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REGULATORY GUIDE

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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). These preliminary draft documents are in support of an October 22, 2015, Category 3 public meeting, and a November 2015 Advisory Committee on Reactor Safeguards (ACRS) subcommittee meeting.

This draft document has not been subject to all levels of NRC management review. Accordingly, it is incomplete and may contain errors 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). ***The following matters are currently under consideration and development by the staff:***

- The staff is developing a more robust technical basis and several sensitivity studies to address the issue of LOCA frequency allocation discussed in Section C.2 and draft Appendix C.
- The staff is developing additional guidance on how to treat partial breaks (e.g., five inch equivalent opening on a 31 inch pipe)
- The staff is developing additional guidance on the applicability of the generic NUREG-1829 data for specific sites. This guidance also addresses site-specific LOCA frequency contributors such as water hammer and seismically-induced LOCA. The level of detail required for this guidance is still being finalized.
- The staff is updating the language on screening to clarify that break locations that produce and transport debris may not be screened from the analysis based strictly on assumed low frequency.
- The staff is updating the section on periodic updates (C.7) to highlight that the NUREG-1829 LOCA frequencies were originally published with an "expiration date" of 15 years.

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

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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 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 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)(4), "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 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. Regulatory Guides 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis" (Ref. 8) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Ref. 9), and 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), required licensees to 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 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, some licensees have made major modifications to their plants to ensure adequate recirculation system performance. For example, some licensees have significantly increased the size of strainers, and some have replaced fibrous insulation with reflective metal insulation, the debris of which is considered less likely to reach or impede flow through strainers. Demonstrating adequate performance of strainers is challenging given the difficulty of testing them such that all conditions (e.g., temperatures, debris amounts and compositions, and operating components of the ECCS and CSS) that might exist during an accident are properly addressed. 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 that include risk-informed options to address 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 provides the specific approach for determining the scope, level of detail, and technical adequacy of the internal events, at power PRA required by 10 CFR 50.46c(e).

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

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, consistent with ASME/ANS RA-Sa 2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Ref. 16), (Probabilistic Risk Assessment (PRA) Standard), as endorsed in RG 1.200. Guidance on screening is contained in NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making" (Ref. 17).

       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 should be screened from the analysis strictly due to its assumed low frequency of occurrence.

c. The increase in risk from 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. For some simplified approaches (e.g., the simplified approach in Appendix B of this RG), changes to this at-power, internal events PRA may not be necessary in order to 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 Tables 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 ΔCDF and ΔLERF should be computed with respect to risk of the plant assuming that there are no debris effects, and should be mean values unless a bounding approach is used. 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 gives 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 ~~applicant may use a~~ detailed approach~~, a~~ of Appendix A or the simplified approach~~, or a hybrid approach~~ of Appendix B are acceptable to the NRC to determine the increase in risk on long-term cooling due to debris.

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 entity 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. ~~This is considered to be a hybrid of the simplified and detailed approach.~~

2. Initiating Event Frequencies. Initiating event frequencies should be developed consistent with the PRA Standard, as endorsed by RG 1.200, to the extent practicable. 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. ~~It~~Therefore, it may be necessary for the analyst to apportion the overall initiating event frequency to specific locations. ~~This is not how a PRA is typically developed but may be appropriate for the systematic risk assessment of debris.~~ 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. ~~Furthermore, the analyst should justify the approach used.~~

a. It is anticipated that many of the initiating events that remain after screening will involve LOCAs~~, and that~~. 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. Further, the analyst may desire to estimate the break frequency associated with a specific weld. In the discussion below, it is assumed that all welds within the scope of the risk assessment have been identified and that the debris that could be generated and transported as a result of a DEGB of each weld has been determined. Any weld rupture which could produce and transport sufficient debris to result in core damage is referred to as a *critical* weld in the following discussion.

(1) Aggregation method: NUREG-1829 does not advocate any specific aggregation method. 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 obtained using arithmetic or mixture distribution aggregation.

(2) Interpolation of NUREG-1829 plant-wide LOCA frequencies: Note that the analyst will need to employ and justify a suitable interpolation scheme for specific break sizes not in the NUREG-1829 tables. A ~~linear-linear~~log-log interpolation is acceptable to the staff. ~~Two approaches for apportioning the NUREG-1829 plant-wide frequencies are addressed below:~~

~~(1) A "bottom-up" approach assumes~~(3) Estimation of LOCA frequency within an interval of break sizes: The NUREG-1829 categories represent exceedance frequencies. For example, Category III contains all LOCAs with an effective break size of three inches *or larger*.

Category	**Effective break size, x (inches)**	**Frequency (NUREG-1829, Table 7.13), per year**
I	½	1.0 E-2
II	1 5/8	3.0 E-3
III	3	7.3 E-5
IV	7	9.4 E-6
V	14	2.4 E-6
VI	31	1.5 E-6

Interval-based estimates may be determined by simply subtracting the estimates for adjacent cumulative LOCA categories. For example, the mean LOCA frequency for the break interval [7", 14"] is equal to 9.4 E-6 – 2.4 E-6 = 7.0 E-6 per year. Additional guidance on this process is contained in NUREG-1829, Section 7.9. When defining intervals, care must be taken to ensure that the intervals fully span the overall range of break sizes (i.e., no "gaps"). The exceedance frequencies for the break size interval end points are determined by interpolating the NUREG-1829 frequencies, if necessary, as described in C.2.a.(2) above.

For example, assume that an analyst wishes to define the following two intervals:

Interval 1: [7", 9")
Interval 2: [9", 14"]

Applying log-log interpolation yields an exceedance frequency for X=9" of 5.7 E-6 per year. Therefore, the frequency for each interval is:

Interval 1: 9.4 E-6 – 5.7E-6 = 3.7 E-6 per year
Interval 2: 5.7 E-6 – 2.4E-6 = 3.3 E-6 per year

Note that the sum of the two intervals (3.7 E-6 + 3.3 E-6 = 7.0 E-6 per year) is equivalent to the overall interval frequency from the original interval. Analysts using this approach should ensure that the overall frequency from NUREG-1829 is preserved when defining intervals and subintervals.

(4) Apportionment of LOCA frequencies: There are several methods that are acceptable for assigning LOCA frequencies to individual break locations. They are presented below in decreasing order of conservatism. Detailed guidance for implementing each method is contained in Appendix C.

(a) Identify the smallest critical break location. 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 CDF. For example, if the smallest critical break location has an effective break size of D_{min}, then $\Delta CDF = f(LOCA\ X \geq D_{min})$.

(b) For each break size interval that contains one or more critical break locations, assign a CCDP of 1.0 to the entire interval. Intervals with no critical break locations are assigned a CCDP of 0. Sum the ΔCDF for each interval to obtain the total ΔCDF.

(c) Assume that the frequency of a LOCA at a specific location is a function not just of break size but of the degradation mechanisms that exist at that location. Examples of degradation mechanisms include thermal fatigue, vibration, stress corrosion cracking, etc. – see NUREG-1829 for a more comprehensive discussion. ~~The number of degradation mechanisms and their severity is assumed to affect LOCA frequency such that two breaks of the same size may have vastly different LOCA frequencies. Under this model, relative LOCA frequency is determined individually for each location and these are summed to obtain a plant-wide frequency. When these frequencies are normalized such that they sum to a pre-existing plant-wide frequency (e.g., NUREG-1829) this approach has sometimes been referred to as a "hybrid top-down, bottom-up" approach. While this RG does not use the term "hybrid" in this manner, it finds bottom-up approaches that are normalized to match NUREG-1829 to be acceptable~~Under this approach, LOCA frequency for each interval is allocated according to broad categories of relative rupture frequency that are assigned according to degradation mechanisms. A sensitivity study is included to ensure that that uncertainty with respect to degradation mechanisms is addressed.

~~Analysts using this approach should justify the selected frequency allocation scheme by describing how the relative allocation of LOCA frequency (i.e., weighting) is consistent with degradation mechanisms and operating experience. One way to accomplish this is to use the inservice inspection (ISI) program to inform the weighting process.~~

~~(2) A "top-down" approach assumes that the frequency of a LOCA at a specific location is a function only of the break size. Therefore, the plant-wide LOCA frequency for a given break size is equally allocated to all locations that could support a break of that size. For example, if the plant-wide LOCA frequency for a particular break size is 1E-6 per year and there are ten locations where a break of this size could occur, a frequency of 1E-7 per year would be allocated to each specific location (1E-6 / 10 = 1E-7 per year). The NRC finds the "top-down" approach acceptable, only if the analyst justifies that the~~

~~frequency assigned to any weld that results in core damage has not been underestimated by the simple averaging of the top-down approach, compared to the degradation mechanisms appropriate to those welds.~~

~~The following example compares the bottom-up and top-down approaches and illustrates one way an analyst might justify the use of the top-down approach.~~

~~Assume the plant-wide LOCA frequency for a certain break size is 1E-6 per year and that there are ten locations that can support a break of this size. The bottom-up approach would consider degradation mechanisms at each location and would allocate the 1E-6 per year accordingly:~~

~~**Bottom-up approach – hypothetical LOCA category with total plant-wide LOCA frequency of 1E-6 per year for a break of a given size**~~							
~~Break Location, x_i~~	~~X_1~~	~~X_2~~	~~$X_3 \ldots X_7$~~	~~X_8~~	~~X_9~~	~~X_{10}~~	~~**Sum**~~
~~Weight, w_i~~	~~0.6~~	~~0.1~~	~~0.01…0.01~~	~~0.05~~	~~0.12~~	~~0.08~~	~~**1.0**~~
~~Frequency, f_i (per year)~~	~~6E-7~~	~~1E-7~~	~~1E-8…1E-8~~	~~5E-8~~	~~1.2E-7~~	~~8E-8~~	~~**1E-6**~~

~~The bottom-up approach can produce frequencies that differ by more than an order of magnitude for breaks of the same size. In this example, the frequency of a LOCA at location X_1 is greater than that at X_3 by a factor of 50. Licensees using this approach should perform a reasonableness check to verify that such differences are logical and consistent with qualitative insights about degradation mechanisms.~~

~~**Top-down approach – hypothetical LOCA category with total plant-wide LOCA frequency of 1E-6 per year for a break of a given size**~~							
~~Break Location, x_i~~	~~X_1~~	~~X_2~~	~~$X_3 \ldots X_7$~~	~~X_8~~	~~X_9~~	~~X_{10}~~	~~**Sum**~~
~~Weight, w_i~~	~~0.1~~	~~0.1~~	~~0.1…0.1~~	~~0.1~~	~~0.1~~	~~0.1~~	~~**1.0**~~
~~Frequency, f_i (per year)~~	~~1E-7~~	~~1E-7~~	~~1E-7…1E-7~~	~~1E-7~~	~~1E-7~~	~~1E-7~~	~~**1E-6**~~

~~The top-down approach assumes that all breaks of the same size have the same frequency. In this example, the plant-wide frequency for a break of this size is 1E-6 per year so the assumed frequency at each location is simply 1E-6 / 10 = 1E-7 per year. This method can produce results that are significantly higher or lower than the bottom-up approach. For example, assume location X_7 was assumed to have a conditional core damage probability (CCDP) of 1.0 and all other welds were determined to have a CCDP of 0. In this case, the bottom-up approach would produce a ΔCDF of 1E-8 x 1.0 = 1E-8 per year. The top-down approach would produce a ΔCDF of 1E-7 per year, a factor of 10~~ ~~*higher*.~~

~~On the other hand, assume X_1 was determined to have a CCDP of 1.0 and all other welds were determined to have a CCDP of 0. In this case, the bottom-up approach would produce a ΔCDF of 6E-7 per year. The top-down approach would (again) produce a CDF of 1E-7 x 1.0 = 1E-7 per year but this time the result would be a factor of six~~ ~~*lower*~~ ~~when compared to the bottom-up approach.~~

~~Therefore, the top-down approach is not, strictly speaking, "conservative" or "nonconservative." Rather, it is a simplification whose effect cannot be determined before identifying which break locations are likely to generate and transport problematic debris loads. It is not necessary for analyses that use this approach to also perform a full bottom-up approach and compare the results quantitatively. However, given the potential effect of the top-down simplification, analysts wishing to use this approach should, at minimum, provide a qualitative comparison between the two approaches for any breaks resulting in core damage.~~

~~For the simplified approach described in Appendix B, this may be done by identifying the break locations where CCDP is assumed to equal 1.0 and discussing whether the relative frequency (i.e., weight) at each of these locations would be no higher than the weight used in the simple averaging approach.~~[1]

~~Arguments for why the relative frequency would be low may include recent weld remediation and/or the lack of degradation mechanisms relative to other break locations of the same size. If all locations where CCDP is assumed to be 1.0 would have a relative frequency lower than average, the top-down method is acceptable. If not, sensitivity studies should be done to address those welds with CCDP = 1.0 and whose expected relative frequency would be higher than average if calculated.~~

~~This type of comparison may or may not be feasible for the detailed method described in Appendix A. Analysts wishing to use a detailed method using the top-down approach should provide a comparison between the top-down results and anticipated bottom-up results to demonstrate that risk is not underestimated.~~

b. Initiating event 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. As stated above, it is anticipated that many of the initiating events that remain after screening will involve LOCAs. The following guidance applies to the initiating event frequencies for LOCA events.

(1) (1) Information in NUREG-1829 should be used to estimate LOCA frequencies in general for piping and non-piping ~~systems~~components in PWRs and BWRs. The analyst should ensure that any LOCA locations included in the analysis and for which the NUREG-1829 information is applied are consistent with the locations assumed in that NUREG. The analyst should confirm that the NUREG-1829 values are applicable to the plant. The NRC considers the LOCA frequencies in NUREG-1829 to be acceptable for licensees using this RG, provided that the licensee demonstrates that the plant in question is not an outlier for which the results of NUREG-1829 would not apply. The NUREG-1829 LOCA frequencies represent generic, or average, estimates for the commercial fleet and

~~[1] In other words, a sound qualitative argument should be presented that the weight assigned to that location—were it to be calculated—could reasonably be expected to be less than the simple average 1/N, where N is the total number of locations for that break size.~~

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 ~~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 plant in question is not an outlier for which the results of NUREG-1829 would not apply. To do this, the licensee should provide a qualitative discussion of the previously mentioned variables (i.e., material conditions, plant geometry, etc) that provides reasonable confidence that no unusual site-specific conditions exist (e.g., one-of-a-kind material, unusual plant geometry, etc).

In addition, 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 should be evaluated separately.

~~(2) NUREG-1829 does not advocate any specific aggregation method. 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 sensitivity analyses. 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 obtained using arithmetic or mixture distribution aggregation.~~

~~(3~~

One acceptable approach for addressing 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, 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 10-5/yr seismic event or 25 percent for a 10-6/yr seismic event for thermally aged stainless steel weld properties) will also bound the value that would be calculated for the specific plant.

The following is one acceptable approach for addressing water hammer:

The applicant should 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 applicant 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.

(2) If the information from NUREG-1829 is not used to estimate LOCA frequencies, the analyst should justify LOCA initiating event frequencies. The analyst should evaluate the effect of alternative selection of frequency ranges or distributions for initiating events on CDF and LERF, and demonstrate that conclusions would not be affected by alternatives.

(43) 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.

~~(5) NUREG-1829, Section 6.3, offers the following guidance on partial breaks that applies to both piping and non-piping LOCA contributors:~~

> ~~"A LOCA of a given size can occur by either a complete break of the smallest component supporting that LOCA size, or a partial break of a larger component. Most panelists believe that complete failure of a smaller component is more likely than partial failure of a bigger component. Therefore most panelists expect that the smallest diameter piping system or subcomponent that could support a particular LOCA size or category is the dominant LOCA frequency contributor."~~

~~An analysis referencing this RG should demonstrate consistency with the above paragraph or justify that deviating from this guidance does not underestimate risk.~~

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, "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, as well as the aggregate 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 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 "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 feasibility of these operator actions and any effect on nondebris operator actions should be discussed (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 approach 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. 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. For example, there may be existing programs that track risk based on equipment availability and reliability. 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. 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. The NRC does not expect an update of the entire PRA in order to fulfill this requirement, but rather an update of the risk-informed analysis of debris to capture any changes in the plant, procedures, operating experience. For example, a periodic update of the PRA failure data in the base PRA model would not be required under the rule unless such data was directly related to the risk assessment of debris (e.g., new information on LOCA frequencies).

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 ~~has increased by more than a small fraction of the risk approved by~~exceeds the NRC acceptance criteria, or in the event that defense-in-depth or safety margins have decreased from the NRC-approved analysis. ~~For power reactor licensees, the NRC considers that an increase in risk attributable to the effect of debris on long-term core cooling of greater than 1E-7 per year for CDF and 1E-8 per year for LERF would represent "greater than a small fraction" and would be the reporting threshold.~~ 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 Paragraph 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. ~~Note that, as the thresholds for~~

~~corrective action are higher than for reporting, a report would likely be required for any condition where corrective action is needed.~~

9. License Application. An entity seeking to use the risk-informed approach is required by10 CFR 50.46c(e)(2) to submit an application that includes: (1) A description of the alternative 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 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[2] 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[3] 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 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.

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2 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.

3 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.

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[4]

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."

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 1, Washington, DC, November 2002.

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4 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.

9. NRC, RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Washington, DC, March 2009.

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.[5]

17. NRC, Draft NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," Revision 1, Washington, DC, ~~March 2009.~~January 2, 2013. (ADAMS Accession No. ~~ML090970525~~ML12346A288).

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).

[5] 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.

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 defense-in-depth, safety margins, and performance measurement as set forth in Section C of this regulatory guide (RG).

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.

Consistent with RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis," the analyst may exclude hazard groups and operating modes from further consideration when the analysis demonstrates, qualitatively or quantitatively, that the corresponding risk contribution of the excluded hazard group or operating mode would not affect the decision being made or overall conclusion of the risk-informed analysis. Any such screening should be done on a plant-specific basis and the analyst should document the basis for each hazard or operating mode not being included in the risk-informed analysis. For screening purposes, these scenarios should be grouped in a logical fashion, for example according to initiating event.

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 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. For excluded failure modes, the analysis should still consider direct and indirect effects of debris on 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 to be used for the risk assessment, 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 will likely need to develop a number of submodels to evaluate strainer and downstream system performance, and then integrate these submodels for the systematic risk assessment. Possible submodels could include the following:

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).

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 to assume that all debris is at the strainer at the initiation of recirculation. 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 split 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 RG).

a. The integrated model should be structured to allow for propagation of relevant parameter uncertainty (see paragraph C.4 of this RG). A model using the Monte Carlo method or other suitable sampling approach could be considered for the propagation of parameter uncertainty. The sampling approach should use variance reduction techniques, such as stratified sampling, to ensure that relevant distribution tails are properly considered and sampled.

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 be consistent with 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 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 to the extent applicable. 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. As stated in A-13.b., 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. This is consistent with the PRA Standard as endorsed in RG 1.200.

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 RG, 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 RG, 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. This is consistent with the Probabilistic Risk Assessment (PRA) Standard as endorsed in RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," dated March 2009.

d. 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 a generic ratio of LERF to CDF, but should justify the LERF estimate by comparing the types of scenarios that lead to LERF in the base PRA model and the types of scenarios in the risk assessment of debris.