PRELIMINARY DRAFT

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REGULATORY RESEARCH

???? 2016
Revision 0

REGULATORY GUIDE

Technical Lead
Steven Laur

Public availability of this draft document is intended to inform stakeholders of the current status of the NRC staff's preliminary draft final rule package and associated documents for § 50.46c of Title 10 of the Code of Federal Regulations (10 CFR). This preliminary draft document is in support of a December 3, 2015, Advisory Committee on Reactor Safeguards (ACRS) full committee meeting.

This draft document has not been subject to all levels of NRC management review. Accordingly, it is incomplete and may be in error in one or more respects. The document may be subject to further revision before the staff provides the final draft rule language package to the Commission (currently scheduled to be provided to the Commission in February 2016).

REGULATORY GUIDE 1.229

(Draft was issued as DG-1322, dated April 2015)

RISK-INFORMED APPROACH FOR ADDRESSING THE EFFECTS OF DEBRIS ON POST-ACCIDENT LONG-TERM CORE COOLING

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes methods and approaches that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for demonstrating compliance with the voluntary, risk-informed alternative for addressing the effects of debris during long-term cooling as required in Title 10 of the *Code of Federal Regulations*, Section 50.46c (10 CFR 50.46c), "Emergency Core Cooling System Performance during Loss-of-Coolant Accidents (LOCA)" (Ref. 1). Regulations in 10 CFR 50.46c require that the emergency core cooling system (ECCS) have the capability to provide long-term cooling of the reactor core following any successful initial operation of the ECCS. The ECCS must be able to remove decay heat so that the core temperature is maintained at an acceptably low value for the extended period of time required by the long-lived radioactivity remaining in the core. The rule contains a provision in 10 CFR 50.46c(e) that allows the voluntary use of a risk-informed approach to address the

Written suggestions regarding this guide or development of new guides may be submitted through the NRC's public Web site under the Regulatory Guides document collection of the NRC Library at http://www.nrc.gov/reading-rm/doc-collections/reg-guides/contactus.html.

Electronic copies of this regulatory guide, previous versions of this guide, and other recently issued guides are available through the NRC's public Web site under the Regulatory Guides document collection of the NRC Library at http://www.nrc.gov/reading-rm/doc-collections/. The regulatory guide is also available through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under ADAMS Accession No. ML1525A125. The regulatory analysis may be found in ADAMS under Accession No. ML12283A188 and the staff responses to the public comments on DG-1322 may be found under ADAMS Accession No. ML15252A131.

effects of debris on long-term cooling. The risk-informed approach is an alternative to deterministic approaches for complying with 10 CFR 50.46c(d)(2)(iii).

This RG describes acceptable methods and approaches for addressing 10 CFR 50.46c(e), "Alternate risk-informed approach for addressing the effects of debris on long-term core cooling," and applicable portions of 10 CFR 50.46c(m), "Reporting, corrective actions, and updates" of 10 CFR 50.46c. While the general risk-informed approach in this RG may be applied to any reactor design within the scope of 10 CFR 50.46c, many of the specific approaches (e.g., WCAP-16530-NP-A for chemical effects) and acceptance criteria (e.g., 15 grams per fuel assembly for hot leg break) were developed for the current fleet of pressurized-water reactors (PWRs). Entities (Licensees or applicants) using this guidance should justify that the application of each approach or method used meets the intent of this guidance.

Applicable Rules and Regulations

- 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 2).

- 10 CFR 50.46c, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors."

- 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 15, "Reactor Coolant System Design" (Ref. 3).

- 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 35, "Emergency Core Cooling" (Ref. 4).

- 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 38, "Containment Heat Removal" (Ref. 5).

- 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 41, "Containment Atmosphere Cleanup" (Ref. 6).

Related Guidance

- RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" (Ref. 7). This RG provides additional information on calculation of net positive suction head.

Purpose of Regulatory Guides

The NRC issues RGs to describe to the public methods that the NRC staff considers acceptable for use in implementing specific parts of the agency's regulations and to provide guidance to licensees and applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required. Methods and solutions that differ from those set forth in regulatory guides will be deemed acceptable if a basis acceptable to the NRC for the specific application is provided and it meets the applicable regulatory requirement.

Paperwork Reduction Act

This regulatory guide contains and references information collections covered by 10 CFR Part 50 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget (OMB), control number 3150-0011.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

B. DISCUSSION

Reason for Issuance

This guide addresses the risk-informed alternative in 10 CFR 50.46c(e) and the corresponding reporting and corrective actions in 10 CFR 50.46c(m). 10 CFR 50.46c(e) allows entities to address the effects of debris on long-term core cooling using a risk-informed approach as an alternative to deterministic approaches, which typically rely on plant-specific or generic performance tests that use conservative test protocols and do not allow credit for nonsafety-related mitigation capabilities. This guide is intended to describe a risk-informed approach acceptable to the NRC that entities can use in addressing the effects of debris on long-term core cooling.

Background

The NRC's risk-informed approach includes consideration of risk, defense in depth, and safety margins, and the NRC expects entities to implement performance measurement strategies to ensure these principles continue to be addressed. This RG does not change these principles, but rather builds on existing guidance and provides additional detail for the specific risk-informed analysis of the effects of debris on ECCS long-term cooling performance. RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis" (Ref. 8); RG 1.200[1], "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Ref. 9); and RG 1.82 are relied upon as set forth in Section C and the appendices of this RG.

The risk-informed alternative for consideration of effects of debris during post-accident long-term core cooling in 10 CFR 50.46c implements Commission direction in the Staff Requirements Memorandum (SRM) related to Commission paper (SECY paper) SECY-12-0093, "Closure Options for Generic Safety Issue – 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance" (Ref. 10) and in the SRM related to SECY-12-0034, "Proposed Rulemaking – 10 CFR 50.46c: Emergency Core Cooling System Performance During Loss-of-Coolant Accidents (RIN 3150-AH42)" (Ref. 11). Without this alternative, entities would need to seek exemptions from the rule to use the risk-informed approach.

Efforts have been focused in the past on ascertaining the reliability of ECCSs in nuclear power plants during design-basis accidents. The performance of sump strainers for recirculation of cooling water could be challenged by the presence of debris - whether already present in the containment or generated as a result of an initiating event such as a LOCA. RG 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," Revision 0 (Ref. 12), stated that licensees should assume a 50-percent blockage for recirculation sump strainers in their analyses. Generic Letter (GL) 85-22, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage" (Ref. 13), later called for replacement of the 50-percent blockage assumption with a more comprehensive requirement to assess debris effects on a plant-specific basis.

A number of events occurred during the 1990s that motivated re-examination of the reliability of ECCS strainers during accident conditions at operating boiling-water reactors (BWRs). The NRC requested that BWR licensees implement appropriate procedural measures, maintenance practices, and

[1] The reference to RG 1.200 is intended to refer to the current revision of RG 1.200. The reference to NUREG-1855 refers to the publicly available pre-publication version of the NUREG; when Revision 1 to NUREG-1855 is published, the final published version of NUREG-1855, Revision 1, should be used.

plant modifications to minimize the potential for the clogging of ECCS suction strainers by debris accumulation following a LOCA. The BWR-related research led to the discovery of issues related to the adequacy of PWR strainer designs in general. The BWR research findings demonstrated that the amount of debris generated by a high-energy line break (HELB) in a PWR could be greater, that the debris could be finer (and thus more easily transportable), and that certain combinations of debris (e.g., fibrous material plus particulate material) could result in a substantially greater head loss through ECCS strainers than an equivalent amount of either type of debris alone. The NRC opened Generic Safety Issue (GSI)-191, “Assessment of Debris Accumulation on PWR Sump Performance” (Ref. 14), to track these issues. The objective of GSI-191 is to ensure that post-accident debris blockage will not impede or prevent the operation of the ECCS or containment spray system (CSS) in recirculation mode at PWRs during LOCAs or other HELB accidents for which recirculation is required.

The NRC issued GL 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors” (Ref. 15), requesting holders of operating licenses for PWRs to address GSI-191. Specifically, licensees were requested to perform a mechanistic evaluation of the recirculation functions and, as appropriate, to take additional actions, such as plant modifications, to ensure system functionality. From the results of testing and analyses, the NRC identified additional issues, such as the combined effect of chemicals and debris on strainer performance and the effects of debris penetration through the strainer and into the reactor vessel and reactor coolant system.

In response to GL 2004-02, most licensees have made major modifications to their plants to provide assurance of adequate recirculation system performance. For example, most licensees have significantly increased the size of strainers, and some have replaced fibrous insulation with reflective metal insulation, which is considered less likely to reach or impede flow through strainers. Demonstrating adequate performance of strainers is challenging given the difficulty of addressing all conditions (e.g., temperatures, debris amounts and compositions, and operating components of the ECCS and CSS) that might exist during an accident. It is also difficult to develop reasonable, reliable, and validated models for strainer performance operating under complex conditions.

The NRC staff prepared SECY-12-0093 and SECY-12-0034 to include risk-informed options for addressing GSI-191. The Commission issued SRMs for SECY-12-0093 and SECY-12-0034 directing the staff to propose revised regulations in 10 CFR 50.46c to contain a provision allowing GSI-191 to be addressed, on a case-by-case basis, using risk-informed alternatives, without the need for an exemption (e.g., under 10 CFR 50.12, “Specific Exemptions”). The objective of this RG is to provide guidance to entities that choose to use the risk-informed approach for addressing the effect of debris on post-accident long-term core cooling. This guidance is consistent with RG 1.174, and it may be used by entities to support the staff’s approval of a risk-informed application.

Harmonization with International Standards

The NRC staff reviewed guidance from the International Atomic Energy Agency, International Organization for Standardization, and International Electrotechnical Commission and did not identify any guidance from these organizations that provided useful information specific to the topic of risk-informed consideration of the effects of debris during post-accident long-term core cooling.

Documents Discussed in Staff Regulatory Guidance

This regulatory guide refers to several industry documents (*e.g.*, topical reports) that contain information that may be used in the risk-informed analysis of debris. These industry documents are not approved by the staff in this RG, unless this RG expressly indicates approval of the identified industry document. The staff approval may be conditioned, as stated in this RG. The bases for any of these conditions are set forth in this RG. NRC approval of these references, including any limitations or conditions, is contained in the safety evaluation for those specific documents, which is either included in the final version of topical reports or separately referenced in this regulatory guide. These referenced industry documents are provided as examples of approaches that may be used for specific portions of the risk-informed analysis as set forth herein. In the future, other topical reports or industry documents may be reviewed and endorsed by the NRC staff. This regulatory guide neither endorses nor modifies the previous NRC approval of these industry documents.

C. STAFF REGULATORY GUIDANCE

Regulations in 10 CFR 50.46c(e) require that an application be submitted to the NRC to request the use of the alternative risk-informed approach for consideration of effects of debris during post-accident long-term core cooling. 10 CFR 50.46c(m) includes requirements for reporting, corrective action, and periodic updating of the risk-informed analysis. RG 1.174 describes a general approach to risk-informed regulatory decision-making and discusses specific topics common to all risk-informed regulatory applications. RG 1.200 describes one acceptable approach for determining the technical adequacy of the PRA.

This regulatory guide (RG 1.229) provides guidance specifically for the risk-informed alternative of 10 CFR 50.46c. This section has descriptions of methods, approaches, and data that the NRC staff considers acceptable for meeting the requirements of the regulations cited in the Introduction. The methods, approaches, or data in these regulatory guidance positions are not requirements.

This section of the RG contains overall guidance for implementing the risk-informed approach and describes in appendices two methods whose primary difference centers around calculation of debris-induced head loss at the strainer. The "detailed approach" in Appendix A calculates head loss as a function of conditions at the strainer, while the "simplified" approach in Appendix B compares the expected debris load for each scenario with a previously-completed test that showed acceptable strainer performance with the tested debris load present. Under this simplified approach, a numeric value for head loss is not calculated for each scenario; rather, scenarios that produce and transport debris in excess of the acceptance criteria determined by the test are assumed to lead to core damage. The following steps are common to both methods:

1. Systematic risk assessment of debris. The rule requires that systematic processes be used to evaluate the risk from debris in terms of core damage frequency (CDF) and large early release frequency (LERF).

    a. The systematic risk assessment should consider all hazards, initiating events, and plant operating modes. It should not be limited to design-basis accidents, licensing basis events, specific plant operating modes, or specific initiating events such as LOCA.

    b. A screening process may be used to justify removing certain hazards, initiating events and plant operating modes based on not being relevant or affected by debris; insignificant contribution, or otherwise not being important to the regulatory decision. One acceptable approach is described in NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making" (Ref. 17). PRA information used to support the screening process should meet the technical adequacy guidance contained in RG 1.200.

        For LOCA events or other scenarios where the effects of debris may be location-dependent, the amount of debris generated and transported for each such location should be determined by the analyst and included as part of the license application (see C.9). In the case of LOCAs, for example, the analyst should determine the locations where a LOCA could occur (both piping and non-piping). Due to inherent uncertainties associated with LOCA frequencies, no break location or LOCA scenario should be screened from the analysis strictly due to its assumed low frequency of occurrence.

c. The increase in risk from debris (i.e., the “risk attributable to debris”) is defined as the difference in risk calculated considering debris effects and the risk calculated assuming debris is not present. This must include both the increase in CDF (ΔCDF) and increase in LERF (ΔLERF). Qualitative or bounding approaches may be used to estimate or bound the increase in CDF and LERF from debris. In using bounding approaches, the analyst must assure that the approach does not result in underestimating the increase in risk estimates.

d. The rule requires that, at a minimum, a plant-specific at-power, internal events PRA be used in the evaluation of the risk of debris. PRA information used to support the systematic risk assessment of debris should be technically adequate, commensurate with its use in this application. RG 1.200 provides one acceptable approach for determining whether the technical adequacy of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results. For some simplified approaches (e.g., the simplified approach in Appendix B of this RG), it may not be necessary to change the existing at-power, internal events PRA to adequately assess or bound the risk of debris (ΔCDF and ΔLERF). It is consistent with the rule and therefore acceptable to assess the risk of debris in a simplified manner and to use the PRA required by the rule to provide insights, for example, to:

    (1) estimate the internal events portion of total plant CDF and LERF, which are used along with ΔCDF and ΔLERF from debris when using Figures 4 and 5 of RG 1.174;

    (2) ensure completeness of internal initiating events considered in the risk evaluation;

    (3) aid in screening of scenarios from further consideration;

    (4) determine relative frequencies of being in a given plant operating condition and

    (5) justify LERF estimates based on the type of sequences affected by debris.

e. The rule allows other risk assessment techniques, including PRA, margin-based methods, bounding risk assessments, or other approaches to address hazards, initiating events, and plant operating modes that are not covered in the required plant-specific, at-power, internal events PRA. Justification should be given for such techniques used as part of the risk evaluation.

f. The risk evaluation may rely on engineering calculations, tests, and other supporting information. The rule considers this supporting information to be part of the evaluation of the risk of debris (i.e., the systematic risk assessment) and therefore subject to the quality assurance, configuration control, performance measurement, reporting, and corrective action requirements in the rule. Guidance on complying with these aspects, which is applicable to the overall risk-informed approach, is found later in this RG.

g. The rule requires any risk increase attributable to debris to be small. Figures 4 and 5 of RG 1.174 provide the acceptance criteria for meeting this requirement for CDF and LERF, respectively. When using Figures 4 and 5 of RG 1.174, the mean values of ΔCDF and ΔLERF should be computed with respect to risk of the plant assuming that there are no debris effects. As stated in RG 1.174, the CDF and LERF on the horizontal axis on

Figures 4 and 5 should be the total risk estimates for the plant as described in Section 2.4 of that regulatory guide.

h. Specific approaches for the systematic risk assessment that are acceptable to the staff for PWRs addressing GSI-191 are contained in Appendices A and B of this RG. Appendix A provides a realistic and comprehensive approach acceptable to the NRC for evaluating the risk of debris. Appendix B offers a simplified approach that is also acceptable to the NRC. Future appendices may be added as additional guidance is sought by entities covered under 10 CFR 50.46c.

The simplified approach determines strainer and in-vessel debris loads above which long-term cooling is assumed to fail. The debris limits should be based on deterministic methods acceptable to the staff. Any scenario that results in an amount of debris greater than the limits is assumed to result in core damage. If the simplified approach shows that no scenarios result in debris greater than the limits for strainer failure or core damage then the analyst should consider using a deterministic approach to comply with 10 CFR 50.46c. Simplified approaches may not require all of the areas in this guidance to be addressed. If the approach selected by the analyst does not rely on specific aspects addressed in the guidance, the application does not need to address these areas. An analyst may also use the simplified approach to screen some scenarios from consideration in the detailed model. If the applicant demonstrates that some scenarios result in debris amounts that will not challenge long-term cooling, these scenarios do not need to be evaluated in the more complex model but can be considered to lead to successful long-term cooling.

2. Initiating Event Frequencies. Initiating event frequencies should be developed consistent with the ASME/ANS PRA Standard, as endorsed by RG 1.200[2]. In general, initiating event frequencies are plant-wide. However, the effect of some initiating events important to generating or transporting debris, such as a pipe break, may be highly location-dependent. Therefore, it may be necessary for the analyst to apportion the overall initiating event frequency to specific locations. The analyst should ensure that any apportioning preserves the overall initiating event frequency and does not result in truncation of sequences based solely on the low frequency resulting from the apportioning. It is anticipated that many of the initiating events that remain after screening will involve LOCAs. NUREG-1829, "Estimating Loss-of-Coolant Accident Frequencies Through the Elicitation Process" (Ref. 18), may be employed as the source document for plant-wide LOCA frequencies. The following guidance should be applied when using the NUREG-1829 LOCA frequencies:

a. Break locations: The analyst may desire to estimate the frequency associated with a specific break location. In the discussion below, it is assumed that all break locations within the scope of the risk assessment have been identified and that the debris that could be generated and transported as a result of the largest possible rupture (e.g., double-ended guillotine break (DEGB) for a pipe) has been determined for each location. Any rupture which could produce and transport sufficient debris to result in core damage is referred to as "*critical*" in the following discussion.

---

[2] The reference to the ASME/ANS PRA standard is intended to refer to the revision of the ASME/ANS PRA standard that is endorsed in the current revision of RG 1.200.

For long-term material degradation in piping components, NUREG-1829 states that "welds are almost universally recognized as likely failure locations because they can have relatively high residual stress, are preferentially attacked by many degradation mechanisms, and are most likely to have preexisting fabrication defects." Therefore, for piping components, the analyst may assume that all breaks occur at weld locations. For non-piping components, the likely failure locations (nozzle, manway, etc.) depend on the type of component. While analysts can refer to NUREG-1829 for guidance, all potential non-piping failure locations should be assessed and justification provided for any locations that are screened out from further analysis.

Some piping and non-piping locations would be unlikely to break due to long-term material degradation under loads consistent with normal operations, but may rupture under acute stresses caused by site-specific LOCA contributors such as a seismic or water hammer event. Additional guidance on analyzing the risk contribution from these initiators is contained in Section C.2.g of this RG.

b. Aggregation method for LOCA frequencies: NUREG-1829 does not advocate any specific aggregation method for combining the results of the expert elicitation, but does note that the LOCA frequency results are sensitive to the selection of aggregation method. For the purposes of this guide, the use of the NUREG-1829 frequencies determined using arithmetic or mixture distribution aggregation is acceptable. Use of LOCA frequencies derived from one of the alternative aggregation methods presented in NUREG-1829 (such as geometric mean aggregation) should be justified by the analyst, and alternatives should be considered in uncertainty analyses (see Section C.4). The analyst should demonstrate that conclusions would not be significantly different from the conclusions reached through the use of alternative aggregation methods by comparing the results with those obtained using arithmetic or mixture distribution aggregation.

c. Interpolation of NUREG-1829 plant-wide LOCA frequencies: The analyst will need to employ and justify a suitable interpolation scheme for specific break sizes not in the NUREG-1829 tables. A semi-log interpolation scheme (i.e., linear interpolation between break sizes and log interpolation between frequencies) is acceptable to the staff since the break sizes are generally within an order of magnitude; the frequencies span several orders of magnitude; and there is approximately a one half order of magnitude difference between successive LOCA categories.

d. Apportionment of LOCA frequencies: The analyst should identify the smallest critical break location and interpolate the NUREG-1829 plant-wide frequency estimates to obtain the overall frequency of having a LOCA of this break size or larger. This represents an upper bound for ΔCDF that is acceptable for estimating the portion of risk attributable to debris. For example, if the smallest critical break location has an effective break size of D_{min}, then $\Delta CDF = f(LOCA\ x \geq D_{min})$. Additional guidance is contained in Appendix C.

e. Parametric uncertainty: LOCA frequencies should be represented by probability density functions to be used in the risk assessment, unless a simplified, bounding risk assessment is used such as set forth in Appendix B of this RG. For the detailed approach in Appendix A, these distributions would be input to the parametric uncertainty. See Section C.4 for additional guidance on uncertainty.

f. Applicability of NUREG-1829 results: The analyst should confirm that the NUREG-1829 values are applicable to the specific unit under evaluation. The NUREG-1829 LOCA frequencies represent generic, or average, estimates for the commercial fleet and are not meant to represent a specific site or design. Furthermore, the uncertainty bounds listed in the NUREG represent the uncertainty of the experts with respect to these generic estimates, rather than bounding values associated with one or two plants.

The experts developed these generic estimates using representative assumptions about important variables such as material conditions, plant geometry, degradation mechanisms, loading, and maintenance practices. The experts also assumed normal plant operational cycles and loading histories (e.g., pressure, thermal, residual). Finally, the experts assumed that plant construction and operation comply with all applicable codes and standards required by regulation and technical specifications. A summary of these assumptions is contained in the executive summary to NUREG-1829, beginning on page xv.

The NRC considers the LOCA frequencies in NUREG-1829 to be acceptable for licensees implementing RG 1.229, provided that the analyst demonstrates that the results of NUREG-1829 would apply to the plant in question; i.e., the plant is not an outlier. To do this, the licensee should provide a qualitative discussion of the previously mentioned variables (e.g., material conditions, plant geometry, loading history, etc.) that provides reasonable confidence that no unusual site-specific conditions exist (e.g., one-of-a-kind material, unusual plant geometry, etc.).

As stated in NUREG-1829, the purposes and context of an application should be considered when determining how to use elicitation results. A qualitative streamlined method for demonstrating applicability of the NUREG-1829 frequencies is appropriate because the scope of 50.46c(e) is limited to a very narrow set of conditions (i.e., the effect of debris on long term core cooling).

Other applications that, if approved, would lead to broader effects on plant operation and design (e.g., power uprates, ECCS design changes, technical specification changes) may require additional analyses to confirm the site-specific applicability of the NUREG-1829 LOCA frequencies. Analysts seeking to use the NUREG-1829 LOCA frequencies for purposes outside of RG 1.229 should refer to application-specific guidance.

g. Site-specific LOCA contributors: The analyst should be aware that the NUREG-1829 LOCA frequencies include only breaks caused by long-term material degradation. Other potential contributors to LOCA frequency such as water hammer and seismically-induced LOCA (both direct and indirect) should be evaluated separately.[3] The contribution from any site-specific LOCA contributor that is shown to be significant should be included in the final calculation of the risk attributable to debris.

One acceptable approach for evaluating direct seismically-induced LOCA is for the analyst to demonstrate that the subject plant is bounded by the results of NUREG-1903. In this bounding analysis, the applicant should demonstrate that the plant-specific primary loop piping (PLP) stresses, materials, material properties (including any aging-related property changes), and site-specific hazard information individually falls within,

[3] A “direct” seismically-induced LOCA involves rupture of a piping or non-piping component caused by the seismic event itself. An “indirect” seismically-induced LOCA is caused by, for example, failure of piping or component supports that leads to the consequential failure of the piping or non-piping component.

or is bounded by, the ranges considered in NUREG-1903. Additionally, the plant-specific combination of PLP stresses, materials, material properties, and site-specific hazard information should also be bounded by evaluations within NUREG-1903. If these conditions are satisfied, the bounding critical flaw depth calculated in NUREG-1903 (i.e., approximately 35 percent for a 1E-5/yr seismic event or 25 percent for a 1E-6/yr seismic event for thermally aged stainless steel weld properties) will also bound the value that would be calculated for the specific plant. The licensee should next assess the likelihood that the flaws leading to rupture under the aforementioned seismic stresses to determine the risk significance of seismically-induced LOCAs.

One acceptable approach for evaluating indirect seismically-induced LOCA is for the analyst to use the method described by NUREG-1903, Section 4.6. This analysis should replace the "representative" values in the NUREG with site-specific fragility and hazard information that, as appropriate, accounts for any effects of material degradation or aging. Alternatively, the analyst may demonstrate that the representative values are bounding for the site in question with consideration of effects due to material degradation or aging.

One acceptable approach for evaluating water hammer is for the analyst to verify that the potential for water hammer is not likely to cause pipe rupture in the break locations that can produce and transport problematic debris. Water hammer includes various unanticipated high-frequency hydrodynamic events, such as steam hammer and water slugging. To demonstrate that component failure risk due to water hammer is acceptably low, the analyst should take the following actions:

- Assess historical frequencies of water hammer events affecting break locations (piping and non-piping) that could generate and transport debris
- Evaluate operating procedures and conditions and demonstrate that they are effective in precluding water hammer.
- Alternatively, the applicant can demonstrate the following:
  - Plant changes, such as the use of J-tubes, vacuum breakers, and jockey pumps, coupled with improved operating procedures, have been used to successfully mitigate water hammer events.
  - Measures used to abate water hammer frequency and magnitude have been effective over the licensing period of the plant.

Again, the acceptability of the approaches for addressing water hammer and seismically-induced LOCAs described in this RG are not necessarily acceptable for other applications involving these site-specific LOCA contributors. Analysts should refer to application-specific guidance.

h. If the information from NUREG-1829 is not used to estimate LOCA frequencies, the analyst should justify LOCA initiating event frequencies, the uncertainty associated with these estimates, and the impact of this uncertainty on the results and conclusions of the analysis.

i. NUREG-1829 contains different summary tables for 0.05, 0.5, and 0.95 LOCA frequency quantiles and mean frequencies derived using different approaches to aggregate elicitations from the individual elicited experts. If a statistical distribution is chosen to represent the uncertainty about parameters in the risk assessment, it should preserve the mean values of the initiating event frequencies from original source documents, such as NUREG-1829.

3. Defense in Depth and Safety Margins. 10 CFR 50.46c(e)(1)(2) requires that sufficient defense in depth and safety margins be maintained. Section C.2.1.1 of RG 1.174 gives seven elements that can be used to demonstrate consistency with the defense-in-depth philosophy. For the risk-informed evaluation of debris effects, additional guidance for each element is given below.

   a. Reasonable balance should be preserved among prevention of core damage, prevention of containment failure or bypass, and mitigation of consequences of an offsite release. The evaluation should address the effect of debris-related failure modes on the ECCS (prevention of core damage), on the containment systems (prevention of containment failure), and on emergency preparedness (consequence mitigation). Examples of defense-in-depth measures can be found in a paper by the Nuclear Energy Institute (NEI), "Example Pressurized Water Reactor Defense-in-Depth Measures For GSI-191, PWR Sump Performance," (Ref. 19). The analyst should consider the effect of debris on the availability and reliability of each level of defense (i.e., prevention of core damage, prevention of containment failure or bypass, and mitigation of the consequences of an offsite release), as well as the combined effect.

   b. There should not be an over-reliance on programmatic activities to compensate for weaknesses in plant design. The analyst should evaluate programmatic activities relevant to the effects of debris including, but not limited to, design controls to limit debris, the inservice inspection (ISI) program, plant personnel training, the reactor coolant system leak detection program, and containment cleanliness inspection activities.

   c. System redundancy, independence, and diversity should be preserved commensurate with the expected frequency of challenges, consequences of failure of the system, and associated uncertainties in determining these parameters. If systems that could be affected by debris are modified, added, or removed, the analyst should address the effect of such changes on redundancy, independence, and diversity. Absent such system changes, the evaluation may conclude that this element of defense in depth is met. Note that common cause failures are addressed by the next element so they do not have to be addressed under this element.

   d. Defenses against potential common cause failures should be preserved and the potential for the introduction of new common cause failure mechanisms are to be assessed and addressed. The analyst should assess the effect of debris on intersystem (e.g., among low-head and high-head injection systems) and intrasystem (e.g., among trains of a given system) availability and reliability. It is recognized that debris is a new common cause failure mechanism compared to a debris-free plant. The analyst should identify the design and operational measures that mitigate the potential for debris-related common cause failures. Some examples could include: securing containment spray pumps and trains of ECCS, refilling the external water source, switching among operating trains, back flushing strainers.

   e. Independence of barriers should not be degraded. As stated in RG 1.174, a *barrier* is a layer of defense against core damage, containment failure, or bypass, and not necessarily a physical barrier. The analysis should include a description of a realistic plant response to each debris-related failure mode identified in the systematic risk assessment. The analyst should assume that a debris-related failure mode has occurred (i.e., a corresponding barrier has failed) and should identify the remaining plant equipment or mitigative measures (i.e., remaining barriers) that can be independently relied upon. For example, if strainer mechanical collapse occurred because of debris, the ECCS may not be sufficient to prevent core damage. The next barrier would be containment structures. Therefore, the analyst should demonstrate, qualitatively or quantitatively, that reasonable confidence exists that the containment would remain as an effective independent barrier for these scenarios.

Examples of defense-in-depth or mitigative measures can be found in NEI's letter "Example Pressurized Water Reactor Defense-in-Depth Measures for GSI-191, PWR Sump Performance." Equipment (e.g., containment fan coolers) and operator actions that would not be compromised by this debris-related failure mode should be described and credited as contributing to barrier independence. When performing this step, analysts may take into account how plant conditions vary over time. For example, when evaluating containment performance following assumed strainer structural failure, the analysis may assume thermal-hydraulic conditions consistent with the time that strainer failure would realistically be assumed to occur. Realistic containment properties (e.g., ultimate tensile strength) may be used for this portion of the analysis.

f. Defenses against human errors should be preserved. The analysis should include a discussion of any operator actions for the plant with debris that would not exist in a debris-free plant. The analyst should discuss the feasibility of these debris-related operator actions and any adverse effect on other required operator actions (e.g., any impact on crew workload). The analyst should justify that any human errors, in general, will not be significantly more likely compared to the clean plant.

g. The intent of the plant's design criteria should be maintained. The analyst should confirm that no debris-related failure could completely disable multiple layers of defense between the fission product source term and the public.

The analysis should demonstrate that sufficient safety margins are maintained when debris is present in the as-built and as-operated plant. This demonstration may be qualitative or quantitative and should address safety margins associated with both the design-basis aspects (e.g., effect on SSCs, flow rates, temperatures, pressures) as well as with any realistic assumptions used in the systematic risk assessment. In a fundamental sense, margin is the difference between some limit and a value that may be attained by a parameter. Assumptions about the limit and actual parameter values should be consistent with licensing-basis calculations unless otherwise justified. For example, if the licensing basis calculations use a given value for the required net positive suction head, then the risk model should also use this value, or a justification should be provided if a different value is considered.

4. Uncertainty

Consistent with RG 1.174, comparisons to the risk acceptance guidelines should be made with appropriate consideration of the uncertainties involved. The fundamental objective of an uncertainty evaluation is to provide confidence that the risk acceptance guidelines are met. For the purposes of this application, NUREG-1855 provides an acceptable method for treating uncertainty. Approaches other than NUREG-1855 that achieve this objective are also acceptable. Analysts should note that "consideration" of a source of uncertainty does not necessarily mean that its effect is quantified. Bounding approaches, screening, and sensitivity studies are examples of alternative methods that are acceptable, provided the guidance in NUREG-1855 is followed.

In addition, portions of the analysis using NRC staff-accepted deterministic methods do not require quantification of uncertainty (model or parametric). The NRC considers the accepted deterministic methods to be conservative enough to compensate for uncertainty. The NRC recognizes that some methods that were accepted in the past are currently not considered to contain significant conservatism; however, the most recent methods, for example those accepted in RG 1.82, are considered to be adequately conservative.

Analysts should apply their chosen approach to all sources of uncertainty that could affect the decision being made (i.e., whether the RG 1.174 risk acceptance guidelines are met). This includes, but is not necessarily limited to:

- initiating event frequency (plant-wide and location-specific)
- debris generation
- debris transport
- head loss at strainer
- chemical effects
- strainer penetration
- downstream effects (in-vessel and ex-vessel)
- calculation of the baseline CDF and LERF (for comparison to risk acceptance guidelines)

NUREG-1855 contains guidance that is applicable to a variety of risk-informed applications. When applying this guidance to the systematic risk assessment, the following considerations apply:

- The methodologies (models, assumptions, etc.) described in the staff's Safety Evaluation of Nuclear Energy Institute (NEI)-04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology. Volume 2 – Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02, Revision 0, December 6, 2004" (Ref. 20), are considered to meet the definition of "consensus models" found in NUREG-1855, Section 7.2.4.

- When no consensus models exist (e.g., choice of the arithmetic mean or geometric mean for aggregation of expert's input to an elicitation process), the analyst's choice of one method over another represents a key assumption and, therefore, a key source of uncertainty. The analyst should follow the guidance in NUREG-1855.

- When considering parameter uncertainty, the analyst should account for any dependency or correlation that might exist among various parameters. For example, a phenomenological model may rely on sump temperatures, containment pressures, and pumping rates. If a correlation exists between any of these parameters, this should be accounted for in the analysis. For the purposes of uncertainty quantification, this correlation must be maintained even across models (e.g., if a parameter is used in both the PRA and a phenomenological model).

5. Monitoring Program

Entities selecting the risk-informed alternative are required by 10 CFR 50.46c(e)(1)(v) to use a monitoring program that ensures the acceptance criteria in 10 CFR 50.46c(e)(1)(i) and (ii) will continue to be met. The monitoring program must assess the effects of design or plant modifications, procedure changes, as-found conditions, identified changes or errors in the analysis, industry operating experience, and any other information that could result in increased risk, or decreased defense in depth or safety margins, under the alternative risk-informed approach.

The implementation and monitoring program may be partially or fully implemented using existing licensee programs, with enhancements if necessary to account for the unique aspects of calculating the portion of CDF and LERF attributable to debris. For example, there may be existing programs that track risk based on equipment availability and reliability. However, such programs may not generally be suited to evaluating, for example, the discovery of a large quantity of degraded coatings that could contribute to the debris source term. Plant work control processes may need to be modified to ensure that planned modifications do not result in unacceptable risk of debris or loss of adequate defense in depth or safety margins. As another example, if the systematic risk assessment of debris credited the removal of a problematic insulation type, plant work control practices should be in place to prevent its future introduction.

Consistent with RG 1.174, the results (e.g., tracking and trending data) of this monitoring program should be retained onsite for inspection.

6. Quality Assurance. 10 CFR 50.46c(e)(1)(iv) requires that the risk-informed approach be performed under a quality assurance program. This should also be the case for the periodic update of the risk-informed analysis, discussed below. The NRC does not expect the risk-informed analysis to be performed under 10 CFR 50, Appendix B, but rather under a QA program that includes at least the four pertinent quality assurance requirements from the Appendix B QA program that are set forth in Section C.5 of RG 1.174. These quality assurance requirements of 10 CFR 50, Appendix B are *pertinent* because the risk-informed analysis of debris (including the systematic risk assessment) is needed to demonstrate that the design of safety-related SSCs meets NRC requirements.

7. Periodic Update of Risk-Informed Analysis. Licensees are required by 10 CFR 50.46c(m)(8) to update the risk-informed analysis within at least 48 months since initial approval by the NRC or since the latest update. The update needs to include all parts of the risk-informed evaluation: the systematic risk assessment, consideration of defense in depth, and consideration of safety margins. As described in 10 CFR 50.46c(e)(1)(iii), the systematic risk assessment includes: internal events, at-power PRA, other risk assessment methods, and the engineering calculations, tests, and other supporting information used in the risk assessment. The intent of the update is to capture the effects of any plant changes, procedure changes, or new information on the risk-informed analysis and to confirm that the acceptance criteria are still maintained. Note that the risk acceptance criteria depend upon the total risk of the plant (i.e., CDF and LERF) and the risk attributable to debris (i.e., ΔCDF and ΔLERF).

   Entities may use their existing PRA maintenance and update process that is consistent with the guidance in RG 1.200 to accomplish the update required by 10 CFR 50.46c(m)(8), but should confirm that, as a minimum, the risk-informed analysis of debris appropriately captures changes in the plant, procedures, and operating experience. The entity should also include any new information on LOCA frequencies that may be developed.

   Analysts using the LOCA frequencies in NUREG-1829 should adhere to the guidance on the effects of plant operating time on total LOCA frequency stated in Section 7.4 of NUREG-1829. The expert-elicited LOCA frequencies were developed using the best available information at the time NUREG-1829 was published (2008) and were assumed to be "stable" for 15 years. Periodic updates that occur after this 15 year period should confirm that these frequencies continue to reflect the latest available information. Examples of factors that may drive future LOCA frequencies up or down include discovery of new degradation mechanisms and additional experience on the effectiveness of primary water stress corrosion cracking mitigation practices. Although the effect of these factors cannot be predicted, the periodic update process should ensure that the LOCA frequencies reasonably reflect any new information that is gained in this area.

8. Reporting and Corrective Actions. Licensees are required by 10 CFR 50.46c(m)(6) to make a report to the NRC in the event that risk of debris exceeds the NRC acceptance criteria, or in the event that defense in depth or safety margins have decreased from the NRC-approved analysis. The rule requires that the report be submitted in accordance with 10 CFR 50.72 or 50.73.

   Corrective action is required under 10 CFR 50.46c(m)(7) if it is determined that either the risk acceptance criteria have been exceeded or the defense in depth or safety margins credited by the assessment have not been maintained. The risk acceptance criteria are those in Section C.1.g of this RG. Timely action to reduce the risk to within the acceptance criteria or to restore defense in depth or safety margins would have to be taken.

9. License Application. An entity seeking to use the risk-informed approach is required by 10 CFR 50.46c(e)(2) to submit an application that includes: (1) A description of the risk-informed approach; (2) A description of the measures taken to assure that the scope, level of detail, and technical adequacy of the systematic processes that evaluate the plant for internal and external events initiated during full power, low power, and shutdown operation are commensurate with the reliance on risk information; and, (3) A description of, and basis for acceptability of, the evaluations of risk, defense in depth, safety margins, and the monitoring program. The entity should include the following in its application:

   a. A description of the systematic risk assessment of debris and the assessment of adequate defense in depth and safety margins.

   b. A description of the measures taken to assure the scope, level of detail, and technical adequacy of the systematic risk assessment and a description of the QA program under which it was performed.

   c. A description of the monitoring program, including how existing programs may be used or modified to fulfill the requirement in 10 CFR 50.46c, and of the periodic update process.

   d. Final Safety Analysis Report (FSAR) or Updated FSAR pages, as appropriate, listing applicable design, plant, and operational capabilities of defense in depth and safety margins with respect to the use of the risk-informed alternative.

   e. Results of the risk assessment, including CDF, ΔCDF, LERF, ΔLERF, and the uncertainty evaluation.

   f. Key aspects of the plant that limit the magnitude of the CDF and LERF when accounting for effects of debris, such as the following:

      - frequency of events that could produce significant amounts of debris
      - amount of debris produced
      - resiliency of strainer system to failure under the presence of debris

- debris filtration by strainers
- resiliency of the system to provide adequate core cooling under debris presence
- resiliency of the system to limit large early releases
- alternate flow paths for cooling of the core, if credited

g. For LOCAs and other initiating events that are location-dependent, a listing of any locations where the amount of debris that could be generated and transported would be expected to fail the strainers or block the core. For example: Assume that, in the simplified approach of Appendix B of this RG, testing shows that 100 pounds mass of a certain type of debris is the threshold for failure. If a pipe-break LOCA could produce and transport, for example, 1000 pounds mass of the debris, failure would be almost certain, even considering the conservatisms in the testing that developed the threshold value. The analyst should specify any compensatory measures or elements of defense in depth that would mitigate the effects of such events.

h. A summary of the evaluation made to address areas in the RG where justifications are noted as necessary to validate the analysis, and a list or table summarizing qualitative assessments, justifying key assumptions and providing a basis for any deviations from this guidance.

D. IMPLEMENTATION

The purpose of this section is to provide information on how applicants and licensees[4] may use this guide and information regarding the NRC's plans for using this regulatory guide. In addition, it describes how the NRC staff complies with 10 CFR 50.109, "Backfitting" and any applicable finality provisions in 10 CFR Part 52 "Licenses, Certifications, and Approvals for Nuclear Power Plants."

Use by Licensees

Licensees may voluntarily[5] use the guidance in this document to demonstrate compliance with the underlying NRC regulations. Methods or solutions that differ from those described in this regulatory guide may be deemed acceptable if they provide sufficient basis and information for the NRC staff to verify that the proposed alternative demonstrates compliance with the appropriate NRC regulations.

Licensees may use the information in this regulatory guide for actions which do not require NRC review and approval such as changes to a facility design under 10 CFR 50.59, "Changes, Tests, and Experiments," that do not require prior NRC review and approval. Licensees may use the information in this regulatory guide or applicable parts to resolve regulatory or inspection issues.

Use by NRC Staff

The NRC staff does not intend or approve any imposition or backfitting of the guidance in this regulatory guide. The NRC staff does not expect any existing licensee to use or commit to using the guidance in this regulatory guide, unless the licensee makes a change to its licensing basis. The NRC staff does not expect or plan to request licensees to voluntarily adopt this regulatory guide to resolve a generic

[4] In this section, "licensees" refers to licensees of nuclear power plants under 10 CFR Parts 50 and 52; and the term "applicants," refers to applicants for licenses and permits for (or relating to) nuclear power plants under 10 CFR Parts 50 and 52, and applicants for standard design approvals and standard design certifications under 10 CFR Part 52.

[5] In this section, "voluntary" and "voluntarily" mean that the licensee is seeking the action of its own accord, without the force of a legally binding requirement or an NRC representation of further licensing or enforcement action.

regulatory issue. The NRC staff does not expect or plan to initiate NRC regulatory action which would require the use of this regulatory guide. Examples of such unplanned NRC regulatory actions include issuance of an order requiring the use of the regulatory guide, requests for information under 10 CFR 50.54(f) as to whether a licensee intends to commit to use of this regulatory guide, generic communication, or promulgation of a rule requiring the use of this regulatory guide without further backfit consideration.

During regulatory discussions on plant-specific operational issues, the staff may discuss with licensees various actions consistent with staff positions in this regulatory guide, as one acceptable means of meeting the underlying NRC regulatory requirement. Such discussions would not ordinarily be considered backfitting even if prior versions of this regulatory guide are part of the licensing basis of the facility. However, unless this regulatory guide is part of the licensing basis for a facility, the staff may not represent to the licensee that the licensee's failure to comply with the positions in this regulatory guide constitutes a violation.

If an existing licensee voluntarily seeks a license amendment or change and (1) the NRC staff's consideration of the request involves a regulatory issue directly relevant to this new or revised regulatory guide and (2) the specific subject matter of this regulatory guide is an essential consideration in the staff's determination of the acceptability of the licensee's request, then the staff may request that the licensee either follow the guidance in this regulatory guide or provide an equivalent alternative process that demonstrates compliance with the underlying NRC regulatory requirements. This is not considered backfitting as defined in 10 CFR 50.109(a)(1) or a violation of any of the issue finality provisions in 10 CFR Part 52.

If a licensee believes that the NRC is either using this regulatory guide or requesting or requiring the licensee to implement the methods or processes in this regulatory guide in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfit appeal with the NRC in accordance with the guidance in NRC Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection" (Ref. 21), and in NUREG-1409, "Backfitting Guidelines," (Ref. 22).

REFERENCES[6]

1. Title 10, Part 50, Section 46c, of the *Code of Federal Regulations* (10 CFR 50.46c), "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors."

2. 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities."

3. 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 15, "Reactor Coolant System Design."

---

6 All NRC documents that are publicly available may be accessed through the NRC Library on the NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or 800-397-4209; fax 301-415-3548; or e-mail to pdr.resource@nrc.gov.

4. 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 35, "Emergency Core Cooling."

5. 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 38, "Containment Heat Removal."

6. 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 41, "Containment Atmosphere Cleanup."

7. NRC, RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 4, Washington, DC, March 2012.

8. U.S. Nuclear Regulatory Commission (NRC), Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, Washington, DC, May 2011.

9. NRC, RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities."

10. NRC, "Staff Requirements – SECY-12-0093 – Closure Options for Generic Safety Issue-191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance." Washington, DC, December, 14, 2012. (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12349A378).

11. NRC, "Staff Requirements – SECY-12-0034 – Proposed Rulemaking - 10 CFR 50.46c: Emergency Core Cooling System Performance during Loss-of-Coolant Accidents (RIN 3150-AH42)." Washington, DC, January, 7, 2013. (ADAMS Accession No. ML13007A478).

12. NRC, RG 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," Revision 0, Washington, DC, June 1974.

13. NRC, Generic Letter (GL) 85-22, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage" Washington, DC, December 3, 1985. (ADAMS Accession No. ML031150731).

14. NRC, NUREG-0933, Section 3, "New Generic Issues-Issue 191, Assessment of Debris Accumulation on PWR Sump Performance," Revision 2, Washington, DC, 2011.

15. NRC, GL 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," Washington, DC, September 13, 2004.

16. ASME/ANS RA-Sa 2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," American Society of Mechanical Engineers, New York, NY, and American Nuclear Society, La Grange Park, IL, 2009.[7]

17. NRC, NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," Pre-Publication Revision 1, Washington, DC, April 10, 2015. (ADAMS Accession No. ML15026A512).

[7] Copies of ASME standards and documents may be obtained from the American Society of Mechanical Engineers, Three Park Avenue, New York, NY 10016-5990; phone 212-591-8500.

18. NRC, NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," Washington, DC, April 2008. (ADAMS Accession No. ML080630013).

19. Nuclear Energy Institute (NEI), "Example Pressurized Water Reactor Defense-in-Depth Measures for GSI-191, PWR Sump Performance," Washington, DC, March 2012. (ADAMS Accession No. ML120730660).

20. NRC, Report NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology. Volume 2 – Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02, Revision 0, December 6, 2004," Washington, DC, December 2004. (ADAMS Accession No. ML050550156).

21. NRC, Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection," Washington, DC, October 9, 2013. (ADAMS Accession No. ML12059A460).

22. NRC, NUREG-1409, "Backfitting Guidelines," Washington, DC, July 1990 (ADAMS Accession No. ML032230247).

23. NRC, "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Coatings Evaluation," Washington, DC, March 2008 (ADAMS Accession No. ML080230462).

24. Westinghouse and NRC, "Final Safety Evaluation by the Office of Nuclear Reactor Regulation, Topical Report WCAP-16530-NP-A 'Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191,'" Washington, DC, March, 2008. (ADAMS Accession Nos. ML081150383 and ML101230629).

25. NRC, "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Plant-Specific Chemical Effect Evaluations," Washington, DC, March 2008. (ADAMS Accession No. ML080380214).

26. NRC, "Safety Evaluation by the Office of Nuclear Reactor Regulation, Topical Report (TR) WCAP-16406-P, Revision 1, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," Pressurized Water Reactor Owners Group , Project No. 694. Washington, DC, December 20, 2007 (ADAMS Accession No. ML073520295).

27. Westinghouse and NRC. "Final Safety Evaluation for Pressurized Water Reactor Owners Group Topical Report WCAP-16793-NP, Revision 2, 'Evaluation of Long-Term Cooling Considering Particulate Fibrous and Chemical Debris in the Recirculating Fluid,'" Washington, DC, July 2013 (ADAMS Accession No. ML13239A111).

APPENDIX A

A DETAILED APPROACH FOR CONDUCTING THE RISK-INFORMED ANALYSIS OF DEBRIS FOR PWRs

This appendix provides guidance for a detailed risk assessment of debris acceptable to the staff that may be used by pressurized water reactor (PWR) licensees to address Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR Sump Performance." This approach supports the systematic risk assessment. The risk-informed approach must also consider the general guidance set forth in Section C of this regulatory guide (RG), e.g., defense in depth, safety margins, and performance measurement.

A-1. Scope: The systematic risk assessment required by Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46c(e) should include all relevant initiating events and plant operating modes for all hazard groups for which debris could adversely affect core damage frequency (CDF) or large, early release frequency (LERF). Therefore, the analyst should identify and group all scenarios that could be mitigated by the activation of sump recirculation. In this context, the term *scenario* means an initiating event followed by a plant response (e.g., combination of equipment successes, failures, and human actions) leading to a specified end state (e.g., success, core damage, large early release). These scenarios should be grouped in a logical fashion, for example according to initiating event.

The analyst may exclude hazard groups and operating modes from further consideration using the guidance in Section C of this RG. An example of screening criteria that could be used for a PWR might be the following:

As a minimum, any scenario or group of scenarios meeting all of the following four inclusion criteria should be included in the risk-informed analysis:

a. The scenario response involves recirculation to maintain core cooling;

b. The scenario involves the potential for debris inside primary containment that could adversely affect structures, systems or components (SSCs) needed for recirculation;

c. The scenario involves a mechanism that could transport the debris to the sump; and,

d. The debris is necessary for the scenario to result in core damage or containment failure.

A-2: Failure Mode Identification: The analyst should identify the debris-related failure modes for each SSC whose successful operation helps to mitigate the postulated scenarios screened as included under Paragraph A-1 of this appendix. For example, it is expected that the emergency core cooling system (ECCS) would be identified during this step. The ECCS may fail because of the following debris-related failure modes (the list is not exhaustive and other failure modes may need to be considered):

a. Excessive head loss at the strainer leads to loss of net positive suction head (NPSH) margin for adequate operation of the pumps;

b. Excessive head loss at the strainer causes mechanical collapse of the strainer;

c. Excessive head loss at the strainer lowers the fluid pressure, causing release of dissolved gases (i.e., degassing) and void fractions in excess of pump limits. Vortexing and flashing may also cause pump failure;

d. Debris in the system downstream of the strainer exceeds ex-vessel limits (e.g., blocks small passages in downstream components or causes excessive wear);

e. Debris results in core blockage and decay heat is not adequately removed from the fuel;

f. Debris buildup on cladding results in inadequate decay heat removal,

g. Debris buildup in the vessel leads to potential excessive boron concentrations within the core caused by reduction of coolant with relatively low boric acid concentration entering the core; and,

h. Debris prevents adequate flow to the strainer or prevents the strainer from attaining adequate submergence.

The analyst may exclude debris-related failure modes from further consideration if a bounding analysis shows that maximum credible debris loads under detrimental configurations would not lead to a given failure mode. For example, if the analyst uses WCAP-16406, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," to evaluate ex-vessel downstream effects (failure mode d) and it is determined that no failures occur, downstream ex-vessel effects can be "excluded" from the final calculation of risk. However, even if a particular failure mode is excluded, the analysis should still consider direct and indirect effects from the debris source term on the SSC performance for other parts of the analysis.

For example, an analysis may show that a bounding amount of debris would not completely block flow through the residual heat removal (RHR) heat exchanger, with a maximum loss in heat transfer rate not sufficient to significantly change cooling rates and cause core damage. In this example, exclusion of this failure mechanism (i.e., flow blockage) for that RHR heat exchanger might be justifiable. However, the estimated percent reduction in heat transfer rates would still need to be considered when computing temperatures of water volumes inside containment (i.e., pool temperatures), which may affect other failure modes. As another example, analysis may show that the strainer can function with the calculated amount of debris; however, in-vessel limits may be exceeded.

A-3. PRA Model Changes: After identifying and screening relevant scenarios and debris-related failure modes of SSCs, the analyst should evaluate failure modes identified from Paragraph A-2 of this Appendix and determine how to incorporate these failure modes into the probabilistic risk assessment (PRA) model, which is used to calculate CDF and LERF. The "baseline" PRA model for assessing the risk increase attributable to debris is one where the effects of debris are assumed to be negligible. For example, the baseline PRA model might not distinguish between successful actuation of one train of ECCS versus two trains, as either would meet the traditional PRA success criterion for a LOCA. When evaluating the effects of debris, however, the distinction between one and two trains may be important as it may affect the distribution of debris (to one versus two strainers) as well as safety injection flow rates and could, therefore, affect the probability and frequency of ECCS debris-related failure modes. Changes to the PRA should be described in the application to the U.S. Nuclear Regulatory Commission (NRC). Any operator actions credited with reducing the CDF or LERF attributable to debris should be described.

A-4. Submodel Development: For this detailed approach, the analyst should develop the submodels necessary to evaluate strainer and downstream system performance, and then integrate these submodels into the systematic risk assessment. Submodels may need to be developed to address the areas below:

a. debris source term (debris generation mechanisms and debris size distribution);

b. debris transport and accumulation on strainers;

c. strainer head loss and criteria for strainer failure (e.g., available head less than the required net positive suction head, flashing, and deaeration);

d. debris penetration through strainers and downstream effects (such as debris accumulation inside the reactor pressure vessel);

e. chemical effects that could increase flow resistance (for example by the formation of chemical precipitates) and head loss through debris beds on strainers and in the vessel; and,

f. effects of safety-related and nonsafety-related system activation to mitigate the event (e.g., strainer blockage, in-vessel effects, and ex-vessel downstream effects).

Guidance for selected submodels is provided in paragraphs A-6 through A-12. Integration is discussed in paragraph A-13.

A-5. Scenario Development: The analyst should develop descriptions of the as-built and as-operated nuclear power plant including the phenomenological, physical, and mathematical models identified under Paragraph A-4 of this Appendix. The analyst should define the following:

a. plant operating modes and operating components that were not screened out of the risk-informed analysis of debris effects;

b. long-term period of performance, including a definition of the safe and stable end-state of the nuclear power plant (i.e., safe state after mitigation of the event); the 24-hour mission time typically used in PRAs may not be applicable if long-term effects (e.g., chemical precipitation) are expected to occur outside of this time frame;

c. human actions that are part of the accident sequence; and,

d. the set of assumptions and considerations relevant to the development of the systematic risk assessment.

A-6. Debris Source Term: The analyst should describe the source term for generation of debris under a postulated event to be mitigated by activation of the recirculation system.

a. The analyst should describe the postulated accidents and debris generation mechanisms (e.g., pipe break and jet impinging on materials within containment) applicable to the as-built and as-operated plant.

b. The analyst should identify the types of debris or materials that could be generated and transported to the strainer and affect its performance, or otherwise affect core or containment cooling. In addition, the analyst should quantify the potential amounts of debris that could be generated by the initiating events and included scenarios identified under Paragraph A-1of this Appendix. Analysts may refer to RG 1.182, Section C.1.3.3, for guidance on identifying debris types and the use of the zone-of-influence concept to estimate debris amounts. RG 1.82, Sections C.1.3.5, C.1.3.6, and C.1.3.10, provide guidance pertinent to coatings debris, latent debris, and chemical effects. The NRC staff review guidance, "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Coatings Evaluation" (Ref. 23), contains guidance focused on coatings as a source of debris.

c. As necessary for quantifying the head loss through a bed of debris on strainers, the analyst should quantify debris characteristics, including material type, size distribution and shape, and density. The analyst should quantify the amount of debris penetration through or bypass around the strainers. The analysis should account for interactions with chemicals in the water when relevant to strainer failure or core damage mechanisms. Safety evaluations in Nuclear Energy Institute (NEI) document NEI 04-07 (Ref. 20), Sections 3.4.3 and 3.5, provide guidance acceptable to the NRC for quantifying debris characteristics.

d. The analyst should combine information from Paragraphs A-6.a through A-6.c of this appendix into a submodel for quantifying debris amounts after a postulated initiating event. The analyst should verify the validity of the model, relying, for example, on tests and empirical data, analogy to other systems, or comparison with other calculations.

e. If delayed debris generation is modeled, the analyst should justify that the timing is appropriate. It is conservative to assume that all debris is generated at the initiation of the event.

A-7. Debris Transport: Once the amount and type of debris is characterized, the analyst should describe the mechanism for debris transport to the strainers.

a. The calculation of debris quantities transported to the ECCS strainers should consider all modes of debris transport, including blowdown, washdown, pool fill, and recirculation. Section C.1.3.4 of RG 1.82 has guidance on the development of deterministic transport analyses and models.

b. The analyst should develop a model for debris transport to be used in the systematic risk assessment that will be used to calculate CDF and LERF. The transport model should be consistent with water inventory balance (e.g., safety injection flow rates, containment spray system flow rates) related to the postulated event under consideration.

c. The analyst should evaluate the validity of the transport model, relying, for example, on tests and empirical data, analogy to other systems, or detailed computational fluid dynamics models.

d. If delayed transport of debris is modeled the analyst should justify that the timing is appropriate. It is generally conservative for strainer head loss assessment to assume that all debris is at the strainer at the initiation of recirculation. However, this assumption may not be appropriate for modeling debris penetration through the strainer.

A-8. Strainer Evaluation: The analyst should evaluate the fluid conditions at the strainer, properly accounting for the pool water level, the water volume displaced by hardware, post-accident pressure, and water inventory holdup in upstream paths. Relevant guidance is given in RG 1.82, Sections C.1.3.1 and C.1.3.7. In general, RG 1.82 recommends conservatively assuming that the containment pressure is equal to the saturation pressure. The analyst should justify use of pressure beyond atmospheric in NPSH computations.

A-9. Impact of Debris: The analyst should develop and use a model for debris accumulation and head loss through the potential debris bed developed on strainers. Head loss should account for chemical effects as described in A-10 below. The output of this approach is a calculated head loss value for each scenario.

a. Guidance on the development of head loss analyses is given in the safety evaluation of Sections 3.4.3 and 3.5 of NEI 04-07 (Ref. 20), and in RG 1.82, Section C.1.3.11.

b. The analyst should develop a model of head loss through the debris bed on strainers to be used in the systematic risk assessment for the computation of CDF and LERF. The model should represent or bound the broad range of possibilities of debris loads and compositions, as well as pertinent accident conditions.

c. The analyst should evaluate the validity of the model, relying, for example, on tests and empirical data, analogous systems, and use of approved guidance. Section C.1.3.12 of RG 1.82 defines prototypical head loss testing that could be used to support models. The model should be validated for the range of plant-specific conditions and debris loads to which it is being applied. Validation should be based on results of prototypical head loss testing using appropriate debris types.

d. The analysis should address the potential for air entrainment in the fluid or voiding caused by vortexing or excessive head loss across the strainer.

A-10. Chemical Effects: The analysis should account for the presence of chemicals in the water and interactions with debris that could change the head loss through debris beds.

a. The Westinghouse topical report, WCAP-16530-NP-A, and the limitations discussed in the associated NRC staff safety evaluation, "Final Safety Evaluation by the Office of Nuclear Reactor Regulation, Topical Report WCAP-16530-NP-A 'Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191'" (Ref. 24), provide an acceptable approach for the evaluation of chemical effects that may occur in a post-accident containment sump pool. The NRC staff review guidance, "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Plant-Specific Chemical Effect Evaluations" (Ref. 25), provides guidance on plant-specific chemical effect evaluations.

b. The analyst should develop a model of chemical effects on flow resistance and head loss through the debris bed on strainers and within the reactor vessel to be used in the systematic risk assessment for the computation of the CDF and LERF. The model should represent or bound the broad range of conditions described in the safety evaluation of WCAP-16530-NP-A.

c. The analyst should evaluate the validity of the model, relying, for example on tests, empirical data, and analogies to other systems. The safety evaluation of topical report

WCAP-16530-NP-A defines testing and analyses that could be used to support models of chemical effects. The chemical effects model should be validated for the full range of plant conditions and debris loads to which it is applied.

d. The timing of chemical effects should be considered and justified in the model.

A-11. Debris Penetration Evaluation: The analyst should evaluate debris penetration through the strainer.

a. The analyst should characterize debris penetration through strainers under potential accident conditions. The analysis should account for all debris penetration mechanisms or mechanisms where debris can bypass the strainer.

b. The analyst should develop a model to estimate the amount of debris penetration through strainers, with the goal of evaluating downstream effects.

c. The analyst should evaluate the validity of the model, relying, for example, on tests, empirical data, and analogies to other systems. Testing to validate the strainer penetration model should be done under conditions that are prototypical or conservative with respect to the as-built and as-operated plant.

A-12. Debris Penetration Effects: The analyst should evaluate the effects of debris strainer penetration inside (in-vessel) and outside (ex-vessel) the reactor vessel.

a. The analyst should evaluate downstream ex-vessel effects of debris (e.g., blockage of flow paths in equipment, and wear and abrasion of surfaces). The safety evaluation for the Topical Report WCAP-16406-P, "Safety Evaluation by the Office of Nuclear Reactor Regulation, Topical Report (TR) WCAP-16406-P, Revision 1, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191" (Ref. 26), and RG 1.82 provide guidance that the analyst may use to evaluate ex-vessel effects of debris.

b. The safety evaluation for the Topical Report WCAP-16793-NP "Final Safety Evaluation for Pressurized Water Reactor Owners Group Topical Report WCAP-16793-NP, Revision 2, 'Evaluation of Long-Term Cooling Considering Particulate Fibrous and Chemical Debris in the Recirculating Fluid'" (Ref. 27), has guidance to evaluate the effect of debris in recirculating fluid on long-term cooling, including in-vessel effects such as blockage of flow clearances through fuel assemblies. The topical report defines in-vessel debris load limits (i.e., 15 grams of fiber per fuel assembly as transported and accumulated during a hot-leg break) below which testing has demonstrated that long-term core cooling is not impeded. WCAP-16793, Revision 2, has been accepted by the NRC staff (with conditions and limitations) as adequately defining in-vessel debris limits. This topical report, or other NRC accepted topical reports or methods, may be used to define in-vessel debris limits.

c. The analyst should address the potential for boric acid precipitation in its analysis. Analysts may refer to NRC-approved methods and topical reports to evaluate boric acid precipitation.

d. The analyst should develop a model for downstream effects, and clearly show how the model would be used in the estimate of CDF and LERF. In particular, the analysis should properly account for the fraction of total flow that enters the core compared to bypass flows; for example, flow diverted to containment sprays, when calculating debris accumulation in the core. A fraction of the flow that carries debris may be considered not to contribute to debris buildup inside the pressure vessel, such as the flow discharged through breaks or through the containment spray system. The analyst should provide a technical basis for any fraction of the flow considered not to contribute to in-vessel debris buildup. However, note that the debris returned to the pool may pass through the strainer again.

e. Chemical effects should be considered in the downstream effects evaluation.

f. The analyst should evaluate the validity of the downstream effects model, relying, for example, on tests, empirical data, and analogies of similar systems or components.

g. If delayed transport of debris to the reactor vessel is modeled the analyst should justify that the timing is appropriate. It is generally conservative to assume that all debris is present in the reactor at the initiation of recirculation.

A-13. Submodel Integration: The analyst should combine the submodels for the debris source term, debris transport, strainer head loss, chemical interactions, debris penetration, and downstream effects into an integrated model to allow computation of failure probabilities in the modified PRA model to evaluate debris effects (implemented in Paragraph A-14 of this appendix).

a. The integrated model should be structured to allow for propagation of relevant parameter uncertainty (see paragraph C.4 of this RG).

b. Inputs to the integrated model should be consistent with inputs to the modified PRA discussed in Paragraph A-14 in this appendix.

c. The analysis should consider failure modes of SSCs, identified in Paragraph A-2 of this appendix, and the corresponding failure probabilities. The analysis should address the failure of piping and non-piping (e.g., valve bodies, pump casings, manways, control rod penetrations, etc.) passive systems considered in NUREG-1829. The analyst should provide a technical basis for allocating plant-wide initiating event frequencies to location-specific events; refer to paragraph C.2 in the body of this RG.

A-14. Systematic Risk Assessment: The analyst should estimate the change in risk attributable to debris.

a. The analyst should modify the baseline PRA model (i.e., the PRA model that assumes any effects of debris are negligible), consistent with Paragraph A-3 of this Appendix, to perform the calculation of the risk (CDF and LERF) for the as-built and as-operated nuclear power plant.

b. The analysis should use commonly accepted methods and approaches to modify the PRA model for the systematic risk assessment of debris. These methods and approaches should be consistent with the guidance in RG 1.200. For example, if new operator actions are added to the PRA model to account for debris, the human reliability analysis would typically be the same as is used in the base PRA model. Similarly, event tree and fault tree changes made to account for debris would typically use the same approach as used in the peer-reviewed PRA model. The changes made and methods employed to effect those changes should be well described in the application.

c. Changes to the PRA should include revisions of failure frequencies and probabilities and reliability data in general to account for the presence of debris.

d. New human failure events (HFEs) should be added to the model as appropriate. Debris effects on the HFEs in the PRA model should be determined and human error probabilities adjusted accordingly. The dependency among multiple human errors in the same accident sequence, including new HFEs added to the model to account for debris presence, should be assessed and accounted for in the quantification of the PRA model.

e. The inputs to the modified PRA should be consistent with inputs and information used by the various submodels that comprise the integrated model of A-13 of this appendix.

   (1) Common input distributions should be consistently sampled in the modified PRA and in the submodels.

   (2) Common information of the modified PRA and the submodels should be consistently treated, including the use of correlations where needed.

f. Plant states and configurations that are screened into the analysis (e.g., with the procedure in Paragraph A-1 of this appendix) but are not explicitly treated in the systematic risk assessment should be assumed to lead to core damage. Examples may include scenarios that meet all of the aforementioned screening criteria but can be shown to have very low frequency. Rather than model plant performance for these scenarios, the analyst may choose to assign a CCDP of 1.0. The contribution to the CDF and LERF for these unaccounted states and configurations should be quantified.

g. The modified PRA, together with the various submodels that comprise the integrated model of A-13 of this Appendix, should be used to quantify the mean values of CDF and LERF, accounting for debris effects.

APPENDIX B

A SIMPLIFIED APPROACH FOR CONDUCTING THE RISK-INFORMED ANALYSIS OF DEBRIS FOR PWRs

This appendix provides guidance on a simplified risk assessment of debris acceptable to the staff that may be used by pressurized-water reactor (PWR) licensees to address Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR Sump Performance." Note however, that much of the guidance in Appendix A also applies to this simplified approach. The key feature of the simplified approach is that the results of testing are used to set bounding thresholds for the debris amounts below which the effect of debris on structure, system and component (SSC) performance can be assumed to be negligible (i.e., conditional core damage probability (CCDP) = 0). This approach supports the systematic risk assessment. The risk-informed approach must also consider defense in depth, safety margins, and performance measurement as set forth in Section C of this Regulatory Guide (RG).

B-1. Scope, Failure Modes, Scenarios, and Debris: The analyst would use the following guidance of Appendix A to this RG in order to determine the overall scope of the risk assessment.

a. A-1, Scope

b. A-2, Failure Mode Identification

c. A-5, Scenario Development

d. A-6, Debris Source Term

e. A-7, Debris Transport – the analyst may evaluate transport or assume that all of the debris generated, for which limits are necessary, transports to the strainer.

B-2. Impact of Debris: The analyst determines a threshold value for each debris type below which that debris cannot adversely affect SSCs within the scope of the risk analysis. These threshold values are derived from testing that demonstrates that long-term core cooling will be maintained under those debris loads. For each scenario and corresponding debris source term, the analyst compares the debris source term to the appropriate threshold values. Scenarios which produce debris at the strainer exceeding this limit would be assigned a CCDP of 1.0. Scenarios which produce less debris than this limit would be bounded by the test results and would be assigned a CCDP of 0. The analyst may account for changes in thermal-hydraulic conditions over time, such as flow and temperature. For this simplified approach:

a. The analyst should define a range of loads, debris types, debris combinations, debris arrival sequences, and interactions with chemicals in the fluid where the strainer is not expected to fail and net positive suction head (NPSH) margins can be maintained. Strainer failure may be structural or be attributed to degasification or voiding that results in excessive void fraction at the pumps. Testing should be done per guidance in RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Section 1.3.12, to support conclusions of strainer performance.

b. The analyst should determine when an initiating event could result in debris loads that are predicted to cause debris loads at the strainer greater than those shown acceptable under

Paragraph B-2.a of this appendix, and assume system failure whenever those conditions are predicted to occur.

c. The analyst should evaluate the effects of debris on boric acid precipitation within the reactor. Entities may refer to U.S. Nuclear Regulatory Commission (NRC)-approved methods and topical reports to evaluate boric acid precipitation.

d. The analyst should define a range of debris loads, debris types, debris combinations, debris arrival sequences, and interactions with chemicals in the fluid where adequate flow to the core is maintained. Debris load limits should be defined by testing. WCAP-16793, Revision 2, 'Evaluation of Long-Term Cooling Considering Particulate Fibrous and Chemical Debris in the Recirculating Fluid," has been accepted by the NRC staff (with conditions and limitations) as adequately defining in-vessel debris limits. This topical report, or other NRC-accepted topical reports or methods, may be used to define in-vessel debris limits. Analysis may be used to show that water can reach the core via alternate flow paths. The analysis should demonstrate that the alternate flow paths supply adequate coolant flow and cannot be blocked by debris.

e. The analyst should determine conditions when an initiating event can result in in-vessel debris limits or loads that are greater than those found acceptable under Paragraph B-2.d of this appendix, and assume system failure whenever those conditions are predicted to occur.

f. If the analyst considers timing in the assessment of debris effects it should be demonstrated that the assumptions on timing are appropriate. The assumptions about timing may need to be changed depending on the aspect of the assessment being performed. For example, early debris arrival at the strainer may maximize the challenge to NPSH margin, but may decrease debris penetration through the strainer. Timing assumptions may have to be different for each aspect of the assessment.

B-3. Systematic Risk Assessment: The analyst should develop a bounding estimate of the contribution of debris on core damage frequency (CDF) as follows:

a. For all accident scenarios in scope of the risk assessment, those that generated less debris than the applicable threshold determined by testing would be assumed to have negligible effect on CDF and would be assigned a CCDP of 0.

b. The remaining in-scope accident scenarios would be assumed to result in core damage with a probability of 1.0.

c. Plant states and configurations not explicitly treated in the simplified approach and which did not screen out under paragraph B-1 of this appendix should be assumed to lead to core damage

e. The bounding CDF attributable to debris (i.e., ΔCDF) may be used to estimate the bounding large, early release frequency (LERF) attributable to debris (i.e., ΔLERF). Typically, simplified LERF methods assume that LERF is some fraction of CDF (e.g., 0.1). Since debris may introduce additional dependency between the emergency core cooling system and containment spray system, the analyst should not assume such a simplified relationship between LERF and CDF, but should instead justify the LERF estimate by comparing the risk from the scenarios that lead to LERF in the base PRA model with the scenarios in the risk assessment of debris.

APPENDIX C

PARTITIONING PLANT-WIDE LOCA FREQUENCIES

Cautions, Limitations, and Definitions:

- This appendix refers to the RG 1.174 risk acceptance guidelines in terms of core damage frequency (CDF). It is understood that the large early release frequency (LERF) guidelines must also be met.
- In the context of this appendix, a "critical" break location is one that can produce and transport sufficient debris to cause core damage (e.g., by blocking the ECCS strainer).
- The examples in this appendix involve piping components. LOCA contribution from the failure of non-piping components must also be considered as stipulated in C.2.a of this RG if these breaks in these locations can generate and transport debris.

Methodology

When determining the risk attributable to debris, it may be necessary to partition plant-wide LOCA frequency so that it may be allocated to individual break locations. This may be done using the following bounding approach:

Analysts should identify the critical break location with the smallest effective break size, D_{min}. As stated in NUREG-1829, Section 3.7, this effective break size corresponds to a partial fracture for pipes with larger diameters than the break size, a complete single-ended rupture in pipes with the same inside diameter, or a double-ended guillotine break (DEGB) in pipes having inside diameters $1/\sqrt{2}$ times the break size. Assuming that <u>all</u> breaks of this size or larger lead to core damage, even those that are non-critical, provides a bounding estimate of the risk attributable to debris (i.e., ΔCDF). Expressed mathematically:

$$\Delta CDF = f(LOCA\ x \geq D_{min})$$

If this method yields a ΔCDF that meets the risk acceptance guidelines, then the risk attributable to debris has been shown to be acceptable. Because this is meant to be an upper bound estimate of risk, the mean LOCA frequency values using arithmetic mean (AM) aggregation should be used (NUREG-1829, Table 7.13). Semi-log interpolation of the NUREG-1829 LOCA frequencies is acceptable because of the overall trend in frequency vs break size

For example, consider a plant where the smallest critical break location has an effective break size of 8 inches. Using semi-log interpolation and AM aggregation, the analyst would calculate an exceedance frequency of f(LOCA X ≥ 8 inches = 7.7 E-6 per year.

In this example, the analysis produces results that meet the RG 1.174 risk acceptance guidelines. If the risk acceptance guidelines were not met, the analyst must demonstrate that the risk attributable to debris is small (i.e., meets the RG 1.174 guidelines) through additional analysis or mitigation. NUREG-1855 contains guidance for performing this step and provides the following options:

- Redefine application

- Refine PRA
- Implement compensatory measures and/or performance monitoring

Licensees should select one or more of these options in accordance with the guidance in NUREG-1855, Stage F.