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DRAFT REGULATORY GUIDE

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(Proposed New Regulatory Guide)

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

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes methods and approaches that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for demonstrating compliance with the voluntary, risk-informed alternative for addressing the effects of debris during long-term cooling in 10 CFR 50.46c, “Emergency core cooling system performance during loss-of-coolant accidents (LOCA)” (Ref. 1) 10 CFR 50.46c requires that the 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 section (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 paragraph (d)(2)(iii) of the rule.

This RG describes acceptable methods and approaches for addressing paragraph 50.46c(e), “Alternate risk-informed approach for addressing the effects of debris on long-term core cooling,” and paragraph (m)(4), “Updates to risk-informed consideration of debris in long-term cooling,” 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 PWRs. 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

This regulatory guide is being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. It has not received final staff review or approval and does not represent an official NRC final staff position. Public comments are being solicited on this draft guide and its associated regulatory analysis. Comments should be accompanied by appropriate supporting data. Comments may be submitted through the Federal-rulemaking Web site, <http://www.regulations.gov>, by searching for Docket ID: NRC-2015-0095. Alternatively, comments may be submitted to the Rules, Announcements, and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Comments must be submitted by the date indicated in the *Federal Register* notice.

Electronic copies of this draft regulatory guide, previous versions of this guide, and other recently issued guides are available through the NRC’s public Web site under the Regulatory Guides document collection of the NRC Library at <http://www.nrc.gov/reading-rm/doc-collections/reg-guides/>. The draft regulatory guide is also available through the NRC’s Agencywide Documents Access and Management System (ADAMS) at <http://www.nrc.gov/reading-rm/adams.html>, under Accession No. ML15023A025. The regulatory analysis may be found in ADAMS under Accession No. ML12283A188.

- Title 10, Part 50, of the *Code of Federal 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).

Related Guidance

The NRC’s risk-informed approach includes consideration of risk, defense-in-depth, and safety margins, and the NRC expects licensees 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. The following RGs are relied upon in large measure as set forth in Section C of this RG:

- RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis” (Ref. 5).
- RG 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities” (Ref. 6).
- RG 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident” (Ref. 7).

Purpose of Regulatory Guides

The NRC issues RGs to describe to the public methods that the NRC staff (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 RG contains information collection requirements covered by 10 CFR 50 that the Office of Management and Budget (OMB) approved under OMB control number 3150-0011. The NRC may neither conduct nor sponsor, and a person is not required to respond to, an information collection request or 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). This section of the rule allows licensees 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 non-safety-related mitigation capabilities. This guide is intended to describe a risk-informed approach acceptable to the NRC that licensees can use in addressing the effects of debris on long-term core cooling.

Background

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 SECY-12-0093, “Closure Options for Generic Safety Issue – 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance” (Ref. 8) 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. 9). Without this alternative, licensees would need to seek exemptions from the rule to implement 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. 10) 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. 11), 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 helped to identify issues related to the adequacy of pressurized water reactor (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,” 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. 12) 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, a number of licensees have implemented 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 two Commission papers (SECY papers) (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 licensees that choose to implement 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 licensees 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 provide 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

10 CFR 50.46c(e) requires 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. This section provides descriptions of the 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.

1. The risk assessment required by 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 application 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, the licensee may exclude hazard groups and operating modes from further consideration when the licensee 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 performed on a plant-specific basis and the licensee 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 provide core cooling;
 - b. The scenario involves the potential for debris inside primary containment that could adversely impact 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.
2. The licensee should identify the debris-related failure modes for each SSC whose successful operation helps to mitigate the postulated scenarios screened as included under Paragraph C.1 of this RG. For example, it is expected that the 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 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 gasses (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 core heat transfer limits are exceeded;
- f. Debris buildup on cladding exceeds heat transfer limits; and,
- g. Debris buildup in the vessel leads to potential excessive boron concentrations within the core caused by reduction of fresh coolant entering the core.

The licensee may exclude debris-related failure modes from further consideration if a bounding analysis shows that maximum credible debris loads under detrimental configurations (e.g., fine and compact debris filling the void space of a fibrous debris bed) would not lead to a given failure mode. For excluded failure modes, the licensee 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.

3. After identifying and screening relevant scenarios and debris-related failure modes of SSCs, the licensee should evaluate failure modes identified from Paragraph C.2 of this RG and identify 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 impact 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 clearly described in the application to the NRC. Any operator actions credited with reducing the CDF or LERF attributable to debris should be clearly described.
4. It is anticipated that licensees pursuing a risk-informed approach to evaluate the effects of debris on the ECCS and CSS functions will utilize integrated models to evaluate strainer and downstream system performance, including 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 non-safety-related system activation to mitigate the event (e.g., strainer blockage, in-vessel effects, and ex-vessel downstream effects).

Integrated models should account for uncertainty in parameters and phenomenological models, as well as for the frequency of initiating events (e.g., frequency of large break LOCAs) and intensity of those events (e.g., size of the pipe break causing a LOCA). Paragraphs C.5 to C.13 of this RG provide details on guidance of the integrated model components.

- 5. The licensee should develop descriptions of the as-built and as-operated nuclear power plant (i.e., accounting for the effects of debris) system evaluated by phenomenological, physical, and mathematical models identified under Paragraph C.4 of this RG. The licensee should define the following:
 - a. power plant operating modes and operating components important to 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 integrated model.
- 6. The licensee should describe the source term for generation of debris under a postulated event to be mitigated by activation of the recirculation system.
 - a. The licensee 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 licensee 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 licensee should quantify the potential amounts of debris that could be generated by the initiating events and included scenarios identified under Paragraph C.1 of this RG. Licensees may refer to Section C.1.3.3 of RG 1.82 for guidance on identifying debris types and the use of the zone of influence concept to estimate debris amounts. Sections C.1.3.5 and C.1.3.6 of RG 1.82 provide guidance pertinent to coatings debris and latent debris. The NRC staff review guidance, “NRC

Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Coatings Evaluation,” provides guidance focused on coatings as a source of debris (Ref. 13).

- c. As necessary for quantifying the head loss through a bed of debris on strainers, the licensee should quantify debris characteristics, including material type, size distribution and shape, and density. The licensee should quantify the amount of debris penetration through or bypass around the strainers. The licensee should account for interactions with chemicals in the water when relevant to strainer failure or core damage mechanisms. Safety evaluations in Sections 3.4.3 and 3.5 of the Nuclear Energy Institute (NEI) document, NEI 04-07 (Ref. 14), provide guidance acceptable to the NRC for quantifying debris characteristics, including latent debris, which the licensee should consider in the development of risk-informed analysis.
 - d. The licensee should integrate information from Paragraph C.6.a to C.6.c of this RG into a model for quantifying debris amounts after a postulated initiating event. The licensee 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. The licensee should identify relevant data and model uncertainties, and propagate those uncertainties into the integrated model to compute the CDF and LERF for the as-built and as-operated plant (i.e., plant including debris). The licensee should describe where in its analysis conservative deterministic approaches (i.e., approaches that tend to overestimate the CDF and LERF) from NRC-approved guidance are used. Doing so will identify areas where propagation of uncertainty is not required. RG 1.174 contains additional guidance on uncertainty quantification and propagation.
7. Once the amount and type of debris is characterized, the licensee 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 provides guidance on the development of deterministic transport analyses and models.
 - b. The licensee should develop a model for debris transport to be used in the integrated model 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 licensee 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. The licensee should identify relevant data and model uncertainties, and propagate those uncertainties into the integrated model to compute the total CDF and LERF. The licensee should describe where in its analysis conservative deterministic approaches (i.e., approaches that tend to overestimate the CDF and LERF) from NRC-approved guidance are used. Doing so will identify areas where propagation of uncertainty is not required.

8. The licensee 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 provided 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 licensee should justify use of pressure beyond atmospheric in NPSH computations.
9. Using this RG, the licensee should perform analyses for each debris-related failure mode identified in Paragraph C.2, accounting for the debris sources in Paragraph C.6 that are assumed to be transported according to mechanisms identified in Paragraph C.7, with the goal to estimate CDF and LERF. A simplified approach to this part of the analysis may be performed, as described in paragraphs C.9.a through C.9.d below, in which case a licensee could then skip paragraphs C.10 through C.13.

For the simplified approach to analyzing debris-related failure modes, the licensee may select a simplified “go/no-go” approach by assuming a justifiable range of debris loads and demonstrating through testing that long-term core cooling will be maintained under those debris loads and pertinent conditions of the as-built and as-operated plant. This option would not seek to calculate time-dependent flow conditions at the strainer or in the vessel but would instead compare each scenario to a threshold value, determined by testing. Scenarios which produce debris exceeding this limit would be assigned a conditional core damage probability (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. If this approach is selected:

- a. The licensee should define a range of loads, debris types, debris combinations, debris arrival sequences, and interactions with chemicals in the fluid (see Paragraphs C.11 and C.13 of this regulatory guide), where the strainer is not expected to fail. Testing should be conducted per guidance in RG 1.82, Section 1.3.12, to support conclusions of strainer performance.
 - b. The licensee should determine when an initiating event could result in debris loads that are predicted to cause head losses greater than those shown acceptable under Paragraph C.9.a of this RG, and assume system failure whenever those conditions are predicted to occur.
 - c. The licensee 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 (Ref. 15), 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 flowpaths. The analysis should demonstrate that the alternate flowpaths provide adequate coolant flow and cannot be blocked by debris.
 - d. The licensee 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 C.9.c of this RG, and assume system failure whenever those conditions are predicted to occur.
10. Licensees who do not select the simplified approach described in Paragraph C.9 should develop and implement a model for debris accumulation and head loss through the potential debris bed developed on strainers. The output of this approach is a calculated head loss value and in-vessel debris load for each scenario.

- a. Guidance on the development of head loss analyses is provided in the safety evaluation of Sections 3.4.3 and 3.5 of NEI 04-07 and in RG 1.82, Section C.1.3.11.
 - b. The licensee should develop a model of head loss through the debris bed on strainers to be used in the integrated model 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 licensee 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 licensee should identify relevant data and model uncertainties, and propagate those uncertainties into the integrated model to compute the total CDF and LERF. The licensee should describe where in its analysis conservative deterministic approaches (i.e., approaches that tend to overestimate the CDF and LERF) from NRC-approved guidance are used, to identify areas where propagation of uncertainty is not required.
11. The licensee 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 (Ref. 16) 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. 17) provides guidance on plant-specific chemical effect evaluations.
 - b. The licensee should develop a model of chemical effects on flow resistance and head loss through the debris bed on strainers to be used in the integrated model 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 licensee 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 licensee should identify relevant data and model uncertainties, and propagate those uncertainties into the integrated model to compute the total CDF and LERF. The licensee should describe where in its analysis conservative deterministic approaches (i.e., approaches that tend to overestimate the CDF and LERF) from NRC-approved guidance are used. Doing so will identify areas where propagation of uncertainty is not required.
12. The licensee should evaluate debris penetration through the strainer.

- a. The licensee should characterize debris penetration through strainers under potential accident conditions. The licensee should account for all debris penetration mechanisms or mechanisms where debris can bypass the strainer.
 - b. The licensee should develop a model to estimate the amount of debris penetration through strainers, with the goal of evaluating downstream effects.
 - c. The licensee 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 conducted under conditions that are prototypical or conservative with respect to the as-built and as-operated plant.
 - d. The licensee should identify relevant data and model uncertainties, and propagate those uncertainties into an integrated model to compute the total CDF and LERF. The licensee should describe where in its analysis conservative deterministic approaches (i.e., approaches that tend to overestimate the CDF and LERF) from NRC-approved guidance are used. Doing so will identify areas where propagation of uncertainty is not required.
13. The licensee should evaluate the effects of debris strainer penetration inside (in-vessel) and outside (ex-vessel) the reactor vessel.
- a. The licensee should evaluate downstream ex-vessel effects of debris (e.g., blockage of flowpaths in equipment, and wear and abrasion of surfaces). The safety evaluation for the Topical Report WCAP-16406-P (Ref. 18) and RG 1.82 provide guidance that the licensee may use to evaluate ex-vessel effects of debris.
 - b. The safety evaluation for the Topical Report WCAP-16793-NP provides 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 (g) 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. Licensees may use the debris load limits described in WCAP-16793 or other NRC-approved values as acceptance limits for in-vessel debris loading without additional justification.
 - c. The licensee should address the potential for boric acid precipitation in its analysis.
 - d. The licensee should develop a model for debris penetration effects, and clearly identify how the model would be used in the estimate of CDF and LERF. In particular, the licensee should properly account for the fraction of the debris-carrying flow passing through the core in mass balance computations for the amount of in-vessel debris accumulation. 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 licensee 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 evaluation of the effects of debris penetration.

- f. The licensee should evaluate the validity of the penetrated debris effects model, relying, for example, on tests, empirical data, and analogies of .
 - g. The licensee should identify relevant data and model uncertainties, and propagate those uncertainties into the integrated model to compute the total CDF and LERF. The licensee should describe where in its analysis conservative deterministic approaches (i.e., approaches that tend to overestimate the CDF and LERF) from NRC-approved guidance are used. Doing so will identify areas where propagation of uncertainty is not required.
14. The licensee should combine the submodels for the debris source term, debris transport, strainer model, chemical interactions with debris, debris strainer penetration, and in-vessel effects into the integrated model with the goal of computing failure probabilities in the modified PRA model to evaluate debris effects (implemented in Paragraph C.15 of this RG). These submodels were discussed in this RG's paragraphs C.5 to C.9 for the simplified approach and paragraphs C.5 to C.15, excluding C.9, otherwise.
- a. The integrated model should be structured to allow for propagation of relevant parameter and model uncertainties identified in Paragraphs C.5 to C.13 of this RG. A model implementing the Monte Carlo method or other suitable sampling approach could be considered for the propagation of parameter uncertainty. The sampling approach should implement variance reduction techniques, such as stratified sampling, to ensure that relevant distribution tails are properly considered and sampled.
 - b. Uncertainties that can be represented as distributions (i.e., probability density functions or probability mass functions) in input parameters should be identified and properly justified. Effort should be aimed at selecting distributions that properly reflect the state of knowledge of each parameter so that CDF and LERF can be calculated accurately.
 - c. Inputs to the integrated model should be consistent with inputs to the modified PRA discussed in Paragraph C.15 of this RG, and sampled in the integrated model and the modified PRA model consistent with any known statistical correlation or dependence between the uncertainty distributions.
 - d. Initiating event frequencies should be represented by distributions to be used as an input to the modified PRA and the integrated model.
 - (1) Information in NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies through the Elicitation Process" (Ref. 19) is acceptable to estimate LOCA frequencies in general for piping and non-piping passive systems in PWRs and BWRs. The licensee should ensure that HELB locations identified in paragraph C.6 of this RG are consistent with the locations assumed in NUREG-1829. The licensee should confirm that the NUREG-1829 values are applicable to its plant.
 - (2) NUREG-1829 provides 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. NUREG-1829 does not advocate any specific aggregation method. The licensee should select frequencies that would not underestimate the mean CDF, LERF, Δ CDF, and Δ LERF as compared to alternative methods. The NRC finds that the LOCA frequencies from NUREG-1829 derived using the arithmetic mean aggregation and mixed

distribution aggregation methods are acceptable. Use of LOCA frequencies derived from alternative aggregation methods (such as geometric mean aggregation) should be justified by the licensee, and alternatives should be considered in uncertainty analyses. The licensee should demonstrate that conclusions would not be significantly different from the conclusions reached through the use of alternative aggregation methods by comparing the licensee's conclusions to the results obtained using arithmetic or mixture distribution aggregation.

- (3) If the information from NUREG-1829 is not used to estimate LOCA frequencies, the licensee should justify LOCA and non-LOCA initiating event frequencies. The licensee should evaluate the impact of alternative selection of frequency ranges or distributions for initiating events on CDF and LERF, and demonstrate that conclusions would not be impacted by alternatives.
 - (4) Statistical distributions chosen to represent the uncertainty about parameters in the integrated model and in the modified PRA should preserve the mean values of the initiating event frequencies from original source documents, such as NUREG-1829.
- e. The licensee should consider failure modes of SSCs, identified in Paragraph C.2 of this RG, and the corresponding failure probabilities. The licensee's 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.
- (1) The licensee should provide a technical basis for allocating plant wide LOCA frequencies to individual locations (e.g., pipe weld). One acceptable approach is to use information from the licensee's In-Service Inspection (ISI) program. In general, the LOCA frequency assigned to each location should be informed by the known degradation mechanisms at that location.
 - (2) Assumptions that are made when allocating plant-wide LOCA frequencies to individual locations (e.g., relative likelihood of a complete rupture of a small pipe compared to an equivalent size opening in a larger pipe) should be identified and their impact on CDF and LERF should be quantified.
15. The licensee should estimate the change in risk attributable to debris.
- a. The licensee should make modifications to the baseline PRA model (i.e., the PRA model that assumes any effects of debris are negligible), consistent with Paragraph C.3, to perform the calculation of the risk (CDF and LERF) for the as-built and as-operated nuclear power plant.
 - b. Licensees should use commonly accepted methods and approaches to implement changes to the PRA model for the debris risk assessment. 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 implement those changes should be well described in the license amendment request.

- 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. Inputs to the modified PRA should be consistent with inputs and information used by the integrated model.
 - (1) Common input distributions should be consistently sampled in the modified PRA and in the integrated model.
 - (2) Common information of the modified PRA and the integrated model should be consistently treated, including the use of correlations where needed.
 - f. Nuclear power plant states and configurations not explicitly treated in the modified PRA or in the integrated model, and which are not excluded following the screening procedure in Paragraph C.1 of this RG, should be assumed to lead to core damage. The contribution to the CDF and LERF for these unaccounted states and configurations should be quantified.
 - g. The modified PRA, together with the integrated model, should be used to quantify the mean values of CDF and LERF, accounting for debris effects, and compared to risk regions in Figures 4 and 5 of RG 1.174.
 - (1) The Δ CDF and Δ LERF should be computed with respect to risk of the plant assuming that debris effects are negligible. The CDF and LERF on the horizontal axis on Figures 4 and 5 of RG 1.174 should be interpreted as the total risk estimates for the plant as described in Section 2.4 of that regulatory guide.
 - (2) The mean value resulting from a propagation of parametric uncertainty in the PRA model quantification is the appropriate point-estimate for comparison to the RG 1.174 risk acceptance guidelines.
 - (3) The mean values of CDF/ Δ CDF and LERF/ Δ LERF should meet the risk acceptance guidelines of Figures 4 and 5, respectively, of RG 1.174.
16. The licensee should provide a summary description of the plant response to debris. The objective of this summary is to ensure that the overall model produces reasonable results and that there are no counter-intuitive or non-physical modeling artifacts.
- a. The licensee should identify key aspects of the plant that limit the magnitude of the CDF and LERF when accounting for effects of debris, such as the following:
 - (1) frequency of events that could produce significant amounts of debris

- (2) amount of debris produced
 - (3) resiliency of strainer system to failure under the presence of debris
 - (4) debris filtration by strainers
 - (5) resiliency of the system to provide adequate core cooling under debris presence
 - (6) resiliency of the system to limit large early releases
 - (7) alternate flow paths for cooling of the core, if credited
- b. The identification of key aspects of the plant should be complete and consistent with other paragraphs in this RG, including the submodels in Paragraphs C.5 to C.13, the integrated model in Paragraph C.14, and the modified PRA in Paragraph C.15.
- c. The licensee should summarize the technical basis of the identified key aspects of the plant.
- (1) The extent of the technical basis should be commensurate with the risk significance of the identified key aspects and the associated uncertainties.
 - (2) The summary technical basis should be consistent with this RG including the submodels in Paragraphs C.5 – C.13, the integrated model in Paragraph C.14, the modified PRA in Paragraph C.15, and their uncertainties.
 - (3) The technical basis should describe any action taken for prevention (e.g., removal of fiber and insulation, increase in strainer areas to capture debris), and how key aspects under Paragraph C.16.a of this RG would mitigate debris effects. Relevant operator actions to reduce CDF and LERF should be described, such as hot leg switchover and securing of containment spray system pumps. The description should account for relevant plant states and operating SSCs.
 - (4) The licensee should summarize key aspects of the analysis where margins are applied to arrive at conservative estimates of the CDF and LERF, such as strainer failure criteria, in-vessel failure criteria, zones of influence to estimate debris sources, and tests conditions that would overestimate detrimental effects.
17. Section C.2.1.1 of RG 1.174 provides 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 provided 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 licensee should address the impact 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. 20). The assessment should consider the impact of debris on the availability and reliability of each level of defense, as well as the aggregate impact.
 - b. There should not be an over-reliance on programmatic activities to compensate for weaknesses in plant design. The licensee 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 impacted by debris are modified, added, or removed, the licensee should address the effect of such changes on redundancy, independence, and diversity. Absent such system changes, the licensee may conclude that this element of defense-in-depth is addressed. Note that common cause failures are addressed by the next element so do not have to be addressed under this one.
- d. Defenses against potential common cause failures should be preserved and the potential for the introduction of new common cause failure mechanisms are assessed and addressed. The licensee should assess the impact of debris on inter-system (e.g., among low-head and high-head injection systems) and intra-system (e.g., among trains of a given system) availability and reliability. The licensee should justify, qualitatively or quantitatively, that any increase in common-cause failure rates across systems impacted by debris is very small compared to other failure rates that are not debris-related.
- 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 licensee should provide an evaluation describing a realistic plant response to each debris-related failure mode identified in Paragraph C.2 of this RG. This evaluation 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 licensee 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, licensees may take into account how plant conditions vary over time. For example, when evaluating containment performance following assumed strainer structural failure, licensees may assume thermal-hydraulic conditions consistent with the time that strainer failure would reasonably be assumed to occur.

- f. Defenses against human errors should be preserved. The licensee should discuss 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 non-debris operator actions should be discussed (e.g., any impact on crew workload). The licensee 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 licensee should confirm that no debris-related failure could completely disable multiple layers of defense between the fission product source term and the public.
18. The licensee 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 integrated analysis. 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. The demonstration of safety margins should be consistent with guidance at Paragraph C.16.c(4). For example, if the licensing basis calculations use a given value for the required NPSH, then the integrated model should also use this value, or a justification should be provided if a different value is considered.
19. The licensee should ensure that the risk assessment for the evaluation of debris was performed under a QA program that meets the guidance of RG 1.174, Section 5. The use of the pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 as set forth in RG 1.174 is justified because the risk-informed analysis of debris (including the integrated model and supporting computations) is needed to demonstrate that the design of safety-related SSCs meets NRC requirements.
20. Licenses selecting the risk-informed alternative in 50.46c(e) must demonstrate that the risk attributable to debris is small and that defense in depth and safety margins are maintained. Consistent with RG 1.174, the licensee should develop an implementation and monitoring program that will ensure the long term validity of these conclusions. This program should provide reasonable assurance that future planned or unplanned changes to the plant (e.g., modifications, discovery of additional latent debris) or changes to the PRA¹ (e.g., change in LOCA frequencies) are analyzed to ensure that the original conclusions from the LAR remain valid.

Consistent with application-specific guidance for other risk-informed initiatives, the implementation and monitoring program may be partially or fully comprised of existing licensee programs. Licensees wishing to credit existing programs should describe how these programs are suited (or have been modified) to account for the unique challenges of calculating the portion of CDF and LERF attributable to debris. For example, many programs used to track risk are based on equipment availability and reliability. Such programs would not generally be suited to evaluating – for example – the discovery of a large quantity of degraded coatings that could contribute to the source term. Licensees should also describe updates to existing programs that will prevent or mitigate addition of known problematic debris sources into containment. For example, if a licensee's evaluation credits the removal of Marinite®, the licensee should describe how plant work control practices 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. Consistent with 50.46c, licensees are not required to report these results to the NRC unless it is determined that the acceptance criteria in 50.46c(e)(1)

¹ In this context, "PRA" also includes any complementary analyses (e.g., debris evaluation model, human reliability analysis) that are used to calculate the increase in risk attributable to debris.

are no longer met. In this case, existing licensee programs for reporting (e.g., 10 CFR 50.72, 10 CFR 50.73) may be used.

D. IMPLEMENTATION

The purpose of this section is to provide information on how applicants and licensees² 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 voluntarily³ 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.

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" means 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).

REFERENCES⁴

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. Title 10, Part 50, of the *Code of Federal Regulations* (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. 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,” Washington, DC.
6. NRC, RG 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Washington, DC, March 2009.
7. NRC, RG 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident,” Revision 4, March 2012, Washington, DC.
8. 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.
9. 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.
10. NRC, RG 1.82, “Sumps for Emergency Core Cooling and Containment Spray Systems,” Revision 0, June 1974, Washington, DC.
11. 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 No. ML ML031150731)
12. 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.

4 Publicly available NRC-published documents are available electronically 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.

13. NRC, "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Coatings Evaluation," Washington, DC. March 2008. (ADAMS No. ML080230462)
14. 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 No. ML050550156)
15. Westinghouse and NRC. "Final Safety Evaluation for Pressurize 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 No. ML13239A111)
16. 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 Nos. ML081150383 and ML101230629)
17. 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 No. ML080380214)
18. 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 No. ML073520295)
19. NRC, NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," Washington, DC. April 2008. (ADAMS No. ML ML080630013)
20. Nuclear Energy Institute (NEI), "Example Pressurized Water Reactor Defense-in-Depth Measures For GSI-191, PWR Sump Performance," Washington, DC. March 2012. (ADAMS No. ML120730660)
21. NRC, Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection," Washington, DC. October 9, 2013.
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