated with having any pair of CS pumps inoperable by more than two orders of magnitude. Current technical specifications permit any two RHR pumps or any CS subsystem to be inoperable for 7 days. Based on the current allowed outage time (AOT) for the RHR pumps, the results from Appendix H indicate that it may be possible to extend the AOT for the CS pumps. In addition, current technical specifications only permit two core spray pumps within the same subsystem to be inoperable for up to 7 days. A possible change that is supported by the results in Appendix H would be to extend this technical specification to any pair of CS pumps, not just those in the same subsystem. Alternatively, the

technical specifications could be changed to permit any pair of low-pressure injection/spray pumps to be inoperable for up to 7 days; however, the single outage risks associated with having various pairs of CS and RHR pumps inoperable would have to be computed to support this alternative.

Another change that could be contemplated would be to lengthen the maximum duration for having low-pressure injection/spray pumps inoperable. The single outage risks for a 31 day limit are also included in Appendix H, and generally indicate that such a change may be within the CDF acceptance guidelines specified in Section 4.1 (though no attempt has been made to estimate the yearly frequency and average duration of RHR or CS pump outages).

An issue exists regarding the LPCI success criteria for a large break in a recirculation line. The core may only be reflooded to two-thirds core height for large LOCAs with break flows equivalent to the capacity of one LPCI (RHR) pump. If only one or two RHR pumps operate and if flow is not controlled, the water in the vessel may be subcooled and there will not be steam cooling of the top third of the core. This may be less of a problem for LPCI systems that inject into the shroud area. Current ECCS evaluation models do not proceed this far in time.

To address this issue, a variation of the LPCI fault tree was developed to apply a 3-out-of-4 pump success criterion for large LOCAs. When this alternative LPCI fault tree was applied to large-break LOCAs, the point-estimate total plant CDF increased by less than 0.2%. Although the large-break LOCA CDF increases by 40% with the alternative success criteria, the results presented in Appendix H indicate that there is no dramatic increase in the overall single outage risks for pairs of low-pressure injection/spray pumps due to the alternative success criteria.

Analysis of Low-Pressure ECCS Redundancy

A number of sensitivity cases were developed to investigate the feasibility of decreasing the redundancy of low-pressure ECCS (LPCI/RHR and CS). Each change was evaluated using the base case (1-out-of-4 pumps) and alternative (3-out-of-4 pumps) success criteria for LPCI in large-break LOCAs. Based on the results provided in Appendix H, with credit for the non-ECC injection systems available at Peach Bottom and most other plants in the group, eliminating or reducing the redundancy of the core spray system (or reducing the level of regulatory treatment for this system) appears consistent with the CDF acceptance guidelines specified in Section 4.1. Appendix H also shows that the increase in CDF associated with removal of a pair of RHR pumps is considerably greater than that associated with completely removing the CS system, since the other modes of RHR provide long-term decay heat removal, which is important for a broad spectrum of accidents.

Analysis of Design-Basis LOOP Event for Large- and Medium-Break LOCAs

General Design Criterion 35 of 10 CFR Part 50, Appendix A, 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 starting and loading times that provide significant stresses on the diesels and may reduce their reliability.

The feasibility of eliminating the LOCA-LOOP design requirement for large- and medium-break LOCAs was investigated. When all large-break LOCAs that induce LOOP are assumed to lead to core damage (per Section 3.1.2), the mean CDF increases by a little more than 10^{-6} per critical year. This is the mean frequency of large breaks (7×10^{-5} per critical year) times the mean

probability of LOOP given a break initiator (2×10^{-2}). When both large- and medium-break LOCAs that induce LOOP are assumed to lead to core damage, the mean CDF increases by approximately 4×10^{-6} per critical year. This is the mean frequency of large and medium breaks ($7\times10^{-5} + 1\times10^{-4}$ per critical year) times the mean conditional probability of LOOP given a break initiator (2×10^{-2}).

Concluding Remark

For most units in the BWR 3/4 group the risk increments associated with the aforementioned changes are consistent with the CDF acceptance guidelines; however, the current effort focused on one representative plant and full-power operation. Further work would be required to confirm the results obtained for plants with different support system designs and to examine the impact of proposed changes on accidents initiated in other modes of operation or by external events. In addition there is another issue that arises from current work on pressurized thermal shock (PTS).

4.6 Additional Analyses

The initial generic calculations have identified current regulatory requirements that could potentially be changed for the BWR 3/4 plant group. This plant group was selected for the initial generic analyses because the high degree of low-pressure ECC redundancy that exists for BWR 3/4 plants. Generic calculations similar to those presented in Appendix H will, in the future, be performed for other plant groups.

To support possible rulemaking, analyses of the type presented in Appendix H will be extended to 1) assure coverage of plant variations within each group, 2) evaluate the impact of proposed changes on LERF, 3) evaluate the impact of proposed changes on accidents at low power and shutdown, and 4) quantitatively address uncertainties, where practical. The types of analyses envisioned are discussed in the following paragraphs.

The analyses presented in Appendix H adapted an internal events PRA for one facility, Peach Bottom 2&3. The baseline CDF for Peach Bottom is less than that for other plants in the BWR 3/4 group even after removing unique features such as backup AC power from the Conawingo dam. Plant-specific features that may cause differences in CDF or LERF for plants within a group will be identified and delineated. The initial analysis will be verified by adapting PRA model(s) representative of plants with significantly different features (e.g., two versus four core spray pumps and two versus four AC power divisions per unit). Plant-specific features that could preclude proposed changes to current ECC requirements will be identified. For example, plants that cannot vent containment in the event of a LOOP tend to have much greater baseline CDFs than other plants in the BWR 3/4 group.

The initial generic analyses addressed the impact of proposed changes on CDF but not LERF. Future analyses would assess the impact of proposed changes on LERF. It is anticipated that this will be accomplished by examining the core damage cutsets that are introduced or whose frequency is increased due to a potential change and assessing their potential to cause a large early release. It is anticipated that, in many cases, bounding analyses may provide sufficient characterization of the ΔLERF associated with a proposed change.

As discussed in Section 3.1.3, when a PRA is not full scope, it is necessary to address the out-of-scope items. Generation of low power/shutdown (LPSD) models may be required to evaluate the

minimum ECCS requirements. Generic SPAR templates exist for creating these models. An effort is currently underway that uses these templates to produce LPSD SPAR models for the various classes of PWRs and BWRs. Alternatively, stricter configuration control requirements may be delineated to manage LPSD risks. These requirements could take the form of technical specifications or risk limits for shutdown configurations.

Finally, as noted in Section 3.1.3, the appropriate numerical measures to use in comparing PRA results to acceptance guidelines are mean values. Future analyses will include formal propagation of uncertainties as well as sensitivity analyses required to address any additional modeling issues that are identified.

There are also a number of areas where additional work by various NRC organizatons may be required. The initial generic analyses for the BWR 3/4 plant group indicate several deterministic analyses that could potentially be relevant to ECCS-related regulatory change for the group. First, deterministic analyses could be used to better define the low-pressure coolant injection (LPCI) success criteria for long term decay heat removal in recirculation-line-break LOCAs. As discussed in the previous section, if only one or two RHR pumps operate and if flow is not controlled, the water in the vessel may be subcooled and there will not be steam cooling of the top third of the core. This may be less of a problem for LPCI systems that inject into the shroud area. Current ECCS evaluation models do not proceed this far in time; however, MELCOR analyses could be used to better define the number of RHR pumps required for long-term decay heat removal or flow control measures that could be applied to assure steam cooling of upper regions of the core. Second, deterministic analyses could potentially be performed to establish the extent to which current diesel generator start times could be lengthened while still complying with current ECCS acceptance criteria of 10 CFR 50.46. The feasibility of performing such analyses in a reasonable time frame has not yet been assessed. Third, it was conservatively assumed in Appendix H, that were the simultaneous LOOP eliminated as a design-basis requirement for large-break LOCAs, all large-break LOCAs that induce LOOP would proceed to core damage. This is a bounding analysis. In reality, diesel generators would still have to ramp up, albeit over a longer time interval, in order to cope with other accidents. As long as the pumps capable of making up vessel inventory at low pressure are still powered by the emergency buses, and the diesel generators are still capable of supplying these loads, then low-pressure make-up could still be initiated manually. MELCOR analyses could be used to better assess the extent of core-damage progression in large-break LOCAs as a function of diesel generator start time.

The analyses in Appendix H examined the significance of credit given for injection by non-ECC systems. For the BWR 3/4 plant group, credit for at least some non-ECC systems is needed to justify risk-informed changes to current regulatory requirements for ECC systems. Future analyses will identify a common set of non-ECC systems to be credited for all plants in a group. These systems should exist at all of the plants that would be permitted to implement a proposed change. The non-ECC systems so credited may need to be subjected to an increased level of regulatory treatment.

For internal fire and flood and external events, risks depend not only on the system configuration but also on the plant layout and location. As further discussed in Section 5, appropriate qualitative requirements may need to be delineated to address internal fire and flood and external events.

5. PRINCIPAL TECHNICAL AND POLICY ISSUES

Any attempt to establish the ECCS reliability for an individual plant or group of plants must consider the limitations of the PRAs used in the evaluations. These may include the required scope of the PRA, its adequacy, and the need to include specific issues that could influence the ECCS reliability. There may be cases where the PRAs used by an individual licensee in the plant-specific approach or by the staff in the generic approach are not capable of fully establishing the ECCS reliability and thus there is a need to identify a method or process for compensating for known inadequacies. Potential difficulties with placing plants in generic groups represents an additional concern for the use of a generic approach to identify alternative ECCS configurations that meet established risk guidelines. These issues and potential solutions are identified in this section.

5.1 Plant Grouping Considerations

The generic approach for identifying alternative ECCS configurations that meet established risk guidelines requires that the operating nuclear power plants be placed in groups that have similar configurations. The grouping must not only consider the ECCS configurations but also their required support system configurations. In addition, the credit for alternative coolant injection systems can vary at the plants depending upon the alternative system capacities and alignments.

Although plant grouping by nuclear steam supply vendor (NSSS) is possible, the plants within each vendor group are not standardized. Furthermore, the various architect and engineering (A&E) firms that constructed the plants generated different balance-of-plant and support system configurations. In many cases, the same A&E firm generated different plant configurations for plants with the same NSSS design. The identification of generic plant groups must take these variables into consideration.

Consideration of the differences in the accident mitigating system configurations and success criteria can result in the need to subdivide the vendor/vintage groups. The ECCS configurations within each vendor/vintage group do not vary substantially and it is thus possible to identify a reasonable number of variations. However, comparison of SPAR model success criteria with IPE success criteria revealed some differences. The NRC has underway activities which address differences between sources of plant information; one example is the SPAR model QA process explained previously. The definition of the break size ranges for different LOCA size categories can vary substantially from plant-to-plant within a vendor/vintage group. This can have a significant impact on the assumed success criteria. In addition, there is substantially more variation in support systems configurations which severely impacts the grouping process. Thus, in order to limit the number of plant groups to a manageable level, it is desirable to select limiting system configurations (with regard to the impact on CDF and LERF). Where possible, the selected configurations should be representative of the majority of the plants in the group. In some cases, variations in system configurations may not be important and thus can be ignored. In these cases, the bases for the assumed low impact on the ECCS evaluations must be firmly established either qualitatively or quantitatively.

For alternative coolant injection systems, only those credited by the majority of the plants in a group are credited. Sensitivity studies are performed to evaluate the affect of crediting those systems on the minimum ECCS requirements for the plant group. The results of these sensitivity studies are used to establish the generic ECCS requirements and whether or not to modify those requirements for specific plants in the group.

5.2 PRA Scope Limitations

The scope and level of PRA analysis necessary to support the evaluation of proposed ECCS-related changes to the current design basis will depend on the type of change being proposed. Ideally, an analysis would be based on a full-scope PRA that considered both internal and external initiators, including fires and flooding, and all modes of plant operation including low power and shutdown. On the other hand, for some changes, a full-power internal-event PRA may suffice if the significance of external events or other modes of operation can be demonstrated to be small based on alternative methods or reasoning. Acceptable approaches for dealing with scope limitations are outlined in Regulatory Guide 1.174.

ECCS systems are used to prevent core damage. A Level 1 PRA analysis may suffice to justify proposed ECCS-related changes that clearly would not impact large release probabilities conditional on the occurrence of core damage. ECCS-related changes that could also impact the response of containment systems would require Level 2 evaluations to establish impacts on large release probabilities. When proposed ECCS-related changes could impact online maintenance or the decay heat removal function, an assessment of shutdown risk should be considered.

As discussed in Section 4.6, when a PRA is not full scope, it is necessary to address the significance of the out-of-scope items. For ECCS-related changes, this is difficult due to the fact that the ECCS is required for mitigating potential accidents during different modes of operation. The importance of assessing the contribution of the out-of-scope portions of the PRA to the base case estimates of CDF and LERF is related to the margin between the as-calculated values and the acceptance guidelines. When the contributions from the modeled contributors are close to the guidelines, the argument that the contribution from the missing items is not significant must be convincing, and will likely require additional PRA analyses. When the margin is significant, a qualitative argument may be sufficient. The contribution of the out-of-scope portions of the model to the change in metric may be addressed by bounding analyses, detailed analyses, or by a demonstration that the change has no impact on the unmodeled contributors to risk. In addition, it should also be demonstrated that changes based on a partial PRA do not disproportionally change the risk associated with those accident sequences that arise from the modes of operation not included in the PRA.

The risk contribution from internal fire and flood and external events is dependent not only on the system configurations but also the plant layout and location. Thus, the risk from these events is very plant-specific and it would be very difficult to generically establish the ECCS importance to these events. This implies that for the generic option, the minimum ECCS requirements can only be quantitatively established based on a PRA for internal events. This fact raises an issue on how to account for external events in establishing generic ECCS requirements.

One possible approach to address this issue is to require that the minimum ECCS configuration meet some qualitative requirements that would ensure adequate reliability for internal fires and floods and external events. Examples of these qualitative requirements include:

- the minimum ECCS must be spatially separated such that an internal fire or flood or an external event can only fail half of the ECCS (the reliability of the remaining half of the ECCS

multiplied by the combined frequency of the external events should be significantly less than the CDF acceptance criteria),

- functional redundancy should be maintained such that a common failure mode can not fail the entire ECCS (e.g., maintain both high and low-pressure systems), and
- the minimum ECCS and required support systems and structures should have a high confidence of low probability of failure < 0.1 for external events with frequencies greater than 10^{-5}/yr.

With the above approach, the generic risk results would only apply to internal events; therefore, some additional margin between the calculated risk values and the acceptance guidelines would be necessary to account for potential contributions from external events.

Alternatively, the generic option could be used to establish the minimum ECCS requirements for internal events. However, each licensee would have the burden to show that the risk from external events for the configuration is acceptable.

Another issue for the generic option is the unavailability of low power/shutdown (LPSD) PRA models. Since the conditional risk during cold shutdown can be similar to that at power and the dominant sequences during LPSD include LOCAs and drain down events, the ECCS is important during these modes. However, the absolute risk from these modes of operation is generally small due primarily to the relatively short time the plant is in these modes. Generation of LPSD models would be required to quantitatively evaluate the minimum ECCS requirements for these modes (generic SPAR templates exist for creating these models, and an effort to use these templates to produce LPSD models for the various PWR and BWR classes is now underway). Alternatively, stricter configuration control requirements could be imposed to manage the risk during these modes. These requirements could take the form of technical specifications or risk limits for shutdown configurations.

The PRA evaluations must also address the LERF acceptance criteria. Therefore, a plant-specific evaluation would likely require a PRA of sufficient detail to establish the baseline LERF and change in LERF from any change in the ECCS configuration or requirements. For the generic approach, the new LERF SPAR models currently being developed could be used, if available in a time frame consistent with possible rulemaking (current plans estimate that the first peer-reviewed LERF SPAR model for a lead plant in a plant class, as defined by containment type, will be completed by the end of next year). Alternatively, based on the known contributors to LERF in the IPEs and NUREG-1150 PRAs, it may be possible to establish that for specific containment types, changes to ECCS configurations would have little impact on LERF. Still another approach would be to use the simplified LERF models documented in NUREG/CR-6595 [Ref. 14].

5.3 PRA Adequacy

Regulatory Guide 1.174 defines PRA quality as measuring the adequacy of the actual PRA modeling. A PRA used in risk-informed regulation should be performed correctly, in a manner that is consistent with accepted practices, and commensurate with the scope, level of detail, and technical acceptability required as discussed above. An acceptable approach for assessing the technical acceptability of a PRA for use in ECCS reliability applications is to perform a peer review of the PRA.

In the plant-specific approach, an ECCS reliability application should document the peer review process, including the approach and standard or guidelines to which the PRA is compared, and should address the qualification of the reviewers, the scope of the review, and the review findings. The review findings should address any limitations of the analysis that are expected to impact the conclusion regarding the acceptability of the proposed change to the ECCS or its design bases. The licensee's resolution of the findings of the peer review, when performed, should also be submitted. The resolutions should indicate whether the PRA was modified or justify why no change to the PRA was necessary to support decisionmaking for the proposed change.

Specific peer review requirements have been established in the ASME PRA Standard and a PRA peer review process has been established by the Nuclear Energy Institute (NEI) [Ref. 15]. Both are currently under review by the NRC staff for endorsement that will be documented with any exceptions in a new regulatory guide. Based on the peer review or certification process and on the findings from this process, the licensee should justify why the PRA is adequate for the ECCS reliability application in terms of scope, level of detail, and technical acceptability.

As mentioned above, an ASME PRA Standard is available for use in identifying the technical requirements needed in a PRA. Current plans are for the NRC to endorse this standard and identify exceptions to the PRA technical requirements in a new regulatory guide. When completed, this regulatory guide should provide the bases for establishing the PRA adequacy necessary for ECCS reliability applications. Additional PRA standards are being developed for the analysis of external events (e.g., seismic events), internal fire events, and low power and shutdown conditions.[(2)] Ultimately PRA standards will be developed and endorsed by the NRC that are suitable for risk-informed ECCS-related changes based on full-scope PRAs.

For the generic approach, the Level 1, Revision 3i (or Revision 3, if available) SPAR models are being utilized. The adequacy requirements for these models would need to be as strict as those for plant-specific applications if calculations performed with them are to support rulemaking. The SPAR Model Development Program includes a systematic QA process, which was approved by the interoffice SPAR Model Users Group. Each of the 72 Revision 3i SPAR models is being subjected to an onsite QA review, which includes benchmarking the SPAR model against the licensee's plant PRA model. In addition, one objective of the SPAR Model Development Program is that the Revision 3i SPAR models will meet the requirements of the proposed ASME Standard on PRA to the extent required, commensurate with the intended use and limited scope of the models.

The SPAR models for full-power operation address all relevant initiators that can occur from random events. They include the frontline mitigating systems typically modeled in a PRA including alternative systems. However, the modeling of support systems is typically limited to electrical

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[2] The American Society of Mechanical Engineers (ASME) has developed a standard, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications"; it is for Level 1 and Level 2 (LERF only) PRAs for internal events (excluding fire) that occur during full-power operations.

The American Nuclear Society (ANS) is developing a draft standard for external events (e.g., seismic events, including seismic margins, wind, flood), "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications: External Events." The ANS is also developing a draft standard for low-power and shutdown conditions, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications: Low Power and Shutdown." In addition, the ANS is considering developing a fire PRA standard.

power and cooling water systems. Other support systems such as HVAC and instrument air are usually not modeled. The lack of these support systems in the model can influence the results to some unknown extent.

The SPAR models use generic data, much of which is taken from the Idaho National Engineering and Environmental Laboratory (INEEL) system studies. For plants with significantly higher component failure probabilities, the resulting ECCS reliability evaluations using this generic data may be inaccurate. The SPAR models have the capability to propagate uncertainty in the probabilities associated with equipment performance, but point estimates are presently used for all human error probabilities. The human reliability analysis methodology used in the Revision 3i SPAR models is currently being enhanced to provide uncertainty analysis capability.

The INEEL system studies determined recovery probabilities obtained from recovery actions that have occurred during unplanned system demands. In the approach currently incorporated in the Revision 3i SPAR models, the system recovery events are included in the system model and given the same default probability for all accident scenarios. Accident sequence-dependent recovery probabilities are subsequently adjusted by the analyst. Recent user experience with the Revision 3i models indicates that this approach should be re-evaluated and modified, if necessary, to reduce the level of effort required by the user. For the current application, a detailed look at all sequences containing recovery actions has not been performed; therefore, the appropriateness of each recovery action probability has not been confirmed.

5.4 Modeling Issues

The level of detail required of the PRA is that which is sufficient to model the impact of the proposed ECCS-related change. In characterizing the proposed change, establishment of a cause-effect relationship can help to identify portions of the PRA affected by the ECCS application under consideration. Due to the use of the ECCS for mitigating most postulated accidents, most ECCS configuration changes are expected to require full-scale application and quantification of the PRA. Changes related to the ECCS design basis (for example eliminating large-break LOCAs with coincident LOOP from the 10 CFR 50.46 evaluations) may only require more limited PRA evaluations.

The specific PRA requirements necessary for an ECCS-related application should be determined using a process similar to that given in the ASME PRA Standard [Ref. 16]. This process should identify the systems, structures, components and activities affected by the proposed change including the cause-effect relationship; identify the PRA scope needed to evaluate the change; and determine the requirements for each PRA element necessary to evaluate the change. Efforts are underway to identify the necessary requirements for different types and scopes of PRAs. The ASME PRA Standard focuses on internal events during full-power operation. A regulatory guide endorsing this standard and identifying exceptions is planned and should be used to establish the specific PRA element requirements for evaluating ECCS-related changes.

Attachment 2 to SECY-01-0133 identified three areas where technical analysis would be required to support development of the regulatory guidance for ECCS-reliability applications:

- LOCA frequencies
- Conditional probability of LOOP given a LOCA

- ECCS risk/reliability analyses

As discussed in Section 3.2, updated LOCA frequencies are being developed by the staff by means of an expert elicitation process, and are scheduled to be available in the time frame projected for possible rulemaking. A process for establishing a plant-specific conditional probability of LOOP given a LOCA is identified in Section 3.3 (and Appendix D). It is possible that very few plants may have the requisite operational data to implement this plant-specific process. Generic probabilities for consequential LOOP following a reactor trip and following a LOCA are provided in Section 4.3 (with details in Appendix G). PRA modeling issues that must be considered in both plant-specific and generic ECCS risk/reliability analyses are identified in Appendix B. The approaches taken to address these issues in the generic evaluations are outstanding issues.

5.5 Treatment of Uncertainty

Because of the way the acceptance guidelines were developed, the appropriate numerical measures to use in the initial comparison of the PRA results to the acceptance guidelines are mean values. The mean values referred to are the means of the probability distributions that result from the propagation of the uncertainties on the input parameters and those model uncertainties explicitly represented in the model. A formal propagation of the uncertainty is the best way to correctly account for state-of-knowledge uncertainties that arise from the use of the same parameter values for several basic event probability models.

While the analysis of parametric uncertainty in PRAs is fairly mature, the analysis of the model and completeness uncertainties cannot be handled in such a formal manner. Whether the PRA is full scope or only partial scope, it will be necessary to demonstrate that the choice of models or methods do not significantly change the assessment of the ECCS-related change. This demonstration can take the form of well formulated sensitivity studies. Model uncertainties that would drive the result toward unacceptableness should be identified and sensitivity studies performed or reasons given as to why they are not appropriate for the current ECCS application or for the particular plant. In general, the results of the sensitivity studies should confirm that the guidelines are still met even under the alternative model or assumptions.

The treatment of uncertainty is consistent with the guidance provided in RG 1.174.[(3)] In the generic approach, parameter uncertainty was propagated through the models, and sensitivity studies were performed to test the robustness of the ECCS evaluations to uncertainties in the modeling.[(4)] The model uncertainties evaluated include credit for alternative coolant injection systems and the LPCI success criteria for large-break LOCAs.

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[3]Additional guidance on treatment of uncertainties is being developed and is expected to be included in the next revision of the framework for risk-informing 10 CFR Part 50. This additional guidance could be used to support any possible rulemaking associated with risk-informing the ECCS reliability requirements.

[4]At the time of this report, propagation of parameter uncertainty and sensitivity studies have only been completed for the base case.

6. REFERENCES

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APPENDIX A

RELATED REGULATIONS AND IMPLEMENTING DOCUMENTS

APPENDIX A RELATED REGULATIONS AND IMPLEMENTING DOCUMENTS

APPENDIX B

GENERAL PRA MODELING CONSIDERATIONS

APPENDIX B GENERAL PRA MODELING CONSIDERATIONS

APPENDIX C

QUANTITATIVE GUIDELINES FROM THE FRAMEWORK FOR RISK-INFORMING 10 CFR PART 50

APPENDIX C QUANTITATIVE GUIDELINES FROM THE FRAMEWORK FOR RISK-INFORMING 10 CFR PART 50

APPENDIX D

PLANT-SPECIFIC PROBABILITY OF A CONSEQUENTIAL LOOP: A METHOD OF ASSESSMENT

APPENDIX D PLANT-SPECIFIC PROBABILITY OF A CONSEQUENTIAL LOOP: A METHOD OF ASSESSMENT

APPENDIX E

PLANT GROUPING

APPENDIX E PLANT GROUPING

APPENDIX F

INTERIM LOCA FREQUENCIES

[To Be Supplied]

APPENDIX F INTERIM LOCA FREQUENCIES

APPENDIX G

GENERIC PROBABILITY OF A CONSEQUENTIAL LOOP: AN EVALUATION

APPENDIX G GENERIC PROBABILITY OF A CONSEQUENTIAL LOOP: AN EVALUATION

APPENDIX H

INITIAL PRA RESULTS FOR THE BWR3/4 PLANT GROUP

APPENDIX H INITIAL PRA RESULTS FOR THE BWR 3/4 PLANT GROUP