Southern Nuclear

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10 CFR 50.90
10 CFR 50.12

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U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

Vogtle Electric Generating Plant Units 1&2
Docket Nos. 50-424, 50-425

Subject: Exemption Request and License Amendment Request for a Risk-Informed Resolution to GSI-191

Pursuant to 10 CFR 50.12, Southern Nuclear Operating Company (SNC) hereby requests an exemption from certain requirements in the 10 CFR 50.46 regulation for Vogtle Electric Generating Plant (VEGP) Units 1 and 2. In addition, pursuant to 10 CFR 50.90, SNC hereby requests an amendment to Operating Licenses NPF-68 and NPF-81 for VEGP Units 1 and 2. The proposed amendment will revise the licensing basis as described in the VEGP Final Safety Analysis Report (FSAR) to allow the use of a risk-informed approach to address safety issues discussed in Generic Safety Issue (GSI)-191, “Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance.” In addition, the proposed amendment adds a new Technical Specification (TS) 3.6.7, "Containment Sump," and adds an Action to address the condition of the containment sump made inoperable due to containment accident generated and transported debris exceeding the analyzed limits.

Enclosure 1 provides the proposed exemption request. The proposed license amendment request (LAR) is presented in Enclosures 2, 3, and 4. Enclosure 2 provides a Summary Description, Detailed Description, and Technical Evaluation for the SNC request to implement a risk-informed approach for addressing GSI-191. Enclosure 3 provides a Description and Assessment for the SNC request to add a containment sump TS following the model application in TSTF-567, Revision 1. Enclosure 4 provides the Regulatory Evaluation and Environmental Consideration for the requested actions in Enclosures 2 and 3.

Approval of the proposed amendment is requested within one year of the date of this letter to support implementation of the Unit 1 strainer modifications during the Fall 2021 outage. The amendment will be implemented after the strainers have been modified in both units and prior to the conclusion of the Spring 2022 Unit 2 refueling outage.

In accordance with 10 CFR 50.91, a copy of this application is being provided to the designated Georgia Official.

This letter contains no new regulatory commitments. If you have any questions, please contact Jamie Coleman at 205.992.6611.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 17th day of August, 2020.

Respectfully submitted,

C. A. Gayheart
Director, Regulatory Affairs
Southern Nuclear Operating Company

CAG/RMJ

Enclosures:

1. Request for Exemption

2. Implementation of a Risk-Informed Approach for Addressing GSI-191
   Enclosure 2 Attachments:
   A1 - Proposed FSAR Changes (Mark-Up) for Information Only
   A2 - Guidance for Supporting Operability Evaluations for Information Only
   A3 - Updated Evaluation for In-Vessel Effects and Coatings

3. Proposed Changes to the Technical Specifications
   Enclosure 3 Attachments:
   A1 - Proposed Technical Specification Changes (Mark-Up)
   A2 - Revised Technical Specification Pages
   A3 - Proposed Technical Specification Bases Changes (Mark-Up) for Information Only

4. Regulatory Evaluation and Environmental Consideration

cc: Regional Administrator, Region ll
NRR Project Manager – Vogtle 1 & 2
Senior Resident Inspector – Vogtle 1 & 2
State of Georgia Environmental Protection Division
RType: CVC700

Vogtle Electric Generating Plant
Licensing Submittal for a Risk-Informed Resolution of Generic Letter 2004-02

Enclosure 1

Request for Exemption

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1.0 GENERAL

1.1 Introduction

In support of the Alvin W. Vogtle Electric Generating Plant (VEGP) risk-informed approach to addressing Generic Safety Issue (GSI) 191 and response to the United States Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2004-02 (Reference 1), this enclosure provides Southern Nuclear Operating Company's (SNC's) request for exemption under Title 10 of the Code of Federal Regulations (CFR) Section 50.12 (10 CFR 50.12) from certain requirements in 10 CFR 50.46. This exemption request complements a license amendment request (LAR) provided in Enclosures 2 through 4 of this submittal. Enclosure 2 proposes methodology changes that will be incorporated in the VEGP Units 1 and 2 Final Safety Analysis Report (FSAR) based on NRC acceptance of the risk-informed method and results (Reference 2). In addition, changes to the VEGP Technical Specifications (TS) will be made as described in Enclosure 3 of this submittal. The regulatory evaluation and environmental consideration are addressed in Enclosure 4 of this submittal for the proposed license amendment to incorporate both the risk-informed methodology and TS changes.

The specific exemption request pertains to requirements associated with the emergency core cooling system (ECCS) function for core cooling following a postulated loss of cooling accident (LOCA). The scope and key elements of the requested exemption are described in Section 2.0.

Approval of the exemption will allow the use of a risk-informed method to account for the probabilities and uncertainties associated with mitigation of the effects of debris following postulated LOCAs. The method evaluates concerns raised by GSI-191 related to the effects of post-accident debris on the containment sump recirculation strainers and reactor core blockage due to debris in the recirculating fluid. To confirm acceptable sump design, the risk associated with loss of core cooling due to the effects of debris is evaluated. The risk-informed approach is designed to be consistent with the guidance in Regulatory Guide (RG) 1.174 (Reference 3).

The general methodology for the VEGP approach is provided in Enclosure 3 of the July 2018 submittal (Reference 4). The VEGP approach is the risk-informed part of an overall graded approach that is based on the amount of debris in the plant, as discussed in SECY-12-0093 (Reference 5). The VEGP risk-informed approach addresses the five key principles in RG 1.174 (Reference 3). The resulting risk metrics (i.e., CDF, LERF, ΔCDF, and ΔLERF) are used to determine whether plant modifications are warranted to ensure acceptable sump performance. The VEGP risk quantification for Units 1 and 2 showed that ΔCDF and ΔLERF are below the threshold for RG 1.174 Region III, "Very Small Changes," without significant plant modifications (Reference 4). Therefore, the risk-informed approach provides an equivalent level of assurance for sump performance without incurring significant cost and occupational dose associated with removing, replacing, or reinforcing insulation in containment.

Approval of the requested exemption will support the application of the risk-informed approach.

One physical modification that VEGP will implement is to reduce the overall height of the residual heat removal (RHR) sump strainers by removing the two top disks from each stack of the RHR strainer assemblies. In addition, emergency operating procedures were revised to inject additional refueling water storage tank (RWST) inventory for breaks that do not initiate containment sprays. These physical and procedural modifications ensure that the RHR strainers are completely submerged for an increased number of postulated LOCA scenarios, which reduces the risk associated with post-accident debris effects. These modifications are described in detail in Enclosure 2 of the July 2018 submittal (Reference 4).

1.2 Background and Overview

GSI-191 identifies the possibility that debris generated during a LOCA could clog the containment sump strainers in pressurized water reactors (PWRs) and result in loss of net positive suction head (NPSH) for the ECCS and containment spray system (CSS) pumps, impeding the flow of water from the sump. GL 2004-02 requested licensees to address GSI-191 by demonstrating compliance with the 10 CFR 50.46 ECCS acceptance criteria (Reference 1). In addition, GL 2004-02 required licensees to address downstream effects. As stated in GL 2004-02, licensees were requested to perform analyses using an NRC-approved methodology and to ensure successful operation of the ECCS and CSS during design-basis accidents (DBAs) that require containment sump recirculation (Reference 1):

> "Although not traditionally considered as a component of the 10 CFR 50.46 ECCS evaluation model, the calculation of sump performance is necessary to determine if the sump and the ECCS are predicted to provide enough flow to ensure long-term cooling.
>
> Based on the new information identified during the efforts to resolve GSI-191, the staff has determined that the previous guidance used to develop current licensing basis analyses does not adequately and completely model sump screen debris blockage and related effects. As a result, due to the deficiencies in the previous guidance, an analytical error could be introduced which results in ECCS and CSS performance that does not conform to the existing applicable regulatory requirements outlined in this generic letter. Therefore, the staff is revising the guidance for determining the susceptibility of PWR recirculation sump screens to the adverse effects of debris blockage during DBAs requiring recirculation operation of the ECCS or CSS. In light of this revised staff guidance, it is appropriate to request that addressees perform new, more realistic analyses and submit information to confirm the functionality of the ECCS and CSS during DBAs requiring recirculation operations."

In addition, GL 2004-02 identified the following regulatory requirement (Reference 1):

> "NRC regulations in Title 10, of the Code of Federal Regulations Section 50.46, 10 CFR 50.46, require that the ECCS has the capability to provide long-term cooling of the reactor core following a LOCA. That is, 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 required by the long-lived radioactivity remaining in the core."

As described in Enclosure 2 of the July 2018 submittal (Reference 4), compensatory and mitigative measures have been implemented in response to Bulletin 2003-01 (Reference 6) and GL 2004-02 (Reference 1) for VEGP Units 1 and 2 to address the potential for sump strainer clogging and related GSI-191 concerns. This includes installation of larger containment sump strainers that greatly reduce the potential for loss of NPSH for the RHR and CSS pumps. Defense-in-depth measures are described in Enclosure 4 of the July 2018 submittal (Reference 4).

The Commission issued Staff Requirements Memorandum (SRM) SECY-10-0113 directing the staff to consider alternative options for resolving GSI-191 (Reference 7). Subsequently, SECY-12-0093 outlined a few different options that PWR licensees can use to address GSI-191, including deterministic and risk-informed approaches (References 5 and 8). In a letter to the NRC on May 16, 2013 (Reference 9), VEGP selected Option 2, the full risk-informed resolution path.

The VEGP risk-informed methodology is described in Enclosure 3 of the July 2018 submittal (Reference 4). Based on the guidance in RG 1.174, this approach requires an exemption from certain requirements of 10 CFR 50.46 in accordance with 10 CFR 50.12.

2.0 EXEMPTION REQUEST

Pursuant to 10 CFR 50.12, SNC is submitting this request for exemption from certain requirements of 10 CFR 50.46(a)(1), "other properties," as it relates to using specific deterministic methodology to evaluate the effects of debris on long-term core cooling.

10 CFR 50.46(a)(1) is shown below with the "other properties" portion for which exemption is requested in bold.

> "(a)(1)(i) Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, **and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated. Except as provided in paragraph (a)(1)(ii) of this section, the**

> **evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident.** Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. Appendix K, Part II Required Documentation, sets forth the documentation requirements for each evaluation model. This section does not apply to a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted.
>
> (ii) Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of appendix K ECCS Evaluation Models."

The scope of the exemption applies to all debris effects addressed in the risk-informed element of the VEGP methodology described in SNC's July 2018 submittal (Reference 4). The debris effects are associated with those breaks that potentially generate and transport debris that exceeds the analyzed debris limit. The key elements of the exemption request are listed as follows. SNC is requesting exemption for these breaks to allow evaluation of the debris effects using a risk-informed methodology.

1. The exemption will apply only to the effects of debris as described in Enclosures 2 and 3 of the July 2018 submittal (Reference 4).
2. The exemption will apply to any breaks that can generate and transport debris that is not bounded by VEGP-specific analyzed limits, provided that the ΔCDF and ΔLERF remain in RG 1.174 Region III (Reference 3).

This exemption request is complemented by the accompanying LAR (see Enclosures 2 through 4), which seeks NRC approval to amend the licensing basis based on acceptable design of the containment sump. The risk-informed method provides a high probability of assurance for acceptable sump performance and debris mitigation as calculated by the ECCS evaluation model.

The VEGP risk-informed approach to addressing GSI-191 and responding to GL 2004-02 is consistent with the NRC safety evaluation (SE) for NEI 04-07 that discussed the modeling of sump performance as follows (Reference 11):

> "While not a component of the 10 CFR 50.46 ECCS evaluation model, the calculation of sump performance is necessary to determine if the sump and the residual heat removal system are configured properly to provide enough flow to ensure long-term cooling, which is an acceptance criterion of 10 CFR 50.46. Therefore, the staff considers the modeling of sump performance as the validation of assumptions made in the ECCS evaluation model. Since the modeling of sump

> performance is a boundary calculation for the ECCS evaluation model, and acceptable sump performance is necessary for demonstrating long-term core cooling capability (10 CFR 50.46(b) (5)), the requirements of 10 CFR 50.46 are applicable."

This exemption request is consistent with the provisions of the proposed ECCS rule change. The following statement, found on Page 85 of the proposed 10 CFR 50.46c final rule change package attached to SECY-16-0033 (Reference 12), applies to the new 10 CFR 50.46c(d), which will replace the current 10 CFR 50.46(a)(1):

> "Demonstration of consideration of such factors may also be achieved through analytical models that adequately represent the empirical data obtained regarding debris deposition. The final rule alternatively allows the use of risk-informed approaches to evaluate the effects of debris on localized coolant flow and delivery of coolant to the core during the long-term cooling (post-accident recovery) period."

The proposed ECCS rule change will allow use of a risk-informed approach, addressed in 10 CFR 50.46c(e), in lieu of a deterministic evaluation. Similar to the proposed new rule change, VEGP's risk-informed approach is an alternative to the current deterministic evaluation required by 10 CFR 50.46(a)(1). VEGP requires exemption from 10 CFR 50.46(a)(1) "other properties" since there currently is no risk-informed evaluation alternative. VEGP requests an exemption from those deterministic requirements in order to enable the use of a risk-informed method to demonstrate acceptable sump performance and debris mitigation, and to validate assumptions in the ECCS evaluation model.

3.0 REGULATORY REQUIREMENTS INVOLVED

By regulatory precedent, licensees are required to demonstrate compliance with the relevant regulations by the use of a bounding calculation or other deterministic method. SNC seeks exemption to the extent that 10 CFR 50.46(a)(1) "other properties" requires deterministic calculations or other analyses to address the concerns raised by GSI-191 related to acceptable plant performance during the recirculation mode following a LOCA. The proposed changes to the licensing basis and technical specifications, submitted for NRC approval through a LAR (see Enclosures 2 through 4), address GSI-191 and GL 2004-02 for VEGP, Units 1 and 2, on the basis that the associated risk is shown to meet the acceptance guidelines in RG 1.174 and adequate defense-in-depth and safety margin are demonstrated.

This exemption request is to allow the use of a risk-informed method to demonstrate acceptable mitigation of the effects of debris following postulated LOCAs. Prior to the risk-informed approach, deterministic methods were used to evaluate the effects of accident-generated and transported debris in order to meet the current licensing basis assumptions for analyzing the effects of post-accident debris blockage in the sump and in-vessel. However, these evaluations did not address debris effects fully for the as-built, as-operated plant conditions. The risk-informed approach evaluates the debris

effects as part of the assessment of the residual risk associated with GSI-191 concerns. Based on confirmation of acceptable ECCS design as determined by the resulting risk meeting the acceptance guidelines in RG 1.174, the licensing basis for ECCS compliance with 10 CFR 50.46(a)(1) can be amended.

The exemption request to support closure of GL 2004-02 for VEGP is intended to address ECCS cooling performance design as presented in 10 CFR 50.46(a)(1) as it relates to imposing the deterministic requirements in “other properties.” For the purposes of demonstrating the balance of the acceptance criteria of 10 CFR 50.46, the design and licensing basis descriptions of accidents requiring ECCS operation remain unchanged, as documented in VEGP FSAR Chapters 6 and 15, including analysis methods, assumptions, and results. The performance evaluations for accidents requiring ECCS operation described in FSAR Chapters 6 and 15 are based on the Appendix K large-break loss-of-coolant accident (LBLOCA) analysis and demonstrate that, for breaks up to and including a double ended guillotine break (DEGB) of a reactor coolant pipe, the ECCS will limit the clad temperature to below the limit specified in 10 CFR 50.46 and the core will remain in place and substantially intact with its essential heat transfer geometry preserved.

The requirements of 10 CFR 50.46(a)(1) remain applicable to the model of record that meets the required features of Appendix K. Approval of the requested exemption does not impact the current ECCS evaluation. This evaluation model remains the licensing basis for demonstrating that the ECCS calculated cooling performance following postulated LOCAs meets the acceptance criteria.

The VEGP risk-informed approach determines strainer and core blockage conditional failure probabilities that are input into the plant PRA model to determine the ΔCDF and ΔLERF associated with debris-related failures (Reference 4). The results show that VEGP meets the acceptance guidelines defined in RG 1.174 (Reference 3). The exemption request is specific to the requirement for demonstrating ECCS cooling performance design as required by 10 CFR 50.46(a)(1) as it pertains to the requirements for deterministic analyses described in “other properties.” It is not intended to be applicable to other requirements provided in 10 CFR 50.46 or Appendix K to 10 CFR 50.

As noted in Section 1.2, the NRC staff considers the modeling of sump performance to be an input to the ECCS evaluation model, and therefore the requirements of 10 CFR 50.46 are applicable. Consistent with this, the requirements and attributes for the proposed VEGP risk-informed method include a full spectrum of postulated breaks, up to and including DEGBs on the largest reactor coolant system (RCS) pipes in containment (Reference 4).

Engineering analyses and evaluations used to perform plant-specific prototypical testing consider a wide range of effects, including those addressed in NEI 04-07 (Reference 10) and its associated NRC SE (Reference 11) for evaluation of sump performance. The requested exemption does not affect any of the 10 CFR 50.46(a)(1) or Appendix K

requirements for an acceptable ECCS evaluation model and does not change the ECCS acceptance criteria in 50.46(b) as it applies to the calculated results. Application of the exemption request allows the use of a risk-informed approach to evaluate the effects of debris. The results of the risk-informed method demonstrate that the risk associated with GSI-191 meets the acceptance guidelines of RG 1.174 (Reference 3). The current licensing basis for addressing the adequacy of the ECCS to meet the criteria of 10 CFR 50.46, including the Appendix K large-break LOCA analysis and the associated Chapter 15 accident analysis for LOCA, remains in place.

4.0 BASIS FOR THE EXEMPTION REQUEST

Under 10 CFR 50.12, a licensee may request and the NRC may grant exemptions from the requirements of 10 CFR 50 that are authorized by law, will not present an undue risk to the public health and safety, are consistent with the common defense and security, and when special circumstances are present.

The exemption request meets a key principle of RG 1.174, which states, “The proposed change meets the current regulations unless it is explicitly related to a requested exemption” (Reference 3). This exemption request is provided in conjunction with the proposed changes provided in the risk-informed LAR (see Enclosures 2 through 4).

As required by 10 CFR 50.12(a)(2), the Commission will not consider granting an exemption unless special circumstances are present. Special circumstances are present whenever one of the listed items (i through vi) under 10 CFR 50.12(a)(2) are applicable.

SNC has evaluated the requested exemption against the conditions specified in 10 CFR 50.12(a) and determined that this requested exemption meets the requirements for granting an exemption from the regulation, and that special circumstances are present. The information supporting the determination is provided below.

4.1 Applicability of 10 CFR 50.12(a)(1)

Pursuant to 10 CFR 50.12, “Specific exemptions,” the NRC may grant exemptions from the requirements of this part provided the following three conditions are met as required by 10 CFR 50.12(a)(1):

1. The exemption is authorized by law.

The NRC has authority under the Atomic Energy Act of 1954, as amended, to grant exemptions from its regulations if doing so would not violate the requirements of law. This exemption is authorized by law as 10 CFR 50.12 provides the NRC authority to grant exemptions from 10 CFR 50 requirements with provision of proper justification. Approval of the exemption from 10 CFR 50.46(a)(1) would not conflict with any provisions of the Atomic Energy Act of 1954, as amended, any of the Commission’s regulations, or any other law.

2. The exemption does not present an undue risk to the public health and safety.

The purpose of 10 CFR 50.46 is to establish acceptance criteria for ECCS performance to provide a high confidence that the system will perform its required functions. The requested exemption does not involve any modifications to the plant that could introduce a new accident precursor or affect the probability of postulated accidents, and therefore the probability of postulated initiating events is not increased. The PRA and engineering analysis demonstrate that the calculated risk is very small and consistent with the intent of the Commission's safety goal policy statement, which defines an acceptable level of risk that is a small fraction of other risks to which the public is exposed.

As discussed in previous 10 CFR 50.46 rulemaking, the probability of a large break LOCA is sufficiently low. Application of a risk-informed approach shows a high probability with low uncertainty that the ECCS will meet 10 CFR 50.46 requirements (Reference 4), rather than using deterministic methods to achieve a similar understanding. This is applicable to evaluating acceptable containment sump design in support of ECCS and CSS recirculation modes.

The proposed change is to apply a risk-informed method rather than a deterministic method to establish a high probability of success for performance of ECCS in accordance with the requirements in 10 CFR 50.46(a)(1). The risk-informed approach involves a complete evaluation of the spectrum of LOCA breaks up to and including DEGBs on the largest pipe in the reactor coolant system (Reference 4).

The risk-informed approach analyzes LOCAs, regardless of break size, using the same methods, assumptions, and criteria in order to quantify the uncertainties and overall risk metrics (Reference 4). This ensures that large break LOCAs with a low probability of occurrence and smaller break LOCAs with higher probability of occurrence are both considered in the results. Because the design-basis requirement for consideration of a DEGB of the largest pipe in the reactor coolant system is retained, the existing defense-in-depth and safety margin established for the design of the facility are not reduced.

This exemption only affects 10 CFR 50.46(a)(1) requirements that a licensee is able to demonstrate, using a bounding calculation or other deterministic method, that the ECCS and CSS are capable of functioning during a design basis event. This exemption does not impact the adequacy of the acceptance criteria for cladding performance, which is important to maintain adequate safety margins.

3. The exemption is consistent with the common defense and security.

The exemption involves a change to the licensing basis for the plant that has no relation to the control of licensed material or any security requirements that apply to VEGP Units 1 and 2. Therefore, the exemption is consistent with the common defense and security.

4.2 Applicability of 10 CFR 50.12(a)(2)

This section discusses the presence of special circumstances as related to 10 CFR 50.12(a). 10 CFR 50.12(a)(2) states that the NRC will not consider granting an exemption to the regulations unless special circumstances are present. Special circumstances are present whenever one of the listed items (i through vi) under 10 CFR 50.12(a)(2) are applicable.

Such special circumstances are present in this instance to warrant exemption from the requirements in 10 CFR 50.46(a)(1) “other properties,” which use deterministic calculation methods as the design basis for acceptable sump performance to validate the results of the ECCS evaluation model. Approval of this exemption request would allow the use of a risk-informed method to amend the design basis for acceptable performance of the containment emergency sump, as a validation of inputs in the ECCS evaluation model, and in support of the existing licensing bases for compliance with 10 CFR 50.46.

As described below, special circumstances in 10 CFR 50.12(a)(2)(ii) and 10 CFR 50.12(a)(2)(iii) are present as required by 10 CFR 50.12(a)(2) for consideration of the request for exemption.

4.2.1 Applicability of 10 CFR 50.12(a)(2)(ii)

10 CFR 50.12(a)(2)(ii) applies:

> “Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.”

The intent of 10 CFR 50.46(a)(1) is to ensure ECCS cooling performance design requirements imposed by 10 CFR 50.46 are determined by a rigorous method that provides a high level of confidence in ECCS performance. This exemption request is consistent with that purpose in that use of the proposed risk-informed approach accounts for the effect of debris on the ECCS cooling performance and supports a high probability of successful ECCS performance, based on the risk results meeting the acceptance guidelines of RG 1.174 (Reference 3).

The need for this exemption is based on the requirements in the regulations for using deterministic methods to demonstrate acceptable design. Regulatory requirements are largely based on a deterministic framework, and are established for DBAs, such as the LOCA, with specific acceptance criteria that must be satisfied. Licensed facilities must be provided with safety systems capable of preventing and mitigating the consequences of DBAs to protect public health and safety. The deterministic regulatory requirements were designed to ensure that these systems are highly reliable. The LOCA analysis was established as part of this deterministic regulatory framework.

In comparison, the risk-informed approach considers nuclear safety in a more comprehensive way by examining the likelihood of a broad spectrum of initiating events and potential challenges, considering a wide range of credible events and assessing the risk based on mitigating system reliability.

An objective of 10 CFR 50.46 is to maintain low risk to the public health and safety through a reliable ECCS, as supported by the containment sump. The supporting analysis demonstrates that a risk-informed approach to sump performance is consistent with the Commission's Safety Goals for nuclear power plants and supports ECCS operation with a high degree of reliability. Consequently, the special circumstances described in 10 CFR 50.12(a)(2)(ii) apply.

4.2.2 Applicability of 10 CFR 50.12(a)(2)(iii)

10 CFR 50.12(a)(2)(iii) applies:

> "Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated."

In order to meet a deterministic threshold value for sump debris loads, the debris sources in containment would need to be significantly reduced. The amount of radiological exposure received during the removal and/or modification of insulation from the VEGP Units 1 and 2 containments is dependent on the scope of the changes.

Due to uncertainties in radiation levels, contamination levels, and the required modification scope, it is difficult to predict the total occupational dose associated with insulation removal and/or modifications. Dose estimates for removal of insulation from South Texas Project (STP) are described in some detail in the STP pilot submittal (Reference 13). STP is considered representative of VEGP, since both plants are four-loop Westinghouse designs with similar insulation types and quantities. The expected total dose for replacing insulation in VEGP Units 1 and 2 is estimated generically to be about 200 rem (100 rem per unit) based on the STP pilot submittal.

For the above estimates, the highest dose contributor is personnel work hours in close proximity to high dose sources (Reference 13). The estimate considered person-hours required to erect and remove scaffolding and the dose associated with removal of insulation. However, the estimate did not consider dose associated with disposal of the removed insulation or dose associated with insulation modifications for small-bore piping (Reference 13).

The dose considerations discussed above demonstrate that compliance would result in substantial personnel exposure due to insulation modifications in containment, which is not commensurate with the expected safety benefit based on the risk evaluation results showing that the risk associated with post-accident debris effects is less than the threshold for Region III in RG 1.174 (Reference 4). Consequently, the special

circumstances described in 10 CFR 50.12(a)(2)(iii) apply to the exemption requested by SNC.

4.3 Environmental Consideration

Pursuant to the requirements of 10 CFR 51.41, “Requirement to submit environmental information,” and 10 CFR 51.21, “Criteria for and identification of licensing and regulatory actions requiring environmental assessments,” the following information is provided. As demonstrated below, SNC has determined that this exemption is eligible for categorical exclusion as set forth in 10 CFR 51.22, “Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review,” paragraph (c)(9).

4.3.1 Environmental Impacts Consideration

The requested exemption has been evaluated and determined to result in no significant radiological environmental impacts. This conclusion is based on the following.

The requested exemption is to allow the use of a risk-informed method to demonstrate that the design and licensing bases for the ECCS are not significantly affected by accident-generated and transported debris. The intent of the proposed change is to quantify the risk associated with GSI-191 concerns. This quantification, provided in the form of risk metrics using the guidance in RG 1.174 (Reference 3), demonstrates that the risk is less than the threshold for Region III, “Very Small Changes” (Reference 4). Therefore, the requested exemption supports a change that represents a very small ΔCDF and ΔLERF, which is consistent with the Commission’s Safety Goals for public health and safety.

Since the risk-informed analysis demonstrated that the increases in risk are very small, the requested exemption has a negligible effect on the consequences of an accident, and adequate assurance of public health and safety is maintained. The requested exemption does not involve any changes to the facility or facility operations that could create a new accident or release path, or significantly affect a previously analyzed accident or release path. Therefore, the requested exemption would not cause changes in the types or quantities of radiological effluents, or the permitted effluent release paths.

The requested exemption does not impact the release of radiological effluents during and following a postulated LOCA. The design-basis LOCA radiological consequence analysis in the current licensing basis is a deterministic evaluation based on the assumption of a major rupture of the reactor coolant system piping and a significant amount of core damage as specified in RG 1.4 (Reference 14). The current licensing basis analysis shows the resulting doses to the public, control room, and technical support center personnel are acceptable. The requested exemption does not change the radiological analysis for a LOCA. Therefore, the requested exemption does not affect the amount of radiation exposure resulting from a postulated LOCA.

The requested exemption does not involve any changes to non-radiological plant effluents or any activities that would adversely affect the environment. The requested exemption only pertains to the licensing basis for components located within the restricted area of the facility, to which access is limited to authorized personnel. Therefore, the requested exemption would not create any significant non-radiological impacts on the environment in the vicinity of the plant.

Once the planned strainer height modifications are complete, no additional physical changes to the facility will be needed. As a result, there is no possibility of irreversible or irretrievable commitments of resources. Similarly, the requested exemption does not involve the use of any resources not previously considered by the NRC in its past environmental statements for issuance of the facility operating licenses or other licensing actions for the facility. Therefore, the requested exemption does not involve any unresolved conflicts concerning alternative uses of available resources.

4.3.2 Categorical Exclusion Consideration

SNC has evaluated the requested exemption against the criteria for identification of licensing and regulatory actions requiring environmental assessments in accordance with 10 CFR 51.21. It was determined that the requested exemption meets the criteria and is eligible for categorical exclusion as set forth in 10 CFR 51.22, “Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review,” paragraph (c)(9).

This determination is based on the fact that this exemption request is from requirements under 10 CFR 50 with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, specifically to authorize a change to the licensing basis for ECCS as it relates to acceptable containment sump performance in the recirculation mode following a postulated LOCA. The requested exemption has been evaluated to meet the following criteria under 10 CFR 51.22(c)(9).

<u>(i) The exemption involves no significant hazards consideration.</u>

An evaluation of the three criteria set forth in 10 CFR 50.92(c) as applied to the exemption is provided below. The evaluation is consistent with the no significant hazards consideration determination provided in the LAR (see Enclosure 4).

(1) The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to allow an exemption from 10 CFR 50.46(a)(1) to implement a risk-informed evaluation methodology does not initiate an accident and therefore, the proposed change does not increase the probability of an accident occurring.

Approval of the requested exemption and accompanying LAR would allow the results of a risk-informed evaluation to be included in the FSAR. The evaluation concludes that

the ECCS and CSS will serve their safety functions with a high probability following a LOCA. The evaluation considers the impacts of accident-generated and transported debris on the containment emergency sump strainers in recirculation mode, as well as core blockage due to in-vessel effects.

A physical change will be made to the VEGP Units 1 and 2 RHR sump strainers to reduce the overall strainer height by removing the top two disks from each stack of the RHR strainer assemblies. The RHR sump strainer modifications will ensure the strainers are fully submerged for an increased number of postulated LOCA scenarios (Reference 4). These changes reduce the risk associated with post-accident debris effects because a partially submerged strainer has a head loss acceptance criterion of one half the submerged strainer height, which is significantly less (and therefore more limiting) than the full NPSH margin (Reference 4). The strainer modification is credited in the risk evaluation.

The risk evaluation concludes that the risk associated with the proposed change is very small and within Region III as defined by RG 1.174, for both CDF and LERF (Reference 4). As a result, the required systems, structures, and components (SSCs) supported by the containment sumps will perform their safety functions with a high probability, and the proposed change does not alter or prevent the ability of SSCs to perform their intended function to mitigate the consequences of an accident previously evaluated within the acceptance limits. The safety analysis acceptance criteria in the FSAR continue to be met for the proposed change. Additionally, in accordance with the guidance of RG 1.174, there is substantial safety margin and defense-in-depth that provide additional confidence that the design-basis functions are maintained.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of any of the accidents previously evaluated in the FSAR.

(2) The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change is allowance of a risk-informed analysis of debris effects from accidents that are already evaluated in the VEGP FSAR. No new or different kind of accident is created by the proposed change. VEGP will reduce the height of the RHR strainers by removing the top two disks to ensure the RHR strainers are completely submerged for an increased number of postulated LOCA scenarios. As described previously, this change reduces the risk associated with post-accident debris effects. This change does not introduce any new failure mechanisms or malfunctions that can initiate an accident.

Therefore, the proposed change does not create the possibility for a new or different kind of accident from any accident previously evaluated.

(3) The proposed change does not involve a significant reduction in a margin of safety.

The proposed change does not involve a change in any functional requirements or method of performing functions of plant SSCs. The effects of debris are analyzed for a full spectrum of LOCAs, including DEGBs and partial breaks for all RCS piping sizes. Appropriate redundancy, consideration of loss of offsite power, and worst-case single failure are retained, such that defense-in-depth is maintained.

Application of the risk-informed methodology concludes that the increase in risk from the contribution of debris effects is very small as defined by RG 1.174 (Reference 3) and that there is adequate defense-in-depth and safety margin (Reference 4). Consequently, VEGP determined that the containment sumps would continue to support the safety-related components to perform their design functions when the effects of debris are considered.

The proposed change does not alter the manner in which safety limits are determined or the acceptance criteria associated with a safety limit. The proposed change does not implement any significant changes to plant operation that can challenge an SSCs' capability to safely shut down the plant or maintain the plant in a safe shutdown condition. The proposed change does not affect the existing safety margins in the barriers for the release of radioactivity. There are no changes to any of the safety analyses in the FSAR. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

(ii) The exemption involves no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

No physical modifications (except for the reduced strainer height, which is credited for decreasing the risk associated with post-accident debris effects) or changes to operating requirements are proposed for the facility, including any SSCs relied upon to mitigate the consequences of a LOCA. No changes are made to the safety analyses in the FSAR. Approval of the exemption will require the calculated risk associated with post-accident debris effects to meet the Region III acceptance guidelines in RG 1.174 (Reference 3), thereby maintaining public health and safety. Therefore, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

(iii) The proposed exemption involves no significant increase in individual or cumulative occupational radiation exposure.

No new operator actions are implemented that could affect occupational radiation exposure. No physical modifications (except for reducing the strainer height) or changes to operating requirements are proposed for the facility, including any SSCs relied upon to mitigate the consequences of a LOCA. No changes are made to the safety analyses in the FSAR. Therefore, with respect to installation or use of a facility component

located within the restricted area, approval of this exemption request will not result in a significant increase in individual or cumulative occupational radiation exposure.

Based on the above, SNC concludes that the requested exemption meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

5.0 TECHNICAL JUSTIFICATION FOR THE EXEMPTION

Technical justification for the risk-informed method is provided in the accompanying LAR (see Enclosures 2 through 4), and in Enclosure 3 of the July 2018 submittal (Reference 4).

The proposed risk-informed approach meets the key principles in RG 1.174 (Reference 3) in that it is consistent with the defense-in-depth philosophy, maintains sufficient safety margins, results in a very small increase in risk, and is monitored using performance measurement strategies. The requested exemption to allow use of the risk-informed method is consistent with the key principle in RG 1.174 that requires the proposed change to meet current regulations unless explicitly related to a requested exemption.

The VEGP risk evaluation results (Reference 4) show that the risk associated with post-accident debris effects is within RG 1.174 Region III acceptance guidelines as a “Very Small Change,” and, therefore, is consistent with the Commission’s Safety Goals for public health and safety.

6.0 CONCLUSION

Approval of the requested exemption to allow the use of a risk-informed approach will not present an undue risk to the public health and safety and is consistent with the common defense and security as required by 10 CFR 50.12(a)(1). Furthermore, special circumstances required by 10 CFR 50.12(a)(2) are present for 10 CFR 50.12(a)(2)(ii) and 10 CFR 50.12(a)(2)(iii). The requested exemption has been evaluated and determined to result in no significant radiological environmental impacts. Based on the determination that the risk of the exemption meets the acceptance guidelines of RG 1.174 (Reference 4), the results demonstrate reasonable assurance that the ECCS will function in the recirculation mode and that the public health and safety will be protected.

7.0 REFERENCES

1. NRC Generic Letter (GL) 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents for Pressurized-Water Reactors,” September 13, 2004.
2. ML19120A469, “Final Staff Evaluation for Vogtle Electric Generating Plant, Units 1 and 2, Systematic Risk-Informed Assessment of Debris Technical Report (EPID L-2017-TOP-0038),” September 30, 2019.

3. Regulatory Guide 1.174, “An Approach for Using Probabilistic Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Revision 3, January 2018.
4. SNC Letter NL-18-0915 (ML18193B163 and ML18193B165), “Vogtle Electric Generating Plant – Units 1 and 2, Supplemental Response to NRC Generic Letter 2004-02,” July 10, 2018.
5. SECY-12-0093 (ML121310648), “Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance,” July 9, 2012.
6. NRC Bulletin 2003-01, “Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors,” June 9, 2003.
7. ML103570354, “Staff Requirements – SECY-10-0113 – Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance,” December 23, 2010.
8. ML12349A378, “Staff Requirements – SECY-12-0093 – Closure Options for Generic Safety Issue – 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance,” December 14, 2012.
9. SNC Letter NL-13-0953 (ML13137A130), “Vogtle Electric Generating Plant Proposed Path to Closure of Generic Safety Issue-191, ‘Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance’,” May 16, 2013.
10. NEI 04-07 Volume 1, “Pressurized Water Reactor Sump Performance Evaluation Methodology,” Revision 0, December 2004.
11. NEI 04-07 Volume 2, “Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02,” Revision 0, December 2004.
12. SECY-16-0033 Enclosure 1 (ML15238B016), “Federal Register Notice, Draft Final Rulemaking: Performance-Based Emergency Core Cooling Systems Cladding Acceptance Criteria,” February 16, 2016.
13. STP Letter NOC-AE-13002986 (ML13175A211), “South Texas Project, Units 1 and 2, Docket Nos. STN 50-498 and STN 50-499, Revised STP Pilot Submittal and Requests for Exemptions and License Amendment for a Risk-Informed Approach to Resolving Generic Safety Issue (GSI)-191 (TAC Nos. MF0613 and MF0614)”, June 19, 2013.
14. Regulatory Guide 1.4, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors,” Revision 2, June 1974.

Vogtle Electric Generating Plant
Licensing Submittal for a Risk Informed Resolution of Generic Letter 2004-02

Enclosure 2

Implementation of a Risk-Informed Approach for Addressing GSI-191

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Enclosure 2
Implementation of a Risk-Informed Approach for Addressing GSI-191

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, Southern Nuclear Operating Company (SNC) requests an amendment to Operating Licenses NPF-68 and NPF-81 for Vogtle Electric Generating Plant (VEGP) Units 1 and 2. The proposed amendment will revise the licensing basis as described in the VEGP Final Safety Analysis Report (FSAR) to allow the use of a risk-informed approach to address safety issues discussed in Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance."

The proposed changes would allow the use of a risk-informed approach to address Generic Letter (GL) 2004-02 (Reference 1) for VEGP. The risk-informed approach is consistent with the guidance of NRC Regulatory Guide (RG) 1.174 (Reference 2) and SECY-12-0093 (Reference 4). The proposed changes will apply only for the effects of accident-generated debris as described in GSI-191 and GL 2004-02 (Reference 1).

In addition, as described in Enclosure 3 of this submittal, SNC proposes to amend the VEGP Unit 1 and Unit 2 operating licenses to revise the technical specifications (TS) to address containment accident generated and transported debris and the potential impact on the containment sump (Reference 3).

The regulatory evaluation and environmental consideration for incorporating the risk-informed methodology and technical specification changes are addressed in Enclosure 4 of this submittal.

2.0 DETAILED DESCRIPTION

GSI-191 identifies the possibility that debris generated during a loss of coolant accident (LOCA) could clog the containment recirculation sump strainers in pressurized water reactors (PWRs) and result in loss of net positive suction head (NPSH) for the emergency core cooling system (ECCS) and containment spray system (CSS) pumps, impeding the flow of water from the sumps. Additionally, debris that passes through the strainer could affect safety functions of the components downstream of the strainer or challenge long-term core cooling due to debris accumulation in the reactor core.
GL 2004-02 requested licensees to address GSI-191 issues and demonstrate compliance with the ECCS acceptance criteria in 10 CFR 50.46 by performing analyses using an NRC-approved methodology. SNC's supplemental response to GL 2004-02 was provided in Enclosures 2 and 5 of the July 2018 submittal (Reference 5) and additional details are provided in Attachment 3 of this enclosure.

2.1 System Design and Operation

A fundamental function of the ECCS is to recirculate water that has collected in the containment sump following a break in the reactor coolant system (RCS) piping to ensure long-term removal of decay heat from the reactor fuel. Leaks from the RCS in excess of the plant's normal makeup capability (scenarios known as LOCAs), are part of

a nuclear power plant's design bases. Hence, nuclear plants are designed and licensed with the expectation that they are able to remove reactor decay heat following a LOCA to prevent core damage. Long-term cooling following a LOCA is also a basic safety function for nuclear reactors. The recirculation sump located in the lower areas of the reactor containment structure provides a water source to the ECCS in a PWR once the initial water source has been depleted and the systems are switched over to recirculation mode for extended cooling of the core.

ECCS

The ECCS at VEGP consists of the centrifugal charging pumps; safety injection (SI) pumps; residual heat removal (RHR) pumps; accumulators, boron injection tank (Unit 1 only); RHR heat exchangers; refueling water storage tank (RWST); and the associated piping, valves, instrumentation, and other related equipment. The primary function of the ECCS following an accident is to remove the stored and fission product decay heat from the reactor core such that fuel rod damage, to the extent that it would impair effective cooling of the core, is prevented.

As stated in the VEGP FSAR:

> The ECCS is designed to cool the reactor core and to provide additional shutdown capability following initiation of the following accident conditions:
>
> A. LOCA including a pipe break or a spurious relief or safety valve opening in the RCS which would result in a discharge larger than that which could be made up by the normal makeup system.
>
> B. Loss-of-secondary-coolant accident including a pipe break or a spurious relief or safety valve opening in the secondary steam system which would result in an uncontrolled steam release or a pipe break in the secondary feedwater system.
>
> C. A steam generator tube rupture accident.
>
> Emergency core cooling following a LOCA is divided into three phases:
>
> A. Short-Term Core Cooling/Cold Leg Injection Phase
>
> The cold leg injection phase is defined as that period during which borated water is delivered from the RWST and accumulators to the RCS cold legs.
>
> B. Long-Term Core Cooling/Cold Leg Recirculation
>
> The cold leg recirculation phase is that period during which borated water is recirculated from the containment emergency sump to the RCS cold legs.

C. Long-Term Core Cooling/Hot Leg Recirculation Phase

The hot leg recirculation phase is that period during which borated water is recirculated from the containment emergency sump to both the RCS hot legs and RCS cold legs.

In the event of an accident, the RHR pumps are started automatically on receipt of an SI signal. The RHR pumps take suction from the RWST during the injection phase and are automatically realigned to the containment emergency sump during the recirculation phase, although manual action is required to close the suction path from the RWST. In the event of an accident, the SI pumps are started automatically on receipt of an SI signal. These pumps deliver water to the RCS from the RWST during the injection phase and from the containment emergency sump via the RHR pumps during the recirculation phase. When a predetermined low RWST level is reached, the SI and charging pumps are manually aligned to take suction from the RHR pump discharge headers.

The RWST serves as a source of emergency borated cooling water for injection and containment spray.

Containment Spray System

The CSS at VEGP consists of two pumps, spray ring headers and spray nozzles, valves, and connecting piping. Initially, water from the RWST is used for the containment spray followed by water recirculated from the containment emergency sump. The recirculated spray is mixed with trisodium phosphate in the containment sump region. As the RWST empties, containment spray pumps switchover to the recirculation mode of operation.

Description of Planned Sump Modification

The Vogtle ECCS sump strainer consists of stacked disk strainers designed by General Electric. The currently installed strainers for RHR and CSS consist of four parallel, vertically stacked, modular disk strainer assemblies that are connected to a plenum installed over each sump. There are four separate strainers, one for each RHR pump, and one for each CS pump. Each of the two RHR strainer assemblies provides approximately 765 ft^2 of perforated plate surface area and 179 ft^2 of circumscribed surface area per sump. Each of the two CS strainer assemblies provides approximately 590 ft^2 of perforated plate surface area and 139 ft^2 of circumscribed surface area.

The RHR strainers will be modified to reduce the overall height by approximately 6 inches. The modified RHR strainer assemblies will each provide approximately 677.6 ft^2 of perforated plate surface area and 159 ft^2 of circumscribed surface area. All of the analyses in the July 2018 submittal (Reference 5) to support use of the risk-informed methodology were performed with the modified strainer configuration.

The new strainer design does not involve backflushing or any other active approach.

2.2 Current Licensing Basis Requirements

10 CFR 50.46(a)(1) requires that the criteria set forth in 10 CFR 50.46 paragraph (b), be demonstrated with ECCS cooling performance calculated in accordance with an acceptable evaluation model and includes a requirement for “other properties” with regard to the methodology for showing those requirements are met. The methodology is governed by 10 CFR 50.46(a)(1) and is deterministic with no provision for a risk-informed approach. This LAR supports an exemption to 10 CFR 50.46(a)(1) as described in Enclosure 1.

2.3 Reason for the Proposed Change

In order to meet a deterministic threshold value for sump debris loads, the debris sources in containment would need to be significantly reduced. The amount of radiological exposure received during the removal and/or modification of insulation from the VEGP Units 1 and 2 containments is dependent on the scope of the changes. As discussed in Enclosure 1 of this submittal, the expected total dose for replacing insulation in VEGP Units 1 and 2 is estimated to be about 200 rem (100 rem per unit). This estimate does not include dose associated with disposal of the removed insulation or dose associated with insulation modifications for small-bore piping.

The dose considerations discussed above demonstrate that compliance would result in substantial personnel exposure due to insulation modifications in containment, which is not commensurate with the expected safety benefit based on the risk evaluation results showing that the risk associated with post-accident debris effects is less than the threshold for Region III in RG 1.174 (Reference 2).

2.4 Description of the Proposed Change

The proposed change in methodology in this license amendment request (LAR) is to use a risk-informed approach to address the effects of accident-generated and transported debris on the containment emergency sumps instead of a deterministic approach. The details of the risk-informed approach are provided in Enclosure 3 of the July 2018 submittal (Reference 5). The debris analysis covers a full spectrum of postulated LOCAs, including double-ended guillotine breaks (DEGBs), to provide assurance that the most severe postulated LOCAs are evaluated.

Attachment 1 to this enclosure provides markups to the FSAR, which includes revision of applicable FSAR safety system and design bases descriptions that take credit for the risk-informed evaluation described in Appendix 6A. The markups are provided for information only.

3.0 TECHNICAL EVALUATION

The methodology change affects the analysis of systems and functions that are susceptible to the effects of accident-generated debris. The affected systems are those

supported by the containment recirculation sumps and strainers during the recirculation phase of LOCA mitigation. These include the ECCS and CSS.

The risk-informed approach identifies LOCA break scenarios that fail any one of the following GSI-191 acceptance criteria, as determined by break-specific analysis: debris limits (based on VEGP-specific head loss testing), strainer structural margin (i.e., differential pressure limit), un-submerged strainer head loss limit, void fraction limit, flashing limit, pump NPSH margin, and core blockage limits.

VEGP quantified the change in core damage frequency (ΔCDF) and change in large early release frequency (ΔLERF) using the plant probabilistic risk assessment (PRA) model. The debris related failures were computed for each break analyzed and were binned into categories based on break size and whether the failure was a strainer or core failure. The conditional failure probability (CFP) was calculated for each category by allocating the overall plant-wide LOCA frequency to individual welds and break sizes using a top-down methodology. The CFPs were input into the GSI-191 PRA model, which was used to calculate the ΔCDF and ΔLERF due to GSI-191 considerations. The resulting ΔCDF and ΔLERF were compared with the acceptance guidelines in RG 1.174 (Reference 2). The results of the evaluation show that the risk from the proposed change is "very small" in that it is in Region III of RG 1.174. These results meet the requirement for the risk from debris to be small in paragraph (e) of the proposed 10 CFR 50.46c rule change (Reference 6) and associated draft RG 1.229 (Reference 7). See Enclosure 3 of the July 2018 submittal (Reference 5) for a more detailed description of the risk quantification. The methodology includes conservatisms in the plant-specific testing and in the assumption that all unbounded breaks would result in failure of the ECCS to provide long term core cooling.

The proposed VEGP FSAR Appendix 6A (see Attachment 1) describes the risk-informed approach used to confirm that the ECCS and CSS will operate with a high probability following a LOCA when considering the impacts of accident-generated debris. A physical modification to reduce the strainer height will be implemented by VEGP to achieve full submergence of the RHR strainers for an increased number of LOCA scenarios. This change reduces the risk associated with the post-accident debris effects, as described in Enclosure 3 of the July 2018 submittal (Reference 5) and is credited in the risk evaluation.

The performance evaluations for accidents requiring ECCS operation are described in FSAR Chapters 6 and 15 (Reference 8), based on the VEGP Units 1 and 2 Appendix K large-break LOCA analysis. System redundancy, independence, and diversity features are not changed for those safety systems credited in the accident analyses. No new programmatic compensatory activities or reliance on manual operator actions are required to implement this change.

3.1 Engineering Analysis Overview

The design and licensing basis descriptions of accidents requiring ECCS and CSS operation, including analysis methods, assumptions, and results provided in FSAR Chapters 6 and 15 remain unchanged. This is based on the functionality of the ECCS and CSS during design basis accidents being confirmed by demonstrating that safety margin and defense-in-depth (DID) are maintained with high probability (Reference 5).

The methodology for calculating the risk associated with GSI-191 concerns evaluates a full spectrum of breaks up to and including DEGBs for all RCS pipe sizes. The results show that the risk associated with GSI-191 concerns for VEGP Units 1 and 2 is "very small" as defined by Region III in RG 1.174 (Reference 2). The detailed technical description of the risk quantification process is presented in Enclosure 3 of the July 2018 submittal (Reference 5).

This LAR is requesting a change to the licensing basis such that the effects of LOCA generated and transported debris can be evaluated using a risk-informed methodology.

Detailed evaluations of DID and safety margin are presented in Enclosure 4 of the July 2018 submittal (Reference 5). The evaluations determined that there is substantial DID and safety margin that provides a high level of confidence that the calculated risk is conservative and that the actual risk is likely much lower.

3.2 Resolution of Limitations and Conditions

VEGP provided a response to GL 2004-02 with a risk-informed GSI-191 evaluation in the July 2018 submittal (Reference 5). The NRC reviewed and approved the proposed risk-informed evaluation methodology for VEGP with certain limitations and conditions described in the staff evaluation report (Reference 9). The resolution of these limitations and conditions is described in the following sections.

3.2.1 Limitation/Condition 1

The applicability of the NRC's acceptance is limited to the structures, systems, and components; plant configurations; and operations described in Enclosures 2, 3, and 4 of SNC's letter dated July 10, 2018 and the strainer design described in the Section entitled, "16-Disk ECCS Suction Strainer Summary," of Enclosure 2.

<u>Resolution</u>

Two plant modifications were described in the July 2018 submittal (Reference 5). The modification of the operating procedure to inject additional RWST water for breaks that do not initiate containment sprays has been implemented. The modification to remove the top two disks on the RHR strainers is currently scheduled for implementation during the Fall 2021 refueling outage for Unit 1 (V1R23) and the Spring 2022 refueling outage for Unit 2 (V2R22).

3.2.2 Limitation/Condition 2

The applicability of the NRC's acceptance is limited to the Vogtle assessment of risk attributable to debris described in Enclosures 1 and 3 of SNC's letter dated July 10, 2018.

Resolution

The requested exemption described in Enclosure 1 of this submittal and the license amendment described in Enclosures 2 through 4 of this submittal are based on the NRC-accepted risk assessment. Note that there were minor changes to the risk assessment to correct the coatings error as described in Attachment 3 of this enclosure.

3.2.3 Limitation/Condition 3

Describe in-vessel analysis, establish in-vessel acceptance criteria, and demonstrate the criteria are met.

Resolution

The in-vessel analysis, acceptance criteria, and results are described in Attachment 3 of this enclosure.

3.2.4 Limitation/Condition 4

Address Key Principle 1 (i.e., the proposed licensing basis change meets the current regulations unless it is explicitly related to a requested exemption) and Key Principle 5 (i.e., the impact of the proposed licensing basis change should be monitored using performance measurement strategies) in RG 1.174, Revision 3.

Resolution

Key Principle 1 is addressed with a request for exemption to certain aspects of 10 CFR 50.46(a)(1), which is provided in Enclosure 1 of this submittal. Key Principle 5 is addressed as described below.

VEGP has implemented procedures and programs for monitoring, controlling, and assessing changes to the plant that have a potential impact on plant performance related to GSI-191 concerns. These procedures and programs provide the capability to monitor the performance of the sump strainers and assess impacts to the inputs and assumptions used in the PRA and the associated engineering analysis that support the proposed change. Programmatic requirements ensure that the potential for debris loading on the sump strainers does not materially increase. These include:

- The technical requirements manual (TRM) Surveillance 13.5.1.1 implemented by VEGP procedures requires visual inspections of all accessible areas of the containment to check for loose debris, and each containment sump to check for debris.
- The VEGP general engineering guidance procedure includes provisions for managing potential debris sources such as insulation, qualified coatings, addition of aluminum or zinc, and potential effects of post-accident debris on recirculation flow paths and downstream components. The procedure has been augmented to explicitly require that changes involving any work or activity inside containment be evaluated for the potential to affect the following:
  - Insulation inside containment
  - Fire barrier material inside containment
  - Coatings (qualified and unqualified) inside containment
  - Inactive volumes in containment
  - Labels inside containment
  - Buffer changes (iodine and pH control)
  - Structural changes (i.e., choke points) in containment
  - Downstream effects (piping components downstream of the ECCS sump screens)
- A 10 CFR 50.59 screening or evaluation is required to be completed for all design changes.
- The coatings program ensures that protective coatings used inside containment are procured, applied, and maintained in compliance with applicable regulatory requirements.
- As part of the VEGP condition reporting process, condition reports are written when adverse conditions are identified during containment inspections or surveillances of the containment emergency sumps and strainers.
- The VEGP Maintenance Rule program includes performance monitoring of the high safety significant functions associated with the ECCS and CSS. The Maintenance Rule program provides continued assurance of the availability and reliability for performance of the required functions.
- The on-line configuration risk management procedure establishes the administrative controls for performing on-line maintenance of structures, systems, and components (SSCs) to enhance overall plant safety and reliability.
- The VEGP quality assurance (QA) program is implemented and controlled in accordance with the Quality Assurance Topical Report (QATR) and is applicable to SSCs to an extent consistent with their importance to safety. The QA program complies with the requirements of 10 CFR 50, Appendix B and other program commitments as appropriate.

The proposed change does not involve any changes to ASME Section XI inspection programs or mitigation strategies that have been shown to be effective in early detection and mitigation of weld and material degradation in Class I piping applications.

3.2.5 Limitation/Condition 5

Identify key elements of the risk-informed analysis (e.g., methods, approaches, and data) that will be described in the Vogtle UFSAR.

Resolution

Key elements of the risk-informed analysis include:

1. The methodology used to quantify the amount of debris generated at each break location, including the assumed zone of influence (ZOI) size based on the target destruction pressure and break size, and the assumed ZOI shape (spherical or hemispherical) based on whether the break is a DEGB or partial break. This requirement applies to the methodology, but not the tool or program used.
2. The methodology used to evaluate debris transport to the RHR and CSS strainers. This requirement applies to the methodology, but not the tool or program used.
3. The methodology used to quantify chemical precipitates, including the refinements to WCAP-16530-P-A, application of the solubility correlation, and application of the WCAP-17788-P autoclave testing. This requirement applies to the methodology, but not the tool or program used.
4. The strainer debris limits shown in TS Bases Table B 3.6.7-1 (see Attachment 3 of Enclosure 3 in this submittal), which are based on tested and analyzed debris quantities.
5. The methodology and acceptance criteria used to assess ex-vessel component blockage and wear.
6. The methodology used to assess in-vessel fiber accumulation and the associated limits (see Attachment 3 of this enclosure). This requirement applies to the methodology, but not the tool or program used.
7. The methodology used to quantify CFPs, ΔCDF and ΔLERF. This requirement applies to the methodology, but not the tool or program used.

These key elements are discussed in FSAR Appendix 6A as shown in Attachment 1 of this enclosure.

3.2.6 Limitation/Condition 6

Identify key elements of the risk-informed analysis and corresponding methods, approaches, and data that, if changed, would constitute a departure from the method used in the safety analysis as defined by 10 CFR 50.59.

Resolution

The key elements of the risk-informed analysis are described in Section 3.2.5. Changes to these key elements, which were approved by the NRC (Reference 9), are to be evaluated as a potential "departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses" in accordance with 10CFR50.59(c)(2)(viii).

The strainer debris limits, shown in TS Bases Table B 3.6.7-1 (see Attachment 3 of Enclosure 3 in this submittal), are included as a key element of the risk-informed methodology. Any changes to the debris limits in Table B 3.6.7-1 are subject to review under 10 CFR 50.59 and 10 CFR 50.71(e) reporting requirements.

3.2.7 Limitation/Condition 7

Identify the relevant elements of the risk-informed assessment that may need to be periodically updated. The licensee must describe the program or controls that will be used to ensure relevant elements of the risk-informed assessment are periodically updated.

Resolution

Limitation/Condition 4 describes the procedures and programs for monitoring, controlling, and assessing changes to the plant that have a potential impact on plant performance related to GSI-191 concerns. Nonconformances with existing evaluations, or problem identification, will be entered into the station corrective action program for evaluation and corrective actions, as appropriate.

In addition, periodic updates will be made to the risk-informed assessment in accordance with SNC risk-informed engineering procedures. The risk-informed assessment will be updated within at least 48 months following initial NRC approval of the LAR or the most recent update. The periodic updates will include the following elements:

- Addition, removal, or replacement of materials in containment that affect debris quantities.
- Physical or procedural modifications that affect inputs to the NARWHAL model including pump flow rates, sump water level, sump pH, sump temperature, etc.
- Changes to PRA inputs including LOCA frequencies and success criteria. As stated in Enclosure 1 of the July 2018 submittal (Reference 5), reliability data, unavailability data, initiating events frequency data, human reliability data, and other such PRA inputs are reviewed approximately every two fuel cycles and updated as necessary to maintain the PRA consistent with the as-operated plant.

3.2.8 Limitation/Condition 8

Describe a reporting and corrective action strategy for addressing situations in which an update to the risk-informed assessment reveals that the acceptance guidelines described in Section 2.4 of RG 1.174, Revision 3, have been exceeded.

Resolution

Non-conforming conditions will be addressed in accordance with the VEGP corrective action program.

Guidance for supporting a prompt operability evaluation for conditions related to post-accident debris and containment sump operability, including resolution strategies, is described in Attachment 2 of this enclosure and is provided for information only. The operability evaluation guidance will be placed in station procedures. Nonconforming conditions that make the containment sump(s) inoperable for longer than the required TS completion time (see Enclosure 3 of this submittal) will meet the 10 CFR 50.73 reporting criteria for a condition prohibited by TS. Conditions that cause the containment sump(s) to be inoperable and result in the debris-related ΔCDF or ΔLERF to be greater than the RG 1.174 Region III acceptance guidelines are to be reported in accordance with 10 CFR 50.72 and 10 CFR 50.73, as applicable.

3.2.9 Limitation/Condition 9

Correct the error concerning the evaluation of transported coatings debris loads described in SNC's letter dated December 4, 2018. Specifically, provide corrected coating debris volumes and describe how coating debris loads on the strainers are determined. In addition:

a. Verify that the use of the corrected coating debris volumes has a limited impact on strainer head loss and the head loss is acceptable. Also, the licensee must describe the method of verification.

b. Verify that the use of the corrected coating debris volumes has a limited impact on CDF and does not result in exceeding the acceptance guidelines for very small change in risk, as described in Section 2.4 of RG 1.174, Revision 3. Also, the licensee must describe the method of verification.

Resolution

The corrected coatings analysis, results, and methods for verification are described in Attachment 3 of this enclosure.

3.3 Technical Evaluation Conclusion

The technical evaluation shows that the functionality of the ECCS and CSS during design basis accidents is confirmed by demonstrating that safety margin and defense-in-depth are maintained with high probability (Reference 5).

4.0 REFERENCES

1. NRC Generic Letter (GL) 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents for Pressurized-Water Reactors,” September 13, 2004.
2. Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Revision 3, January 2018.
3. TSTF-567 (ML17214A813), “Add Containment Sump TS to Address GSI-191 Issues,” Revision 1.
4. SECY-12-0093 (ML121310648), “Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance,” July 9, 2012.
5. SNC Letter NL-18-0915 (ML18193B163 and ML18193B165), “Vogtle Electric Generating Plant – Units 1 and 2, Supplemental Response to NRC Generic Letter 2004-02,” July 10, 2018.
6. ML15238A947, “SECY-16-0033: Final Draft Rulemaking - 10 CFR 50.46c: Emergency Core Cooling Systems Performance During Loss-of-Coolant Accidents,” April 4, 2016.
7. Draft Regulatory Guide 1.229 (ML16062A016), “Risk-Informed Approach for Addressing the Effects of Debris on Post-Accident Long-Term Core Cooling,” March 2016.
8. Alvin W. Vogtle Electric Generating Plant FSAR, Revision 23, November 2019, Sections 6 and 15.
9. ML19120A469, “Final Staff Evaluation for Vogtle Electric Generating Plant, Units 1 and 2, Systematic Risk-Informed Assessment of Debris Technical Report (EPID L-2017-TOP-0038),” September 30, 2019.

Vogtle Electric Generating Plant

Licensing Submittal for a Risk-Informed Resolution of Generic Letter 2004-02

Enclosure 2

Implementation of a Risk-Informed Approach for Addressing GSI-191

Attachment 1

Proposed FSAR Changes (Mark-Up) for Information Only

1.9.80 REGULATORY GUIDE 1.80, JUNE 1974, PREOPERATIONAL TESTING OF INSTRUMENT AIR SYSTEMS

1.9.80.1 Regulatory Guide 1.80 Position

Withdrawn. Refer to paragraph 1.9.68.3.

1.9.81 REGULATORY GUIDE 1.81, REVISION 1, JANUARY 1975, SHARED EMERGENCY AND SHUTDOWN ELECTRIC SYSTEMS FOR MULTI-UNIT NUCLEAR POWER PLANTS

1.9.81.1 Regulatory Guide 1.81 Position

This guide describes an acceptable method for complying with NRC requirements with respect to the sharing of onsite emergency and shutdown electric systems for multiunit nuclear power plants.

1.9.81.2 VEGP Position

Conform. Refer to paragraph 8.3.1.1.2M for further discussion.

1.9.82 REGULATORY GUIDE 1.82, REVISION 3, WATER SOURCES FOR LONG-TERM RECIRCULATION COOLING FOLLOWING A LOSS-OF-COOLANT ACCIDENT

1.9.82.1 Regulatory Guide 1.82 Position

This guide describes an acceptable method for designing, fabricating, and testing of sump or suction inlet conditions for pumps in the emergency core cooling and containment spray systems.

1.9.82.2 VEGP Position

Conform except for the following exceptions:

1. The subject guide describes requirements for a trash rack to perform the following:

    a) Protect the inner screen from missiles that may be generated by a LOCA or by trash

    b) Prevent large debris from entering the screen.

VEGP is taking exception to the above requirements based on the following reasons:

There are no high-energy line breaks postulated to occur near the screens, and there are no missiles generated in the vicinity of the suction strainers; therefore, there are no jet loads, no pipe whip restraint loads, nor missiles applicable to the screens. ~~The screens are designed to withstand the loading for the largest postulated debris quantity, pieces, and types.~~ The design of the stacked disk screen prevents large debris from reaching the perforated inner area of the screens due to small slots between the strainer disks.

2. The subject guide suggests that a vertically mounted screen be provided:

VEGP takes exception to the vertically mounted screen for the following reasons:

The vertical modular stacked disk screen does not allow gravitationally-influenced settling on the perforated flow area; the top surface is a solid stainless steel plate and protects the perforated plates below it. Therefore, the horizontal screens are functionally equivalent to vertical screens.

See subsections 6.2.2, ~~and~~ 6.3.2*, and Appendix 6A* for further discussion *on how*~~. Note that some nuclear fuel used at the VEGP units may contain design features that provide for flow passage dimensions smaller than those of the containment sump screen. This condition could lead to flow blockage downstream of the containment sump. The fuel flow passage dimensions have been evaluated and the conclusion has been reached that excessive flow blockage in the fuel will not occur. Therefore,~~ long-term core cooling requirements and 10 CFR 50.46 emergency core cooling system (ECCS) acceptance criteria ~~will continue to be met~~ *are addressed for containment post-accident debris effects*.

1.9.83 REGULATORY GUIDE 1.83, REVISION 1, JULY 1975, INSERVICE INSPECTION OF PRESSURIZED WATER REACTOR STEAM GENERATOR TUBES

1.9.83.1 Regulatory Guide 1.83 Position

This guide describes a method acceptable to the NRC for implementing the applicable general design criteria by reducing the probability and consequences of steam generator tube failures through periodic inservice inspection for early detection of defects and deterioration.

1.9.83.2 VEGP Position

The steam generators are designed to permit access to tubes for inspection and/or repair or plugging (if necessary). Plugging will be accomplished by either welded plugs or nonwelded mechanical plugs. The inservice inspection program is discussed in subsection 5.4.2 and the Technical Specifications.

6.2.2.2 Containment Spray System

6.2.2.2.1 Design Bases

6.2.2.2.1.1 Safety Design Bases.

A. The containment spray system is designed to withstand the effects of natural phenomena such as earthquakes.

B. The containment spray system is automatically placed in operation on receipt of two out of four containment pressure (high-3) signals.

C. The containment spray system is designed so that a single failure of any active component, assuming loss of offsite power, cannot impair the capability of the system to perform its safety function during the injection phase.

A single active or passive failure cannot impair the capability of the system to perform its safety function during the recirculation phase.

D. Active components of the containment spray system are capable of being tested during plant operation. Provisions are made for inspection of major components at appropriate times specified in ASME Boiler and Pressure Vessel Code, Section XI.

E. The containment spray system components are designed to remain functional during the accident environment and to withstand the dynamic effect of the accident.

F. The containment spray system in conjunction with the containment cooling system is capable of removing sufficient thermal energy and subsequent decay heat from the containment atmosphere following the postulated LOCA or MSLB accident to maintain the containment pressure below design values.

G. The containment spray system is designed and fabricated to codes consistent with Regulatory Guide 1.26 as described in table 3.2.2-1 and Seismic Category 1 in accordance with Regulatory Guide 1.29. The power supply and control functions are in accordance with Regulatory Guide 1.32.

H. The design basis for the CSS with regard to the effects of debris on the emergency sump strainers is a risk-informed analysis, which shows the risk associated with the effects of debris is very small as defined by Regulatory Guide 1.174. The conclusion is based on plant-specific testing and analyses using inputs and assumptions that provide safety margin and defense-in-depth.

Details of the design basis for the effects of debris on the function of the emergency sump strainers are provided in FSAR Appendix 6A.

6.2.2.2.1.2 Power Generation Design Bases. The containment spray system has no power generation design bases.

6.2.2.2.2.3 Component Description. The mechanical components of the containment spray system are described in this section. Component design parameters are given in table 6.2.2-4. Parts of the system in contact with borated water are stainless steel or an equivalent corrosion-resistant material.

Corrosion tests have been performed on the materials that the spray would come in contact with, e.g., the paint on the inside of the containment structure. (Tests are detailed in WCAP-7825 and NUREG CR-3803.) These tests have shown that no significant amount of corrosion products is produced. Those corrosion products or any chemical precipitation of appreciable size that does occur is trapped by the sump filter screen. The screen size is smaller than the line piping, residual heat removal heat exchanger tubes, and the spray nozzles, so that particles which could potentially block the system will be filtered out. The spray nozzle material (stainless steel, SA351) was chosen for its resistance to corrosion. Tests have been performed on this material in the same type of environment that the nozzle would see during spray actuation. (Corrosion tests of austenitic stainless steel are detailed in WCAP-7803 and WCAP-11611.) The resulting corrosion levels were very low.

Stress corrosion does not present a problem since it only becomes a factor under a combination of conditions of high stress levels at high temperatures for extended periods of time. The stress levels in the pumps during operation are relatively low; the working temperature of the pumps is less than 212°F.

The pumps are located outside the containment. The external surfaces of the pumps, as well as the pump motors, are not subjected to the corrosive atmosphere of the spray solution. Only the internal stainless steel surfaces of the pumps are exposed to a corrosive atmosphere.

6.2.2.2.2.3.1 Refueling Water Storage Tank. This tank serves as a source of emergency borated cooling water for injection and containment spray. It is normally used to fill the refueling canal for refueling operations. During all other plant operating periods, it is aligned to the suction of the emergency core cooling pumps and the containment spray pumps. The tank is a concrete tank lined with type 304 stainless steel plates. The nominal tank volume is 715,000 gal. The contents of this tank are protected from freezing by a sludge mixing system which includes an electric circulation heater. Lines and appurtenances to the RWST serving a safety-related function are heat traced as necessary to prevent freezing.

6.2.2.2.2.3.2 Containment Spray Pumps. The containment spray pumps are of the horizontal centrifugal type, driven by electric motors powered from the emergency buses.

The design head of the pumps is sufficient to continue at rated capacity with a minimum level in the RWST against a head equivalent to the sum of the design pressure of the containment, the head to the uppermost nozzles, line losses, and nozzle pressure losses. The containment spray system is designed so that adequate net positive suction head (NPSH) is provided to the containment spray pumps, in accordance with Regulatory Guide 1.1.

~~To demonstrate that adequate NPSH is provided for the containment spray pumps, it is only necessary to demonstrate that the NPSH is adequate under the worst limiting conditions.~~ *The recirculation mode of operation at -0.3 psig containment pressure and 211°F sump temperature results in the limiting NPSH.* ~~NPSH for the containment spray pumps is evaluated for both the~~

~~injection and recirculation modes of operation for the DBA. The recirculation mode of operation gives the limiting NPSH requirement for the containment spray pumps, and the NPSH available is determined from the following equation:~~

$$\cancel{NPSH_{actual} = (h)_{containment\ pressure} - (h)_{vapor\ pressure} + (h)_{static\ head} - (h)_{loss}}$$

~~To evaluate t~~*T*he adequacy of the available NPSH~~, the debris generation from a high energy line break within the containment and the resultant impact on the containment spray pump performance~~ was evaluated *as described in Appendix 6A* ~~using the following~~.

~~I. The minimum flood level inside the containment based on the RWST water discharged during injection and switchover to recirculation, plus the water volume of 3 accumulator tanks, is at 177 ft 0 in. The minimum flood level also takes into account the flooding of the reactor cavity/incore instrument tunnel via the penetrations in the primary shield.~~

~~J. The containment spray pump suction elevation is 121 ft 5 in.~~

~~K. The calculated maximum line losses which include losses through pipe fittings, valves, entrance and exit are 10.3 ft at 3200 gal/min, containment spray pump design flow.~~

~~L. The quantity of insulation debris generated by the double-ended rupture of the RCS hot leg at its connection to the steam generator, the limiting case, is 432 ft^3.~~

~~M. Hydraulic model studies performed on a scale model of the VEGP containment emergency sump configuration showed that the approach velocities to the four sumps during two train operation was essentially the same. Therefore, it is assumed that the volume of debris transported and deposited on one sump screen is one quarter of the total debris generated by the postulated pipe break event. For one RHR and CS pump train operating the volume of debris transported and deposited on one CS sump screen is 41.6% of the total debris.~~

~~However, the volume of debris transported to the sumps is also evaluated as a pump flow-weighted ratio. The most limiting case is reflected in paragraph 6.2.2.2.2.3.2.~~

~~N. The maximum containment spray emergency sump screen head loss, assuming 41.6% of the total debris generated is evenly deposited on the sump screen is less than 16.65 ft.~~

~~O. The vapor pressure of the pumped liquid is assumed to be in equilibrium with the containment ambient pressure (i.e., no credit is taken for subcooling of the sump fluid) for sump fluid temperatures of equal to or greater than 211°F:~~

$$\cancel{h_{containment\ ambient\ pressure} = h_{vapor\ pressure}}$$

~~The NPSH equation for the recirculation mode when suction is taken from the containment emergency sump becomes:~~

$$\cancel{NPSH_{available} = h_{static\ head} - h_{line\ loss} - h_{sump\ screen\ loss}}$$

~~Using the above values, the calculated containment spray pump available NPSH, after considering debris effects, is greater than 36 ft. The required NPSH at 3200 gal/min is 19.5 ft. Therefore, adequate available NPSH margin is provided for proper containment spray pump operation.~~

Design parameters for these pumps are presented in table 6.2.2-4.

6.2.2.2.2.3.3 Spray Nozzles. The hollow cone spray nozzles are not subject to clogging by particles less than 1/4 in. in size and produce a drop size spectrum with a mean diameter of less than 700 mm at 40 psi differential pressure. During spray recirculation operation, the water is screened through perforated plates with 3/32-in. diameter holes (a small~~%age~~ *percentage* – 124 holes – are larger than 3/32-in. diameter but < ¼-in. diameter – see section 6.1.2.) before leaving the containment emergency sump. With the spray pump operating at design conditions and the containment at design pressure, the pressure drop provided across the nozzles exceeds 40 psi. The spray nozzles used in the construction of the containment spray system are designed to withstand differential pressures in excess of those expected to occur as a result of an accident.

The spray nozzles are stainless steel and have a 3/8-in.-diameter orifice.

6.2.2.2.2.3.4 Spray Additive Tank. The spray additive tank has been abandoned in place.

6.2.2.2.2.3.5 Spray Additive Eductors. The spray additive eductors serve only to maintain the spray system pressure boundary integrity.

6.2.2.2.2.3.6 Piping, Valves, and Containment Sumps. The piping used in the construction of the containment spray system is designed to withstand differential pressures in excess of those expected to occur as a result of an accident. The containment sump recirculation lines that run from the containment sump to the containment spray pumps are enclosed within guard pipes from the containment emergency sump floor to the first valve outside the containment. The pipe and guard pipe are capable of withstanding containment pressure and temperature. The guard pipe prevents leakage from the containment if the recirculation line ruptures.

The ~~two~~ containment ~~ECCS~~ sumps are designed in accordance with the requirements of Regulatory Guide 1.82*, as described in Appendix 6A. The risk-informed methodology, applied to evaluate the risk associated with effects of accident-generated debris, shows that the increase in risk associated with debris that would exceed the design limits of the sump strainers is very small, in accordance with the acceptance criteria of Regulatory Guide 1.174*. These sumps are located in a manner that protects them from the effects of high-energy line breaks, and they are separated from each other. The elevation of the containment emergency sumps are selected to allow optimum use of the available coolant. The sump intakes are protected by vertically stacked disk screens. The size of the opening in the screens is based on the minimum flow area through the components that receive coolant from the emergency sumps.

Analyses were conducted to ensure that effects such as reduction of NPSH and screen blockage will not result in degraded pump or system performance. The screens are installed in a manner to facilitate inspection of the structures and pump suction intakes.

There are two Containment Spray (CS) sumps. A screen is installed on each sump. The CS screens are composed of four stacks of 14 disks that are 30-in. long by 30-in. wide by ~40-in. high, four of which provide 590 ft^2 of perforated plate area and 133 ft^2 of circumscribed surface area per sump. Each typical screen disk is a welded assembly of two perforated plates and their structural support components. ~~The screens are designed to withstand the loading for the largest postulated debris pieces, types, and amount.~~ The plate-hole (perforation) diameter of the screen is 3/32 in. (a small~~%tage~~ *percentage* - 124 - holes are larger than 3/32-in.diameter but none are larger than ¼-in. diameter – see section 6.1.2.) The screen is mounted over the

containment sump. ~~Because of these dimensions, it is not considered credible that a screen can plug sufficiently to impede pump suction. In the remote case that~~ *Any particles that penetrate* ~~traverse~~ the screen~~, they would~~ *will* pass through the piping pumps and valves as well as the 3/8-in. diameter containment spray nozzle openings without difficulty. The screens bolt to the floor and may be removed by unbolting individual screen sections for inspection during shutdown periods, or the dedicated inspection port may be used.

Water sprayed in the refueling canal from the containment sprays may escape back to the elevation of the emergency sump through two 12-in. drain pipes located at the lowest point of the refueling canal. The water passes from the canal to a passageway on the containment floor. Spray water from 33 out of 171 nozzles during 1-train operation or a maximum of 66 out of a total of 342 nozzles from 2 containment spray trains in operation falls into the refueling canal for a maximum canal fill rate of approximately 500 gal/min and 1000 gal/min, respectively. This represents less than 20% of the total spray rate in either case.

The drain layout is such that each 12-in. drain line is capable of passing approximately 2000 gal/min; therefore, there is no danger of starving the sump via the refueling canal. The drain piping is isolated during refueling and left open during normal reactor operation. Plant refueling procedures ensure that these drain pipes are opened after refueling prior to plant startup.

One containment spray pump is provided for each train. A single failure therefore leaves one of the two trains in service. The containment spray pumps are located within compartments sealed by watertight doors; a postulated rupture in one train cannot flood the other.

The first of the two motor-operated valves in series on the recirculation lines are totally enclosed in protective chambers, ensuring that all liquid escaping from a damaged or leaky valve does not escape to the outer environment. All materials that can come in contact with recirculation fluid are austenitic stainless steel. The spray headers are located in the proximity of the containment liner of the dome. The spray headers are anchored to the concrete through the liner plate. Each system is designed for SSE and appropriate thermal and dead weight loading conditions. The spray system is designed for maximum coverage in the containment, with the nozzles located and oriented so that the spray will not be blocked by structures or equipment.

6.2.2.2.2.3.7 Motors for Pumps and Valves. The motors for the containment spray system components will be designed in accordance with specifications discussed for the motors in the SI system. (See section 6.3.)

6.2.2.2.2.3.8 Electrical Supply. Details of the emergency bus power sources are discussed in chapter 8.

6.2.2.2.2.4 <u>System Operation</u>. The spray system is actuated by a signal initiated manually from the control room or automatically on coincidence of two of four containment pressure (high-3) signals. These signals start the containment spray pumps and open the discharge valves to the spray headers.

6.3 EMERGENCY CORE COOLING SYSTEM

6.3.1 DESIGN BASES

The emergency core cooling system (ECCS) is a Seismic Category 1 safety-related system. It consists of the centrifugal charging pumps, safety injection pumps, residual heat removal pumps, accumulators, boron injection tank (Unit 1 only), residual heat removal heat exchangers, refueling water storage tank (RWST), and the associated piping, valves, instrumentation, and other related equipment. Nuclear plants employing the same ECCS design as VEGP are given in section 1.3.

The primary function of the ECCS following an accident is to remove the stored and fission product decay heat from the reactor core such that fuel rod damage, to the extent that it would impair effective cooling of the core, is prevented.

The ECCS is designed to cool the reactor core and to provide additional shutdown capability following initiation of the following accident conditions:

A. Loss-of-coolant accident (LOCA) including a pipe break or a spurious relief or safety valve opening in the reactor coolant system (RCS) which would result in a discharge larger than that which could be made up by the normal makeup system.
B. Loss-of-secondary-coolant accident including a pipe break or a spurious relief or safety valve opening in the secondary steam system which would result in an uncontrolled steam release or a pipe break in the secondary feedwater system.
C. A steam generator tube rupture accident.

The acceptance criteria for the consequences of each of these accidents are described in chapter 15 in the respective accident analyses sections.

The bases used in design and for selection of ECCS functional requirements are derived from 10 CFR 50, Appendix K, limits for fuel cladding temperature, etc., following any of the above accidents as delineated in 10 CFR 50.46. The subsystem functional parameters are selected so that, when integrated, the Appendix K requirements are met over the full spectrum of break sizes and single failure assumptions.

The design basis for the ECCS with regard to the effects of debris on the emergency sump strainers, to the extent that the strainers support the ECCS element of the core cooling function, is a risk-informed analysis that shows there is a high probability that the ECCS can perform its design basis functions.

The conclusion that ECCS will perform its design basis functions with high probability is based on plant-specific testing using assumptions that provide safety margin and defense-in-depth. The risk from breaks that do not meet one (or more) of the GSI-191 acceptance criteria is very small and acceptable in accordance with the guidelines of RG 1.174.

Details of the design basis for the effects of debris on the function of the emergency sump strainers are provided in FSAR Appendix 6A.

Portions of the ECCS also operate in conjunction with the other systems of the cold shutdown design. The primary function of the ECCS during a safety grade cold shutdown is to ensure a means for injecting and throttling boration and makeup flow. Details of the cold shutdown design bases are discussed in subsection 5.4.7.

The primary function of the safety injection system (SIS) is to provide emergency core cooling in the event of a LOCA resulting from a break in the primary RCS or to provide emergency

The net positive suction head (NPSH) of the RHR pumps was evaluated for normal plant cooldown operation and for both the injection and recirculation modes of operation for the DBA.

The recirculation mode of operation at *-0.3 psig containment pressure and 211°F sump temperature results in* ~~gives~~ the limiting NPSH. ~~requirement for the RHR pumps, and the NPSH available as determined from the following equation:~~

$$NPSH_{available} = h_{ambientpressure} + h_{statichead} - h_{vapor} - h_{systempressurelosses}$$

~~To evaluate t~~*T*he adequacy of the available NPSH~~, the debris generation from a high energy line break within the containment and the resultant impact on the RHR pump performance~~ was evaluated ~~using the following~~ *as described in Appendix 6A*.

~~A. The minimum flood level inside the containment, based on the RWST water discharged during injection and switchover to recirculation, plus the water volume of 3 accumulator tanks, is at el 177 ft 0 in. The minimum flood level also takes into account the flooding of the reactor cavity/incore instrument tunnel via the penetrations in the primary shield.~~

~~B. The RHR pump suction elevation is 124 ft 3 ¾ in.~~

~~C. The calculated maximum line losses which include losses through pipe fittings, valves, entrance and exit are 17.0 ft at 4500 gal/min, RHR pump runout flow.~~

~~D. The quantity of insulation debris generated by the double-ended rupture of the RCS hot leg at its connection to the steam generator, the limiting case, is 432.0 ft^3.~~

~~E. For one RHR and one CS pump train operation, the volume of debris transported and deposited on one RHR sump screen is 58.4% of the total debris. This is based on the ratio of individual pump flowrate to total recirculation flowrate.~~

~~F. The maximum RHR emergency sump screen head loss, assuming 58.4% of the total debris generated is evenly deposited on the trash guard-screen cage, is 16.65 ft.~~

~~G. The vapor pressure of the pump liquid is assumed to be in equilibrium with the containment ambient pressure (i.e., no credit is taken for subcooling of the sump fluid) for sump fluid temperatures of equal to or greater than 211°F.~~

$$h_{containmentambientpressure} = h_{vaporpressure}$$

~~The NPSH equation for the recirculation mode when suction is taken from the containment emergency sump becomes:~~

$$NPSH_{available} = h_{statichead} + h_{statichead} - h_{lineloss} - h_{sumpscreenloss}$$

~~Using the above values, the calculated final RHR pump available NPSH is 26.6 ft. The RHR pump required NPSH at 4500 gal/min is 19 ft. Therefore adequate available NPSH margin is provided for proper RHR pump operation.~~

6.3.2.2.5 Centrifugal Charging Pumps

In the event of an accident, the centrifugal charging pumps are started automatically on receipt of an SI signal and are automatically aligned to take suction from the RWST during the injection

TABLE 6.3.2-1 (SHEET 2 OF 3)

Available NPSH at maximum flow (ft) from RWST	59

RHR pumps (See figure 6.3.2-2.)

Number	2
Design pressure (psig)	600
Design temperature (°F)	400
Design flow (gal/min)	3000
Design head (ft)	375
Maximum flow (gal/min)	4500
Design head at maximum flow (ft)	325
Design head at shutoff (ft)	450
Motor rating (hp)	400
Required NPSH at maximum flow (ft)	(See figure 6.3.2-2.)
Available NPSH at maximum flow (ft)	
From RWST	92
From emergency sumps	~~34.7~~ *Variable (see Appendix 6A)*

Residual heat exchangers

(See subsection 5.4.7 for design parameters.)

Boron injection tank (Unit 1 only)

Number	1
Total volume (gal)	900
Usable volume at operating conditions, solution (gal)	900
Boron concentration, nominal (ppm)	0-2600
Design pressure (psig)	2735
Operating pressure (psig)	2684
Design temperature	300
Operating temperature	Ambient
Heaters	Determinated

APPENDIX 6A

Resolution of NRC Generic Letter 2004-02

6A.1 Introduction and Risk-Informed Approach Summary

NRC Generic Letter (GL) 2004-02 (Reference 1) required licensees to perform an evaluation of the emergency core cooling system (ECCS) and containment spray system (CSS) recirculation functions, and the flow paths necessary to support those functions, based on the potential susceptibility of sump strainers to debris blockage during design basis accidents requiring recirculation operation of ECCS or CSS. This generic letter resulted from Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance." As a result of the evaluation required by GL 2004-02, and to ensure system function, sump strainer design modifications were implemented.

The plant licensing basis considers long-term core cooling (LTCC) following a loss of coolant accident (LOCA) as identified in 10 CFR 50.46(b)(5). Long-term cooling is supported by the ECCS, which includes the charging, safety injection (SI), and residual heat removal (RHR) systems. These systems and the CSS are subject to the effects of accident-generated debris because they rely on the containment emergency sump in the recirculation mode. Debris generated from non-LOCA initiating events (secondary side breaks inside containment that result in a consequential LOCA) are also considered. The risk-informed evaluation analyzes the following events:

1. *Large, medium, and small LOCAs due to:*
   i. *Pipe breaks*
   ii. *Failure of non-piping components*
   iii. *Water hammer*
2. *Secondary side breaks inside containment that result in a consequential LOCA upon failure to terminate safety injection or a stuck open PORV requiring sump recirculation*
3. *Seismically-induced LOCAs*

A risk-informed evaluation was performed to respond to GL 2004-02. The evaluation provides confidence that the sump design supports LTCC following a LOCA. The evaluation meets the acceptance guidelines for a very small risk impact as defined in Regulatory Guide (RG) 1.174 (Reference 2).

The licensing basis with regard to effects of debris is determination of a high probability that the ECCS and CSS can perform their design basis functions based on VEGP-specific testing using an NRC-approved methodology. The risk from breaks that could generate debris and that do not meet one (or more) of the GSI-191 acceptance criteria is very small and is, therefore, acceptable in accordance with the RG 1.174 guidelines (Reference 2).

The use of a risk-informed method, rather than the deterministic methods prescribed in the regulation, required an exemption to 10 CFR 50.46(a)(1), which has been granted pursuant to 10 CFR 50.12.

The risk-informed approach identifies scenarios that fail any one of the following GSI-191 acceptance criteria, as determined by break-specific analysis:

1. *Strainer debris limits (based on VEGP-specific head loss testing),*
2. *Strainer structural differential pressure limit,*
3. *Partially-submerged strainer head loss limit,*
4. *Pump void fraction limit,*
5. *Flashing limit*
6. *Pump net positive suction head (NPSH) margin*
7. *Core blockage limits.*

The results of the break-specific analysis are used to calculate conditional failure probabilities (CFPs). The resulting CFPs are used with the VEGP PRA model to calculate the change in core damage frequency (ΔCDF) and change in large early release frequency (ΔLERF) related to the effects of debris. The ΔCDF and ΔLERF values are calculated based on the difference between the calculated CFP values and a hypothetical condition with no debris related failures (CFPs of 0). The ΔCDF and ΔLERF values are used for comparison to the guidelines in RG 1.174 (Reference 2). The results of the evaluation show that the risk from the proposed change is "very small" (i.e., in Region III of RG 1.174). The methodology includes conservatisms in the plant-specific testing and in the assumption that all unbounded breaks result in loss of core cooling.

Key aspects of the risk-informed evaluation include:

1. *The methodology used to quantify the amount of debris generated at each break location, including the assumed zone of influence (ZOI) size based on the target destruction pressure and break size, and the assumed ZOI shape (spherical or hemispherical) based on whether the break is a double-ended guillotine break or partial break. This requirement applies to the methodology, but not the tool or program used.*
2. *The methodology used to evaluate debris transport to the residual heat removal and containment spray strainers. This requirement applies to the methodology, but not the tool or program used.*
3. *The methodology used to quantify chemical precipitates, including the refinements to WCAP-16530-P-A (Reference 6), application of the solubility correlation, and application of the WCAP-17788-P autoclave testing (Reference 7). This requirement applies to the methodology, but not the tool or program used.*
4. *The strainer debris limits shown in TS Bases Table B 3.6.7-1, which are based on tested and analyzed debris quantities.*

5. *The methodology and acceptance criteria used to assess ex-vessel component blockage and wear.*
6. *The methodology used to assess in-vessel downstream effects and the associated limits. This requirement applies to the methodology, but not the tool or program used.*
7. *The methodology used to quantify conditional failure probabilities, and ΔCDF and ΔLERF. This requirement applies to the methodology, but not the tool or program used.*

6A.2 Debris Generation

Post-accident debris includes insulation, fire barrier, and coatings debris generated within the ZOI of the pipe break, as well as latent debris, unqualified coatings, and miscellaneous debris in containment. To support the debris generation evaluation, containment walkdowns were performed using the guidance of NEI 02-01 (Reference 3).

The pipe break characterization followed the methodology of NEI 04-07 (Reference 4) and associated NRC safety evaluation (SE) (Reference 5), with the exception that it characterized a full range of breaks rather than just the worst-case breaks as suggested by NEI 04-07. Double-ended guillotine breaks (DEGBs) and partial breaks on every in-service inspection (ISI) weld within the Class 1 pressure boundary were considered. All break sizes and locations were characterized by placing each analyzed break into three high-level categories: small-break LOCAs (SBLOCAs) – breaks smaller than 2 inches, medium-break LOCAs (MBLOCAs) – breaks greater than or equal to 2 inches and less than 6 inches, and large-break LOCAs (LBLOCAs) – breaks greater than or equal to 6 inches with the largest break being a DEGB of the 31 inch crossover leg.

In the debris generation calculation, a three-dimensional CAD model of the Unit 1 containment building was used to model the ZOI for each postulated break. Note that the Unit 1 model was used to represent both units because the containment buildings are almost identical. ZOIs representing possible breaks on the reactor coolant system (RCS) piping were modeled at each ISI weld.

DEGBs are modeled using a spherical ZOI with a radius proportional to the pipe inner diameter. Partial breaks are any breaks smaller than a DEGB and are modeled using a hemispherical ZOI with a radius proportional to the equivalent break size. Break sizes ranging from ½ inch up to a DEGB were modeled at each weld. In addition, because the orientation of partial breaks can have a significant effect on the results, partial breaks were modeled every 45 degrees around the circumference of the pipe at each weld. Credit was taken for shielding by concrete walls. While DEGBs on main loop piping are typically bounding with regard to the volume of debris generated, smaller breaks are more likely to occur.

Although the probability of occurrence is low, a secondary side break inside containment could require ECCS recirculation. Therefore, secondary side breaks from

the steam generator feedwater lines and main steam lines were characterized. Because secondary side breaks occur at lower pressure and temperature than the primary side breaks, the ZOI size corresponding to the insulation destruction pressure would be smaller, compared with the primary side breaks. The appropriate ZOI sizes were calculated based on the ANSI jet methodology described in Appendix I of NEI 04-07 Volume 2 (Reference 5). Breaks were postulated along the main steam and feedwater pipes. All secondary side breaks were assumed to be DEGBs. Only Nukon insulation was considered for the secondary side breaks because there is no fire barrier within the vicinity of the main steam and feedwater lines, and the coating quantities would be bounded by the primary side breaks.

Since different material types have different destruction pressures, a ZOI was determined for each type of material. The quantity of generated debris for each break case was calculated using these material specific ZOI sizes.

Unqualified coatings considered in the analysis include coatings within containment that do not have a specified preparation, application, or inspection compliant with plant specifications; previously qualified coatings that have noticeably deteriorated; coatings inaccessible for inspection; and coatings applied by vendors on vendor-supplied items that cannot be qualified. There are several types of unqualified coatings applied over numerous substrates within containment, including epoxy, inorganic zinc, and alkyd coatings. Unqualified coatings were conservatively assumed to fail at the start of sump recirculation for all postulated breaks.

The total amount of latent debris calculated based on walkdown data was 60 lbm; however, 200 lbm is assumed in the strainer evaluation. This conservatively bounds the 60 lbm of actual latent debris with ample operating margin. Per the guidance in NEI 04-07 Volume 2, latent debris is assumed to consist of 15 percent fiber and 85 percent particulate by mass (Reference 5).

A total of 2 ft^2 of foreign materials, such as labels, tags, stickers, placards and other miscellaneous materials, were identified via walkdown. However, 50 ft^2 of miscellaneous debris is assumed in the strainer evaluation to account for foreign materials. Per the guidance in NEI 04-07 (Reference 4) and the SE (Reference 5), the total surface area of miscellaneous debris was assumed to block an equivalent surface area of the sump strainers after allowance for 25% overlap.

6A.3 Debris Transport to the Sump Strainers

The debris transport analysis determines the fraction of each type and size of debris that could be transported to the sump strainers. The evaluation considers debris transport during the blowdown, washdown, pool fill, and recirculation phases based on plant-specific layout and flow conditions. For the recirculation phase, computational fluid dynamic (CFD) modeling was used to determine the sump pool flow conditions and transport of debris inside the pool for different break locations and pump lineups (e.g., number of ECCS and CSS trains in service). Debris accumulation on the two ECCS

strainers and two CSS strainers is assumed to be proportional to the flow split across the four strainers.

Potential upstream blockage points in containment were reviewed. Specifically, the refueling canal drains and doorways through the secondary shield wall were qualitatively evaluated and it was concluded that blockage would not occur at these locations.

6A.4 Chemical Effects

The post-LOCA sump strainer chemical effects analysis methodology includes:

- *Calculation of plant-specific chemical precipitate (sodium aluminum silicate and calcium phosphate) loading using the WCAP-16530-NP-A (Reference 6) base methodology with modification to the aluminum release rate by crediting phosphate passivation of aluminum.*
- *Consideration of aluminum solubility to determine when to account for sodium aluminum silicate precipitate loading.*

All Class 1 weld locations on the primary RCS piping upstream of the first isolation valve were evaluated for chemical product generation. The amount of chemical precipitate generated was calculated for each of the breaks using the break-specific debris generation quantity. Other plant-specific inputs, such as pH, temperature, pool volume, aluminum quantity, and spray time, were used to calculate the amount of chemical precipitate. The amount of precipitate was scaled by the ratio of test strainer area to plant strainer area and compared to the chemical precipitate amounts in the strainer testing to determine the analyzed head loss across the strainer as a function of time.

Calcium release was determined using the WCAP-16530-NP-A (Reference 6) release rates. Calcium phosphate is assumed to precipitate as soon as the calcium is released into solution. Aluminum release from non-metallic sources was determined using the WCAP-16530-NP-A (Reference 6) release rates. For metallic aluminum sources, an aluminum release rate that was developed based on testing performed at the University of New Mexico in a trisodium phosphate buffered environment was used. It was assumed that aluminum in the post-LOCA pool precipitates once the dissolved aluminum concentration reaches a solubility limit, as calculated using a solubility equation that was developed based on testing at Argonne National Laboratory (ANL). For each pipe break that does not reach the calculated aluminum solubility limit before 24 hours, it was conservatively assumed that all aluminum in solution precipitates at 24 hours.

The phosphate passivation and aluminum solubility modifications to the base WCAP-16530-NP-A (Reference 6) methodology were validated using autoclave test results documented in WCAP-17788-P, Volume 5 (Reference 7). Additionally, the following assumptions are considered essential parts of the phosphate passivation and aluminum solubility modifications to the base WCAP-16530-NP-A (Reference 7) methodology:

- *A double-ended pump suction LOCA with minimum safeguards temperature profile is used to determine chemical release, which promotes greater aluminum release.*
- *The pH was analytically combined to use a maximum sump pool pH of 7.8 for aluminum release with a less than design-basis minimum pH of 7.0 for solubility. The way the pH values were combined is not physically possible and bounds potential pH profile variations.*
- *Unsubmerged aluminum is treated as fully submerged or fully wetted in the containment spray solution.*
- *No credit is taken for aluminum that remains soluble after precipitation is predicted to occur.*

6A.5 Sump Strainer Evaluations

There are several failure criteria considered in the sump strainer evaluation: tested debris limits, strainer partial submergence, vortexing, void fraction, flashing, pump NPSH, and strainer structural margin. A postulated break that exceeds one or more of these criteria for the RHR or CSS strainers/pumps is treated as a failure of the corresponding ECCS or CSS train(s). Each of the failure criteria was evaluated during the LOCA transient to determine if an ECCS or CSS failure would occur.

The containment sump pool water volume following a LOCA was determined by considering all water sources (i.e., the refueling water storage tank, reactor coolant system, and accumulators) and subtracting the various holdup volumes. The holdup volumes include dead volumes inside the containment, filling of empty pipe, water in transit, and steam holdup. The sump pool volume was used to determine the pool water level using a correlation between pool water depth and volume.

Strainer Head Loss Test Limits

Head loss tests were performed to measure the head losses of the conventional debris (fiber and particulate) and chemical precipitate debris generated and transported to the sump strainers following a LOCA. The test program used a test strainer, debris quantities, and flow rates that were prototypical to Vogtle. Different test cases were performed with the thin bed and full debris load protocols, following the 2008 NRC staff review guidance (Reference 8).

A sump strainer is conservatively assumed to fail for any break where the transported quantity of fiber, particulate, and/or chemical precipitate debris exceeds the tested debris quantities (scaled from the test strainer area to the plant strainer area with adjustments for partial submergence and/or blockage by miscellaneous debris).

The results of the head loss tests provided a matrix of head loss data for various combinations of conventional and chemical debris loads. This matrix was used to determine the head loss for the debris load associated with each break scenario. A rule-based approach was used to calculate head loss based on the results of head loss

testing. If the fiber debris load at the strainer is less than the tested quantity from the thin bed test, the maximum thin bed conventional debris head loss was returned. If the quantity was greater than what was tested in the thin bed test, the conventional head loss of the full-load test was returned. A similar rule-based approach was applied for chemical debris head loss. If chemical products are generated, the maximum head loss from testing was applied.

Partially Submerged Strainer Criteria

If the strainers are partially submerged, the risk-informed calculation assumed that the strainer would fail if the head loss across the debris bed and strainer is equal to or greater than half of the submerged strainer height per RG 1.82 (Reference 9). Note that the pump NPSH and strainer structural limits are also applicable for a partially submerged strainer. The calculation tracks time-dependent accumulation of debris on the strainer. When the strainer is partially submerged, the evaluation only credits the surface area of the submerged portion of the strainers for flow and debris accumulation. Based on the calculated sump pool water level, the sump strainers are fully submerged for most breaks.

Strainer Vortexing Criteria

In lieu of vortexing calculations, testing was conducted to identify under which conditions vortexing and air ingestion is expected to occur in the plant for both clean strainer and debris laden conditions. The vortex test used a prototype strainer assembly with a conservatively high approach velocity and low strainer submergence. Comparison of test results with plant conditions showed no vortex formation for the plant strainers up to the tested debris limits.

Void Fraction Criteria

A pump failure due to degasification was assumed if the calculated steady-state gas void fraction at the pump is greater than 2 percent by volume. The quantity of air released from a given volume of water across the strainer was determined by calculating the difference between the concentration of air dissolved in the sump water and the concentration of air dissolved in water downstream of the strainer. A small amount of accident pressure was credited in the gas void calculation. Note that the pump NPSH required was adjusted to account for the void fraction in accordance with the guidance in RG 1.82 (Reference 9).

Flashing Criteria

The acceptance criterion is zero flashing as the fluid experiences a pressure drop across the debris bed and strainer. Strainer flashing was calculated by comparing the internal strainer pressure at the top of the strainer to the sump water saturation pressure. If the internal strainer pressure is less than the saturation pressure, the water flashes and the strainer is assumed to fail. By crediting a small amount of accident

pressure (i.e., above saturation pressure), no strainer failure due to flashing was recorded.

Pump NPSH Criteria

The RHR and CS pump NPSH margin was calculated based on the NPSH available minus the NPSH required for the respective pumps. NPSH available was defined without considering strainer head loss. This was calculated as a time-dependent parameter based on containment pressure, sump temperature, water level, losses in the pump suction piping (which are a function of the friction factor), and vapor pressure. The NPSH required was also calculated as a time-dependent parameter based on flow rate and gas void fraction.

Containment accident pressure is not credited in the analysis for pump NPSH. For containment pressure, the saturation pressure at the sump temperature was assumed for sump temperatures greater than 211°F. Note that the temperature of 211°F corresponds to the saturation temperature at the Technical Specification (TS) minimum containment pressure of -0.3 psig (or 14.396 psia). For sump temperatures below 211°F, the minimum containment pressure of -0.3 psig was used to calculate the pump NPSH available.

Because NPSH available is calculated without considering strainer head loss, the acceptance criterion is that the total strainer head loss (i.e., clean screen head loss plus conventional debris head loss plus chemical debris head loss) must be less than the pump NPSH margin. This was evaluated on a time-dependent basis.

Because the SI pumps and centrifugal charging pumps (CCPs) take suction from the RHR pumps during recirculation, only the NPSH margins of the RHR and CS pumps are calculated.

Strainer Structural Criteria

The strainers are located outside the secondary shield wall between the secondary shield wall and the containment wall and, as such, are not exposed to pipe whip, jet impingement, or postulated missiles generated from the LOCA event.

The analyzed strainer structural limit for each strainer is 24.0 ft. The head loss across each of the RHR and CS strainers was compared to this value to ensure that the structural margin is not exceeded.

6A.6 Downstream Effects – Components and Systems

An analysis was performed to evaluate the impact of debris on the wear or blockage of the ECCS and CSS piping and components downstream of the strainer (excluding reactor vessel) following a LOCA. This ex-vessel downstream effects evaluation used the methodology presented in WCAP-16406-P-A (Reference 10). The analyzed effects

of debris ingested through the containment sump strainers during the recirculation mode include erosive wear, abrasion, and potential blockage of downstream flow paths.

The smallest clearance for the VEGP heat exchangers, orifices, and spray nozzles in the recirculation flow paths is larger than the sump strainer hole size. Therefore, no blockage of the containment spray flow paths is expected.

ECCS and CSS instrumentation tubing was evaluated for potential debris accumulation in the sensing lines. The transverse velocity past this tubing was determined to be sufficient to prevent debris settlement in the instrument lines; therefore, debris effects will not cause an instrument failure.

The heat exchangers, orifices, and spray nozzles were evaluated for the effects of erosive wear for a bounding debris concentration over the 30-day mission time. The erosive wear on these components was determined to be insufficient to affect the system performance.

The effects of debris ingestion were evaluated for three aspects of pump operability including hydraulic performance, mechanical shaft seal assembly performance, and the mechanical performance (vibration) of the pump. These performances were determined to not be affected by the recirculating sump debris.

Evaluations of the system valves showed that the minimum recirculation flow rates are adequate to preclude debris sedimentation in all cases. All of the valves that are subject to being blocked pass the plugging criteria at their current positions, since the strainer mesh size is smaller than the minimum valve clearance. All of the valves that are subject to erosion pass the acceptable criteria for the 30-day mission time.

6A.7 Downstream Effects – Fuel and Vessel

During the post-LOCA sump recirculation phase, debris that passes through the ECCS sump strainers could accumulate at the reactor core inlet or inside the reactor vessel, potentially challenging LTCC. In-vessel downstream effects were analyzed using WCAP-17788-P (Reference 11) and plant-specific penetration test data.

Testing was conducted to collect time-dependent fiber penetration data for a prototypical strainer under various conditions (e.g., approach velocity and water chemistry) and strainer configurations (e.g., number of strainer disks). The test results were used to derive a model that was used to quantify fiber penetration for the RHR and CSS strainers at plant conditions. This model defines the time dependent downstream debris source term used to calculate the fiber accumulation in the reactor vessel.

Methods and acceptance criteria contained in WCAP-17788-P, Revision 1 (Reference 11) were used to evaluate the accumulation of fiber inside the reactor vessel. An evaluation was performed to demonstrate applicability of the WCAP-17788-P methods and results to VEGP in accordance with the NRC staff review guidance for in-vessel

effects (Reference 12). The applicability evaluation compares the values of key parameters assumed in the WCAP-17788 analysis to VEGP-specific values. The quantity of fiber accumulation inside the reactor vessel was calculated using break specific debris quantities and the appropriate accumulation based on the break location (i.e., hot leg break vs. cold leg break scenarios). The accumulated debris was compared to the debris limits defined in WCAP-17788-P (Reference 11). In-vessel debris limit failures were determined to be bounded by strainer debris limit failures (see Section 6A.8).

6A.8 Analyzed Debris Limits

Containment accident generated and transported debris is defined as the quantity of debris calculated to arrive at the containment sump strainers. As described in the previous sections, the evaluation of the effects of debris includes strainer head loss, downstream ex-vessel effects, and downstream in-vessel effects. Of these three aspects of the evaluation, strainer head loss has the bounding debris limits.

Based on the tested and analyzed debris quantities, strainer debris limits were defined as shown in TS Bases Table B 3.6.7-1. These debris limits cannot be exceeded for breaks smaller than or equal to 10 inches for two RHR train operation or breaks smaller than or equal to 6 inches for single RHR train operation. Larger breaks may exceed these debris limits without exceeding the RG 1.174 Region III acceptance guidelines (Reference 2).

If debris quantities greater than the analyzed debris limits are identified, the containment sump LCO (TS 3.6.7) would not be met and Condition A would be entered. Immediate action would be initiated to mitigate the condition and restore the sump to operable status in accordance with the TS and as described in the TS Bases.

6A.9 References

1. *NRC Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents for Pressurized-Water Reactors," September 13, 2004.*
2. *Regulatory Guide 1.174, "An Approach for Using Probabilistic Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, January 2018.*
3. *NEI 02-01, "Condition Assessment Guidelines: Debris Sources Inside PWR Containments," April 2002.*
4. *NEI 04-07 Volume 1, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Revision 0, December 2004.*
5. *NEI 04-07 Volume 2, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02," Revision 0, December 2004.*
6. *WCAP-16530-NP-A, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," March 2008.*

7. *WCAP-17788-P Volume 5, "Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090) – Autoclave Chemical Effects Testing for GSI-191 Long-Term Cooling," Revision 1, December 2019.*
8. *ML080230038, "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Strainer Head Loss and Vortexing," March 2008.*
9. *Regulatory Guide 1.82, "Water Sources for Long Term Recirculation Cooling Following a Loss of Coolant Accident," Revision 3, November 2003.*
10. *WCAP-16406-P-A, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," Revision 1, March 2008.*
11. *WCAP-17788-P Volume 1, "Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090)," Revision 1, December 2019.*
12. *ML19228A011, "U.S. Nuclear Regulatory Commission Staff Review Guidance for In-Vessel Downstream Effects Supporting Review of Generic Letter 2004-02 Responses," September 4, 2019.*

Vogtle Electric Generating Plant

Licensing Submittal for a Risk-Informed Resolution of Generic Letter 2004-02

Enclosure 2

Implementation of a Risk-Informed Approach for Addressing GSI-191

Attachment 2

Guidance for Supporting Operability Evaluations for Information Only

A2-1 Risk-Informed GSI-191 Design Basis

With approval of the license amendment request (LAR) to use a risk-informed approach to address GSI-191, the new design basis for Vogtle will be that the risk increase associated with post-accident debris effects is within Regulatory Guide (RG) 1.174 Region III (i.e., ΔCDF less than 1E-06 yr^{-1} and ΔLERF less than 1E-07 yr^{-1}) (Reference 1). Note that the ΔCDF guideline is more limiting for Vogtle than the ΔLERF guideline because the calculated ΔLERF is more than two orders of magnitude lower than the calculated ΔCDF (Reference 2, Enclosure 3, Section 14.1). Therefore, ΔLERF would not be exceeded without also exceeding ΔCDF.

As described in Enclosure 3 of the July 2018 submittal (Reference 2), containment sprays are only expected to initiate for hot leg breaks larger than 15 inches. Therefore, for smaller break sizes, only two strainer configurations are applicable based on whether there are one or two RHR pumps operating. The probability of any configuration with one RHR pump failing is less than 5% (Reference 2, Enclosure 3, Section 6.3).

Using log-linear interpolation of the 25-year geometric mean loss of coolant accident (LOCA) frequencies (Reference 3) and a 95/5 split for the equipment configuration, failure of all breaks greater than or equal to 10 inches for cases where both RHR pumps are running and failure of all breaks greater than or equal to 6 inches for cases where only one RHR pump is running would result in a ΔCDF value of 7.7E-07 yr^{-1} (0.95 × 6.56E-07 + 0.05 × 2.85E-06 = 7.66E-07). Therefore, the risk quantification would remain in RG 1.174 Region III (i.e., a ΔCDF less than 1E-06 yr^{-1}) even if all breaks larger than 10 inches fail when both RHR pumps are available and all breaks larger than 6 inches fail when only 1 RHR pump is available, as long as none of the breaks smaller than these thresholds fail. This is illustrated in Figure A2-1.

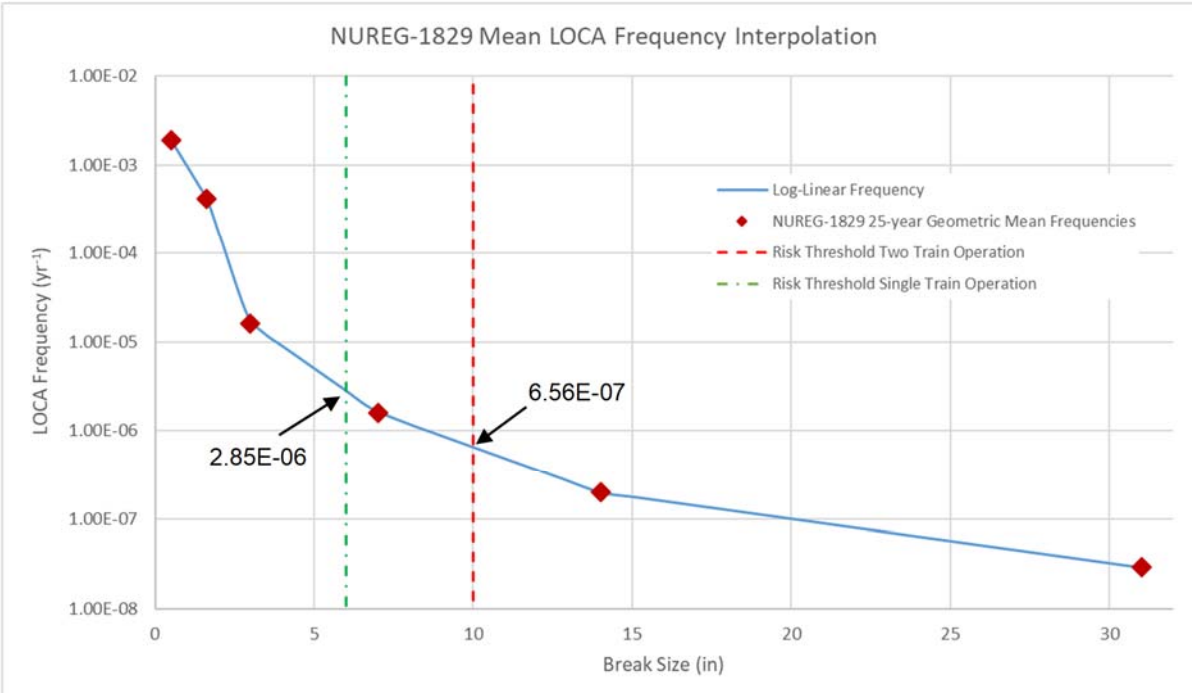

Figure A2-1: 25-Year Geometric Mean LOCA Frequencies with Log-Linear Interpolation

Therefore, to assess an emerging issue (e.g., an operability evaluation) related to GSI-191, it is not necessary to define acceptable limits associated with breaks larger than 10 inches. However, it is necessary to define acceptable limits for breaks smaller than or equal to 10 inches for two train operation and 6 inches for single train operation to ensure that an identified issue would not cause any of these breaks to exceed the limits and potentially push the risk up into RG 1.174 Region II.

A2-2 Available Debris Margin

Available debris margin can be determined based on the difference between the strainer debris limits and the quantity of debris for the relevant break sizes. The fiber and particulate debris limits are based on the tested quantity of debris scaled to the full surface area of one RHR strainer minus the area covered by miscellaneous debris. Note that using the single train scaling factor is more conservative for the purpose of deriving debris limits because the miscellaneous debris would be split across the two RHR strainers for two train operation, which would result in a slightly higher scaling factor. The acceptable debris limits based on the tested and analyzed debris quantities are shown in Table A2-1.

As discussed above, the debris limits for the insulation fiber, coatings, and fire barrier debris in Table A2-1 are based on tested debris loads scaled to the net surface area of one RHR strainer. These debris limits are consistent with the test debris quantities shown in Table 3.f.5-1 of Enclosure 5 in the July 2018 submittal (Reference 2) but are

broken down differently. In the July 2018 submittal table, the quantity of latent fiber and fiberglass insulation debris was combined and the quantity of latent particulate and coatings particulate debris was combined. For the new debris limits in Table A2-1, separate limits are presented for insulation fiber, coatings, and latent debris. Also, the scaled values in the July 2018 submittal table were based on the net surface area of one RHR strainer, whereas the values in Table A2-1 are scaled based on the net surface area of one RHR strainer minus the area blocked by miscellaneous debris.

Table A2-1: Containment Sump Debris Limits Applicable to Breaks ≤ 6 inches for 1 Train Operation and to Breaks ≤ 10 inches for 2 Train Operation

Debris Type	Acceptable Limit
Insulation Fiber Debris*	95.2 ft^3
Qualified and Unqualified Coatings*	16.8 ft^3
Fire Barrier Debris*	284.7 lb_m
Latent Debris**	200 lb_m
Miscellaneous Debris (Tags, Labels, etc.)**	50 ft^2

* Acceptable quantity of debris transported to one RHR strainer

** Total quantity of debris in containment, which would predominantly transport to the active RHR strainer for single train operation and split proportionally for two train operation

The available margins for insulation fiber, coatings and fire barrier were derived from the transported debris quantities and for latent debris and miscellaneous debris from the analyzed quantities. Because the total fiber and particulate debris quantity transported to one RHR strainer is greater for the single train operation case, it is conservative to use these debris quantities to calculate the available margin for breaks smaller than or equal to 10 inches. These debris margins are shown in Table A2-2.

Table A2-2: Debris Margin for Breaks ≤ 10 Inches

Debris Type	Current Quantity	Limit	Available Margin
Fiberglass Insulation Debris	5.3 ft^3	95.2 ft^3	89.9 ft^3
Qualified and Unqualified Coatings Debris	15.3 ft^3	16.8 ft^3	1.5 ft^3
Fire Barrier Debris	0 lb_m	284.7 lb_m	284.7 lb_m
Latent Debris	60 lb_m	200 lb_m	140 lb_m
Miscellaneous Debris	2 ft^2	50 ft^2	48 ft^2

A2-3 Operability Evaluation Guidance

The values in Table A2-2 are similar to the debris limits and margins for a deterministic design basis. These values can be used to support a prompt operability determination following discovery of an unanalyzed debris source.

As stated in Section A2-1, risk quantification for breaks larger than 10 inches (when both RHR pumps are available) or larger than 6 inches (when only one RHR pump is available) is not necessary. Even if all of these breaks were to fail, the frequency of occurrence is so low that the risk contribution and resultant ∆CDF and ∆LERF remain in Regulatory Guide 1.174, Region III, which is acceptable. Evaluation of smaller breaks must be performed, however.

For example, if it was determined that insulation previously thought to be RMI was actually fiberglass insulation, the quantity of fiberglass could be compared to the available margin to ensure that the total quantity does not exceed the acceptable limit for breaks smaller than or equal to 10 inches. A very simplistic assessment could be performed with a conservative assumption that the entire quantity of unanalyzed fiberglass fails as fine debris and transports to the strainer. Alternatively, a more refined assessment could be performed to determine the quantity of insulation within a bounding zone of influence (ZOI) in the vicinity of the insulation (for a break up to 10 inches) and/or determine realistic transport fractions for the newly identified debris source.

If the quantity of additional debris does not exceed the available margin, the sump can be declared operable. During the next outage, the debris source could be removed, or the design basis calculation could be updated to reflect the reduction in available margin.

However, if the quantity of additional debris exceeds the available margin, the sump would be declared inoperable, and Technical Specification (TS) 3.6.7 Condition A would be entered. Required Action A.3 allows 90 days to restore the sump to an operable condition. This additional time can be used to refine the debris generation and transport analysis to show that the debris quantities are within the limits, or to revise the risk quantification and submit an exigent or emergency LAR if the risk is sufficiently low to justify continued operation. Note that any Table A2-1 debris limit exceedance for the applicable break sizes, even if the resulting ∆CDF and ∆LERF are still within RG 1.174 Region III, requires prior NRC approval to accept the condition beyond 90 days (TS 3.6.7, Condition A).

In the unlikely situation where the risk quantification shows that the ΔCDF associated with GSI-191 is unacceptably high (i.e., within RG 1.174 Region I), it would be necessary to shut down and remove the problematic source of debris or otherwise correct the identified issue. This process is illustrated in Figure A2-2 for a newly identified source of debris.

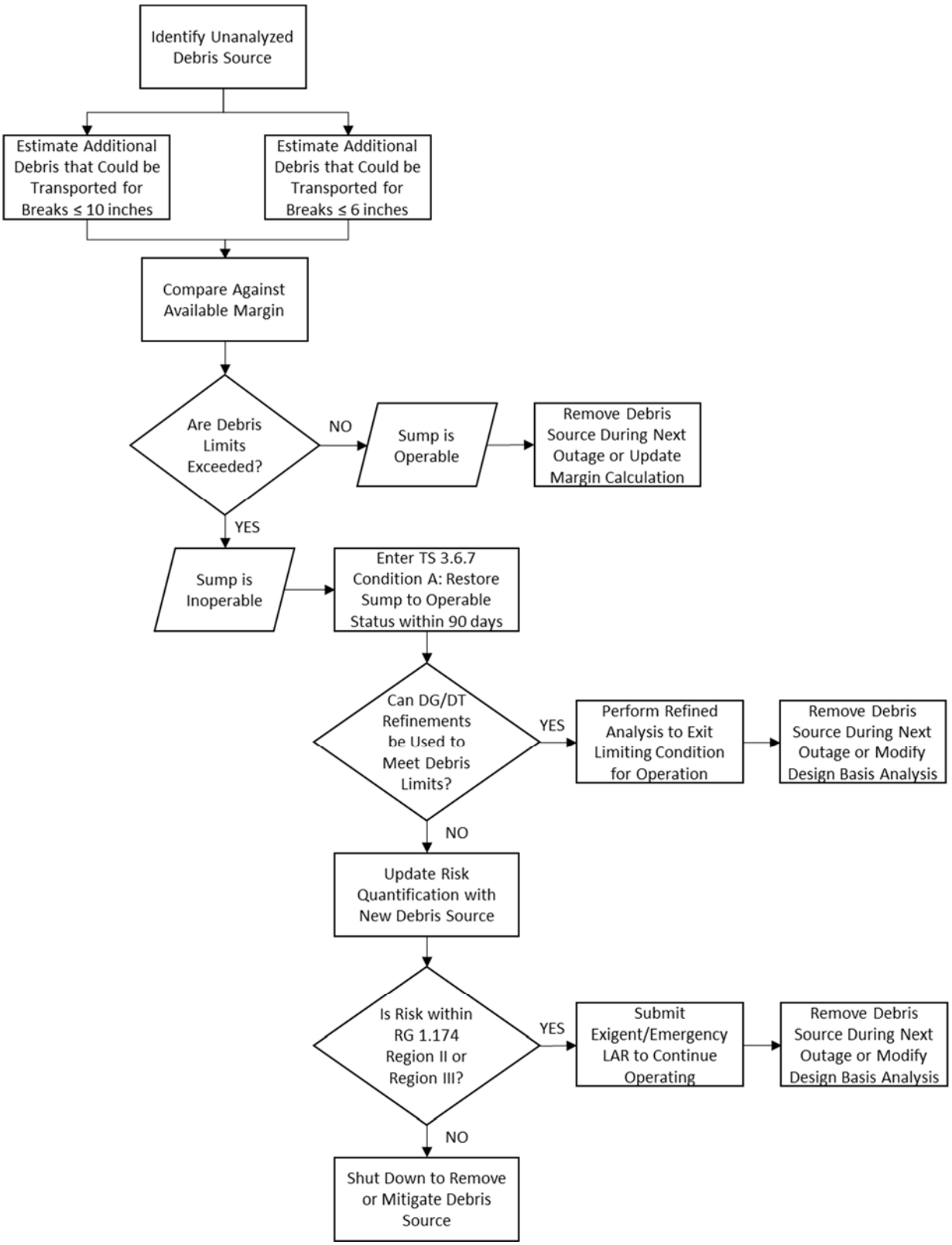

Figure A2-2: Illustration of operability guidance for an unanalyzed debris source

Attachment 2
Guidance for Supporting Operability Evaluations

The methodology for addressing design modifications with respect to GSI-191 is similar as illustrated in Figure A2-3.

***Other GSI-191 inputs include:**

- Pump flow rates
- Pump NPSH available
- Pump NPSH required
- ECCS/CS Setpoints
- Sump water volume
- Sump pH
- Sump temperature
- Containment temperature
- Containment pressure
- Insulation type
- Buffer type
- Strainer geometry
- Strainer structural margin
- Rated core thermal power

Figure A2-3: Illustration of design modification process with respect to GSI-191 parameters

A2-4 References

1. Regulatory Guide 1.174, “An Approach for Using Probabilistic Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Revision 3, January 2018.
2. SNC Letter NL-18-0915 (ML18193B163 and ML18193B165), “Vogtle Electric Generating Plant – Units 1 and 2, Supplemental Response to NRC Generic Letter 2004-02,” July 10, 2018.
3. NUREG-1829 Volume 1, “Estimating Loss-of-Coolant Accident (LOCA) Frequencies through the Elicitation Process,” April 2008.

Vogtle Electric Generating Plant

Licensing Submittal for a Risk-Informed Resolution of Generic Letter 2004-02

Enclosure 2

Implementation of a Risk-Informed Approach for Addressing GSI-191

Attachment 3

Updated Evaluation for In-Vessel Effects and Coatings

Attachment 3
Updated Evaluation for In-Vessel Effects and Coatings

A3-1 BACKGROUND AND OVERVIEW

Introduction

Southern Nuclear Company (SNC) prepared a technical report addressing Generic Safety Issue (GSI) 191 for Vogtle Electric Generating Plant (VEGP) and submitted it for Nuclear Regulatory Commission (NRC) staff review in July 2018 (Reference 1). This technical report described the use of a risk-informed approach to address GSI-191. The NRC issued a staff evaluation (Reference 3) that identified nine limitations and conditions. The purpose of this enclosure is to address Limitations and Conditions 3 and 9, as described below:

- *Describe in-vessel analysis, establish in-vessel acceptance criteria, and demonstrate the criteria are met.*
- *Correct the error concerning the evaluation of transported coatings debris loads described in SNC's letter dated December 4, 2018. Specifically, provide corrected coating debris volumes and describe how coating debris loads on the strainers are determined. In addition:*
    - *Verify that the use of the corrected coating debris volumes has a limited impact on strainer head loss and the head loss is acceptable. Also, the licensee must describe the method of verification.*
    - *Verify that the use of the corrected coating debris volumes has a limited impact on CDF and does not result in exceeding the acceptance guidelines for very small change in risk, as described in Section 2.4 of RG 1.174, Revision 3. Also, the licensee must describe the method of verification.*

In-Vessel Debris Effects

The July 2018 submittal (Reference 1) included an evaluation of in-vessel debris downstream effects based on the methodology in WCAP-17788-P Revision 0 (Reference 7). However, that methodology was not approved for use by the NRC at the time of the submittal. As a result, the staff reviewed the fiber penetration testing and analysis along with the transport to the reactor core, but did not review the in-vessel evaluation (Reference 3). The NRC's technical evaluation report (TER) of in-vessel debris effects (Reference 4) concluded that post-LOCA debris inside the reactor vessel has low safety significance. However, the evaluation did not address the subject of regulatory compliance per 10 CFR 50.46(b)(5). Subsequent to the TER, the NRC issued their staff review guidance for in-vessel downstream effects (Reference 5). The review guidance outlined approaches that the NRC would deem sufficient to demonstrate compliance with the requirements of 10 CFR 50.46(b)(5) for addressing in-vessel effects. Although the NRC staff did not approve WCAP-17788-P for use, the staff did expect that many of the methods developed in WCAP-17788 may be used by licensees to demonstrate adequate long-term core cooling (LTCC) (Reference 5). The staff developed review criteria, based on the information, evaluations, and analyses summarized in the TER (Reference 4), to determine the level of plant-specific review activity needed to establish compliance. Section A3-2 of this attachment follows the

NRC staff review guidance (Reference 5) and PWROG guidance (Reference 6) to describe the in-vessel analysis, establish in-vessel acceptance criteria, and demonstrate that the criteria are met.

Error Concerning the Evaluation of Transported Coatings Debris Quantities

As stated in the NRC staff evaluation, an error was identified in the transported coatings debris quantities. By letter dated December 4, 2018 (Reference 2), SNC summarized the impact of the error and the plan to correct it. Section A3-3 of this attachment describes SNC's investigation of the error and provides updated tables in the submittal with corrected coatings debris volumes. As stated in Section A3-3, the correction has a limited impact and does not change the conclusion of the risk quantification.

A3-2 RESOLUTION OF IN-VESSEL DEBRIS EFFECTS

WCAP-17788-P, Revision 1 provides evaluation methods to address in-vessel downstream effects (Reference 8). As discussed in the TER (Reference 4), although the NRC staff did not approve WCAP-17788-P for use, the staff expects that many of the methods developed in WCAP-17788-P may be used by PWR licensees in demonstrating adequate LTCC (Reference 5). SNC has elected to use methods and analytical results developed in WCAP-17788-P, Revision 1 (Reference 8) to address in-vessel downstream debris effects for VEGP and has evaluated the applicability of the methods and analytical results from WCAP-17788-P, Revision 1 for VEGP.

A3-2.1 Sump Strainer Fiber Penetration

Sump strainer fiber penetration was evaluated based on plant-specific penetration testing, which was used to develop a model of fiber penetration through the strainer over time. The test program and fiber penetration model were described in the July 2018 submittal (Reference 1) and accepted by the NRC (Reference 3).

A3-2.2 In-Vessel Effects Analysis and Resolution

As described in the NRC staff review guidance (Reference 5), there are four different paths that may be taken to resolve in-vessel effects. The paths are illustrated in Figure 1 of the NRC staff review guidance and referred to as Box 1 through Box 4. VEGP is selecting the Box 4 path to demonstrate adequate treatment of in-vessel effects. Per Section 3.0 of the NRC staff review guidance (Reference 5), it is necessary to confirm that VEGP is within the key parameters of the WCAP-17788 (Reference 8) methods and analysis. Each of the key parameters is presented in the table below.

Table A3-1 – Summary of In-Vessel Effects Parameters

Parameters	Values from WCAP-17788-P, Revision 1	VEGP Values
Nuclear Steam Supply System (NSSS) Design	Various	Westinghouse 4-loop
Fuel Type	Various	Westinghouse 17 x 17 VANTAGE 5, VANTAGE+ fuel
Barrel/Baffle Configuration	Various	Upflow
Minimum Chemical Precipitation Time	t_{block} from WCAP-17788 143 minutes	24 hours
Maximum HLSO Time	N/A	8 hours
Maximum Core Inlet Fiber Load for Hot Leg Break (HLB)	WCAP-17788, Volume 1, Table 6-3	90.61 g/FA
Total In-Vessel Fiber Limit for HLB	WCAP-17788, Volume 1, Section 6.4	N/A
Minimum Sump Switchover (SSO) Time	20 minutes	31.9 minutes
Maximum Rated Thermal Power	3658 MWt	3625.6 MWt
Maximum Alternate Flow Path (AFP) Resistance	WCAP-17788, Volume 4, Table 6-1	WCAP-17788, Volume 4, Table RAI-4.2-24
ECCS Flow per FA	8 – 40 gpm/FA	15.5 gpm/FA

A detailed comparison of VEGP values to WCAP-17788-P, Revision 1 values (Reference 8) is provided below.

Comparison of VEGP Chemical Precipitation Time with HLSO Time and t_{block}

For VEGP, chemical precipitation was shown to occur after the latest HLSO time and after the time that complete core inlet blockage can be tolerated, which is defined in WCAP-17788 as t_{block}.

1. VEGP chemical precipitation time (t_{chem}) – Chemical precipitation is shown not to occur within 24 hours following the accident based on the autoclave testing in WCAP-17788, Volume 5. Table A3-2 presents key VEGP parameters for

evaluating chemical precipitation timing, which were compared with the testing conditions of various test groups from the WCAP. Test Group 44 was shown to be the most representative of the VEGP post-LOCA environment. Test Groups 30, 39, 40, 41, and 42 were also considered to address bounding post-LOCA environments (high pH, low pH, high calcium, high aluminum, low silicon, etc.). None of the tests performed at VEGP-representative chemical conditions in the above test groups showed precipitation prior to 24 hours. Therefore, for VEGP t_{chem} is 24 hours.

Table A3-2 - Key Parameter Values for Chemical Precipitation Timing

Parameter	VEGP Value
Buffer	Trisodium Phosphate
Sump pH (Long-term)	7.0 - 7.8
Minimum Sump Volume	57,787 ft^3
Maximum Sump Pool Temperature	251.44°F
Maximum Calcium Silicate	0 g
Maximum E-Glass	2,440,388 g
Maximum Silica	0 g
Mineral Wool	0 g
Maximum Aluminum Silicate	0 g
Maximum Concrete	Not Determined*
Maximum Interam™	18,325 g (40.4 lbm)
Aluminum (unsubmerged + submerged)	1275 ft^2 (926.6 ft^2 + 348.4 ft^2)
Galvanized Steel	~211,000 ft^2

*Concrete is not a significant contributor to chemical precipitation

2. VEGP HLSO time – VEGP's maximum HLSO time is 8 hours after the event.
3. Time of t_{block} used in WCAP-17788 – VEGP is a Westinghouse NSSS plant with an upflow barrel/baffle design. WCAP-17788 used a t_{block} of 143 minutes.

<u>Comparison of VEGP In-Vessel Fiber Load with WCAP-17788 Limit</u>

The maximum amount of fiber that may arrive at the core inlet for VEGP exceeds the core inlet fiber limit but is less than the total in-core fiber limit presented in WCAP-17788.

1. WCAP-17788 core inlet fiber limit – The core inlet fiber limit that is applicable for VEGP (i.e., Westinghouse NSSS and upflow barrel/baffle design and

Westinghouse fuel) is in Table 6-3 of WCAP-17788 Volume 1. Since VEGP has a 17x17 fuel design and a fuel assembly pitch that matches the value in Section 6.3 of WCAP-17788 Volume 1, Revision 1 (Reference 8), no adjustment to this fiber limit is necessary.

2. WCAP-17788 total in-core fiber limit – The total in-core fiber limit is in Section 6.4 of WCAP-17788, Volume 1.
3. VEGP in-vessel fiber load – The methodology and fiber penetration test data that VEGP used to evaluate fiber accumulation at the core inlet have been reviewed and accepted by the NRC (Reference 3). Per the latest NRC review guidance (Reference 5), the fiber split between the core inlet and heated core was not accepted. If no credit is taken for flow diversion through AFPs, the maximum amount of fiber that may reach the VEGP reactor core inlet for an HLB is 90.61 g/FA. Various combinations of input parameters (e.g., ECCS flow rate, CS flow rate, CS duration, transported fiber load) were analyzed to ensure the maximum core inlet fiber load is captured.

The NRC staff did not accept the methodology of diverting flow and debris from the core inlet to the AFPs (References 4 and 5) as described in WCAP-17788 because it was based on the assumption that debris accumulates uniformly at the core inlet. In reality, the debris bed at the core inlet would not be uniform due to non-uniform flow, and it would take more debris than determined by WCAP-17788 to completely block the core inlet and activate the AFPs (see Reference 5, Appendix B).

Because of the expected non-uniform debris loading, the debris head loss at the core inlet would be lower than predicted in WCAP-17788. Lower head loss would allow additional fiber accumulation beyond the core inlet fiber limit where complete core blockage is predicted to occur in the WCAP. By definition, if the head loss at the core inlet is not high enough to activate flow through the AFPs, the core is continuing to receive sufficient flow for LTCC through the core inlet. At some point, if enough debris is added to the RCS, the resistance at the core inlet will be high enough to activate the AFPs. As described in WCAP-17788 (Reference 8), LTCC is assured as long as the total amount of fiber to the RCS remains below the total in-core fiber limit. Therefore, it is reasonable to use the total in-core fiber limit as the acceptance criterion for HLBs. For VEGP, the maximum quantity of fiber predicted to reach the reactor core (90.61 g/FA) is lower than the WCAP-17788 total in-core fiber limit (Reference 8, Volume 1, Section 6.4). As a result, the accumulation of fiber inside the reactor core will not challenge LTCC.

<u>Comparison of VEGP SSO Time with that Assumed in WCAP-17788</u>

The earliest SSO time for VEGP is greater than that assumed in the WCAP-17788 analysis.

1. VEGP SSO time – The SSO time marks the beginning of sump recirculation and fiber accumulation inside the reactor vessel. For VEGP, the shortest duration for injection from the RWST is 31.9 minutes.

2. The SSO time assumed in the WCAP-17788 analysis is 20 minutes (Reference 8, Volume 4, Table 6-1).

Comparison of VEGP Maximum Thermal Power with that Assumed in WCAP-17788

VEGP maximum rated thermal power is less than the analyzed power level in WCAP-17788 for a Westinghouse NSSS with an upflow barrel/baffle design.

1. VEGP rated thermal power – VEGP maximum rated thermal power is 3625.6 MW_t.
2. Thermal power assumed in WCAP-17788 – The WCAP analysis used a thermal power of 3658 MW_t for a Westinghouse upflow barrel/baffle plant design as shown in Table 6-1 of WCAP-17788, Volume 4 (Reference 8).

Comparison of VEGP Reactor AFP Resistance with that Assumed in WCAP-17788

The VEGP reactor AFP resistance is less than that analyzed in WCAP-17788.

1. VEGP reactor AFP resistance – The VEGP AFP resistance is presented in Table RAI-4.2-24 of WCAP-17788, Volume 4 (Reference 8).
2. AFP resistance assumed in WCAP-17788 – The AFP resistance used in the WCAP-17788 analysis that is applicable for VEGP (i.e., Westinghouse upflow barrel/baffle design) is provided in Table 6-1 of WCAP-17788, Volume 4 (Reference 8).

Comparison of VEGP ECCS Flow Rate with that Analyzed in WCAP-17788

The VEGP ECCS flow per fuel assembly is within the range of flow rates analyzed in WCAP-17788.

1. VEGP ECCS flow rate – The VEGP ECCS flow rate per fuel assembly is 15.5 gpm/FA.
2. ECCS flow rates analyzed in WCAP-17788 – For a Westinghouse NSSS with an upflow barrel/baffle design, the analyzed ECCS flow rate is 8 gpm/FA to 40 gpm/FA (Reference 8, Volume 4, Table 6-1).

Based on the comparisons shown above, in-vessel downstream effects due to accumulation of debris inside the reactor core will not challenge LTCC at VEGP.

A3-3 ERROR CORRECTION IN THE EVALUATION OF TRANSPORTED COATINGS DEBRIS LOADS

SNC's investigation of the error in the transported coatings debris load showed that the washdown transport fractions, which were correctly identified in Table 3.e.6-2 of Enclosure 5 of the July 2018 submittal (Reference 1), were incorrectly applied for

unqualified coatings debris in the NARWHAL runs for risk quantification, and sensitivity and uncertainty analyses. This error resulted in non-conservative transported particulate debris loads in Tables 3.e.6-15 and 3.e.6-16 of Enclosure 5 of the July 2018 submittal (Reference 1). The selection of the four worst-case breaks that do not fail any acceptance criteria, as shown in Tables 3.b.4-2, 3.e.6-16, and 3.h.5-2 of the July 2018 submittal (Reference 1), were also impacted by this error. Additionally, this error affected the risk quantification results for the base case, and the sensitivity and uncertainty cases. SNC has revised the calculations affected by this error.

A3-3.1 Updates to Debris Transport Quantities

An updated version of Tables 3.b.4-2, 3.e.6-15, 3.e.6-16, and 3.h.5-2 from the July 2018 submittal are provided with corrected coatings debris volumes in the tables below:

Table A3-3 - Corrected Enclosure 5, Table 3.b.4-2: The Four Worst-Case Breaks that Do Not Fail Any Acceptance Criteria for the Single Train Failure Equipment Configuration

Break Location		11201-008-3-RB	11201-005-3-RB	11201-007-7-RB	11201-008-3-RB
Break Size		25"	24"	27"	24"
Break Type		Partial (135°)	Partial (225°)	Partial (225°)	Partial (135°)
Nukon (ft^3)	Fine	60.65	59.94	59.11	59.43
	Small	229.04	225.31	221.25	223.43
	Large	38.58	41.16	43.18	40.66
	Intact	41.64	44.43	46.62	43.89
Fire Barrier (ft^3)	Fine	0.0	0.0	0.0	0.0
	Small	0.0	0.0	0.0	0.0
Fire Barrier (lbm)	Particulate	0.0	0.0	0.0	0.0

Table A3-4 - Corrected Enclosure 5, Table 3.e.6-15: Transported Debris for the Four Overall Worst-Case Breaks

Break Location		11201-004-6-RB	11201-001-5-RB	11201-001-3-RB	11201-004-4-RB
Break Size		29"	29"	29"	29"
Break Type		DEGB	DEGB	DEGB	DEGB
Fiber (ft^3)	RHR	639.8	635.4	618.0	605.8
	CS	279.5	277.6	272.7	267.4
Coatings and Latent Particulate (ft^3)	RHR	23.03	23.03	22.96	22.93
	CS	9.94	9.94	10.01	9.99
Calcium Phosphate (lbm)	RHR	73.5	73.5	73.5	73.5
	CS	42.3	42.2	42.1	42.1
Sodium Aluminum Silicate (lbm)	RHR	86.2	86.1	85.8	85.8
	CS	0.5	0.5	0.5	0.5
Fire Barrier Particulate (lbm)	RHR	0.0	18.7	18.5	0.0
	CS	0.0	8.1	8.1	0.0
Fire Barrier Fiber[1] (ft^3)	RHR	0.0	2.6	2.6	0.0
	CS	0.0	1.1	1.1	0.0

[1] The volumes of fire barrier fiber debris shown here were converted from their mass values using the density of low density fiberglass of 2.4 lbm/ft^3. For example, the 2.6 ft^3 of fire barrier fiber debris corresponds to a mass of 6.2 lbm.

Table A3-5 - Corrected Enclosure 5, Table 3.e.6-16: Transported Debris for the Four Worst-Case Breaks that Do Not Fail the Acceptance Criteria

Break Location		11201-008-3-RB	11201-005-3-RB	11201-007-7-RB	11201-008-3-RB
Break Size		25"	24"	27"	24"
Break Type		Partial (135°)	Partial (225°)	Partial (225°)	Partial (135°)
Fiber (ft^3)	RHR	45.8	45.6	45.4	45.3
	CS	1.43	1.44	1.43	1.42
Coatings and Latent Particulate (ft^3)	RHR	16.30	16.29	16.50	16.29
	CS	0.00	0.00	0.01	0.00
Calcium Phosphate (lbm)	RHR	51.6	51.7	51.6	51.3
	CS	0.0	0.0	0.0	0.0
Sodium Aluminum Silicate (lbm)	RHR	24.7	24.8	24.7	24.7
	CS	0.0	0.0	0.0	0.0
Fire Barrier Particulate (lbm)	RHR	0.0	0.0	0.0	0.0
	CS	0.0	0.0	0.0	0.0
Fire Barrier Fiber (ft^3)	RHR	0.0	0.0	0.0	0.0
	CS	0.0	0.0	0.0	0.0

Table A3-6 - Corrected Enclosure 5, Table 3.h.5-2: Coatings Debris for the Four Worst-Case Breaks that Do Not Fail the Strainer Acceptance Criteria

Break Location		11201-008-3-RB	11201-005-3-RB	11201-007-7-RB	11201-008-3-RB
Break Size		25"	24"	27"	24"
Break Type		Partial (135°)	Partial (225°)	Partial (225°)	Partial (135°)
Qualified Epoxy (ft^3)		0.158	0.129	0.832	0.130
Qualified IOZ (ft^3)		0.066	0.054	0.144	0.054
Unqualified Epoxy (ft^3)	**UC**	17.671	17.671	17.671	17.671
	LC	12.711	12.711	12.711	12.711
Unqualified Alkyd (ft^3)	**UC**	0.0	0.0	0.0	0.0
	LC	0.516	0.516	0.516	0.516
Unqualified IOZ (ft^3)	**UC**	0.117	0.117	0.117	0.117
	LC	0.265	0.265	0.265	0.265

The updated coating debris loads were determined using the methodology described in the July 2018 submittal (Reference 1) with a correction to the NARWHAL model.

A3-3.2 Updates to Risk Quantification of Base Case

The affected strainer head losses were determined using the same methodology as described in the July 2018 submittal (Reference 1). While correcting the error resulted in greater transported coatings debris loads, it had an insignificant impact on the risk quantification results. Due to the increased debris loads, the conditional failure probability (CFP) for large break LOCAs, reported in Table 3-5 of Enclosure 3 of the July 2018 submittal as "Train B Failures", increased from 0.0736 to 0.1017.

Table A3-7 – Corrected Enclosure 3, Table 3-5 – NARWHAL CFP Results for Large Break LOCAs

Equipment Configuration	Core	Strainer A and B	Strainer A only	Strainer B only
No Equipment Failure	0	0.0118	0	0
RHR Pump B Failure	0	N/A	0.0679	N/A
Charging Pump B Failure	0	0.0118	0	0
SI Pump B Failure	0	0.0118	0	0
Train B Failure	0	N/A	0.1017	N/A
CS Pump B Failure	0	0.0139	0	0
Both CS Pumps Failure	0	0.0177	0	0

Using the updated CFP results, the change in risk calculated in the VEGP GSI-191 PRA model is shown in the updated table below.

Table A3-8 - Corrected Enclosure 3, Table 3-8: VEGP Total Risk Impact due to GSI-191 Failures

Case	ΔCDF (ry^{-1})	ΔLERF (ry^{-1})
Risk increase from GSI-191 failures for high-likelihood LOCA configurations	2.33×10^{-8}	3.12×10^{-11}
Bounding risk increase from GSI-191 failures for unlikely LOCA configurations	1.42×10^{-9}	4.12×10^{-12}
Risk increase from GSI-191 failures for seismically-induced LOCAs	2.00×10^{-9}	2.00×10^{-10}
Risk increase from GSI-191 failures for SSBIs	1.39×10^{-9}	8.25×10^{-11}
Total risk increase associated with GSI-191	**2.81×10^{-8}**	**3.18×10^{-10}**

These CDF, LERF, ΔCDF, and ΔLERF values fall well within the Regulatory Guide (RG) 1.174 Region III guidelines (Reference 9), and these values did not significantly increase from the values reported in the July 2018 submittal. Therefore, correcting the error did not change the conclusion that the effects of debris have very low risk at VEGP.

A3-3.3 Updates to Risk Quantification of Sensitivity and Uncertainty Cases

The error was also corrected in the NARWHAL models for the sensitivity and uncertainty analyses. As described in the July 2018 submittal (Reference 1), a simplified approach was used to calculate ΔCDF using the CFP values calculated for each sensitivity and uncertainty case. Using this simplified approach and the corrected CFP values, the base case ΔCDF was calculated to be 2.57×10^{-8} yr^{-1}. This is very close to the risk increase calculated using the PRA model excluding the contribution of secondary side breaks and seismically-induced LOCAs (see the corrected Table 3-8 above).

A3-3.3.1 Parametric Sensitivity Cases

The corrected risk quantification results for the parametric sensitivity analysis are provided in the updated table and figure below. These ΔCDF values are compared with that of the base case: 2.57×10^{-8} yr^{-1}. Correcting the coatings error did not change the order for the top 10 parameters that have the most significant effect on ΔCDF.

Table A3-9 - Corrected Enclosure 3, Table 3-12: Results of Parametric Sensitivity Analysis

Input Parameter	ΔCDF at Minimum Input	ΔCDF at Maximum Input
Simulation Time	2.56E-08	2.57E-08
Initial RWST Level	1.24E-08	2.57E-08
RHR Pump Flow Rate	2.32E-08	8.09E-08
CS Pump Flow Rate	2.82E-08	2.36E-08
Pressure and Temperature Profiles	1.19E-08	2.56E-08
Sump pH	1.18E-08	2.57E-08
ZOI Debris Quantity	9.73E-09	3.43E-08
Latent Debris Quantity	2.41E-08	2.58E-08
Miscellaneous Debris Quantity	2.33E-08	2.57E-08
Submerged Aluminum Surface Area	2.57E-08	2.57E-08
Unsubmerged Aluminum Surface Area	2.57E-08	2.57E-08
Debris Head Loss	2.57E-08	2.60E-08
Strainer Debris Limits[1]	4.01E-08	6.46E-09
Containment Accident Pressure	2.57E-08	2.57E-08
Strainer Penetration Fractions	2.59E-08	3.08E-08
Containment Spray Duration	2.94E-08	2.54E-08
Containment Spray Duration of 1439 Minutes	2.57E-08	---
Reactor Vessel Hot Leg Break Fine Fiber Limit	1.10E-07	2.57E-08
Reactor Vessel Cold Leg Break Fine Fiber Limit	2.57E-08	2.57E-08
Geometric LOCA Frequency Values	4.26E-11	6.41E-08
Unqualified Coatings Quantity	2.55E-08	2.97E-08

[1] Two typos were identified in Table 3-12 of Enclosure 3 in the July 2018 submittal (Reference 1). The ΔCDF values for the minimum and maximum "Strainer Debris Limits" sensitivity cases should have been 3.47E-08 and 5.31E-09, respectively (compared with the incorrect values shown in the submittal: 3.47E-07 and 5.31E-08). Correcting the coatings error increased the ΔCDF from 3.47E-08 to 4.01E-08 for the minimum input case, and from 5.31E-09 to 6.46E-09 for the maximum input case.

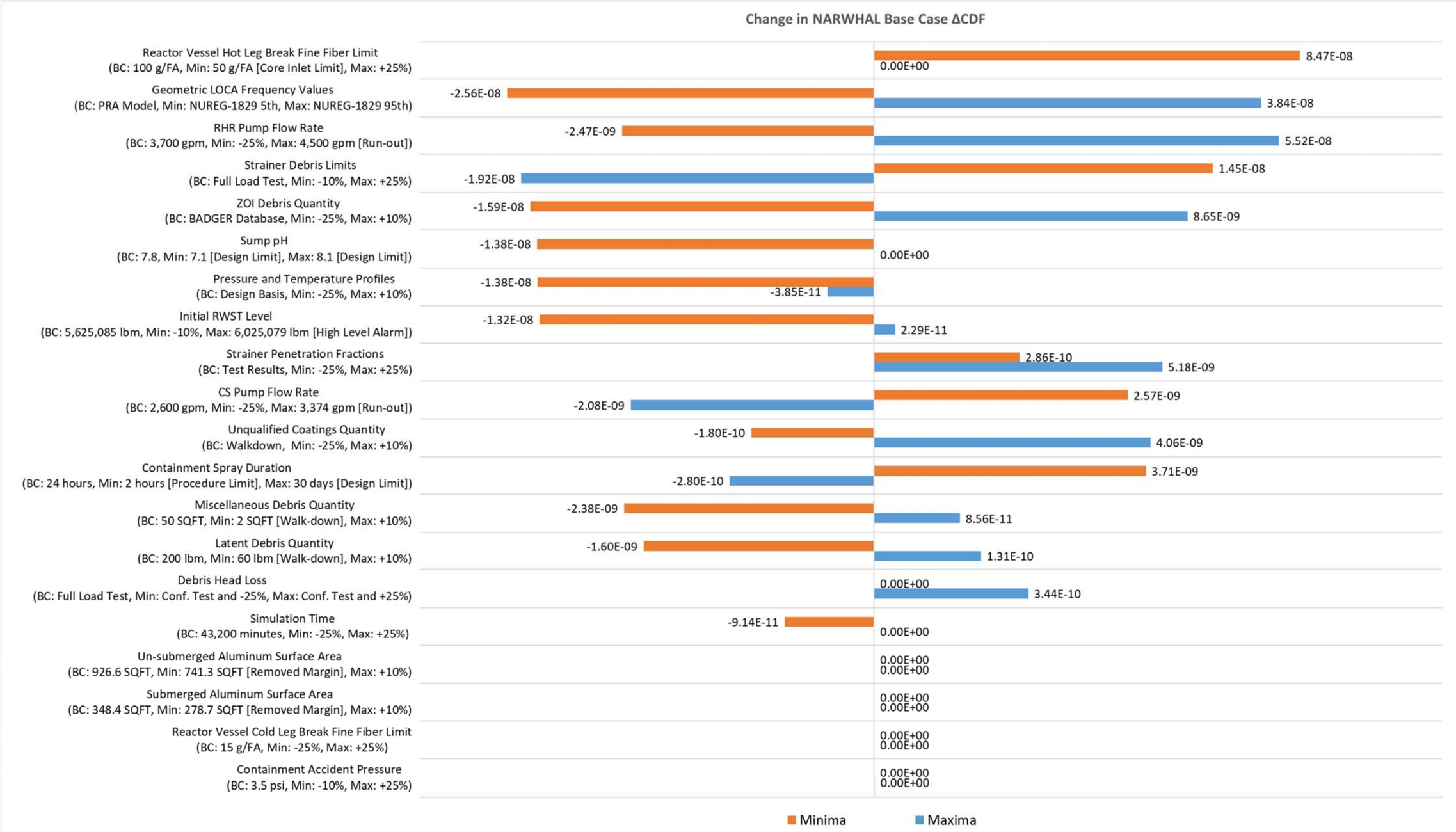

Figure A3-1 – Corrected Enclosure 3, Figure 3-9 – Tornado Diagram Showing Risk Sensitivity Ranking

It should be noted that the ΔCDF values for the minimum initial RWST level and minimum particulate debris limit sensitivity cases were refined by eliminating the particulate debris limit failures for those breaks with a theoretical uniform fiber bed thickness less than 0.45 in. This approach is supported by the thin-bed head loss test results, which showed negligible head losses by particulates when the fiber bed thickness is equal to or less than 0.45 in (see Figure 3.f.10-4 in Enclosure 5 of the July 2018 submittal).

The sensitivity run for the minimum initial RWST level resulted in failures for some of the small breaks due to the particulate debris loads on the strainer exceeding the debris limit. Such failures were only observed for the equipment lineup with single-train failure. Note that the particulate debris limit is derived in NARWHAL by multiplying the tested particulate debris load by the ratio between the submerged plant strainer surface area and test strainer surface area. For the sensitivity run with the minimum initial RWST level, some of the small breaks had partially submerged strainers, resulting in the scaled particulate debris limit being less than the transported particulate debris load. The refinement described above was applied to the single-train failure case of this sensitivity run only to eliminate the failures for the small breaks that do not have a theoretical uniform bed thickness of 0.45 in. This bed thickness was calculated in NARWHAL based on the actual amount of fiber debris transported to the strainer and the submerged strainer surface area.

Similarly, the sensitivity run for the minimum particulate debris limit case resulted in small break failures due to their transported particulate debris loads for the single train failure case exceeding the minimum particulate debris limit. The refinement was applied in the same way as that described above.

A3-3.3.2 Parametric Uncertainty Cases

The corrected risk quantification results for the parametric uncertainty cases are provided in the updated table and figure below. These ΔCDF values are compared with that of the base case: 2.57×10^{-8} yr^{-1}.

Table A3-10 – Corrected Enclosure 3, Table 3-15 – Results of Parametric Uncertainty Quantification

Sensitivity Case	Description	ΔCDF	Change in ΔCDF from NARWHAL Base Case
Strainer Case 1	Min water volume and min CS duration	1.79E-07	1.54E-07
Strainer Case 2	Min water volume and max CS duration	1.76E-07	1.50E-07
Strainer Case 3	Max water volume and min CS duration	1.09E-07	8.30E-08
Strainer Case 4	Max water volume and max CS duration	1.08E-07	8.26E-08
Core Case 1	Min water volume and min RHR flow rate	7.26E-08	4.69E-08
Core Case 2	Min water volume and max RHR flow rate	1.64E-07	1.38E-07
Core Case 3	Max water volume and min RHR flow rate	7.37E-08	4.80E-08
Core Case 4	Max water volume and max RHR flow rate	9.41E-08	6.84E-08

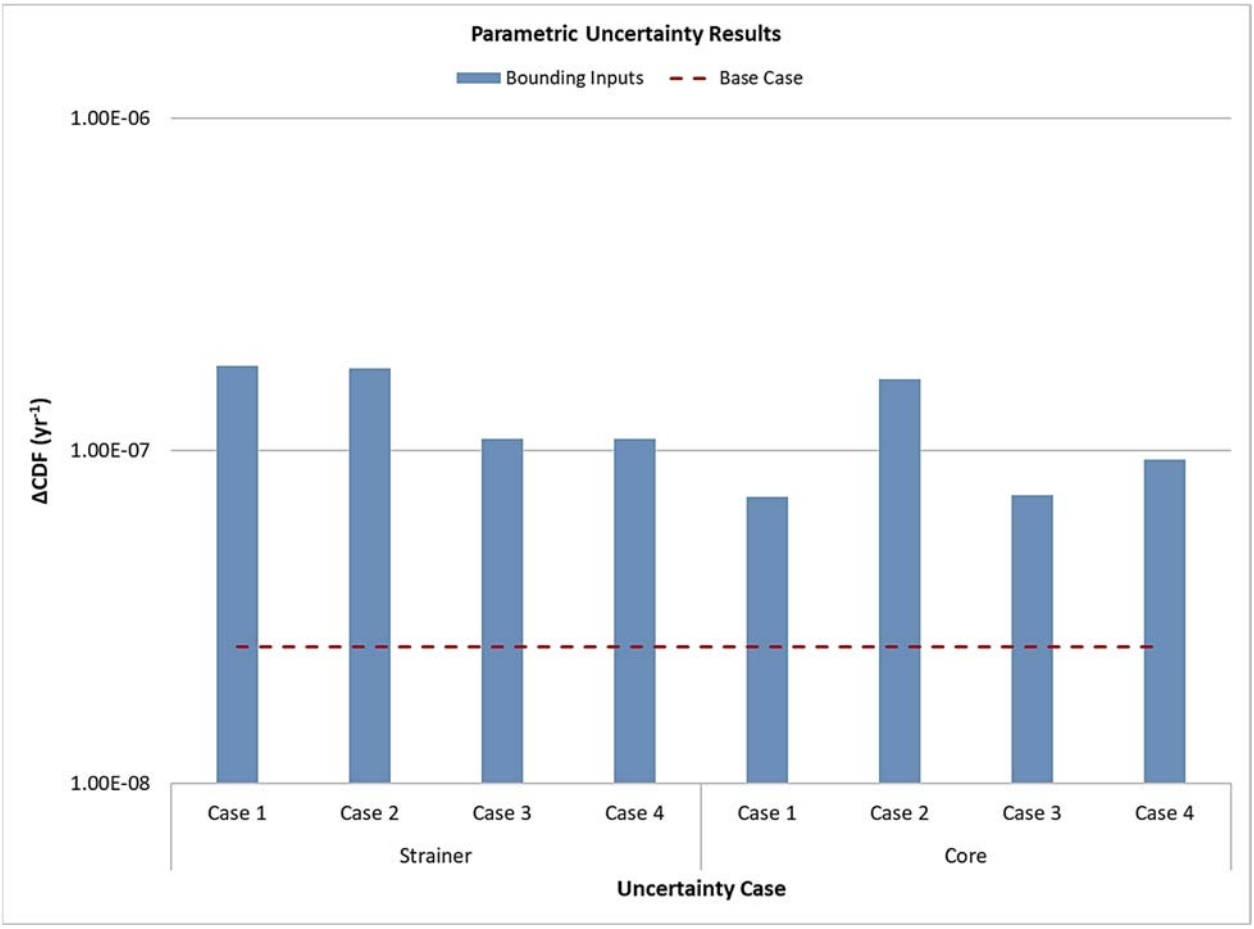

Figure A3-2 – Corrected Enclosure 3, Figure 3-10 – Comparison of Parametric Uncertainty Cases to the NARWHAL Base Case

A3-3.3.3 Model Uncertainty Cases

The corrected risk quantification results for the model uncertainty cases are provided in the updated table and figures below. These ΔCDF values are compared with that of the base case: 2.57×10^{-8} yr^{-1}.

Table A3-11 – Corrected Table 3-17 – Results of Model Uncertainty Quantification

Model with No Consensus	Sensitivity Case	ΔCDF	Change in ΔCDF from NARWHAL Base Case
Continuum Break Model	DEGB-Only Model	3.18E-08	6.15E-09
Top-Down LOCA Frequency Allocation with Values from the VEGP PRA	Top-Down LOCA Frequency Allocation with NUREG-1829 Arithmetic Mean Values	5.35E-07	5.10E-07
Methodology to Allocate LOCA Frequency to Welds	Hybrid Allocation Methodology: Skewed to High Rupture Probability Welds	5.04E-11	-2.56E-08
	Hybrid Allocation Methodology: Skewed to High and Medium Rupture Probability Welds	3.67E-09	-2.20E-08
	Hybrid Allocation Methodology: Spread Equally Across all Welds (top-down)	2.57E-08	0.00E+00
Breaks Activating Containment Sprays	All Breaks >15"	2.79E-08	2.26E-09
	All Breaks >6"	4.94E-08	2.37E-08
	All Breaks >2"	9.96E-07	9.70E-07
	No Breaks	2.70E-08	1.39E-09
UNM Aluminum Metal Release Equation	WCAP-16530 Equation	2.57E-08	0.00E+00
0.45-inch Fiber Thickness Required for Chemical Head Loss	0-inch Fiber Thickness Required for Chemical Head Loss	6.75E-08	4.19E-08
LBLOCA Size Range Discretization (6-15, 15-25, and 25-43.84 inches)	Bias 1 (6-10, 10-15, and 15-43.84 inches)	5.23E-08	2.66E-08
	Bias 2 (6-20, 20-27, and 27-43.84 inches)	2.54E-08	-2.91E-10

Attachment 3
Updated Evaluation for In-Vessel Effects and Coatings

Model with No Consensus	Sensitivity Case	ΔCDF	Change in ΔCDF from NARWHAL Base Case
Chemical Effects Sensitivities	Case 1 (Howe Solubility, WCAP-16530 Aluminum Metal Release, Best Estimate pH, Best Estimate Temperature Profiles)	1.53E-08	-1.04E-08
	Case 2 (No Calcium Phosphate Debris Limit Failure)	9.87E-09	-1.58E-08
	Case 3 (Howe Solubility, WCAP-16530 Aluminum Metal Release, Full Solubility)	2.57E-08	0.00E+00
NARWHAL Time Step Size	2 minutes	2.58E-08	9.76E-11
	3 minutes	2.55E-08	-1.26E-10
	4 minutes	2.59E-08	2.66E-10
	5 minutes	2.51E-08	-5.36E-10
	15 minutes	4.58E-08	2.01E-08

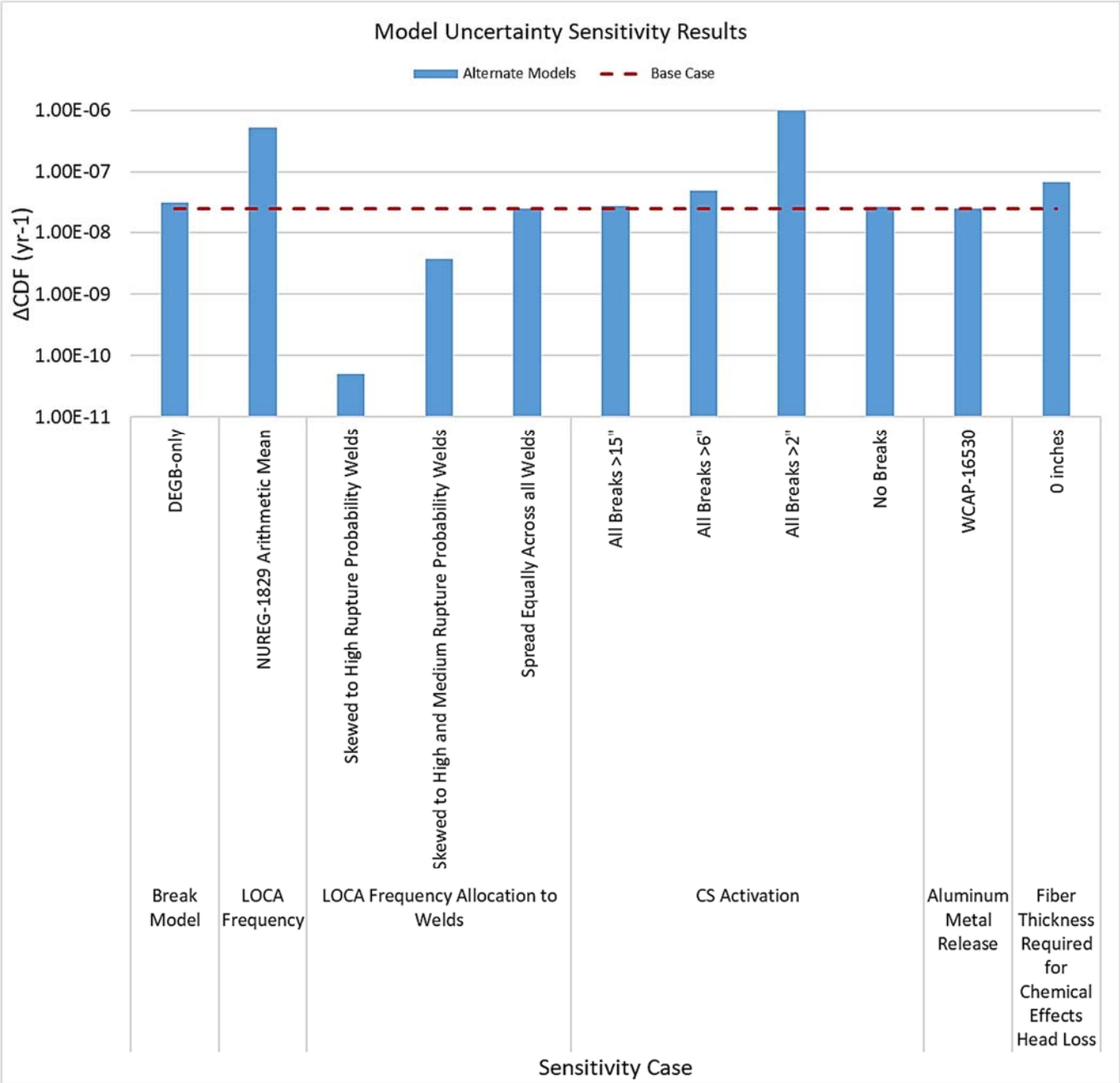

Figure A3-3 – Corrected Enclosure 3, Figure 3-11 – Comparison of Model Uncertainty Cases to the Base Case (1 of 2)

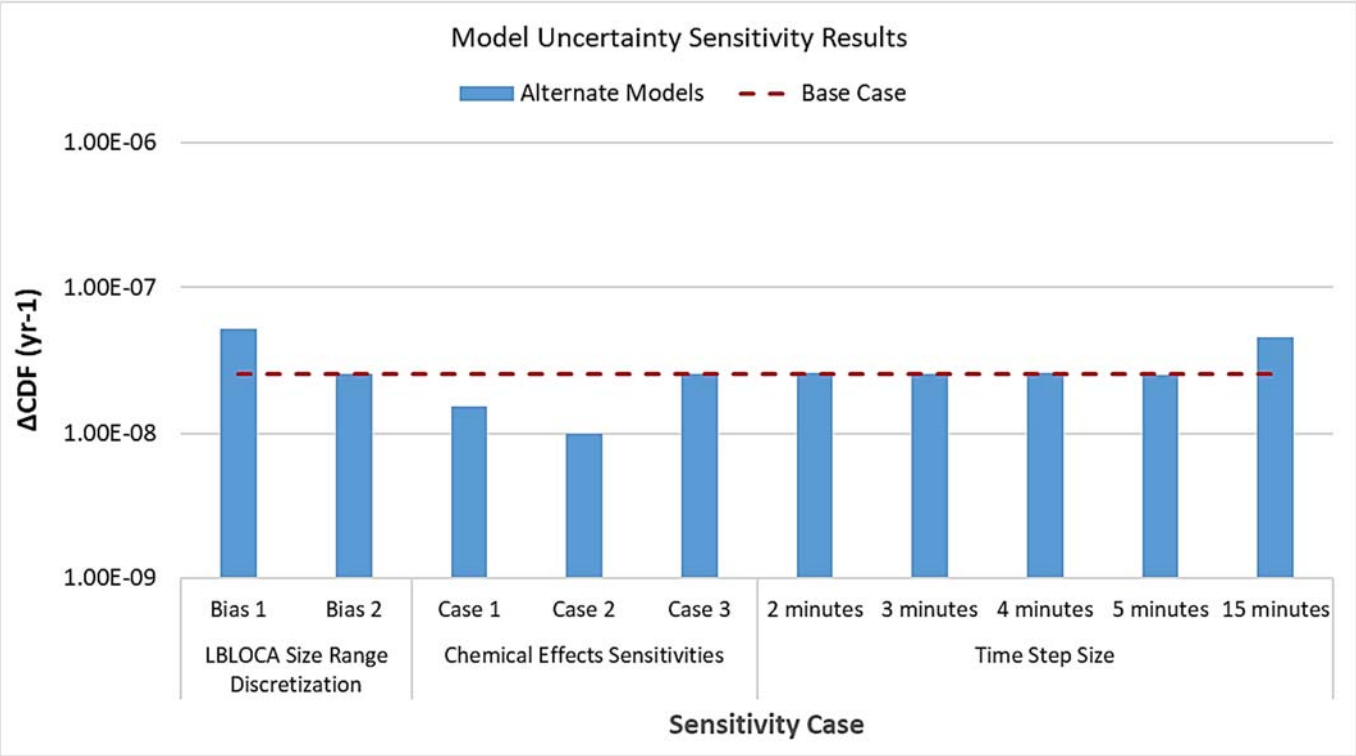

Figure A3-4 – Corrected Enclosure 3, Figure 3-12 – Comparison of Model Uncertainty Cases to the Base Case (2 of 2)

The results show that the ΔCDF for each of the parametric and model uncertainty cases is within Region III as defined in RG 1.174 (Reference 9). Therefore, it is concluded with high confidence that the risk associated with GSI-191 is very low as defined by the acceptance guidelines in RG 1.174. Correcting the coatings error does not change this conclusion from the July 2018 submittal (Reference 1).

A3-4 REFERENCES

1. SNC Letter NL-18-0915 (ML18193B163 and ML18193B165), “Vogtle Electric Generating Plant – Units 1&2, Supplemental Response to NRC Generic Letter 2004-02,” July 10, 2018.
2. SNC Letter NL-18-1447 (ML18338A497), “Vogtle Electric Generating Plant Units 1&2, Systematic Risk-Informed Assessment of Debris Technical Report Supplemental Information,” December 4, 2018.
3. ML19120A469, “Final Staff Evaluation for Vogtle Electric Generating Plant, Units 1 and 2, Systematic Risk-Informed Assessment of Debris Technical Report (EPID L-2017-TOP-0038),” September 30, 2019.
4. ML19178A252, “Technical Evaluation Report of In-Vessel Debris Effects,” June 13, 2019.
5. ML19228A011, “U.S. Nuclear Regulatory Commission Staff Review Guidance for In-Vessel Downstream Effects Supporting Review of Generic Letter 2004-02 Responses,” September 4, 2019.

6. PWROG-16073-P, “TSTF-567 Implementation Guidance, Evaluation of In-Vessel Debris Effects, Submittal Template for Final Response to Generic Letter 2004-02 and FSAR Changes,” Revision 0.
7. WCAP-17788-P, Volumes 1 – 6, “Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090),” Revision 0.
8. WCAP-17788-P, Volumes 1 – 6, “Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090),” Revision 1.
9. Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Revision 3.

Vogtle Electric Generating Plant
Licensing Submittal for a Risk-Informed Resolution of Generic Letter 2004-02

Enclosure 3

Proposed Changes to the Technical Specifications

Table of Contents

Enclosure 3
Proposed Changes to the Technical Specifications

1.0 DESCRIPTION

Southern Nuclear Operating Company (SNC) requests revision to the Vogtle Electric Generating Plant (VEGP), Units 1 and 2, Technical Specifications (TS) to address post-accident debris effects on the containment sump. The selected TS changes follow the model application in TSTF-567, "Add Containment Sump TS to Address GSI-191 Issues," Revision 1 (Reference 1).

The proposed amendment adds a new Technical Specification (TS) 3.6.7, "Containment Sump," and adds an Action to address the condition of the containment sump made inoperable due to containment accident generated and transported debris exceeding the analyzed limits. The Action provides time to correct or evaluate the condition in lieu of an immediate plant shutdown. This Action is placed in a new specification on the containment sump that otherwise retains the existing Technical Specifications requirements. An existing Surveillance Requirement (SR) is moved from TS 3.5.2 to the new specification. The requirement to perform the SR in TS 3.5.3 is deleted.

The regulatory evaluation and environmental consideration for the proposed TS changes are addressed in Enclosure 4 of this submittal.

2.0 ASSESSMENT

2.1 Applicability of Safety Evaluation

SNC has reviewed the safety evaluation for TSTF-567 provided to the Technical Specifications Task Force in a letter dated July 3, 2018 (Reference 2). This review included the NRC staff's evaluation, as well as the information provided in TSTF-567. As described herein, SNC has concluded that the justifications presented in TSTF-567 and the safety evaluation prepared by the NRC staff are applicable to VEGP, Units 1 and 2, and justify this amendment for the incorporation of the changes to the VEGP TS.

2.2 Variations

SNC is proposing the following variations from the TS changes described in TSTF-567 or the applicable parts of the NRC staff's safety evaluation. These variations do not affect the applicability of TSTF-567 or the NRC staff's safety evaluation to the proposed license amendment.

- The VEGP TS utilize different numbering than the Standard Technical Specifications on which TSTF-567 was based. Specifically, SR 3.5.2.8 in NUREG-1431 (Reference 3) is SR 3.5.2.7 in the VEGP TS and the new containment sump specification, TS 3.6.19, in TSTF-567 (Reference 1) is TS 3.6.7 in the VEGP TS. These differences are administrative and do not affect the applicability of TSTF-567 to the VEGP TS.

- The VEGP Technical Specifications contain a Surveillance Frequency Control Program. Therefore, the Frequency for Surveillance Requirement 3.6.7.1 is "In accordance with the Surveillance Frequency Control Program."

- The required action and Notes of proposed Condition B in TSTF-567, Revision 1, are revised to require declaring the affected ECCS and CSS trains inoperable immediately instead of a requirement to restore the containment sump to operable status within a specific completion time and two notes requiring entry into the associated ECCS and CSS TS actions. In addition, Condition C is revised to state, “Required Action and associated Completion Time of Condition A not met,” since the proposed required actions of Condition B to “declare” affected trains inoperable are immediate and can easily be accomplished rendering Condition C unnecessary. The TS Bases markups have also been revised to reflect the changes to the actions. These changes have minimal impact and are further explained, as follows.

  When the containment sump is inoperable for reasons other than Condition A, such as blockage, structural damage, or abnormal corrosion that could prevent recirculation of coolant, one or more ECCS or CSS trains are rendered inoperable; therefore, declaring the affected trains inoperable immediately will ensure appropriate restrictions are implemented in accordance with the required actions of the ECCS and CSS TS.

  As indicated in TSTF-567, Revision 1, the completion time of TSTF-567 Required Action B.1 is specified as either 7 days or 72 hours depending on the completion time established for a single inoperable ECCS or CSS train. This action is redundant to an action that will also be required when one ECCS or CSS train is inoperable since TSTF-567, Required Action B.1, Note 1, will require the actions of TS 3.5.2 and TS 3.5.3 to be applied and Note 2 will require the actions of TS 3.6.6 to be applied. In addition, if more than one train of ECCS or CSS is inoperable or a combination of ECCS and CSS trains are inoperable as a result of the inoperable containment sump, the TSTF-567 Condition B notes will require more restrictive actions than Required Action B.1. The proposed required actions (Required Actions B.1 and B.2) achieve the same goal while providing simplified action requirements.

  This plant-specific variation is considered administrative since the proposed requirements will result in equivalent action taken for the condition and, therefore, does not affect the applicability of the TSTF-567 model application or the NRC staff's model safety evaluation for the proposed TS changes.

- The containment sump debris limits are provided in the TS Bases, Table B 3.6.7-1, instead of the Final Safety Analysis Report (FSAR). Any changes to the debris limits in Table B 3.6.7-1 are subject to review under 10 CFR 50.59 and 10 CFR 50.71(e) reporting requirements. This is an administrative change to place the debris limits in a more convenient location for the operators.

3.0 REFERENCES

1. ML17214A813, TSTF-567, “Add Containment Sump TS to Address GSI-191 Issues,” Revision 1, August 2, 2017.
2. ML18109A071, Final Safety Evaluations of Technical Specifications Task Force Traveler TSTF-567, Revision 1, “Add Containment Sump TS to Address GSI-191 Issues” (EPID: L-2017-PMP-0005), July 3, 2018; Enclosure 1 (ML18116A606), Enclosure 2 (ML18116A599), Enclosure 3 (ML18117A453).
3. NUREG-1431, Volume 1, “Standard Technical Specifications, Westinghouse Plants,” Revision 4.0, April 2012.

Vogtle Electric Generating Plant

Licensing Submittal for a Risk-Informed Resolution of Generic Letter 2004-02

Enclosure 3

Proposed Changes to the Technical Specifications

Attachment 1

Proposed Technical Specification Changes (Mark-Up)

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
~~SR 3.5.2.7~~	~~Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion.~~	~~In accordance with the Surveillance Frequency Control Program~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	----------NOTE---------- An RHR train may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned to the ECCS mode of operation. ---------- The following SRs are applicable for all equipment required to be OPERABLE: SR 3.5.2.3 ~~SR 3.5.2.7~~ SR 3.5.2.4	In accordance with applicable SRs

3.6 CONTAINMENT SYSTEMS

3.6.7 Containment Sump

LCO 3.6.7 Four containment sumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment sumps inoperable due to containment accident generated and transported debris exceeding the analyzed limits.	A.1 Initiate action to mitigate containment accident generated and transported debris.	Immediately
	AND	
	A.2 Perform SR 3.4.13.1.	Once per 24 hours
	AND	
	A.3 Restore the containment sumps to OPERABLE status.	90 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more containment sumps inoperable for reasons other than Condition A.	B.1 Declare affected Emergency Core Cooling System train(s) inoperable. AND B.2 Declare affected containment spray train(s) inoperable.	Immediately Immediately
C. Required Action and associated Completion Time of Condition A not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.7.1	Verify, by visual inspection, the containment sumps do not show structural damage, abnormal corrosion, or debris blockage.	In accordance with the Surveillance Frequency Control Program

Vogtle Electric Generating Plant

Licensing Submittal for a Risk-Informed Resolution of Generic Letter 2004-02

Enclosure 3

Proposed Changes to the Technical Specifications

Attachment 2

Revised Technical Specification Pages

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	----------NOTE---------- An RHR train may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned to the ECCS mode of operation. ---------- The following SRs are applicable for all equipment required to be OPERABLE: SR 3.5.2.3 SR 3.5.2.4	In accordance with applicable SRs

3.6 CONTAINMENT SYSTEMS

3.6.7 Containment Sump

LCO 3.6.7 Four containment sumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment sumps inoperable due to containment accident generated and transported debris exceeding the analyzed limits.	A.1 Initiate action to mitigate containment accident generated and transported debris.	Immediately
	AND	
	A.2 Perform SR 3.4.13.1.	Once per 24 hours
	AND	
	A.3 Restore the containment sumps to OPERABLE status.	90 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more containment sumps inoperable for reasons other than Condition A.	B.1 Declare affected Emergency Core Cooling System train(s) inoperable. AND B.2 Declare affected containment spray train(s) inoperable.	Immediately Immediately
C. Required Action and associated Completion Time of Condition A not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.7.1	Verify, by visual inspection, the containment sumps do not show structural damage, abnormal corrosion, or debris blockage.	In accordance with the Surveillance Frequency Control Program

Vogtle Electric Generating Plant

Licensing Submittal for a Risk-Informed Resolution of Generic Letter 2004-02

Enclosure 3

Proposed Changes to the Technical Specifications

Attachment 3

Technical Specification Bases Changes (Mark-Up) for Information Only

(continued)

BASES

BACKGROUND (continued)

The centrifugal charging subsystem of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive accumulators, ~~and~~ the RWST, *and the containment sump,* covered in LCO 3.5.1, "Accumulators," ~~and~~ LCO 3.5.4, "Refueling Water Storage Tank (RWST)," *and LCO 3.6.7, "Containment Sump,"* provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI and RWST level low-low (for RHR semiautomatic switchover to the containment sump) signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~SR 3.5.2.7~~

~~Periodic inspections of the containment sump suction inlet ensure that it is unrestricted and stays in proper operating condition. There are no high-energy line breaks postulated to occur near the screens, and there are no missiles generated in the vicinity of the suction screens; therefore, there are no jet loads, no pipe whip restraint loads, nor missiles applicable to the screens. The screens are designed to withstand the loading for the largest postulated debris quantity, pieces, and types. The design of the stacked disk screen prevents large debris from reaching the perforated inner area of the screens due to small slots between the screen disks. Structurally, the stacked disk screen is designed as an integral screen and trash rack. Thus, inspection of the stacked disk screens includes the structural aspects of the trash rack. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. FSAR, Section 6.3, ECCS.
4. FSAR, Chapter 15, "Accident Analysis."
5. NRC Memorandum to V. Stello, Jr., from R.L. Baer," Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
6. IE Information Notice No. 87-01.
7. ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code).

(continued)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS — Shutdown

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS—Operating," is applicable to these Bases, with the following modifications.

In MODE 4, the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and residual heat removal (RHR) (low head).

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) *and the* ~~or~~ containment ~~emergency~~ sump can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that certain automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

Only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation. The ECCS trains satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

In MODE 4, one of the two independent (and redundant) ECCS trains is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Containment Sump

BASES

BACKGROUND

The containment sumps provide a borated water source to support recirculation of coolant from the containment sumps for residual heat removal, emergency core cooling, containment cooling, and containment atmosphere cleanup during accident conditions.

The containment sumps supply both trains of the Emergency Core Cooling System (ECCS) and the Containment Spray System (CSS) during any accident that requires recirculation of coolant from the containment sumps. The recirculation mode is initiated when the pump suction is transferred to the containment sumps on low Refueling Water Storage Tank (RWST) level, which ensures the containment sumps have enough water to supply the net positive suction head to the ECCS and CSS pumps. There are four containment sumps with two sumps providing suction for the two independent ECCS trains and two sumps providing suction for the two independent CSS trains.

The containment sumps contain strainers to limit the quantity of the debris materials from entering the sump suction piping. Debris accumulation on the strainers can lead to undesirable hydraulic effects including air ingestion through vortexing or deaeration, and reduced net positive suction head (NPSH) at pump suction piping.

While the majority of debris accumulates on the strainers, some fraction penetrates the strainers and is transported to downstream components in the ECCS, CSS, and the Reactor Coolant System (RCS). Debris that penetrates the strainer can result in wear to the downstream components, blockages, or reduced heat transfer across the fuel cladding. Excessive debris in the containment sump water source could result in insufficient recirculation of coolant during the accident, or insufficient heat removal from the core during the accident.

APPLICABLE SAFETY ANALYSIS

During all accidents that require recirculation, the containment sumps provide a source of borated water to the ECCS and CSS pumps. As such, it supports residual heat removal, emergency core cooling, containment cooling, and containment atmosphere cleanup during an accident. It also provides a source of negative reactivity (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "ECCS - Operating," B 3.5.3, "ECCS - Shutdown," and B 3.6.6, "Containment Spray and Cooling Systems."

BASES

APPLICABLE SAFETY ANALYSIS (continued)

FSAR Appendix 6A (Ref. 2) describes evaluations that confirm long-term core cooling is assured following any accident that requires recirculation from the containment sump.

The containment sumps satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Four containment sumps are required to ensure a source of borated water to support ECCS and CSS OPERABILITY. A containment sump consists of the containment drainage flow paths, the associated containment sump strainer, and the inlet to the ECCS or CSS piping. An OPERABLE containment sump has no structural damage or abnormal corrosion that could prevent recirculation of coolant and will not be restricted by containment accident generated and transported debris.

Containment accident generated and transported debris consists of the following:

a. Accident generated debris sources - Insulation, coatings, and other materials which are damaged by the high-energy line break (HELB) and transported to the containment sump. This includes materials within the HELB zone of influence and other materials (e.g., unqualified coatings) that fail due to the post-accident containment environment following the accident;

b. Latent debris sources – Pre-existing dirt, dust, paint chips, fines or shards of insulation, and other materials inside containment that do not have to be damaged by the HELB to be transported to the containment sump; and

c. Chemical product debris sources – Aluminum, zinc, carbon steel, copper, and non-metallic materials such as paints, thermal insulation, and concrete that are susceptible to chemical reactions within the post-accident containment environment leading to corrosion products that are generated within the containment sump pool or are generated within containment and transported to the containment sump.

Containment debris limits are defined in Table B 3.6.7-1 and additional discussion is provided in FSAR Appendix 6A (Ref. 2).

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, containment sump OPERABILITY requirements are dictated by the ECCS and CSS OPERABILITY requirements. Since both the ECCS and the CSS must be OPERABLE in MODES 1, 2, 3, and 4, the containment sumps must also be OPERABLE to support their operation.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the containment sumps are not required to be OPERABLE in MODES 5 or 6.

ACTIONS

A.1, A.2, and A.3

Condition A is applicable when there is a condition which results in containment accident generated and transported debris exceeding the analyzed limits. Containment debris limits are defined in Table B 3.6.7-1 and additional discussion is provided in FSAR Appendix 6A (Ref. 2). Immediate action must be initiated to mitigate the condition. Examples of mitigating actions are:

- *Removing the debris source from containment or preventing the debris from being transported to the containment sump;*
- *Evaluating the debris source against the assumptions in the analysis;*
- *Deferring maintenance that would affect availability of the affected systems and other LOCA mitigating equipment;*
- *Deferring maintenance that would affect availability of primary defense-in-depth systems, such as containment coolers;*
- *Briefing operators on LOCA debris management actions; or*
- *Applying an alternative method to establish new limits.*

While in this condition, the RCS water inventory balance, SR 3.4.13.1, must be performed at an increased Frequency of once per 24 hours. An unexpected increase in RCS leakage could be indicative of an increased potential for an RCS pipe break, which could result in debris being generated and transported to the containment sump. The more frequent monitoring allows operators to act in a timely fashion to minimize the potential for an RCS pipe break while the containment sump is inoperable.

BASES

ACTIONS (continued)

For the purposes of applying LCO 3.0.6 and the Safety Function Determination Program while in Condition A, the four containment sumps are considered a single support system for all ECCS and CSS trains because containment accident generated and transported debris issues that would render one sump inoperable would render all of the sumps inoperable.

The inoperable containment sump must be restored to OPERABLE status in 90 days. A 90-day Completion Time is reasonable for emergent conditions that involve debris in excess of the analyzed limits that could be generated and transported to the containment sump under accident conditions. The likelihood of an initiating event in the 90-day Completion Time is very small and there is margin in the associated analyses. The mitigating actions of Required Action A.1 provide additional assurance that the effects of debris in excess of the analyzed limits will be mitigated during the Completion Time.

B.1

When one or more containment sumps are inoperable for reasons other than Condition A, such as blockage, structural damage, or abnormal corrosion that could prevent recirculation of coolant, the affected ECCS and CSS trains are rendered inoperable; therefore, the affected ECCS and CSS trains must be immediately declared inoperable. Declaring the affected trains inoperable ensures appropriate restrictions are implemented in accordance with the Required Actions of the ECCS and CSS Specifications.

C.1 and C.2

If the containment sump cannot be restored to OPERABLE status within the associated Completion Time for Condition A, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Periodic inspections are performed to verify the containment sumps do not show current or potential debris blockage, structural damage, or abnormal corrosion to ensure the operability and structural integrity of the containment sumps (Ref. 1).

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. *FSAR, Chapter 6 and Chapter 15.*
2. *FSAR Appendix 6A, Resolution of NRC Generic Letter 2004-02.*

Table B 3.6.7-1 (page 1 of 1)
Containment Sump Debris Limits Applicable to Breaks ≤ 6 inches for 1 Train Operation and to Breaks ≤ 10 inches for 2 Train Operation

Debris Type	Acceptable Limit
Insulation Fiber Debris*	95.2 ft^3
Qualified and Unqualified Coatings Debris*	16.8 ft^3
Fire Barrier Debris*	284.7 lb_m
Latent Debris**	200 lb_m
Miscellaneous Debris (Tags, Labels, etc.)**	50 ft^2

* Acceptable quantity of debris on one RHR strainer
** Total quantity of debris in containment, which would predominantly transport to the active RHR strainer for single train operation and split proportionally for two train operation.

Vogtle Electric Generating Plant
Licensing Submittal for a Risk-Informed Resolution of Generic Letter 2004-02

Enclosure 4

Regulatory Evaluation and Environmental Consideration

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1.0 REGULATORY EVALUATION

1.1 Applicable Regulatory Requirements/Criteria

Approval of the proposed amendment is contingent upon approval of the request for exemption from certain aspects of 10 CFR 50.46(a)(1) as provided and justified in Enclosure 1 of this submittal.

1.1.1 Regulatory Guide 1.174

NRC Regulatory Guide (RG) 1.174 (Reference 3) provides the NRC staff's recommendations for using risk information in support of licensee-initiated licensing basis changes to a nuclear power plant that require NRC review and approval. This RG describes an acceptable approach for assessing the nature and impact of proposed licensing basis changes by considering engineering issues and applying risk insights.

In implementing risk-informed decision-making, licensing basis changes are expected to meet a set of key principles. These principles include the following:

(1) <u>The proposed change meets the current regulations unless it is explicitly related to a requested exemption (i.e., a specific exemption under 10 CFR 50.12, "Specific Exemptions").</u>

The exemption requested in Enclosure 1 of this submittal complies with this requirement.

(2) <u>The proposed change is consistent with a defense-in-depth philosophy.</u>

Defense-in-depth (DID) is presented in detail in Enclosure 4 of the July 2018 submittal (Reference 2). The proposed change is consistent with the DID philosophy in that the following aspects of the facility design and operation are unaffected:

- Functional requirements and the design configuration of systems
- Existing plant barriers to the release of fission products
- Design provisions for redundancy, diversity, and independence
- Plant response to transients or other initiating events
- Preventive and mitigative capabilities of plant design features

The Vogtle Electric Generating Plant (VEGP) risk-informed approach analyzes a full spectrum of loss of coolant accidents (LOCAs), including double-ended guillotine breaks (DEGBs) for all piping sizes up to and including the largest pipe in the reactor coolant system. By requiring that mitigative capability be maintained in a risk-informed evaluation of GSI-191 for a full spectrum of LOCAs, the approach ensures that DID is maintained.

(3) The proposed change maintains sufficient safety margins.

As described in Enclosure 4 of the July 2018 submittal (Reference 2), sufficient safety margins associated with the design will be maintained by the proposed change.

(4) When proposed changes result in an increase in CDF or risk, the increases should be small and consistent with the intent of the Commission's safety goal policy statement.

The proposed change involves evaluation of the risk associated with effects of accident-generated and transported debris using a risk-informed methodology. Using engineering analysis and the probabilistic risk assessment (PRA), this risk has been calculated and shown to be "very small" as defined by Region III in RG 1.174 (Reference 3) and is therefore consistent with the Commission's safety goal policy statement.

(5) The impact of the proposed change should be monitored using performance measurement strategies.

Performance monitoring is discussed in Section 3.2.4 of Enclosure 2.

1.1.2 Regulatory Guide 1.200

NRC RG 1.200 (Reference 5) describes one acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for light water reactors. As described in Enclosure 1 of the July 2018 submittal (Reference 2), the VEGP PRA model used for the risk-informed GSI-191 evaluation complies with RG 1.200 (Reference 5).

1.2 Precedent

The proposed licensing change for VEGP is very similar to the license amendment and 10 CFR 50.46(a)(1) exemption granted to South Texas Project Nuclear Operating Company for implementation of the risk-informed approach to address GSI-191 concerns at South Texas Project (STP), Units 1 and 2 (References 6 and 7).

VEGP requests implementation of a risk-informed methodology for resolution of GSI-191 that is similar to the approach used by STP. Key similarities include, but are not limited to:

- Use of RG 1.174 acceptance guidelines and key principles.
- Identification of key methods and approaches in the risk-informed methodology that, if changed after implementation, are to be evaluated as a potential

"departure from a method of evaluation described in the FSAR" under 10 CFR 50.59.
- Associated request for exemption from 10 CFR 50.46(a)(1) "other properties."
- Technical Specification (TS) changes that provide for additional time to address the effects of debris on emergency core cooling system (ECCS) and containment spray system (CSS) operability.

Key differences include, but are not limited to:

- Software used for the VEGP risk analysis (NARWHAL) differs from the software used for the STP analysis (CASA Grande).
- The methodology used for the Vogtle risk quantification (conditional failure probability approach) differs from the methodology used by STP (RoverD approach). The NRC conducted a review of the analysis methodology used by VEGP and documented it in a Staff Evaluation (Reference 4).
- STP requested exemption from 10 CFR 50, Appendix A, General Design Criteria 35, 38 and 41. Southern Nuclear Operating Company (SNC) determined that an exemption from the general design criteria is not necessary. The approval of SNC's exemption to 10 CFR 50.46(a)(1) and of this LAR provides an acceptable alternative to meet the intent of the stated General Design Criteria.
- The VEGP TS changes follow the TSTF-567, Revision 1 model application, which was not available at the time of the STP LAR submittal.

1.3 No Significant Hazards Consideration

The proposed amendment implements a risk-informed approach to address the effects of accident-generated and transported debris on the containment emergency sumps.

The proposed amendment also proposes changes to the VEGP, Unit 1 and Unit 2, Technical Specifications (TS). The proposed TS changes add a new Technical Specification (TS) 3.6.7, "Containment Sump," and add an Action to address the condition of the containment sump made inoperable due to containment accident generated and transported debris exceeding the analyzed limits. The Action provides time to correct or evaluate the condition in lieu of an immediate plant shutdown. This Action is placed in a new specification on the containment sump that otherwise retains the existing Technical Specifications requirements. An existing Surveillance Requirement (SR) is moved from TS 3.5.2 to the new specification. The requirement to perform the SR in TS 3.5.3 is deleted.

SNC has evaluated whether a significant hazards consideration is involved with the proposed change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

(1) Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed change is a methodology change for assessment of debris effects that adds the results of a risk-informed evaluation to the VEGP licensing basis. This is a viable approach for the resolution of GSI-191 per SECY-12-0093 (Reference 1). The analysis that supports the methodology change concludes that the functionality of the ECCS and CSS during design basis accidents is confirmed by demonstrating that safety margin and DID are maintained with high probability (Reference 2).

There is no significant increase in the probability of an accident previously evaluated. The proposed change addresses mitigation of LOCAs and has no effect on the probability of the occurrence of a LOCA. The proposed methodology change does not implement any changes in the facility or plant operation that could lead to a different kind of accident.

The proposed change does not involve a significant increase in the consequences of an accident previously evaluated. The methodology change confirms that required structures, systems, and components (SSCs) supported by the containment sumps will perform their safety functions with a high probability, as required, and does not alter or prevent the ability of SSCs to perform their intended function to mitigate the consequences of an accident previously evaluated within the acceptance limits. The safety analysis acceptance criteria in the Final Safety Analysis Report (FSAR) continue to be met for the proposed methodology change. The evaluation of the proposed change determined that containment integrity will be maintained. The dose consequences were considered in the assessment and quantitative evaluation of the effects on dose using input from the risk-informed approach shows the increase in dose consequences is very small (Reference 2).

The proposed change also adds a new specification to the TS for the containment sump. An existing SR on the containment sump is moved to the new specification and a duplicative requirement to perform the SR in TS 3.5.3 is removed. The new specification retains the existing requirements on the containment sump and the actions to be taken when the containment sump is inoperable with the exception of adding new actions to be taken when the containment sump is inoperable due to containment accident generated and transported debris exceeding the analyzed limits. The new action provides time to evaluate and correct the condition instead of requiring an immediate plant shutdown.

The containment sump is not an initiator of any accident previously evaluated. The containment sump is a passive component and the proposed change does

not increase the likelihood of the malfunction. As a result, the probability of an accident is unaffected by the proposed change.

The containment sump is used to mitigate accidents previously evaluated by providing a borated water source for the ECCS and CSS. The design of the containment sump and the capability of the containment sump assumed in the accident analysis is not changed. The proposed action requires implementation of mitigating actions while the containment sump is inoperable and more frequent monitoring of reactor coolant leakage to detect any increased potential for an accident that would require the containment sump. The consequences of an accident during the proposed action are no different than the current consequences of an accident if the containment sump is inoperable.

Therefore, the proposed methodology and TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed change is a methodology change for assessment of debris effects from LOCAs and secondary side breaks that are already evaluated in the VEGP FSAR. No new or different kind of accident is being evaluated. The proposed change does not install or remove any plant equipment, or alter the design, physical configuration, or mode of operation of any plant SSCs. The proposed change does not introduce any new failure mechanisms or malfunctions that can initiate an accident.

The proposed change also adds a new specification to the TS for the containment sump. An existing SR on the containment sump is moved to the new specification and a duplicative requirement to perform the SR in TS 3.5.3 is removed. The new specification retains the existing requirements on the containment sump and the actions to be taken when the containment sump is inoperable with the exception of adding new actions to be taken when the containment sump is inoperable due to containment accident generated and transported debris exceeding the analyzed limits. The new action provides time to evaluate and correct the condition instead of requiring an immediate plant shutdown.

The proposed change does not alter the design or design function of the containment sump or the plant. No new systems are installed or removed as part of the proposed change. The containment sump is a passive component and cannot initiate a malfunction or accident. No new credible accident is created that

is not encompassed by the existing accident analyses that assume the function of the containment sump.

Therefore, the proposed methodology and TS changes do not create the possibility of a new or different kind of accident from any previously evaluated.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

The proposed change is a methodology change for assessment of debris effects from LOCAs and secondary side breaks that are already evaluated in the VEGP FSAR. The effects from a full spectrum of LOCAs and secondary side breaks inside containment, including DEGBs, are analyzed (Reference 2). Appropriate redundancy and consideration of loss of offsite power and worst-case single failure are retained, such that DID is maintained.

Application of the risk-informed methodology showed that the increase in risk from the contribution of debris effects is very small as defined by RG 1.174 (Reference 3) and that there is adequate DID and safety margin (Reference 2). DID and safety margin were extensively evaluated. This evaluation showed that there is substantial DID and safety margin that provide a high level of confidence that the calculated risk for the effects of debris is conservative and that the actual risk is likely much lower. Consequently, VEGP determined that the risk-informed method demonstrates the containment sumps will continue to support the ability of safety-related components to perform their design functions when the effects of debris are considered. Note that the risk-informed approach was identified as viable for the resolution of GSI-191 per SECY-12-0093 (Reference 1).

The proposed change does not alter the manner in which safety limits are determined or the acceptance criteria associated with a safety limit. The proposed change does not implement any changes to plant operation and does not significantly affect SSCs that respond to safely shutdown the plant and to maintain the plant in a safe shutdown condition. The proposed change does not significantly affect the existing safety margins in the barriers for the release of radioactivity. There are no changes to any of the safety analyses in the FSAR.

The proposed change also adds a new specification to the TS for the containment sump. An existing SR on the containment sump is moved to the new specification and a duplicative requirement to perform the SR in TS 3.5.3 is removed. The new specification retains the existing requirements on the containment sump and the actions to be taken when the containment sump is inoperable with the exception of adding new actions to be taken when the containment sump is inoperable due to containment accident generated and transported debris exceeding the analyzed limits. The new action provides time

to evaluate and correct the condition instead of requiring an immediate plant shutdown.

The proposed change does not affect the controlling values of parameters used to avoid exceeding regulatory or licensing limits. No Safety Limits are affected by the proposed change. The proposed change does not affect any assumptions in the accident analyses that demonstrate compliance with regulatory and licensing requirements.

Therefore, the proposed methodology and TS changes do not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

1.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the license amendment will not be inimical to the common defense and security or to the health and safety of the public.

2.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

3.0 REFERENCES

1. SECY-12-0093 (ML121310648), “Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance,” July 9, 2012.
2. SNC Letter NL-18-0915 (ML18193B163 and ML18193B165), “Vogtle Electric Generating Plant – Units 1 and 2, Supplemental Response to NRC Generic Letter 2004-02,” July 10, 2018.

3. Regulatory Guide 1.174, “An Approach for Using Probabilistic Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Revision 3, January 2018.
4. ML19120A469, “Final Staff Evaluation for Vogtle Electric Generating Plant, Units 1 and 2, Systematic Risk-Informed Assessment of Debris Technical Report (EPID L-2017-TOP-0038),” September 30, 2019.
5. Regulatory Guide 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Revision 2, March 2009.
6. ML17019A001, “South Texas Project, Units 1 and 2 – Issuance of Amendment Nos. 212 and 198 – Risk-Informed Approach to Resolve Generic Safety Issue 191 (CAC Nos. MF2400 and MF2401),” July 12, 2017.
7. ML17037C871, “South Texas Project, Units 1 and 2, Exemptions from the Requirements of 10 CFR Part 50, Section 50.46 and 10 CFR Part 50, Appendix A, General Design Criteria 35, 38, and 41 (CAC Nos. MF2402-MF2409), July 11, 2017.