UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 30, 2019

Ms. Cheryl A. Gayheart
Regulatory Affairs Director
Southern Nuclear Operating Co., Inc.
3535 Colonnade Parkway
Birmingham, AL 35243

SUBJECT: 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)

Dear Ms. Gayheart:

By letter dated April 21, 2017, as supplemented by letters dated July 11; and November 9, 2017, and January 2, January 9, February 6, February 12, February 21, May 23, July 10, and December 4, 2018, Southern Nuclear Operating Company (SNC) submitted a technical report for U.S. Nuclear Regulatory Commission (NRC) staff review regarding the use of a risk-informed approach to resolve Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR [Pressurized-Water Reactor] Sump Performance," at Vogtle Electric Generating Plant, Units 1 and 2 (Vogtle), and to supplement its response to NRC Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors."

The version of the technical report enclosed with the letter dated July 10, 2018, superseded the version of the technical report enclosed with the letter dated April 21, 2017, as supplemented by letters dated July 11, 2017, and November 9, 2017; and January 2, January 9, February 6, February 12, February 21, and May 23, 2018. The version of the technical report enclosed with SNC's letter dated July 10, 2018, incorporated clarifications identified during the NRC audit and review process, and incorporated SNC responses to NRC requests for additional information.

The NRC staff has found that the technical report enclosed with the letter dated July 10, 2018, is acceptable for use in plant-specfic licensing applications for Vogtle in accordance with the limitations and conditions section and applicability provided in the enclosed NRC staff evaluation. The enclosed staff evaluation provides the basis for the NRC to consider use of the technical report in future licensing applications. Except for downstream effects – fuel and vessel and licensing basis, the NRC staff has concluded that the technical report contains sufficient information to address the information requested in NRC GL 2004-02.

The NRC staff evaluation applies only to material provided in the subject technical report. License amendment requests that deviate from this technical report will be subject to additional review in accordance with applicable review standards.

If you have any questions or require any additional information, please feel free to contact the NRC Senior Project Manager for the review, Michael Marshall, at (301) 415-2871 or michael.marshall@nrc.gov.

Sincerely,

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosure:
Staff Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

STAFF EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

SYSTEMATIC RISK-INFORMED ASSESSMENT OF DEBRIS TECHNICAL REPORT

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

1.0 INTRODUCTION

1.1 Application

By letter dated April 21, 2017 (Reference 1), as supplemented by letters dated July 11 (Reference 2) and November 9, 2017 (Reference 3) and January 2 (Reference 4), January 9 (Reference 5), February 6 (Reference 6), February 12 (Reference 7), February 21 (Reference 8), May 23 (Reference 9), July 10 (Reference 10), and December 4, 2018 (Reference 11), Southern Nuclear Operating Company (SNC, the licensee) submitted a technical report regarding the use of a risk-informed approach to resolve Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR [Pressurized-Water Reactor] Sump Performance," and closeout Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors, " at Vogtle Electric Generating Plant, Units 1 and 2 (Vogtle, VEGP, or the licensee).

The version of the technical report enclosed with the letter dated July 10, 2018, supersedes the version of the technical report enclosed with the letter dated April 21, 2017, as supplemented by letters dated July 11, 2017, and November 9, 2017; and January 2, January 9, February 6, February 12, February 21, and May 23, 2018. The version of the technical report enclosed with SNC's letter dated July 10, 2018, incorporated clarifications identified during the U.S. Nuclear Regulatory Commission (NRC or the Commission) audit and review process, and incorporated SNC responses to NRC requests for additional information (RAIs).

1.2 Background

1.2.1 Challenges to Safety Systems Function from Debris in Containment

The function of the emergency core cooling system (ECCS) is to cool the reactor core and provide shutdown capability following a loss-of-coolant accident (LOCA). The primary functions of the containment spray system (CSS) are to reduce containment pressure and reduce the

concentration and quantity of fission products in the containment building after a LOCA.

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 core cooling (LTCC) 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 for extended cooling of the core in a pressurized water Reactor (PWR) once the initial water source has been depleted and the systems are switched over to recirculation mode.

If a LOCA occurs, piping thermal insulation and other materials located in containment may be dislodged by the two-phase (steam and liquid) water jet emanating from the broken pipe. This debris may be transported by the flow of water and steam from the break or from the CSS to the pool of water that collects in the containment recirculation sump. Once transported to the sump pool, the debris could be drawn toward the ECCS sump strainers, which are designed to prevent debris from entering the CSS and the ECCS. If this debris clogs the strainers, the ECCS could fail, resulting in core damage, or the CSS pumps could fail, resulting in containment pressure or radiation dose increasing beyond deterministic limits. It is also possible that some debris could bypass the sump strainers and get lodged in the reactor core. This could result in reduced core cooling and potential core damage.

1.2.2 <u>Generic Safety Issue - 191</u>

In 1996, the NRC identified an issue associated with the effects of debris accumulation on PWR sump performance during design-basis accidents (i.e., GSI-191). This issue was similar to concerns at boiling-water reactors (BWRs), to new information identified following closure of the actions taken for resolution of the issue at BWRs, and to confirmatory testing conducted by the NRC.

Findings from research and industry operating experience raised questions concerning the adequacy of PWR sump designs. Research findings demonstrated that the amount of debris generated and transported by a high-energy LOCA could be greater than originally anticipated. The debris from a LOCA could also be finer, and thus, more easily transportable, and could be comprised of debris consisting of fibrous material combined with particulate material that could result in a substantially greater flow restriction than an equivalent amount of either type of debris alone. These research findings prompted the NRC to open GSI-191.

The two distinct but related safety concerns are: (1) potential clogging of the sump strainers that results in ECCS or CSS pump failure, and (2) potential clogging of flow channels within the reactor vessel because of debris bypassing the sump strainers, often referred to as in-vessel effects. Clogging at either the strainers or in-vessel channels can result in loss of the LTCC safety function.

More information on the background, testing, and other actions associated with GSI-191 can be found in NUREG-0897, "Containment Emergency Sump Performance: Technical Findings Related to Unresolved Safety Issue A-43," dated October 1985 (Reference 12), and NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors," dated June 9, 2003 (Reference 13).

1.2.3 Generic Letter 2004-02

As part of the actions to resolve GSI-191, in September 2004, the NRC issued GL 2004-02 (Reference 14) to holders of operating licenses for PWRs. In GL 2004-02, the NRC staff requested that licensees perform an evaluation of their ECCS and CSS recirculation functions, considering the potential for debris-laden coolant to be circulated by the ECCS and the CSS after a LOCA or high-energy line break inside containment, and, if appropriate, take additional action to ensure system function. GL 2004-02 required, per Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.54(f), that licensees provide the NRC a written response describing the results of their evaluation and any modifications made, or planned, to ensure ECCS and CSS system function during recirculation following a design-basis event, or any alternate action proposed and the basis for its acceptability.

The staff requirements memorandum (SRM) associated with SECY-10-113, "Closure Options for Generic Safety Issue 191, Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance," dated December 23, 2010 (Reference 15), directed the NRC staff to consider a risk-informed approach for resolution of GSI-191. In 2012, the staff developed three options to resolve GSI-191. These options were documented and proposed to the Commission in SECY-12-0093 (Reference 16). The options are summarized as follows:

- Option 1 allows licensees to demonstrate compliance with 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," through approved models and test methods.

- Option 2 requires implementation of additional mitigating measures and allows additional time for licensees to resolve issues through further industry testing or use of a risk-informed approach.

- Option 3 involves separating the regulatory treatment of the sump strainer and in-vessel effects so that strainer issues can be treated deterministically and in-vessel issues can be risk-informed.

These options allowed industry alternative approaches for resolving GSI-191. The Commission issued SRM-SECY-12-0093 on December 14, 2012 (Reference 17), approving all three options for closure of GSI-191.

By letter dated May 16, 2013 (Reference 18) SNC stated that it would pursue Option 2 for the closure of GSI-191 and GL 2004-02, and intended to use a risk-informed methodology.

1.3 Licensee's Approach

While the licensee's technical report provides its risk-informed approach to evaluate the effects of debris, the analysis includes in-vessel fiber limits that were determined using a methodology that is not approved by the NRC. By letter dated February 14, 2017 (Reference 19), the NRC staff stated that it would not be appropriate for the NRC staff to accept for review a requested licensing action that relied upon methodology being reviewed by the NRC staff. This staff evaluation documents the NRC staff's review of the licensee's risk-informed approach to resolve GL 2004-02 at Vogtle, with the exception of in-vessel fiber limits.

The licensee's technical report enclosed with the letter dated July 10, 2018, considers three of the five key principles of risk-informed integrated decisionmaking, including assessment of

defense-in-depth and safety margins, described in Regulatory Guide (RG) 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (Reference 20). The licensee's overall evaluation of risk attributable to debris for Vogtle is based heavily on physical models that have been used in the past and accepted generically by the NRC for GSI-191 resolution. The licensee provided a detailed description of the plant-specific conditions and models related to GSI-191, as well as a description of the risk quantification using the Break Analysis Debris Generator (BADGER) and Nuclear Accident Risk Weighted Analysis (NARWHAL)[1] computer codes and the Vogtle probabilistic risk assessment (PRA) model.

The licensee described a risk-informed approach for determining the design requirements to address the effects of LOCA-generated debris on ECCS and CSS recirculation functions. The licensee stated that, if implemented, the risk-informed approach would replace the existing deterministic approach described in the Vogtle licensing basis, and consequently, would require amendments to the Vogtle Units 1 and 2, operating licenses to incorporate the revised methodology per the requirements of 10 CFR 50.59. SNC stated that the proposed amendments to the operating licenses will be described in future license amendment requests (LARs). In addition, the licensee stated that exemptions to the overall requirements associated with 10 CFR 50.46(a)(1); General Design Criterion (GDC) 35, "Emergency Core Cooling"; GDC 38, "Containment Heat Removal"; and GDC 41, "Containment Atmosphere Cleanup," will be provided in future LARs.

1.4 Method of NRC Staff Review

The purpose of the NRC staff's review was to evaluate the licensee's assessment of the impact of debris on ECCS and CSS functions following postulated LOCAs at Vogtle. The NRC staff evaluated the licensee's technical report (Reference 21). The NRC staff also conducted an audit of certain information and performed confirmatory calculations in areas deemed appropriate by the NRC staff (Reference 22). A separate audit specifically related to the BADGER and NARWHAL software was completed prior to the licensee's technical report submittal (Reference 23).

In areas where the licensee used NRC-approved or widely accepted methods in performing analyses related to the proposed methodology, the NRC staff reviewed relevant material to ensure that the licensee used the methods consistent with the limitations and conditions placed on the methods. Details of the NRC staff review, audit, and confirmatory calculations are provided in Section 3.0, "Technical Evaluation: Risk-Informed Methodology," of this SE.

The NRC staff did not review analyses or methodologies associated with the potential clogging of flow channels within the reactor vessel because of debris bypassing the sump strainers (i.e., in-vessel effects), with the exception of its possible contribution to risk assessment uncertainty. Further, the NRC staff did not review risk-informed licensing Principle 1 (i.e., the proposed licensing basis change meets the current regulations unless it is explicitly related to a requested exemption) or Principle 5 (i.e., the impact of the proposed licensing basis change should be monitored using performance measurement strategies), which are described in RG 1.174, Revision 3. The NRC staff's review documented below integrates several disciplines (e.g.,

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[1] NARWHAL, "Nuclear Accident Risk Weighted Analysis," is a computer program that evaluates the probability of failures associated with GSI-191 phenomena by holistically analyzing the break-specific consequences in a time-dependent manner. NARWHAL was used by the licensee to quantify the conditional failure probability of Vogtle ECCS sump strainers.

mechanical, structural, thermal-hydraulic, risk) and this SE is organized using the five key principles of risk-informed decision-making as listed in Section 3.0.

2.0 REGULATORY EVALUATION

2.1 Description of Affected Structures, Systems, and Components

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 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 Vogtle Updated Final Safety Analysis Report (UFSAR):

> 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 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. 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 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.

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 inches long by 30 inches wide by ~40 inches high, four of which provide 590 square feet $(ft)^2$ 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 percentage – 124 holes is larger than 3/32 in diameter, but none are larger than ¼ inch diameter (see Section 6.1.2 of this SE.) The screen is mounted over the containment sump. In the case that particles did traverse the screen, they would pass through the piping pumps and valves, as well as the 3/8 inch diameter containment spray nozzle openings, without difficulty. The screens bolt to the floor and may be removed by unbolting individual screen sections.

2.2 Description of Planned Sump Modification

The Vogtle ECCS sump strainer consists of stacked disk strainers designed by General Electric. The licensee stated that 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 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 licensee stated that it will modify the RHR strainers 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. The licensee stated that all of the analyses in its technical report were performed for the modified strainer configuration.

The licensee stated that the new strainer design does not involve backflushing or any other active approach.

2.3 Applicable Regulatory Requirements

The NRC staff's acceptance criteria for ECCS performance following a LOCA are based on 10 CFR 50.46. LOCAs are postulated accidents that would result in the loss of reactor coolant from piping breaks in the reactor coolant pressure boundary at a rate in excess of the capability of the normal reactor coolant makeup system to replenish it. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. The reactor protection and ECCS systems are provided to mitigate these accidents. The NRC staff's review covered the acceptance criteria based on 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria, considering the effects of debris as specified in GL 2004-02.

The following regulatory requirements are applicable to the review of the technical report attached to SNC's letter dated July 10, 2018.

- Section 50.46(a)(1)(i) of 10 CFR requires, in part, each PWR to be provided with an ECCS, and the ECCS performance must be calculated with an acceptable evaluation model. The performance must be calculated for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated.

- Section 50.46(b)(5) of 10 CFR requires licensees of domestic nuclear power plants to provide long-term cooling of the reactor core. 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 of time required by the long-lived radioactivity remaining in the core.

- Section 50.46(c)(2) of 10 CFR defines an evaluation model as the calculational framework for evaluating the behavior of the reactor system during a postulated LOCA. An evaluation model includes one or more computer programs and all other information necessary for applying the calculational framework to a specific LOCA (the mathematical models used, the assumptions included in the programs, the procedure for treating the program input and output information, the parts of the analysis not included in the computer programs, values of parameters, and all other information necessary to specify the calculational procedure). 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.

2.4 Applicable Regulatory Guides, Review Plans, and Guidance Documents

The Nuclear Energy Institute (NEI) developed an evaluation guidance document entitled "PWR Containment Sump Evaluation Methodology," dated May 28, 2004 (Reference 24). On December 6, 2004, the NRC issued an SE for that document that found the NEI document provided an acceptable overall guidance methodology, but that portions needed additional justification and modification (Reference 25). Modifications were made, and the final guidance was provided as NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," in December 2004 (Reference 26). Together, Volume 1 of NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," dated December 2004, and Volume 2, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02," dated December 6, 2004 (Reference 27), describe a method acceptable to the NRC staff, with limitations and conditions for performing the evaluations requested by GL 2004-02.

In addition to the evaluation guidance of NEI 04-07, the industry developed the following topical reports (TRs) to aid licensees in responding to GL 2004-02.

- TR-WCAP-16530-NP-A, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," dated March 2008 (Reference 28).
- TR-WCAP-16406-P-A, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," Revision 1, dated March 2008 (Reference 29).
- TR-WCAP-16793-NP-A, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," Revision 2, dated July 2013 (Reference 30).

The reports listed above, subject to the limitations and conditions contained in the NRC SEs for those TRs, describe methods acceptable to the NRC staff for performing the evaluations and analyses within the scope stated in those documents.

To more clearly communicate the NRC staff's expectations for the level of technical detail in the licensees' submittals, the NRC staff issued documents entitled "Revised Content Guide for Generic Letter 2004-02 Supplemental Responses," dated November 21, 2007 (Reference 31), and "Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02, 'Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized Water Reactors,'" dated March 28, 2008 (Reference 32). The content guide describes the information necessary to be submitted to the NRC for review in each of the following areas:

- corrective actions taken to address GL 2004-02
- break selection
- debris generation and zone of influence
- debris characteristics
- latent debris
- debris transport
- head loss and vortexing

- net positive suction head
- coatings evaluation
- debris source term
- screen modification package
- sump structural analysis
- upstream effects
- downstream effects – components and systems
- downstream effects – fuel and vessel
- chemical effects
- licensing basis

Section 3.8.3, "Concrete and Steel Internal Structures of Steel or Concrete Containments," of the Standard Review Plan (SRP) (Reference 33), lists acceptable codes and standards for design of containment internal structures.

RG 1.82, Revision 4 , "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," dated March 2012 (Reference 34), provides guidance for an evaluation of the effects of debris on ECCS strainers and, more generally, guidance for the evaluation of water sources for long-term recirculation following a LOCA.

RG 1.174, Revision 2, provides guidance on the use of PRA findings and risk insights in support of licensee requests for plant-specific changes to a licensing basis. This RG provides risk acceptance guidelines for evaluating the results of such evaluations. RG 1.174 also provides the five key principles of risk-informed integrated decisionmaking.

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities" (Reference 35), endorses, with clarifications, the American Society of Mechanical Engineers (ASME) and the American Nuclear Society (ANS) PRA Standard ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S 2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (ASME/ANS 2009 Standard) (Reference 36). The ASME/ANS 2009 Standard addresses PRAs for internal events and other hazards. RG 1.200 describes one acceptable approach for determining whether the technical adequacy of the PRA, in total, or the parts that are used to support an application, is acceptable for use in regulatory decisionmaking for light-water reactors.

Guidance on evaluating PRA acceptability is also provided in SRP Chapter 15, Section 15.0.2 (Reference 37) and SRP Chapter 19, Section 19.1, Revision 3, "Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests After Initial Fuel Load," (Reference 38). General guidance for evaluating the technical basis for proposed risk-informed changes is provided in SRP Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance" (Reference 39) Section 19.2 of this SRP references the same criteria as RG 1.174, and states that a risk-informed application should be evaluated to ensure that the proposed changes meet the five key principles of risk-informed decisionmaking.

3.0 TECHNICAL EVALUATION: RISK-INFORMED METHODOLOGY

The NRC staff performed an integrated review of the proposed risk-informed approach, considering three of the five key principles of risk-informed decisionmaking set forth in RG 1.174, Revision 2 (Reference 20). The five key principles are:

1. The proposed change meets the current regulations unless it is explicitly relates to a requested exemption.

2. The proposed change is consistent with the defense-in-depth philosophy.

3. The proposed change maintains sufficient safety margins.

4. When proposed changes result in an increase in risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement – 51 FR 30028 (Reference 40).

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

3.1 Key Principle 1: The Proposed Change Meets Current Regulations Unless it is Explicitly Related to a Requested Exemption

The NRC staff did not consider Principle 1 because Principle 1 was outside the scope of the technical report review.

3.2 Key Principle 2: The Proposed Change is Consistent with the Defense-in-Depth Philosophy

Section C.2.1.1 of RG 1.174, Revision 2, states that the defense-in-depth philosophy consists of a number of considerations, and consistency with the defense-in-depth philosophy is maintained if the following occurs:

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.

- Over-reliance on programmatic activities as compensatory measures associated with the change in the licensing basis is avoided.

- System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers).

- Defenses against potential common-cause failures are preserved, and the potential for the introduction of new common-cause failure mechanisms is assessed.

- Independence of barriers is not degraded.

- Defenses against human errors are preserved.

- The intent of the plant's design criteria is maintained.

In its letter dated April 21, 2017, SNC provided a discussion of how its risk-informed assessment was consistent with the philosophy of defense-in-depth by addressing each of the seven considerations above. Items associated with defense-in-depth that were included in Section 2.1 of Enclosure 4 of the licensee's letter are evaluated in Sections 3.2.1 to 3.2.7. Section 3.2.8, "Additional Defense-in-Depth Considerations" evaluates additional defense-in-depth considerations specific to the debris issue which were also provided in Enclosure 4 of the licensee's submittal.

3.2.1 A Reasonable Balance is Preserved Among Prevention of Core Damage, Prevention of Containment Failure, and Consequence Mitigation

The licensee stated that it performed various physical and procedural changes in response to the concerns raised in GSI-191. Some of these modifications included the installation of new strainers with increased surface areas and a reduced opening size, increased RWST inventory, removal of problematic insulation materials, procedural changes to delay isolation of RHR pumps from the RWST, and program controls to ensure the debris load limits are not exceeded. The licensee also stated it plans additional changes such as modifying the height of the RHR strainers and sump recirculation initiation logic, and that implementing these changes will reduce the risk associated with the effects of LOCA-generated debris as evidenced by its risk-informed analyses.

The NRC staff reviewed the licensee's rationale and concludes that a reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation because of the following:

1. There is a robust plant design to survive hazards and minimize challenges that could result in the occurrence of an event, and the change to adopt a risk-informed approach for assessing the effects of debris does not increase the likelihood of initiating events or create new significant initiating events;

2. Prevention measures are in place if an event occurs with adequate availability and reliability of structures, systems, and components (SSCs), providing the safety functions that prevent plant challenges from progressing to core damage;

3. Existing measures are in place to contain a source term if a severe accident occurs, since the change does not impact the containment function or SSCs supporting that function such as containment sprays; and

4. The change does not reduce the effectiveness of the emergency preparedness program, including the ability to detect and measure releases of radioactivity, notify offsite agencies and the public, and shelter or evacuate the public as necessary.

3.2.2 Over-Reliance on Programmatic Activities as Compensatory Measures Associated with the Change in the Licensing Basis is Avoided

This defense-in-depth consideration evaluates if programmatic activities are substituted for design features to an extent that significantly reduces the reliability and availability of design features to perform their safety functions. The licensee identified that the change would not adversely affect any of the programmatic activities already in place at Vogtle, such as the

inservice inspection (ISI) program, plant personnel training, RCS leakage detection program, or containment cleanliness inspection activities. Additionally, the licensee summarized its previously implemented programmatic activities in Enclosure 5, Section 3.i.: No additional significant activities are associated with implementation of the risk-informed approach. The NRC staff reviewed the licensee's description of programmatic activities and concludes that this defense-in-depth consideration is met because the proposed change does not affect how safety functions are performed, nor does it reduce the reliability or availability of the SSCs that perform those functions. Existing programmatic activities are maintained, and, therefore, there is not an excessive reliance on programmatic activities as compensatory measures related to the risk-informed approach.

3.2.3 System Redundancy, Independence, and Diversity are Preserved, Commensurate with the Expected Frequency, Consequences of Challenges to the System, and Uncertainties

The licensee stated that the planned modifications do not change the redundancy, independence, and diversity of the ECCS or CSS. The licensee analyzed the ECCS and CSS relative to their contribution to nuclear safety using its plant-specific PRA for Vogtle. In addition, the licensee evaluated the risk attributable to debris for the full spectrum of LOCA events. As part of its risk assessment, key assumptions and sources of uncertainty were identified and assessed for impact on the application. The NRC staff reviewed the licensee's evaluation of this defense-in-depth consideration and concludes that it is met because the risk-informed analysis is consistent with the assumptions in the safety analysis for Vogtle and does not significantly increase the expected frequency of challenges to the system, or consequences of failure of the system functions as a result of a decrease in redundancy, independence, or diversity. The licensee performed a comprehensive risk assessment and demonstrated any reduction in redundancy, independence, or diversity of systems does not result in a significant increase in risk, as evidenced by the large margin to RG 1.174 risk acceptance guidelines, including sensitivity cases assessed as part of uncertainty analysis. (see Section 3.4.2.10, "Systematic Risk Assessment" and Section 3.4.2.11, "Sensitivity and Uncertainty Analyses" of SE).

3.2.4 Defenses Against Potential Common-Cause Failures are Preserved, and the Potential for the Introduction of New Common-Cause Failure Mechanisms is Assessed

The licensee stated that the potential for new common-cause failure (CCF) mechanisms has been assessed for the GSI-191 issue. Specifically, the primary failure mechanisms include clogging of the sump strainers or reactor core. The defenses against these clogging mechanisms are not adversely affected by the physical and procedural changes. Additionally, the new risk-informed approach does not introduce any new CCFs or reduce the current plant defenses against CCFs. The NRC staff reviewed the evaluation of this defense-in-depth consideration and concludes that it is met because the risk-informed evaluation does not introduce a new potential CCF cause or event for which a defense is not in place; does not increase the probability or frequency of a cause or event that could cause simultaneous multiple component failures; does not introduce a new coupling factor for which a defense is not in place; or does not weaken or defeat an existing defense against a cause, event, or coupling factor.

3.2.5 Independence of Barriers is Not Degraded

The three barriers to a radioactive release are the fuel cladding, RCS pressure boundary, and reactor containment building. The licensee stated that in its evaluation of a LOCA, the RCS barrier is postulated to be breached, and the proposed change does not affect the design and analysis requirements for the fuel. The licensee also stated the post-LOCA recirculation function is provided by the ECCS located inside the auxiliary building. During the recirculation phase, the RHR pumps take suction from the containment recirculation sumps and supply flow back to the reactor directly and/or through the centrifugal charging pumps (CCP) and SI pumps. The pumps, system piping, and other components on the recirculation flow path serve as the barrier to release. The auxiliary building has a dedicated ventilation system to control airborne radioactivity during emergency conditions, and the building is capable of handling recirculating water leakage. The licensee also stated its analysis confirms that assuming a single failure that results in the loss of one air cooling train and one CS train, the containment fan coolers and the CS system can remove sufficient thermal energy from the containment atmosphere following a LOCA or main steam line break (MSLB) to maintain the peak containment pressure below design values. Based on this rationale, the licensee concluded that the independence of the barriers is maintained and not degraded by implementation of the proposed methodology. The NRC staff reviewed the licensee's evaluation of this defense-in-depth consideration and concludes that it is met because implementation of the methodology does not result in a significant increase in the frequency of existing challenges to the integrity of the barriers or in the failure probability of any individual barrier. Moreover, implementation of the methodology does not introduce new or additional failure dependencies among barriers that significantly increase the likelihood of failure.

3.2.6 Defenses Against Human Errors are Preserved

This consideration evaluates if implementation of the proposed methodology significantly increases the potential for or creates new human errors that might adversely impact one or more layers of defense. The licensee stated that the use of the risk-informed methodology in the GSI-191 analysis does not impose any additional operator actions or increase the complexity of existing operator actions. Thus, the defenses that are already in place with respect to human errors are not impacted by the proposed change. The NRC staff reviewed the evaluation of this defense-in-depth consideration and concludes that it is met because the implementation of the proposed methodology does not reduce the ability of plant staff to perform actions. Specifically, the methodology does not create new human actions that are important to preserving any of the layers of defense for which a high reliability cannot be demonstrated, or significantly increase the probability of existing human errors by affecting performance shaping factors, including mental and physical demands and level of training.

3.2.7 The Intent of the Plant's Design Criteria is Maintained

The licensee described a risk-informed approach for determining the design requirements to address the effects of LOCA-generated debris on ECCS and CSS recirculation functions. The licensee stated that if implemented, the risk-informed approach would replace the existing deterministic approach described in the Vogtle licensing basis, and consequently, would require amendments to the Vogtle, Units 1 and 2, operating licenses to incorporate the revised methodology per the requirements of 10 CFR 50.59. The licensee stated that the proposed amendments to the operating licenses will be described in future LARs. In addition, the licensee stated that exemptions to the overall requirements associated with 10 CFR 50.46(a)(1), GDC 35, GDC 38, and GDC 41, will be provided in future submittals.

The NRC staff reviewed the licensee's evaluation of this defense-in-depth consideration and concludes that the proposed change maintains the intent of the plant's design criteria, because an alternate risk-informed evaluation method provides an acceptable approach that demonstrates LTCC will be maintained following a LOCA, and, therefore, does not result in a reduction in the effectiveness of one or more layers of defense.

3.2.8 Additional Defense-in-Depth Considerations

To augment the discussion about the seven defense-in-depth elements in RG 1.174, the licensee provided additional information about SSCs, plant programs, and design features that support the defense-in-depth philosophy by minimizing the likelihood and consequences of a LOCA, and by ensuring adequate containment performance, even during events where debris is generated.

3.2.8.1 Detecting and Mitigating Adverse Conditions

Adequate defense-in-depth is maintained by ensuring the capability exists for operators to detect and mitigate adverse conditions due to potential impacts of debris blockage such as inadequate flow through the strainers and/or through the reactor core. The licensee's primary means to delay and prevent strainer blockage is to monitor and reduce the flow through the sump strainers, as necessary, and control debris sources inside containment. The licensee has specific measures in place to do this, including emergency operating procedures (EOPs), technical specifications (TSs), and other normal procedures. In addition, the licensee has several measures in place to control the debris sources inside the containment buildings (e.g., awareness training for personnel on containment cleanliness requirements and procedures for containment exit inspections, including those for containment entry and emergency sump inspection procedures).

The licensee has methods in place to detect sump strainer blockage (e.g., indications in the control room for SI, RHR, and CS pump flows and SI and CS pump discharge pressures, core exit thermocouple temperature, and reactor vessel level indications in the control room).

The licensee also has multiple methods to mitigate an inadequate recirculation flow condition caused by the accumulation of debris on the sump strainer (e.g., EOPs to reduce flow and stop pumps taking suction from a clogged sump strainer; procedures to initiate makeup to the RWST from, for example, the spent fuel pool; and diverse and flexible coping strategies to maintain fuel cooling and containment integrity).

The licensee stated that the actions for reducing or controlling flow through the emergency sump strainers during the recirculation phase can have a similar, positive impact on reducing the potential for fuel blockage. The plant design and procedures call for simultaneous hot-leg and cold-leg injection during the recirculation phase. Initially, all ECCS pumps are aligned for cold-leg injection. Then at 7.5 hours after the initiating event, switchover to simultaneous hot/cold-leg injection is made. For this configuration, the RHR and SI pumps provide cooling water through the hot-leg, while the charging pump continues injecting coolant through the cold-leg. The licensee stated that it expects, with most of the injection into the hot-leg, that the flow will reverse and debris at the core inlet will be removed.

The licensee has multiple methods for detection of a core blockage condition as manifested by inadequate RCS inventory or inadequate RCS and core heat removal conditions. The primary

methods include core exit thermocouple temperature indication and reactor water level, as monitored by the reactor vessel level instrumentation system. An additional method for detection of a core blockage condition includes monitoring of containment radiation levels.

The licensee also has multiple proceduralized methods to mitigate an inadequate reactor core flow condition. Upon identification of such a condition, the EOPs direct the operators to take actions to restore cooling flow to the RCS, including: (1) reestablish SI flow to the RCS, (2) reduce RCS pressure by performing rapid secondary depressurization, and (3) restart reactor coolant pumps (RCPs) and open pressurizer power-operated relief valves (PORVs). The licensee has procedures to protect fission product boundaries and return the plant to a controlled stable condition when the EOPs are no longer effective in controlling the event. Cooling can be provided to the reactor core using the flow paths established by the flexible coping strategies or by reinitiating injection through a refilled RWST.

3.2.8.2 Barriers for Release of Radioactivity

The licensee stated for the initial phase of accident mitigation that the proposed licensing basis change for the use of a risk-informed approach described in the technical report to evaluate the effects of debris does not alter the fuel cladding limits or previous analyses and testing programs that demonstrate the acceptability of ECCS.

The licensee stated that the integrity of the RCS pressure boundary is assumed to be compromised for the GSI-191 sump performance evaluation. However, the proposed change does not modify the previous analyses or testing programs that demonstrate the integrity of the RCS. The licensee stated that additional measures are in place to prevent and detect pipe breaks, including the ISI program, RCS overpressure protection, leak detection program, and operator actions outlined in the severe accident mitigation guidelines procedures.

The licensee stated that the containment buildings are designed such that for all break sizes, up to and including, a double-ended guillotine break of an RCS pipe or secondary system pipe, the containment peak pressure is below the design pressure with adequate margin. Therefore, the containment buildings remain a low leakage barrier against the release of fission products for the duration of the postulated LOCAs. The evaluation of post-LOCA debris effects using a risk-informed approach is not part of the analyses that demonstrate containment integrity. The proposed change does not affect the methodology, acceptance criteria, or conclusion of the existing analyses. Therefore, the reactor containment integrity is not affected.

3.2.8.3 Emergency Plan Actions

The licensee stated that the proposal to use the methodology of a risk-informed approach does not involve any changes to the emergency plan. There is no change to the strategies for preventing core damage and containment failure or for consequence mitigation. The licensee also stated that the use of the risk-informed approach does not impose any additional operator actions or complexity, and that implementation of the proposed change would not result in any changes to the response requirements for emergency response personnel during an accident.

3.2.8.4 NRC Staff Review of Additional Defense-in-Depth

The NRC staff reviewed the licensee's additional defense-in-depth actions and programs discussed above and concludes that the licensee's measures to detect and mitigate adverse conditions, barriers for release of radioactivity, and emergency plan actions to address GL 2004-02 provide additional defense-in-depth measures beyond the seven factors defined in RG 1.174, Revision 2.

3.2.9 NRC Staff Conclusion Regarding Key Principle 2: Defense-in-Depth

The NRC staff finds that the defense-in-depth philosophy is maintained under the analysis described in the technical report because the licensee has appropriately addressed each of the seven factors in Section 2.1.1 of RG 1.174, Revision 2, and provided the additional information as discussed above.

3.3 Key Principle 3: The Proposed Change Maintains Sufficient Safety Margins

RG 1.174, Revision 2, states that safety margins are maintained when codes and standards or their alternatives approved for use by the NRC are met and when the safety analysis acceptance criteria in the licensing basis (e.g., UFSAR, supporting analyses) are met, or proposed revisions provide sufficient margin to account for analysis and data uncertainty.

The licensee stated that there are many conservatisms included in the risk-informed evaluation that provide high confidence that the scenarios considered to be mitigated are successful. The licensee provided a list of conservatisms that are considered safety margin in Table 4-1 of Enclosure 4 of the letter dated April 21, 2017. Many of the conservatisms are credited as a result of using NRC staff guidance to perform its evaluation.

The licensee identified margins and conservatisms in the design, analysis, construction, and operation of the plant to show that the proposed risk-informed approach will maintain sufficient safety margins. Specifically, the licensee identified applicable codes and standards (or the pertinent NRC-approved alternatives), explained how Vogtle complies with the codes, and discussed how proposed revisions provide sufficient margin to account for analysis and data uncertainty.

For this section, all descriptions attributed to the licensee's submittal are taken from SNC's letter dated April 21, 2017, unless otherwise noted. The information is primarily from Enclosures 3 and 4 of the letter. Unless otherwise noted, the RAIs and responses to the RAIs discussed in this section are documented the licensee's letter dated November 9, 2017 (Reference 3).

3.3.1 Fabrication, Design, and Construction

In the Vogtle UFSAR Section 3.1.2, the licensee stated that the RCS pressure boundary is designed to accommodate the system pressures and temperatures attained under the expected modes of plant operation, including anticipated transients, with stresses within applicable limits. Consideration is given to loadings under normal operating conditions and to abnormal loadings such as pipe rupture and seismic loadings. The piping is protected from overpressure by means of pressure-relieving devices, as required by ASME Section III. The licensee further stated that reactor coolant chemistry is controlled to protect the materials of construction of the reactor coolant pressure boundary from corrosion.

In UFSAR Section 3.1.4, the licensee stated that all RCS components are designed, fabricated, inspected, and tested in conformance with the ASME Code, Section III. The licensee stated that Vogtle has RCS leakage detection systems.

In UFSAR Section 3.1.4, the licensee stated that close control is maintained over material selection and fabrication for the RCS to assure that the boundary behaves in a non-brittle manner. The RCS materials, which are exposed to the coolant, are corrosion-resistant stainless steel or Inconel reactor coolant pressure boundary materials, and fabrication techniques are such that there is a low probability of gross rupture or significant leakage.

In UFSAR Section 3.1.4, the licensee stated that the RCS pressure boundary is periodically inspected in accordance with the ASME Code, Section XI. The licensee stated that the RCS pressure boundary welds are accessible for ISI to assess their structural and leaktight integrity. Nevertheless, based on operating experience, the NRC staff recognizes that RCS piping is fabricated with metal that will naturally degrade with time because of the interactions of materials, environment, and stresses. The NRC staff notes that the welds in RCS piping generally have a higher probability of degradation than the pipe segments because of inherent weld residual stresses, stresses from applied loads, and the potential for fabrication defects. The RCS pipe segments are joined either by similar metal welds or dissimilar metal welds. Similar metal welds are fabricated with stainless steel filler metal. Dissimilar metal welds are fabricated with nickel-based Alloy 82/182 filler metal.

The NRC staff notes that operating experience has shown that stainless steel welds in the PWR environment are less susceptible to degradation mechanisms. However, operating experience of PWRs has shown that dissimilar metal butt welds that are fabricated with nickel-based Alloy 82/182 filler metal are susceptible to primary water stress corrosion cracking (PWSCC) in the PWR environment. As such, Alloy 82/182 welds may have higher probability of crack initiation and growth than the stainless steel welds in PWRs. The licensee's GSI-191 analysis evaluates the structural integrity of the RCS piping in terms of pipe rupture probability.

3.3.2 ASME Code Class 1 ISI and Testing Program

In Table 3-9 of Enclosure 3 to the letter dated April 21, 2017, the licensee provided a list of information for all ASME Code Class 1 welds that are considered in the GSI-191 analysis. In the letter dated November 9, 2017, the licensee identified the ASME Code, Section XI, Category B-F welds (i.e., pressure retaining dissimilar metal welds in vessel nozzles) nominal pipe size (NPS) 4 inches or larger that are fabricated with Alloy 82/182 filler metal in the hot-leg piping. The licensee stated that it has mitigated the potential for PWSCC in the B-F welds in the hot-leg piping near the reactor pressure vessel by application of mechanical devices that reverse the stress fields. In addition, stainless steel safe ends have been used to eliminate susceptibility to PWSCC for the B-F welds in the hot-legs and cold-legs at the steam generators.

The NRC staff notes that the Alloy 82/182 dissimilar metal welds in the hot-leg piping have been mitigated with the mechanical stress improvement process that reduces crack initiation and growth. In addition, the NRC staff notes that stainless steel safe ends are used in the hot-leg and cold-leg at the steam generator nozzles. The safe ends fabricated with stainless steel are not likely to crack.

The licensee stated that all Alloy 82/182 dissimilar metal welds in piping attached to the pressurizer (including the surge line and pressurizer spray and relief valve piping) have been mitigated from PWSCC by application of a full structural weld overlay. The full structural weld overlay also provides favorable stresses at the inside diameter of the pipe to minimize crack initiation. The NRC staff notes that the full structural weld overlay will reduce the probability of rupture in the pipes attached to the pressurizer.

The licensee stated that the only B-F welds NPS 4 inches or greater that have not been mitigated for PWSCC are the four following cold-leg welds near the reactor pressure vessel: weld 11201-V6-001-W35-R8, weld 11201-V6-001-W38-R8, weld 11201-V6-001-W34-R8, and weld 11201-V6-001-W39-R8. In its letter dated November 9, 2017, the licensee stated that the degradation mechanisms occurred in a weld directly affect the likelihood of a break on that specific weld (i.e., the LOCA frequency at that location). However, the failures associated with the GSI-191 phenomena shown in Table 3-9 of Enclosure 3 to the licensee's letter dated April 21, 2017, are based on the quantity of debris generated for breaks at each weld location (i.e., the analysis assumed the breaks occur without weighting the break frequency according to degradation mechanism). The licensee stated that the four cold-leg welds that have not been mitigated for PWSCC are in the reactor cavity and generate significantly less debris than other break locations. The debris quantities generated at these four locations are sufficiently low that none of these breaks could result in exceeding any of the GSI-191 acceptance criteria. Furthermore, the licensee conducted a sensitivity analysis and determined that weighting break frequency according to degradation mechanism has a negligible effect on the risk attributable to debris (see Sections 3.4.2.2, "Initiating Event Frequencies" and 3.4.2.10, "Systematic Risk Assessment" of this SE).

The NRC staff notes that the four cold-leg welds that are fabricated with Alloy 82/182 filler metal are required to be examined in accordance with 10 CFR 50.55a(g)(6)(ii)(F), which requires licensees of PWRs to implement the requirements of ASME Code Case N-770-2, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated With UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities." ASME Code Case N-770-2, as conditioned in 10 CFR 50.55a(g)(6)(ii)(F), requires the licensee to perform more frequent and rigorous examination of the unmitigated Alloy 82/182 welds than that of the ASME Code, Section XI. Although not mitigated, the NRC staff finds that with frequent examinations, the failure probability of these four cold-leg welds will be reduced. In addition, as the licensee stated above, the debris quantities generated at these four cold-leg weld locations are sufficiently low that none of these breaks exceed any of the GSI-191 acceptance criteria. The NRC staff finds that although these four cold-leg welds are not mitigated, any breaks at these weld locations do not result in exceeding the GSI-191 acceptance criteria. Therefore, the NRC staff finds the licensee's treatment of these four cold-leg welds in the GSI-191 analysis is acceptable.

In Section 2.3.2 of Enclosure 4 of the letter dated April 21, 2017, the licensee stated that the ISI program provides rules for the examination and repair of piping and other RCS components, and plays an important role in the prevention of pipe breaks. In its letter dated November 9, 2017, the licensee further stated that the integrity of the ASME Class 1 welds, piping, and components are maintained at a high level of reliability through the ISI program in accordance with the 2010 Edition of the ASME Code, Section XI, Table IWB-2500-1. Vogtle ISI procedures also ensure that inspections are performed in accordance with the schedule requirements of the ASME Code, Section XI. The licensee reported that no flaws have been detected in any of the unmitigated Alloy 82/182 welds. The NRC staff notes that the licensee has performed periodic examinations of all welds in RCS piping in accordance with specific edition and addenda of the

ASME Code, Section XI, and augmented examinations, as required by 10 CFR 50.55a. The NRC staff recognizes that no flaws have been detected in any of the unmitigated Alloy 82/182 welds. The NRC staff notes that there is sufficient safety margin in the structural integrity of unmitigated welds because the unmitigated Alloy 82/182 welds have so far maintained their structural integrity.

In summary, the NRC staff finds that the RCS piping has maintained sufficient safety margin to minimize the potential for RCS piping rupture because (1) the licensee has performed the required examinations of RCS piping in accordance with the ASME Code, Section XI, and (2) for the Alloy 82/182 dissimilar metal welds in the RCS piping, the licensee has either mitigated the welds or followed the required augmented examinations per 10 CFR 50.55a(g)(6)(ii)(F).

One of the measures to reduce the probability for clogging of the containment building sumps by pipe insulation debris is the early detection of leakage from RCS piping. An RCS pipe failure could result from either a sudden pipe rupture or a slow leak through a crack in the pipe wall thickness. If small leakage can be detected early, the licensee can take corrective actions to avoid catastrophic pipe ruptures. The Vogtle RCS leakage detection systems are described in UFSAR Section 5.2.5, and the detection systems have not been changed or enhanced since UFSAR Section 5.2.5 was last updated in April 2015.

The RCS leakage is classified as either identified or unidentified. Identified leakage such as pump seal, valve packing leakage, or in-line valve leakage is directed to specific drain tanks. The identified leakage will not affect analysis of GSI-191 because the leakage will be drained, identified, and monitored. Identified leakage is not related to pipe ruptures. GSI-191 involves leakage from piping that would be considered as unidentified leakage. Therefore, regarding GSI-191, the NRC staff focuses its evaluation of the RCS leakage detection systems on the detection capability of unidentified leakage.

RCS unidentified leakage can be monitored by the sump level monitoring system, the airborne particulate radioactivity monitoring systems, and the containment fan cooler condensate measuring system. In addition to the above systems, the humidity, temperature, pressure, and radioactive gas monitors provide indirect indications of leakage to the containment.

The licensee stated that the sensitivity and response time of the detection equipment for unidentified leakage is such that a leakage rate or its equivalent of 1 gallon per minute (gpm) can be detected in approximately 1 hour, as specified in RG 1.45, Revision 1, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage," dated May 2008 (Reference 41).

The licensee stated that both the containment airborne particulate radioactivity monitoring system and the containment gaseous radioactivity monitoring system can detect a leak of 1 gpm in approximately 1 hour. The other systems provide additional means of monitoring leakage within the containment building.

The NRC staff notes that the ASME Code Class 1 RCS piping considered in the debris generation analysis is fabricated or mitigated with material that is resistant to cracking such that catastrophic pipe breaks would not likely occur. If cracking does occur, the RCS leakage detection systems will be able to detect leakage, and the operator will take corrective actions in accordance with the requirements of Vogtle TS Section 3.4.13. The NRC staff determines that the subject piping maintains defense-in-depth and safety margin because it satisfies the regulations of 10 CFR 50.55a; GDC 1, "Quality Standards and Records"; GDC 14, "Reactor Coolant Pressure Boundary"; GDC 30, "Quality of Reactor Coolant Pressure Boundary";

GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary"; and GDC 32, "Inspection of Reactor Coolant Pressure Boundary." Therefore, the NRC staff concludes that the piping considered in the debris generation analysis maintains sufficient safety margin to minimize the potential for a large break that would significantly affect the containment sump performance.

3.3.3 Debris Threshold Methodology

The methodology used by the licensee to establish the debris limits for the strainer contains significant conservatism. This conservatism is included in deterministic guidance and methodologies to ensure that uncertainties are accounted for in the analysis. The margins in the evaluation help ensure that scenarios calculated to be successful would result in adequate long-term core cooling (LTCC) with a high level of reliability. The licensee used guidance in RG 1.82, Revision 4 (Reference 34), NEI 04-07 (Reference 26), the associated NRC staff SE (Reference 24), and the March 2008 NRC staff review guidance regarding GL 2004-02 closure (Reference 32). Guidance was issued in the area of strainer head loss and vortexing, coatings, and chemical effects. These guidance documents contain significant conservatism. As is discussed in specific sections of this SE on these topics, the licensee implemented the guidance correctly or used acceptable alternative methods. In addition, the licensee's debris limits are based on head loss tests that resulted in head loss much lower than the net positive suction head (NPSH) margins available. The NRC staff notes that additional debris could be deposited on the strainers before NPSH margins would be reduced significantly.

3.3.4 Debris Generation and Transport Methodology

The guidance for debris generation and transport has been developed to ensure that the debris load predicted to reach the strainer is maximized, considering plant-specific conditions. The margins built into the guidance are intended to assure that the systems will function with a high level of reliability and appropriately consider plant-specific information. The licensee used the approved guidance to perform its debris generation and transport analyses. Therefore, sufficient margins are maintained in these portions of the evaluation.

3.3.5 Strainer Testing Methodology

Strainer testing guidance has been developed to ensure that head losses predicted from testing are reasonably assured to represent the most limiting values for the plant conditions being tested. The guidance also directs that the application of the test results be performed conservatively. The licensee's test program used the maximum chemical debris loads that could be generated by any break when establishing the fiber limit used to determine the risk associated with debris. Relatively high amounts of particulates were also included in the testing.

3.3.6 Plant-Specific Conservatisms

The licensee also incorporated conservatism in its thermal hydraulic analysis by crediting no containment pressure for NPSH calculations and minimal containment pressure for flashing and deaeration calculations. In a realistic calculation, additional pressure may have been credited. All gas voids present in the strainer are assumed to transport to the pump suction without compression. Because there is a large head above the pump suctions, the void fraction at the pumps is over-predicted, which results in a conservative NPSH margin calculation.

Several of the licensee's modeling assumptions were conservative as well. For example, design-basis containment temperatures were used for all break sizes. Even though higher temperatures have both conservative and nonconservative (competing) effects, assuming a higher temperature results in larger risk than using realistic temperatures. In addition, the licensee assumed that 100 percent of epoxy coatings would fail as particulate, even though it is likely that some would not fail and some would fail in pieces too large to transport.

3.3.7 NRC Staff Conclusion Regarding Key Principle 3: Safety Margins

The NRC staff concludes that the proposed approach maintains sufficient safety margin and that the licensee's evaluation assured that the analysis results in a conservative prediction of risk associated with the impact of debris on LTCC.

3.4 Key Principle 4: When Proposed Changes Result in an Increase in Risk, the Increases Should be Small and Consistent with the Intent of the Commission's Safety Goal Policy Statement

This section discusses the licensee's base PRA model for Vogtle, including the calculated total risk values (core damage frequency (CDF) and large early release frequency (LERF)) for each unit and the licensee's risk-informed assessment of debris. A review of this information was necessary in order to determine whether the risk attributable to debris is small and consistent with the Commission's Safety Goal Policy Statement.

3.4.1 Acceptability of the Base PRA Model

Regulatory Position 2.3 of RG 1.174, Revision 2 (Reference 20), states, in part, that the scope, level of detail, and technical adequacy [technical elements] of the PRA are to be commensurate with the application for which it is intended and the role the PRA results play in the integrated decision process.

The acceptability of the PRA is commensurate with the safety implications of the change being requested and the role that the PRA plays in justifying that change. That is, the more the potential change in risk or the greater the uncertainty in that risk from the requested change, or both, the more rigor is placed into ensuring the acceptability of the PRA.

The objective of the NRC staff's review of the Vogtle base PRA model was to determine whether the PRA used in evaluating the risk attributable to debris was of sufficient scope, level of detail, technical elements, and plant representation for this application. The licensee provided information regarding Vogtle PRA acceptability in Enclosure 1 of the letter dated April 21, 2017; letter dated July 11, 2017; and letters dated February 6, February 21, and May 23, 2018. The NRC staff's review of this information focused on the ability of the licensee's PRA model to evaluate the risk of debris in containment.

The licensee used the Vogtle base PRA in the risk analysis to (1) identify accidents requiring recirculation through the ECCS strainers, (2) identify high-likelihood equipment configurations, (3) quantify ΔCDF and ΔLERF associated with GSI-191 failures for the as-built/as-operated plant, (4) provide initiating event frequencies for unlikely scenarios to compute a bounding CDF and LERF for low-likelihood configurations, (5) provide initiating event frequencies for high-likelihood scenarios to perform sensitivity analyses, and (6) provide a basis for estimating a conditional large-early release probability (CLERP), computed as the ratio of the base-case LERF to the base-case CDF (i.e., the LERF to CDF ratio for the case with no

GSI-191 failures). The licensee used CLERP to calculate LERF for bounding GSI-191 risk from low-likelihood configurations and for sensitivity analyses. For sensitivity analyses, the licensee did not exercise the PRA to compute changes to the CDF, but relied on simplified computations that closely approximated outputs of the PRA model.

3.4.1.1 Scope of the Base PRA (Modes/Hazards)

Regulatory Position 2.3.1 in RG 1.174, Revision 2, states that the scope of a PRA is defined in terms of the causes of initiating events and the plant operating modes it addresses. The causes of initiating events are classified into hazard groups, or groups of similar hazards that are assessed in a PRA using common approaches, methods, and likelihood data for characterizing the effect on the plant. Although all plant operating modes and hazard groups should be addressed, a qualitative treatment of some modes and hazard groups may be sufficient when the licensee can demonstrate that their risk contributions would not affect the decision. However, when the risk associated with a particular hazard group or operating mode would affect the decision being made, it is the Commission's policy that, if a staff-endorsed PRA standard exists for that hazard group or operating mode, the risk will be assessed using a PRA that meets that standard (Reference 42).

The licensee stated in Section 2 of Enclosure 1 of the letter dated April 21, 2017:

> Version 5 of the VEGP Units 1 and 2 Internal Events PRA model is used for the GSI-191 risk assessment. The internal events PRA model is an at-power model (i.e., it addresses Modes 1 and 2 of reactor operation). The model includes both CDF and LERF from internal events, including internal flooding.
>
> Version 2 of the VEGP Units 1 and 2 seismic PRA model is used for the assessment of GSI-191 risk from seismically-induced LOCAs. These versions are part of the current VEGP PRA model of record at the time of this analysis.

The licensee also stated that the quantitative risk assessment was performed at full power operation since it assumed full power operation to be equivalent or bounding, compared to the other operating modes. The licensee clarified that at full power, RCS pressure and temperature (key inputs affecting the zone of influence (ZOI)[2] size) are either approximately the same or significantly lower for Modes 2 through 6 and that the flow rate required to cool the core (a key input affecting core blockage) is significantly reduced for low power or shutdown modes.

The licensee described its qualitative evaluation of external hazards other than seismic in the letter dated May 23, 2018 (Reference 9). The licensee stated that fire hazards would not result in pipe breaks, but could cause spurious operations leading to LOCAs with the potential to generate debris in containment. Specifically, the licensee considered fire-induced spurious opening of a pressurizer power-operated relief valve (PORV) or safety valve, spurious reactor head vent discharge, continuous letdown, spurious interfacing system LOCA, and a reactor cooland pump (RCP) seal LOCA due to loss of seal cooling. Of these scenarios, only an RCP seal LOCA was considered to have the potential to generate debris inside containment; however, the licensee determined that the quantity of debris generated would be equivalent to

[2] The ZOI represents the zone or volume in space where a two-phase jet from a high-energy line break can generate debris that may be transported to the sump. The size of the ZOI is defined in terms of pipe diameters and is determined based on the system pressure and the destruction pressure of the insulation impacted by the jet. Higher system pressures result in increased ZOIs. Robust materials have smaller ZOIs than fragile materials.

that generated by small or medium LOCAs, which were determined not to challenge sump strainers. The licensee stated that the following external hazards are also applicable to Vogtle: aircraft impact; extreme winds and tornadoes; external flooding, including intense local precipitation; industrial and military facility accidents; pipeline accidents; transportation accidents; and turbine-generated missiles. The licensee stated that none of these external hazards has the potential to generate debris inside containment.

The NRC staff reviewed the licensee's information regarding the scope of its base PRA and concludes that the risks associated with hazards and operating modes that would affect this application were evaluated using a PRA that meets the applicable PRA standard. Specifically, the NRC staff reviewed the licensee's assessment regarding the scope of the PRA used to support this application and concludes that (1) the at-power risk bounds the shutdown risk of debris because debris ZOI is either approximately the same, or significantly higher, at full power RCS pressure and temperature, and the flow rate required to cool the core is significantly reduced for low power or shutdown modes; and (2) the use of internal events, including internal floods and seismic PRA models, is adequate because the risk contribution from other external hazards does not affect the evaluation of the risk attributable to debris.

Because seismic is the only external hazard that affects this evaluation, the NRC staff reviewed the technical acceptability of the Vogtle internal events, including internal floods and seismic PRAs.

3.4.1.2 Level of Detail of the Base PRA

Regulatory Position 2.3.2 in RG 1.174, Revision 2, states that the level of detail required of the PRA is that which is sufficient to model the impact of the proposed change. The characterization of the problem should include establishing a cause-effect relationship to identify portions of the PRA affected by the issue being evaluated.

The licensee described its PRA models as highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. As described in more detail in Section 3.4.2.4, "Changes to the Base PRA Model," the base PRA model contained sufficient detail such that only "a few relatively minor changes were required for the base PRA model to incorporate the events for the GSI-191 sump strainer and core blockage failures, along with the associated LOCA initiating events and equipment configurations."

The NRC staff reviewed the licensee's description of its base PRA and the cause-effect relationship to identify portions of the PRA affected by the issue being evaluated and concludes that the level of detail of the licensee's base PRA is sufficient to evaluate the risk attributable to debris from GSI-191 sump strainer and core blockage failures, along with the associated LOCA initiating events and equipment configurations.

3.4.1.3 Base PRA Technical Elements

RG 1.200, Revision 2 (Reference 35), describes one approach for determining whether the PRA, in total, or the parts that are used to support an application, is acceptable such that the PRA can be used in regulatory decisionmaking for light-water reactors. RG 1.200, Revision 2, endorses, with comments and qualifications, the use of the ASME/ANS 2009 PRA Standard; NEI 00-02, Revision 1, "Probabilistic Risk Assessment Peer review Process Guidance" (Reference 43) and; and NEI 05-04, Revision 2, "Process for Performing Follow-On PRA Peer reviews Using the ASME PRA Standard," (Reference 44).

RG 1.200, Revision 2, states, in part:

> When used in support of an application, this regulatory guide will obviate the need for an in-depth review of the base PRA by NRC reviewers, allowing them to focus their review on key assumptions and areas identified by peer reviewers as being of concern and relevant to the application.

Therefore, the NRC staff relied on the peer review findings and reviewed the key assumptions in the licensee's PRA in its determination of the acceptability of the technical elements of the base PRA model. The ASME/ANS PRA Standard provides technical supporting requirements (SRs) in terms of three capability categories (CCs). The intent of the delineation of the capability categories within the SRs is generally that the degree of scope and level of detail, the degree of plant specificity, and the degree of realism increase from CC-I to CC-III. In general, the staff anticipates that current good practice (i.e., CC-II of the ASME/ANS Standard) is adequate for the majority of applications. Consistent with the guidance in RG 1.200, Revision 2, and RG 1.174, Revision 2, for this application of the Vogtle PRA to assess the risk associated with GSI-191-related phenomena, the NRC staff considered CC-II to be adequate.

3.4.1.3.1 Vogtle Internal Events PRA

The NRC most recently reviewed the acceptability of the Vogtle internal events PRA (IEPRA) in support of the adoption of the risk-informed TS Initiative 4b at Vogtle (Reference 45). In the letter dated April 21, 2017, the licensee summarized the NRC's review of its IEPRA model as follows:

> The VEGP internal events PRA model (including flooding) was reviewed (along with the fire PRA model) to determine the technical capability for use in supporting the LAR to implement NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines.

The review demonstrated and documented that the Vogtle at power IEPRA model (including flooding) and the fire PRA model conform to the PRA standard at CC-II, which satisfies the guidance of RG 1.200, Revision 2. Since the same IEPRA model and corresponding peer review were used to support the licensee's applications related to both TS Initiative 4b and GSI-191, the NRC staff used its previous TS Initiative 4b-related IEPRA acceptability evaluation to the extent applicable for this review.

The NRC staff reviewed the finding-level Facts and Observations (F&Os) and resolutions provided by the licensee from its 2009 Vogtle IEPRA peer review, which were based on the 2007 Addenda to the PRA standard endorsed in RG 1.200, Revision 1 (Reference 46) . The NRC staff also reviewed a comparison and resolution of differences between 2007 PRA standard SRs, as endorsed by RG 1.200, Revision 1, and SRs in the 2009 PRA Standard, as endorsed by RG 1.200, Revision 2, since RG 1.200, Revision 2, is the most recent guidance.

The peer review of the licensee's IEPRA against the 2007 Addenda considered SRs associated with three F&Os as "Not-Met" (HR-G6-01, QU-D3-01, and LE-G5-01). The licensee stated that its resolution of these findings resulted in SRs HR-G6, QU-D3, and LE-G5 being met at CC-I/II/III. The NRC staff reviewed the licensee's resolution of F&Os and its associated "Not-Met" SRs, in addition to differences between SRs in 2007 and 2009 Addenda, and

concludes that the identified F&Os were adequately resolved by the licensee for this GSI-191 risk assessment, and the licensee's comparison adequately addressed the differences between the 2007 and 2009 Addenda.

The licensee described how it evaluated PRA model assumptions and sources of uncertainty in Section 14.3 of Enclosure 3 of the letter dated April 21, 2017. The licensee stated that the Vogtle IEPRA includes documentation of assumptions and sources of uncertainty, and therefore, these aspects were evaluated as part of the base PRA peer review process. The licensee evaluated pertinent PRA documentation to identify assumptions and uncertainties relevant to GSI-191 risk assessment. Table 3-18 of Enclosure 3 of the licensee's letter dated April 21, 2017, provides a listing of sources of epistemic uncertainty, related assumptions, sensitivities performed (as applicable), and application-specific dispositioning. The licensee stated it used guidance in NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," Revision 1 (Reference 47), and Electric Power Research Institute (EPRI) Report 1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments," (Reference 48) to guide its evaluation. The NRC staff reviewed the identified sources of uncertainty and related assumptions in the licensee's IEPRA pertinent to GSI-191 risk evaluation and concludes that these sources of uncertainty and assumptions were adequately addressed by the licensee because the licensee applied state-of-practice approaches to identify and evaluate application-specific impacts of assumptions and sources of uncertainty, including the performance of sensitivity analysis where applicable.

3.4.1.3.2 <u>Vogtle Seismic PRA</u>

The NRC most recently reviewed the acceptability of Version 2 of the Vogtle, Units 1 and 2, seismic PRA model (SPRA) in support of the licensee's request to incorporate the Vogtle SPRA in its previously-approved implementation of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Plants" Program (Reference 49). Since the same SPRA model and corresponding peer review were used to support the licensee's submittals related to incorporation of SPRA in the licensee's 10 CFR 50.69 and GSI-191 risk-assessment, the NRC staff used its previous 10 CFR 50.69-related SPRA acceptability evaluation to the extent applicable for this review.

In this evaluation, the Vogtle SPRA was used to assess the risk increase from GSI-191 failures from seismically-induced LOCAs. This risk contribution is approximately one order of magnitude smaller than the total risk attributable to debris calculated by the licensee (see Section 3.4.3, "NRC Staff Conclusion Regarding the Increase in Risk" of this SE). As stated above, RG 1.200, Revision 2 (Reference 35), recognizes that the PRA acceptability needed to support regulatory decisions can vary (i.e., that the scope, level of detail, and technical elements of the PRA are to be commensurate with the application for which it is intended and the role the PRA results play in the integrated decision process).

The licensee provided a summary of Vogtle SPRA peer review finding-level F&Os and their disposition, as well as sensitivity analysis performed, to address any findings documented in Attachment 3 to Enclosure 1 of the letter dated April 21, 2017. Specifically, the peer review team determined 67 SRs to be met at CC-II or higher, and 10 SRs were judged to be "Not-Met." The licensee provided the 10 SRs that were judged to be "Not-Met" and the associated finding-level F&Os (16 total).

These peer review findings were generated following a peer review against the 2013 Addenda to the PRA Standard (ASME/ANS RA-Sb-2013 (Addendum B)). However, RG1.200, Revision 2, endorses ASME/ANS PRA Standard Addendum A (ASME/ANS RA-Sa-2009), as discussed in an NRC letter to ASME (Reference 50). In the letter dated, July 11, 2017 (Reference 2), the licensee provided tables for each of the three ASME/ANS PRA Standard Part 5 major technical elements that describe how, if different, analogous Addendum A SRs have been met. The licensee also stated in the letter dated June 22, 2017 (Reference 51), that the peer review was performed using the process defined in NEI 12-13, "External Hazards PRA Peer-Review Process Guidelines" (Reference 52), with no exception.

The licensee stated that its comparison between Addendum A and Addendum B of Part 5 of the PRA Standard demonstrates that most of the SRs in Addendum B are consistent with the SRs in ASME/ANS RA-Sa-2009 Standard (Addendum A). The licensee further concluded that for the few SRs where differences are noted in the tables, the Vogtle SPRA model and documentation meet the analogous Addendum A SRs. Based on a review of the licensee's comparison, the NRC staff finds the licensee's use of Addendum B to be an acceptable alternative to the NRC-endorsed approach for this evaluation because it adequately addresses the technical elements for the development of an SPRA.

The NRC staff reviewed the licensee's SPRA F&Os and supporting information regarding the peer review process and determined additional information was needed to confirm Vogtle SPRA acceptability for use in this application (see RAIs 1, 5, and 6 in References 6 and 8 and RAIs 1, 5, and 7f in References 3 and 4). In the letters dated February 6, 2018, and February 21, 2018, the licensee provided additional information in the following areas related to its SPRA: (1) the peer review process used, (2) the key assumptions and sources of uncertainties associated with the Vogtle SPRA, and (3) the human reliability analysis (HRA) method used.

Regarding the peer review process, the NRC staff requested that the licensee describe how NRC comments on NEI 12-13 by letter dated November 16, 2012 (Reference 53), were addressed by the licensee. The licensee described how its peer review approach met the corresponding requirements in the ASME/ANS 2009 Standard, as endorsed in RG 1.200, Revision 2, related to: (1) the qualifications of the peer review team, (2) the treatment of unreviewed analysis methods, (3) the use of expert judgement, (4) clarification that CC-II of the PRA Standard was met for each SR, and (5) the use of "in-process" peer reviews. The NRC staff reviewed the licensee's response to RAI 1 (Reference 6) related to NRC comments on NEI 12-13 and determined that the licensee's peer review process for SPRA addresses its comments on NEI 12-13 for this evaluation because (1) the licensee adequately addressed qualifications and independence of the SPRA peer review team, (2) no unreviewed analysis methods were identified during the Vogtle peer review, (3) peer reviewers did not identify any use of expert judgement in the licensee's SPRA, (4) the licensee clarified that the Vogtle SPRA was reviewed against CC-II of the PRA Standard for all applicable SRs and that any SRs the reviewers found to meet only CC-I had associated finding-level F&Os, and (5) the licensee stated that the "in-process" review approach was not followed, which eliminates the need for a staff review to determine whether the expectations for such "in-process" reviews were met.

Regarding key assumptions and sources of uncertainties associated with the Vogtle SPRA for this application, the licensee determined two sets of assumptions and associated uncertainties can affect the SPRA results: (1) assumptions in the IEPRA that may affect the SPRA and (2) assumptions made specifically in the SPRA. The licensee provided a table with the results of its assessment of IEPRA assumptions that could impact SPRA results. The table included the potential impact to the SPRA and its impact on GSI-191. The licensee also identified three

assumptions that were specifically associated with the SPRA. The licensee performed a detailed review of those assumptions in the SPRA and determined that only one of these assumptions could impact the GSI-191 assessment. This assumption pertains to the apportionment of potential seismic failures to each of the LOCA categories (small, medium, and large). The licensee performed a sensitivity analysis to evaluate the effect of this uncertainty by increasing the seismic-induced large break LOCA (LBLOCA) frequency by a factor of 3, yielding an increase of 3.75×10^{-8} per year (/yr) in ΔCDF and 1.1×10^{-8} /yr in ΔLERF. The licensee determined that these CDF and LERF increases would not significantly impact the GSI-191 assessment. The NRC evaluated the licensee's assessment and determined that it adequately identified key assumptions and key sources of uncertainties in the licensee's SPRA, evaluated those assumptions, and performed sensitivity analysis to show the impact on this evaluation.

Regarding the licensee's HRA method, the NRC requested that the licensee clarify its resolution to the finding associated with F&O 16-11. This F&O stated that the review of the potential for additional HRA dependencies introduced by the SPRA model was missing and that the dependency analysis had been performed using the EPRI HRA Calculator, which was implemented subsequent to the peer review. The licensee stated that the peer reviewed SPRA model used a manual dependency analysis approach, and the use of the EPRI HRA Calculator was expanded to include the HRA dependency analysis subsequent to the SPRA peer review. However, the licensee also stated that the dependency tool in the EPRI HRA Calculator has a very similar set of rules when compared with the manual process. The licensee provided a comparison of the value of the dependent human error probabilities using the manual analysis approach and the EPRI HRA Calculator for several cases. The licensee described the rationale for any differences and stated that the listed dependent human error probabilities combinations were not significant to seismic CDF or LERF. Based on the information presented by the licensee, the NRC staff concludes that the change from manual dependency analysis to the HRA Calculator dependency analysis does not adversely affect this evaluation. Therefore, the NRC staff notes that the licensee has resolved F&O 16-11 for this evaluation.

Based on review of the SPRA peer review F&Os and their resolutions, the licensee's comparison between Addendum A and Addendum B of the ASME/ANS Standard, and evaluation of key assumptions and sources of uncertainty, the NRC staff concludes that the technical elements of the licensee's SPRA are acceptable for this evaluation because:

- Aspects of the PRA relied upon to evaluate the risk attributable to debris were peer reviewed and found to meet CC-II of the ASME PRA Standard, as endorsed by RG 1.200, Revision 2.

- Aspects of the PRA where findings or key assumptions or sources of uncertainty were identified, the licensee provided a technical justification for why the risk attributable to debris would not be affected.

3.4.1.4 Plant Representation

RG 1.174, Revision 2, states,

> The PRA results used to support an application are derived from a PRA model that represents the as-built and as-operated plant to the extent needed to support the application.

That is, at the time of the application, the PRA should realistically reflect the risk associated with the plant. Section 2.3 of Enclosure 1 of the technical report describes the risk management process for maintaining and updating the PRA to ensure that the Vogtle PRA models used to support its GSI-191 risk assessment accurately reflect the as-built, as-operated plant. Specifically, the licensee stated:

> The VEGP PRA models are controlled in accordance with the SNC procedure for PRA generation, maintenance and updates... To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plants, the PRA maintenance procedure requires the following activities be performed:
>
> - Design changes and procedure changes are reviewed for their impact on the PRA model on an on-going basis.
> - Reliability data, unavailability data, initiating events frequency data, human reliability data, and other such PRA inputs shall be reviewed approximately every two fuel cycles and updated as necessary to maintain the PRA consistent with the as-operated plant.

The licensee stated that it determined there were no significant outstanding changes that would impact the GSI-191 assessment. The licensee provided a summary of plant changes implemented since the cutoff date for the Vogtle SPRA with a qualitative assessment of impact of those changes.

The NRC staff reviewed the changes identified and determined one change is of potential relevance to GSI-191 risk assessment. Specifically, the licensee credited the installation of Westinghouse shutdown seals in its PRA per industry guidance, which was not reviewed and accepted by the NRC at the time of the submittal. The NRC staff considered the potential effects of taking quantitative credit for this change by considering how RCP seal LOCAs contribute to risk attributable to debris. The licensee stated in the letter dated May 23, 2018, that "The quantity of debris generated by an RCP seal LOCA is equivalent to the quantity generated by a small or medium LOCA, which was found to not challenge the sump strainers."

The NRC staff reviewed the licensee's assessment of the effects of RCP seal LOCAs and concludes that the licensee's inclusion of credit for RCP seal installation in its PRA is acceptable for this application because the effects of RCP seal LOCAs would be captured by small and medium LOCAs, which do not contribute additional risk attributable to debris.

The NRC staff concludes that the licensee's PRA model adequately represents the as-built and as-operated plant to the extent needed to support the GSI-191 risk assessment because the licensee's PRA maintenance procedures include an ongoing review of design and procedure changes for their impact on the PRA model, and PRA data or inputs are reviewed and updated, as necessary, on a periodic basis. In addition, the licensee provided a summary of plant changes implemented since the cutoff date for its SPRA and adequately performed a qualitative assessment of the impact of those changes.

3.4.1.5 NRC Staff Conclusion Regarding the Base PRA Model

The NRC staff concludes that the Vogtle base PRA (i.e., IEPRA, including internal floods and SPRA) model used in support of the licensee's GSI-191 risk assessment is acceptable (e.g., has the appropriate scope, level of detail, technical elements, and plant representation) to evaluate the risk attributable to debris because the licensee applied approaches consistent with the guidance in RG 1.174, Revision 2 (Reference 20), and RG 1.200, Revision 2 (Reference 35).

3.4.2 Risk-Informed Approach for Addressing the Effects of Debris on Long-Term Core Cooling

The licensee combined PRA information with traditional engineering analyses to estimate the risk attributable to debris. This integrated analysis is referred to as the "systematic risk assessment."

3.4.2.1 Scope of the Systematic Risk Assessment

This section describes the specific approach used by the licensee to determine all relevant initiating events for which debris could adversely affect CDF or LERF. This includes how relevant scenarios (i.e., an initiating event followed by a plant response such as a combination of equipment successes, failures, and human actions leading to a specified end state, such as event prevention, core damage, or large early release) that could be mitigated by the activation of sump recirculation were identified and considered in the systematic risk assessment.

In the letters dated April 21, 2017, and May 23, 2018, the licensee provided information regarding the scope of its systematic risk assessment that employed a screening process to eliminate scenarios that were deemed not relevant, not affected by debris, or having an insignificant contribution based on the identified failure modes. Screening is a common practice in quantitative risk assessments, and one acceptable approach is discussed in NUREG-1855, Revision 1 (Reference 47). Specifically, NUREG-1855, Revision 1, describes assessment of model and completeness uncertainty, including the identification of sources of uncertainty that are not related to either the parts of the PRA used to generate the results or the significant contributors to the results, and the use of screening and conservative analyses to address non-significant contributors. RG 1.174 recognizes that a screening approach "allows the detailed analysis to focus on the more significant contributions." Information pertaining to the licensee's initial plantwide and location-specific screening approach is described below.

3.4.2.1.1 Initial Plant-Wide Screening

The licensee used Version 5 of the Vogtle IEPRA model and Version 2 of the Vogtle SPRA model to quantitatively evaluate the risk attributable to debris (see Section 3.4.1.1, "Scope of the Base PRA (Modes/Hazards)"). The licensee qualitatively evaluated external hazards other than seismic (e.g., fire hazard, extreme winds, and tornadoes). The licensee's initial plant-wide screening process described in the letter dated May 23, 2018, included the identification of initiating events with the potential to (1) generate debris inside containment, (2) require sump recirculation for mitigation of the event, and (3) result in debris transport to the containment sump. The licensee excluded from its quantitative analysis initiating events that it determined did not meet these criteria.

The licensee stated it considered only scenarios that required recirculation through the ECCS or CSS strainers, since without recirculation, there is no potential for debris-related failures of the strainers, pumps, downstream components, or core. Using this criteria, the licensee considered the following initiating events relevant to GSI-191 risk assessment:

- RCS pipe breaks, resulting in small, medium, and large break LOCAs (SBLOCAs, MBLOCAs, and LBLOCAs, respectively)
- Non-piping LOCAs
- Secondary side breaks inside containment (SSBIs) that result in a consequential LOCA upon failure to terminate SI or a stuck open PORV
- Seismically-induced LOCAs
- Water hammer-induced LOCAs

The licensee stated that among internal initiating events, transients, steam generator tube rupture, inadvertent SI, inadvertent or stuck open PORVs that discharge to the pressurizer relief tank, secondary side breaks outside of containment, and interfacing systems LOCAs that discharge outside of containment do not have the potential to generate debris inside containment.

The licensee quantitatively evaluated the following in it is systematic risk assessment: (1) RCS pipe breaks, resulting in SBLOCAs, MBLOCAs, and LBLOCAs; (2) seismically-induced LOCAs; and (3) MSLB and feedwater line break (FWLB) SSBIs.

The licensee did not explicitly model non-piping LOCAs; however, it qualitatively determined that these LOCAs were bounded by already analyzed breaks at pipe weld locations. The licensee stated that the following non-pipe components were considered: nozzles, pump and valve bodies, pressurizer heater sleeves, manways, control rod drive mechanism penetrations, safety relief valves, RCP seals, the reactor vessel, the pressurizer vessel, the steam generator vessels, welded caps on retired lines, and other components. The licensee stated in its letter dated April 21, 2017, that:

> ...with the exception of non-pipe components that are located in the reactor cavity, all of these non-pipe components are located at or near pipe welds. For example, there are many weld locations in lines around the pressurizer vessel including the surge line, spray lines, and the safety and relief valve lines that could be used to estimate debris generated from non-pipe components in that area of containment. In addition, there are many welds distributed along the cold-legs, including those near the reactor coolant pumps, that could be used to estimate debris generated from non-weld locations in those areas. The modeled welds that are located at the safe ends on the nozzles at the reactor vessel, the pressurizer vessel, and the steam generator vessels are reasonably close to the associated nozzle welds and are close enough to the vessels to produce significant debris from the insulation around those vessels.

The licensee further stated that for non-pipe components associated with the reactor vessel that are not near modeled pipe locations (e.g., control rod drive penetrations, manways, and

instrument lines connected to the reactor vessel), debris generated would be bounded by a reactor vessel nozzle break.

The licensee qualitatively addressed water hammer events in its systematic risk assessment screening process. Specifically, the licensee deemed water hammer events to be an insignificant contributor to GSI-191 risk assessment because of (1) LOCA frequencies determined not impacted by water hammer events for Vogtle since piping is designed to ASME Code, Section III Class 1 standards, (2) the licensee's implementing of an approved gas accumulation prevention/monitoring program, and (3) the lack of historical data for water hammer events for the piping within the scope of the analysis.

In summary, the licensee's initial screening process concluded that SBLOCAs, MBLOCAs, and LBLOCAs from RCS pipe breaks; seismically-induced LOCAs; and MSLB and FWLB SSBIs warranted quantitative analysis as part of its systematic risk assessment. The licensee qualitatively determined that non-piping LOCAs were bounded by already analyzed breaks at pipe weld locations and that non-pipe components associated with the reactor vessel that are not near modeled pipe locations (e.g., control rod drive penetrations, manways, and instrument lines connected to the reactor vessel) are bounded by a reactor vessel nozzle break.

The NRC staff reviewed the licensee's screening approach and concludes that the approach is technically sound and consistent with state-of-practice approaches. Furthermore, the NRC staff concludes that the results of the plantwide screening adequately reflect initiating events relevant to the licensee's systematic risk assessment of GSI-191 phenomena.

3.4.2.1.2 Location-Specific Screening

For some scenarios (e.g., LOCA events), the effects of debris may be location dependent. Therefore, the licensee completed a location-specific analysis to identify accident sequences that could be adversely impacted by debris. This analysis involves identifying all possible break locations (e.g., pipe welds, valve bodies, manways) that could support small, medium, or large break LOCAs and justifying detailed evaluation of only pipe-break LOCAs occurring at weld locations that are important to the risk calculation (i.e., calculation of debris amounts and characteristics generated and transported to the sump). The licensee described its break selection process in Enclosure, 1, 3, and 5, of the letter dated April 21, 2017.

The licensee initially used the guidance from NEI 04-07 (Reference 26) to perform a deterministic break selection process. In general, the NEI 04-07 break selection methodology determines a few break locations that result in the limiting debris load or loads that can reach the strainer. The method is described in more detail in Sections 3.4.2.6, "Debris Source Term Submodel" and 3.4.2.6.1, "Break Selection" of this SE. The licensee later determined that the maximum debris loads predicted by the break selection method were beyond those that the strainer could accommodate. In order to determine which pipe breaks generated debris loads that could be accommodated by the strainers, the licensee performed a break location-specific evaluation for all ASME Class 1 piping welds in the RCS. In the risk-informed analysis, the licensee considered LOCA break frequencies from NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," (Reference 54) which includes contributions of breaks in piping and non-piping locations (see Section 3.4.2.2, "Initiating Event Frequencies" of this SE).

The licensee proposed that any LOCA break that generates and transports more debris of any type to the strainer than was shown to be acceptable by testing is assumed to result in core damage. Breaks that result in less than the amount of tested debris reaching the strainer are

considered to result in successful operation of the ECCS and CSS strainers. The licensee also performs checks to determine whether the head loss that results from the calculated debris load may also cause a failure from some other cause (for example, flashing).

The acceptable debris loading was determined by testing (see Section 3.4.2.8.3, "Head Loss and Vortexing" of this SE). In order to determine the amount of debris generated from each potential break location, the licensee developed a computer-aided design (CAD) model of containment. The CAD model included the locations of each potential break location and locations of debris sources that could be damaged by a LOCA jet. The licensee also calculated the amount of each type of debris that would transport to the strainer, considering the amount of debris generated by each break. Each break location was evaluated by the combination of the debris generation and transport models to determine the largest amount of debris that could arrive at the strainer due to a break at a specific location.

Based on initial screening results, the licensee performed a quantitative analysis of SBLOCAs, MBLOCAs, and LBLOCAs (including seismically-induced LOCAs) from RCS pipe breaks ranging from half-inch partial breaks to DEGBs on every Class 1 ISI weld within the first isolation valve. The licensee also quantitatively assessed SSBIs that result in a consequential LOCA upon failure to terminate SI or a stuck open PORV. For SSBIs, breaks were assumed to be DEGBs occurring approximately every 5 ft on the main steam line and feedwater line.

The licensee clarified how the potential for the failure of piping at locations other than welds (e.g., highly stressed locations, branch connections, and elbows) was considered in the letter dated February 12, 2018. Specifically, the licensee stated that: (1) branch locations and elbows on pipes typically have welds, and thus, these potential break locations were considered in the evaluation; (2) it is possible for a break to occur on a segment of pipe between welds, but evaluating only weld locations meets the intent of NRC guidance; (3) the use of Class 1 ISI welds as break locations is both systematic and thorough because there are multiple ISI welds on every pipe in the RCS and the welds cover the range of possible break locations; (4) a weld is generally closer to equipment that has a large quantity of insulation, compared to a span of straight pipe; and (5) welds are almost universally recognized as likely failure locations. The licensee also stated that since the LOCA frequency was allocated to the individual break locations using the top-down approach. The total plant-wide LOCA frequency is preserved. Therefore, the incorporation of additional non-weld locations would not significantly change the results of the evaluation.

The licensee listed the 413 unisolable ISI welds that it explicitly considered in the analysis in Table 3-9 in Enclosure 3 of the letter dated April 21, 2017. The licensee also considered the effect of ISI welds outside the first isolation valve and determined that there was no significant difference between the type and quantity of debris generated for similar sized breaks upstream or downstream of the first isolation valve.

In addition, the NRC staff requested that the licensee explain whether breaks postulated to occur outside the first isolation valve are in sections of piping that are normally isolated from the RCS and to confirm that the failures of these pipes would not result in RCS leakage rates greater than normal makeup capabilities. As part of the licensee's response dated February 12, 2018, the licensee stated that the largest diameter weld outside the first isolation valve is 10.5 inches, and that a 12-inch break was the smallest break that caused strainer failure inside the first isolation valve for any of the base-case equipment configurations evaluated. Therefore, the licensee concluded that it is not likely that any break downstream of the first isolation valve would cause strainer failure due to the effects of debris. The licensee

also noted that even if those breaks cause any failures from debris, the risk contribution would be negligibly small due to the low likelihood of an isolation valve failing to close, spuriously opening, or developing a large leak (e.g., probability of a normally open valve failing to close is less than 4×10^{-4}, probability of a large leak or spurious operation of an isolation valve is on the order of 10^{-7} or less).

The location-specific screening process refined the quantitative analysis to breaks in ISI welds in the unisolable portion of the Class 1 pressure boundary (i.e., inside the first isolation valve) and SSBIs, namely MSLBs and FWLBs. The NRC staff reviewed the licensee's location-specific screening evaluation and concludes that the licensee identified all locations that could result in a failure of the ECCS LTCC functions, because the full spectrum of possible break locations was considered and systematically assessed for potential effects on the calculation of debris amounts generated and transported to the sump.

NRC Staff Conclusion Regarding the Scope of the Systematic Risk Assessment

The NRC staff reviewed the scope of the systematic risk assessment and finds it sufficient, because the licensee employed a systematic screening process using initial plant-wide and location-specific screening approaches to eliminate scenarios that do not affect GSI-191 risk assessment in a manner, consistent with state-of-practice approaches described in NUREG-1855, Revision 1 (Reference 47).

3.4.2.2 Initiating Event Frequencies

The licensee used the Vogtle PRA as a guide in assigning frequencies to LOCAs and SSBIs that were included in its systematic risk assessment. The licensee stated that the use of initiating event frequencies in the Vogtle base PRA model is consistent with the expectations in the ASME/ANS PRA Standard, as endorsed by RG 1.200, Revision 2," (Reference 35) and confirmed by the Vogtle base PRA model peer review. The licensee's PRA medium and large LOCA frequencies are based on the geometric mean aggregation of NUREG-1829," (Reference 54). NUREG-1829 provides exceedance frequencies for discrete break sizes, which is the annual probability of having a specified break at a given size or larger.

The licensee stated that it used the 1988 - 2010 NRC initiating event database for small LOCAs (Reference 55). Although the value for small break LOCA frequency used by the licensee (4.73×10^{-4} /yr) is lower than that produced from NUREG-1829 data, the GSI-191 risk analysis is not sensitive to the initiating event frequency for small break LOCAs because these breaks are not predicted to generate enough debris to cause strainer or core failures at Vogtle.

For the calculation of small, medium, and large LOCA conditional failure probability (CFP), the LOCA frequencies were allocated to individual pipe welds using a top-down distribution methodology. The top-down LOCA frequency allocation methodology treats all breaks of a similar size as having an equivalent LOCA frequency, regardless of the weld size and specific degradation mechanism.

The licensee used a semi-log interpolation scheme (i.e., linear interpolation between break sizes and logarithmic interpolation between frequencies) to determine frequencies for medium and large break sizes not explicitly listed in NUREG-1829 or in the initiating event database for small break LOCAs. The guidance in NUREG-1829 states that interpolation may be used but does not specify use of any one interpolation scheme. The NRC staff concludes the licensee's use of semi-log interpolation of NUREG-1829 data is acceptable for the following reasons:

(1) the pipe break sizes span approximately one order of magnitude in the NUREG-1829 tables and the non-exceedance frequencies span several orders of magnitude; and (2) the NRC has previously reviewed the effects of using different interpolation schemes in a risk-informed evaluation of GSI-191 phenomena and determined that results are not highly sensitive to choice of interpolation scheme (Reference 54).

NUREG-1829 guidance contains "25-year" or "current" LOCA frequencies and "40-year" or "end of license period" LOCA frequencies. For most LOCA types, the 40-year values are slightly higher due to anticipated aging effects and the possibility of new degradation mechanisms. In some cases, however, the 40-year values are lower, reflecting an expectation that improved mitigation techniques will lower LOCA frequencies. NUREG-1829 recommends the use of the 25-year values for plants that have been operating between 25 and 40 years.

Vogtle, Units 1 and 2, were initially licensed in 1987 and 1989, respectively, and therefore, have been operating between 25 and 40 years. The licensee used "25-year" LOCA frequencies from NUREG-1829. The NRC staff concludes that this is acceptable because the use of "25-year" LOCA frequencies is consistent with the recommendations described in NUREG-1829.

The NUREG-1829 guidance provides LOCA frequencies that are based on a formal expert elicitation process. Several aggregation schemes are presented that combine, or aggregate, the inputs of the individual experts into a single set of frequencies that can be used for decisionmaking. The two primary aggregation schemes are the geometric mean and simple average, or arithmetic mean. Because alternate aggregation methods can lead to significantly different results, NUREG-1829 states that different methods may be appropriate for different applications and recommends that multiple methods and sensitivity studies be considered when selecting an aggregation method.

In the letter dated April 21, 2017, the licensee provided estimates of the risk attributable to debris based on a geometric mean. The licensee performed a sensitivity analysis using the arithmetic mean aggregation to address uncertainty associated with use of geometric aggregation for medium and large LOCA frequencies (see Section 3.4.2.11.1, " Parametric Sensitivity and Uncertainty Analysis" in this SE). The NRC staff reviewed the licensee's sensitivity analysis and concludes that the licensee's approach for evaluating the impact of the aggregation method is an acceptable way to address this source of uncertainty because it is consistent with the recommendation in NUREG-1829.

The NRC staff notes that the frequencies in NUREG-1829 apply only to LOCAs caused by long-term material degradation. The licensee addressed water-hammer-induced and non-piping LOCAs qualitatively. To assess potential effects of alternative material degradation mechanisms, the licensee performed sensitivity analyses with LOCA frequencies skewed to high or high and medium rupture probability welds. These sensitivity analyses showed that alternative frequency weightings related to alternative material degradation mechanisms had a negligible effect on ΔCDF. The NRC staff reviewed the methodology used to evaluate the effect of alternative material degradation mechanisms and concludes that the licensee adequately addressed associated uncertainties consistent with NUREG-1855 guidance pertaining to model uncertainty, because the sensitivity analyses showed the effect of alternative weighting assumptions.

In the letter dated February 12, 2018, the licensee stated that since the LOCA frequency was allocated to the individual break locations using the top-down approach, the total plant-wide LOCA frequency is preserved and does not change with the incorporation of additional break locations. The only difference would be that the current breaks postulated on welds would be allocated a smaller portion of the LOCA frequency in order to allocate some frequency to non-weld locations. The licensee, therefore, concluded that since non-weld locations do not change the overall LOCA frequency and do not tend to have preferentially more debris than breaks on weld locations, analyzing breaks at these locations does not significantly change the results of the evaluation.

The licensee explicitly evaluated MSLB and FWLB side breaks inside containment (SSBI) initiators that result in a consequential LOCA upon failure to terminate SI or a stuck open PORV. The licensee assumed MSLB and FWLBs were DEGBs and separated the MSLB and FWLB frequency due to the significant difference in debris quantities that could be generated by these breaks. The licensee stated that the SSBI frequency used for the GSI-191 analysis was from its base PRA model, which is based on updated industry initiating event data NUREG/CR-6928. "Industry-Average Performance for Components and Initiating Events at U.S. Nuclear Power Plants," (Reference 56) Initiating Event Data Sheets Update 2010, dated January 2012.

In the letters dated April 21, 2017, and February 21, 2018, the licensee discussed initiating event frequency associated with seismically-induced LOCAs. Specifically, the licensee stated that it derived seismic initiating event frequencies by identifying the most susceptible (i.e., lowest seismic capacity) component to seismic events among the nuclear steam supply system components – the RCP. An indirect seismically-induced LOCA may arise due to failure of the RCP column assembly support, which provides the least seismic margin of safety. The licensee derived initiating event frequencies by combining the RCP fragility with site-specific seismic hazards. The licensee stated in the letter dated February 21, 2018, that this alternative initiating event frequency of seismically-induced LOCAs is independent of LOCA frequencies in NUREG-1829. Therefore, uncertainties affecting the regular LOCA frequencies do not affect the seismically-induced LOCA frequencies the licensee estimated.

NRC Staff Conclusion Regarding Initiating Event Frequencies

The NRC staff reviewed the licensee's information on initiating events and concludes that the initiating event frequencies selected by the licensee for this evaluation are acceptable because:

- MBLOCA and LBLOCA frequencies were obtained from NUREG-1829, which is considered to be the most current set of values available. SBLOCA, MSLB, and SSBI initiating event frequencies were obtained from the licensee's base PRA. These frequencies are based on industry updated databases.

- The licensee interpreted the NUREG-1829 data in a manner consistent with the guidance in NUREG-1829.

- The licensee performed sensitivity analyses to address the selection of LOCA frequencies from NUREG-1829 using the arithmetic mean and the geometric mean aggregated frequencies.

- The licensee performed a sensitivity analysis to address potential effects beyond long-term material degradation location-specific LOCA frequency contributors (e.g.,

frequency allocation methods skewed to high or high and medium rupture probability welds).

- The allocation of frequency to non-weld pipe locations inside the first isolation weld would not significantly change the results of the evaluation.

- The licensee employed an acceptable method to establish the frequency of seismically-induced LOCA events, by combining the fragility of the RCP, the most vulnerable component of the nuclear steam supply system, with site-specific seismic hazards.

3.4.2.3 Scenario Development

For the purposes of this SE, the term "scenario" means an initiating event followed by a plant response such as a combination of equipment successes, failures, and human actions leading to a specified end state, such as successful event mitigation, core damage, or large early release.

The licensee modeled the system response to a LOCA break in the presence of debris using NARWHAL based on a 30-day mission time. The model includes a multi-pump and multi-strainer system (four strainers, one strainer per RHR or CS pump) with one ECCS train comprised of one RHR, CS, high head centrifugal charging, and medium head SI pumps, and two SI accumulators. The model further considers injection and recirculation periods with automatic switch from injection to recirculation (after the RWST reaches a minimum level) and automatic activation of CS pumps (occurring only for very large hot-leg breaks). Recirculated coolant is initially injected into cold-legs of the pressure vessel and switched over to simultaneous injection after 7.5 hours. The charging pumps continue injecting into the cold legs, and the RHR and SI pumps are realigned to the hot-legs. The licensee's analysis did not take credit for operator actions such as refilling of the RWST.

In the letter dated May 23, 2018, the licensee described the systematic process it used to determine the high-likelihood equipment configurations included in its detailed NARWHAL analysis and for modeling in its PRA. The process included these high level steps: (1) determine all possible combinations of system or train failures that affect the likelihood of debris-induced failure of ECCS following a LOCA, (2) determine the functional failure probability (FFP) for each configuration and system or train failure state in that configuration using the PRA model to calculate the total probability for each equipment failure combination,(3) determine the annual scenario frequency based on the initiating event frequency and total equipment failure probability for each possible equipment configuration identified for each LOCA size, and (4) retain for further analysis those equipment configurations in each scenario whose frequencies for each LOCA size were identified to contribute to the top 95 percent or individually contributed 1 percent to the total frequency.

The licensee stated in Enclosure 3, Section 6, of the letter dated April 21, 2017, that it performed a detailed computation of the conditional strainer failure probability for the following high-likelihood equipment configurations:

- No equipment failed (all ECCS trains operating)
- One HR pump (RHPA or RHPB) failed
- One charging pump (CPPA or CPPB) failed

- One SI pump (SIPA or SIPB) failed
- One nuclear service cooling water (NSCW) train (NSCWA or NSCWB) failed, causing failure of one ECCS train
- One CS pump (CSPA or CSPB) failed
- Two CS pumps (CSPA and CSPB) failed

The licensee further stated that the cases of one CCP or SI pump failed are equivalent to the case of no equipment failure in its GSI-191 risk assessment because the CCP and SI pumps piggyback off the RHR pumps during recirculation, and, therefore, their failure does not affect flow rates through RHR strainers.

Equipment configurations that did not meet the high-likelihood criteria were identified as low-likelihood configurations. Simplified calculations were used to estimate the ΔCDF and ΔLERF associated with these configurations without explicitly using PRA models. The licensee mapped low-likelihood configurations to high-likelihood configurations on the basis of the number of functional pumps to estimate the associated strainer CFP. When equivalent high-likelihood configurations were not available (e.g., configuration with one RHR pump failed and two CS pumps failed), the licensee assumed that the strainer will fail (i.e., CFP = 1).

The licensee examined MSLB and FWLB SSBIs and implemented detailed computations of the strainer CFP. Specifically, the licensee stated its NARWHAL evaluation assumed all MSLBs and FWLBs were DEGBs and that one or both trains of CS had failed.

The licensee assumed that seismic events may cause a LOCA event and evaluated seismically-induced LOCAs. For seismic-related scenarios, the licensee assumed the system response to these LOCA events to be equivalent to the response to internal event LOCAs that affect GSI-191 risk assessment. Consistent with this assumption, the licensee used the same CFP LBLOCAs of the high-likelihood scenarios to define the conditional strainer failure probability, given a seismically-induced LOCA. The licensee did not implement any specialized NARWHAL simulation to address seismically-induced LOCAs. The licensee used the strainer CFP for LBLOCAs in the SPRA to compute the seismic contribution to the ΔCDF.

<u>NRC Staff Conclusion Regarding Scenario Development</u>

The NRC staff evaluated the licensee's scenario development process and results and concludes that the licensee adequately described the evolution of the system during the mission time and properly considered the important scenarios (including reasonably bounding assumptions) to estimate the risk attributable to debris because the licensee: (1) used a systematic process to identify germane operating components and states (i.e., high- and low-likelihood configurations), (2) considered the long-term period of performance, including a definition of the safe and stable end-state of the nuclear power plant and human actions that are part of the pertinent accident sequences using its PRA, although the licensee did not take credit for operator actions, and (3) evaluated the set of assumptions and considerations relevant to the development of scenarios.

3.4.2.4 Failure Mode Identification

The following are potential debris-related failure modes for the ECCS LTCC function. Each of these failure modes should be considered and specifically evaluated, or shown to be irrelevant, to the risk-informed evaluation. Other potential failure modes should be evaluated, as necessary, for plant-specific conditions. The licensee evaluated each of the phenomena below and did not identify additional failure modes. These failure modes are only those related to debris.

a. Excessive head loss at the strainer leads to loss of NPSH for adequate operation of the pumps. For this case, the licensee also evaluated the strainer at times when it would be unsubmerged by verifying that the head loss does not exceed one-half of the submerged strainer height as recommended by RG 1.82 (Reference 34). For cases in which the debris amount per submerged strainer area exceeded that tested, the licensee assumed a scenario failure.

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

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

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

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

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

g. Debris buildup in the vessel leads to excessive boron concentrations within the core.

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

The licensee listed the failure modes as:

- Debris accumulation in an upstream flow path choke point (e.g., a refueling canal drain) exceeds blockage limits and reduces the available sump volume.

- Strainer head loss exceeds the NPSH margin for the ECCS and CSS pumps when the strainer is fully submerged.

- Strainer head loss exceeds half of the submerged strainer height when the strainer is partially submerged.

- Strainer head loss exceeds the strainer structural margin.

- Gas voids (i.e., water vapor due to flashing or air intrusion due to degasification or vortexing) downstream of the strainers exceed the acceptable void fraction limits of the ECCS and CSS pumps.

- Debris penetration exceeds ex-vessel downstream effects limits for component wear or clogging.

- Debris penetration exceeds in-vessel downstream effects limits for core blockage.

- Buildup of oxides and other chemical precipitates on fuel cladding exceeds heat transfer limits.

- Boric acid concentration in the core exceeds the solubility limit, resulting in boric acid precipitation.

NRC Staff Conclusion Regarding Failure Mode Identification

The NRC staff evaluated the licensee's analysis and compared the licensee's failure modes to those established by the staff and determined that the failure modes evaluated by the licensee include all those that could reasonably expected to lead to debris-induced failure of LTCC. Therefore, the NRC staff concludes that the licensee included the appropriate failure modes in its evaluation.

3.4.2.5 Changes to the Base PRA Model

The licensee provided a description of specific probabilistic risk assessment (PRA) model changes in Enclosure 3, Section 4.0, of the letter dated April 21, 2017. The licensee first described assumptions regarding CS pump actuation for different LOCA initiating events (e.g., small, medium, large) and the modeling of CS pump recirculation suction and charging and SI pump failures in the PRA. The licensee then provided a high-level description of how the Vogtle PRA model represented independent and common cause plugging of ECCS containment sumps A and B prior to GSI-191-related model changes. Lastly, the licensee described changes made to the PRA to model GSI-191-related scenarios for its GSI-191 risk assessment. The licensee modeled each sump strainer and core blockage failure scenario in its PRA as a basic event that is combined with the appropriate LOCA initiating event and pump failure logic to represent the applicable equipment configurations. The seven equipment configurations considered with applicable PRA logic were identified as follows: (1) no equipment failed (all ECCS trains operating), (2) one RHR pump (RHPA or RHPB) failed, (3) one charging pump (CPP) (CPPA or CPPB) failed, (4) one SI pump (SIPA or SIPB) failed, (5) one NSCW train (NSCWA or NSCWB) failed, causing failure of one ECCS train, (6) one CS pump (CSPA or CSPB) failed, and (7) two CS pumps (CSPA and CSPB) failed. The licensee's process for determining these specific configurations is discussed in Section 3.4.2.3, "Scenario Development," of this SE.

The licensee stated logic for GSI-191 RHR sump strainer failures was modeled under new fault tree gates GSI-191-SUMP-A, GSI-191-SUMP-B, and GSI-191-SUMP-AB, with new logic gates included under existing PRA model gates for loss of flow from the ECCS containment sumps associated with specific RHR pump(s) (e.g., ECCS-SUMP-A, ECCS-SUMP-A-ACR, ECCS-SUMP-B, and ECCS-SUMP-B-ACR). The logic developed for GSI-191 core blockage was modeled under new fault tree gate GSI-191-CORE and added under the existing PRA model top gate for core damage (CDF-TOTAL).

In its letter dated April 21, 2017, the licensee stated the following regarding PRA logic

associated with LERF:

> A GSI-191 core blockage logic gate was also added under the existing PRA model gates for LERF end states 01 through 08 (gates LERF-01 through LERF-08). End states LERF-01 through LERF-06 can result from small LOCAs only; therefore, the small LOCA core blockage gate (GSI-191-CORE-SL) was combined with the containment event tree logic associated with each of those end states (e.g., gate LERF01 X ... LERF06X). End state LERF-07 can result from medium or large LOCAs; therefore, the medium and large LOCA core blockage gates (GSI-191-CORE-ML "OR" GSI-191-CORE-LL) was combined with the containment event tree logic for end state LERF-07 (gate LERF07X). Finally, end state LERF-08 can result from small, medium, or large LOCAs; therefore, the core blockage gate for all LOCAs (GSI-191-CORE) was combined with the containment event tree logic for end state LERF-08 (gate LERFOBX).

The licensee also stated that, while specific proceduralized operator actions were included in the risk-informed GSI-191 evaluation, no operator actions were credited to recover from the effects of debris-related failures on the strainer or the core in its ΔCDF and ΔLERF calculation.

The values for CFP that were determined in NARWHAL for each equipment configuration served as inputs to the new PRA model logic for GSI-191 risk assessment (see Section 3.4.2.10, "Systematic Risk Assessment" of this SE). While CFPs for some equipment configurations and initiating events (e.g., SBLOCAs and MBLOCAs) may be zero, the licensee stated it retained GSI-191-related PRA logic even for cases where the CFP may be zero for potential sensitivity studies.

The licensee stated, "Existing sump plugging events may ultimately be removed from the Vogtle PRA model of record and replaced with the more detailed representation of GSI-191 sump strainer failure logic." Consistent with the guidance in RG 1.174, Revision 2, which states that risk-informed applications should include the effects of past applications, if this methodology is later approved by the NRC and adopted, the base PRA used for future submittals will include consideration of the risk attributable to debris unless it can be shown to not affect the decision being made. Consideration of the risk attributable to debris aligns with the overall guidance in RG 1.174, Revision 2, and RG 1.200, Revision 2, both of which state that the PRA should realistically model the as-built, as-operated plant. This consideration also aligns with the ASME/ANS PRA Standard, which explicitly states that phenomenological conditions (e.g., effect of debris on NPSH) should be included in accident sequence or system models.

NRC Staff Conclusion Regarding Changes to the Base PRA Model

The NRC staff reviewed the information on PRA model changes provided by the licensee and concludes that the changes are acceptable since those changes appropriately establish the cause-effect relationship as it relates to GSI-191 phenomena, and are, therefore, sufficient to model the impact of risk attributable to debris.

3.4.2.6 Debris Source Term Submodel

This section describes the debris that may be generated during an initiating event or may be present in the containment prior to the event. It includes a description of the debris sizes and characteristics that may transport to the strainers and affect the ability of the ECCS and CSS to perform their functions. The section also discusses administrative actions used to limit the

sources of potential debris. Additionally, this section evaluates the parts of the deterministic analyses that deal with debris source term to determine whether the licensee used appropriate inputs to the risk-informed analysis.

The licensee conducted strainer head loss tests with amounts of debris that are less than the amounts that may be generated by some initiating events (breaks on pipes leading to LOCAs). The debris amounts used in the testing are considered to be acceptance criteria in the evaluation.

The licensee conducted a debris generation evaluation that considered the sources of debris that may affect system performance. The debris generation evaluation included thousands of debris generation cases based on postulated breaks at all welds that could result in a LOCA, with the exception of those breaks screened out of the analysis. The evaluation considered hundreds of welds with breaks of varying size and orientation at each weld. Any break that generates more debris than that represented in the test is considered to fail a deterministic criterion and is evaluated using a risk-informed analysis.

The acceptance criteria for strainer performance were derived from a head loss test performed by the licensee in 2009. The important information is a listing of the amount of each material included in the testing and an evaluation that determines whether the testing was conducted under realistic plant conditions. Evaluations of the amount of debris that may enter the RCS under various conditions are also important for the risk-informed evaluation.

For the Debris Source Term section, all descriptions attributed to the licensee's submittal are taken from SNC's letters dated April 21, 2017, and February 12, 2018, unless otherwise noted. The information evaluated is primarily from Sections 3.a, 3.b. 3.c, 3.d, 3.h, and 3.i of Enclosure 5 of the licensee's letter dated April 21, 2017. Unless otherwise noted, the RAIs and responses to the RAIs discussed in this section are documented in the licensee's letter dated February 12, 2018 (Reference 7).

3.4.2.6.1 Break Selection

This section describes the licensee's process to identify the break sizes and locations that present the greatest challenge to post-accident sump performance. The licensee provided a summary of the break selection process in Enclosure 5, Section 3.a, of the letter dated April 21, 2017. This section also included significant discussion of debris generation, which is evaluated in Section 3.4.2.6.2, "Debris Generation and Zone of Influence" of this SE. The licensee also considered other potential initiating events (debris generation locations). Some of these initiating events were screened out of the systematic risk assessment and were not considered in the break selection process.

The licensee stated that the debris generation calculation was performed using the methodology of NEI 04-07 and the associated NRC SE (Reference 25). However, instead of focusing on limiting breaks, the licensee evaluated a full range of breaks, including breaks at all Class 1 pressure boundary ISI welds. The licensee stated that both full DEGBs and partial breaks were evaluated. Breaks were split into small (< 2 inch), medium (2 to 6 inch), and large (> 6 inch).

In order to evaluate the potential debris generation from each break, the licensee used BADGER (see Section 3.4.2.6.2, "Debris Generation and ZOI Submodels" in this SE for additional information on BADGER). Due to similarities between units, only the Unit 1 containment was modeled. Break locations that are normally isolated from RCS pressure or

isolated by a check valve were not included in the debris generation analysis. These valves were termed as being "out" in the licensee's submittal.

The ZOI represents the zone or volume in space where a two-phase jet from a high-energy line break can generate debris that may be transported to the sump. The size of the ZOI is defined in terms of pipe diameters and is determined based on the system pressure and the destruction pressure of the insulation impacted by the jet. Higher system pressures result in increased ZOIs. Robust materials have smaller ZOIs than fragile materials.

The licensee stated that although of low probability, breaks in secondary side (ASME Class 2) piping inside containment could require ECCS recirculation. The licensee used a similar methodology for determining debris generation for secondary side piping to primary piping. Because the pressure in the secondary piping is lower than that in the primary piping, the licensee reduced the ZOI sizes for various debris types by calculating a boundary at which the destruction pressure would be the same as that for a larger ZOI originating from a primary break. For secondary breaks, instead of postulating failures at each weld and evaluating different break sizes and orientations, the licensee considered only DEGBs spaced at 5 ft intervals along the piping. The licensee stated that there is no fire barrier material within any secondary break ZOI and that the coatings amounts were bounded by primary break coatings source term values.

The licensee stated that the debris generation results reflect the following conservatisms:

- All debris sources within the reactor cavity were assumed to be available for destruction by breaks within the cavity, even though significant equipment would shadow (i.e., protect) some of the targets.

- All qualified coatings on steel and concrete were assumed to have the worst case coating system (assume thickest coating layer allowed by specification) for the applicable surface.

- Main loop breaks in the steam generator compartments were grouped by loop and truncated in a way that results in conservative debris amounts for some breaks.

The licensee summarized where the worst case (i.e., largest quantity of debris) location for each debris type was located. The licensee also provided a summary of debris generated by debris size for small, medium, and large breaks. The summary table by break classification listed the minimum, maximum, and average amounts of each debris type and size. The licensee also provided graphs of debris amounts for various debris types by break size. The information was provided for both partial breaks and DEGBs.

NRC Staff Conclusion Regarding Break Selection

The NRC staff concludes that the break selection evaluation is acceptable because the licensee evaluated all ASME Code Class 1 weld locations that can result in a LOCA, except as described in Section 3.4.2.1.2, "Location Specific Screening." Although the NEI 04-07 guidance approved by the NRC states (Reference 26) that the licensee should evaluate all pipe locations for potential rupture, the staff concludes that the licensee's evaluation of piping only at welds is acceptable because the weld locations adequately represent the potential debris generation of breaks and are more likely break locations. The NRC staff also notes that longitudinal breaks in piping need not be considered by Vogtle because Vogtle has no longitudinal pipe welds in the

RCS.

The NRC staff concludes that the break selection process and criteria are acceptable because it identifies a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated as part of an acceptable evaluation model as required, in part, by 10 CFR 50.46.

The NRC staff concludes that SNC has provided sufficient information as requested by GL 2004-02 and further described in the revised content guide for GL 2004-02 concerning the break selection criteria. In the technical report the licensee:

- Described and provided the basis for the break selection criteria used in the evaluation.
- Stated whether secondary line breaks were considered in the evaluation (e.g., main steam and feedwater lines) and briefly explained why or why not.
- Discussed the basis for reaching the conclusion that the break size(s) and locations chosen to present the greatest challenge to post-accident sump performance.

3.4.2.6.2 Debris Generation and ZOI Submodels

The licensee conducted a debris generation analysis to calculate the amounts of each type and size of debris that should be added to the 2009 strainer test. The fiber amounts calculated in this analysis were smaller than those that would be calculated using approved assumptions. Therefore, the tested amounts of debris are simply used as acceptance criteria for strainer performance. The NRC staff review did not address the assumptions that were used to develop the test inputs. Instead, the NRC staff considered the evaluation of the amounts of debris that can be generated from each postulated break to ensure that these values were developed appropriately and can be compared with debris values known to be used in the test.

As described in Enclosure 5 of the licensee's technical report, each material's ZOI was defined based on the material destruction pressure. The licensee stated that debris-specific ZOIs were used in the analysis. The destruction pressures and associated ZOI radii for the particulate-based insulating materials in Vogtle containment are discussed below. These assumptions were also carried forward to the risk-informed debris generation calculations.

In the following discussion, L/D defines the damage zone for various materials, where L is the radius of the spherical or hemispherical jet and D is the diameter of the break. For DEGBs, D is equal to the inner diameter of the pipe and a spherical jet is assumed. For single-sided breaks, a hemispherical jet is assumed. More robust materials have higher damage pressures and smaller L/D values.

The licensee stated that NEI 04-07 and the associated staff SE do not recommend a destruction pressure or ZOI for Interam E-50 series fire barrier material installed in the Vogtle containments. The licensee stated that the destruction pressure was assumed to be equivalent to Temp-Mat that has a ZOI radius of 11.7D, which corresponds to a destruction pressure of 10.2 pounds per square inch gauge (psig). The ZOI assumption was based on testing that compared the robustness of Temp-Mat and the fire barrier materials.

The licensee assumed that the ZOI for fibrous insulation was 17D with an associated destruction pressure of 6 pounds per square inch (psi). These values are consistent with

NEI 04-07, as approved. The amount of fiber in the test was much less than the largest potential amount that can be generated from that limiting break. The licensee assumed that qualified epoxy systems had a ZOI of 4D with an associated destruction pressure of 40 psig. For secondary breaks on main steam and feedwater lines, the licensee calculated ZOIs that would result in equivalent pressure at the ZOI boundary, considering the reduced pressure in the secondary piping. The calculations assumed the destruction pressures listed above. Steam line pressure was assumed to be 945 pounds per square inch absolute (psia), and feedwater line pressure was assumed to be 1,150 psia. The ZOIs for steam lines breaks were calculated as 10.5D for Nukon, 7.9D for fire barrier, and 3.0D for qualified coatings. For FWLBs, the ZOIs were calculated as 11.3D for Nukon, 7.2D for fire barrier, and 2.8D for coatings. For MSLBs and FWLBs, the licensee postulated DEGBs and evaluated breaks at 5 ft intervals along the piping.

The licensee stated that robust barriers can be credited to prevent further expansion of the break jet. The volume of a spherical ZOI with a radial dimension extending beyond barriers is truncated at the barrier. The licensee stated that to avoid complications from equipment shadowing, only concrete structures were credited as robust barriers.

The licensee originally intended to show that the amounts of particulates included in the test bounded the amounts that could arrive at the strainer in the plant. After the initial letter dated April 21, 2017, was received by the NRC, the licensee determined that all of the coatings that are assumed to become particulate debris following a LOCA may not have been accounted for in the testing. The NRC staff asked RAIs 24 and 35 (Reference 7) regarding coatings amounts and the way that the coatings were represented in the testing. The licensee stated that the coatings generated amounts were calculated correctly, but that the density correction for testing was applied incorrectly. The licensee corrected the density factor and made changes to the transport calculation to account for the location of unqualified coatings. Coatings are discussed in more detail in Sections 3.4.2.6.5, "Coatings" and 3.4.2.7, "Debris Transport."

The licensee provided tables with debris information for the four break cases that generate the maximum debris quantities and for the four cases that generate debris amounts just below the acceptable debris amounts that were established in its 2009 testing.

The licensee stated that the amount of miscellaneous material (signs, labels, etc.) that could block strainer surface area was 2 ft^2. However, the licensee conservatively included 50 ft^2 of miscellaneous debris in the NARWHAL computations. The miscellaneous debris was modeled in NARWHAL as a reduction of the active strainer area.

The licensee stated that the total amount of latent debris in the containments was about 60 pounds (lbs.) based on walkdown data. However, the analysis included 200 lbs. of latent debris. The licensee assumed that 15 percent of the latent debris was fiber. Therefore, the assumed latent fiber amount was 30 lbs,-mass (lbm), and the assumed latent particulate amount was 170 lbm. These assumptions are based on the NRC staff-approved guidance in NEI 04-07.

The licensee stated that the debris generation calculation methodology followed the approved guidance of NEI 04-07. However, a full range of breaks was evaluated instead of assessing only the limiting breaks. All unisolable welds within the Class 1 ISI pressure boundary were evaluated, including DEGBs and partial breaks. In order to calculate thousands of break scenarios, the licensee used a CAD model of containment in conjunction with the BADGER software to automate the process. By automating the process, the licensee was able to

calculate debris amounts for full breaks and partial breaks of various sizes and orientations. The licensee stated that breaks from 1/2 inch to full DEGBs were evaluated at eight 45-degree angular increments (for each break size) for each weld. The break sizes were evaluated at ½ inch intervals for the smallest breaks, then 2 inch intervals from 2 through 14 inches, 1 inch intervals from 15 to 27 inches, and 1/2 inch intervals from 27.5 to 31 inches. These intervals were chosen to ensure that risk associated with potential breaks was captured accurately, without using excessive resources for the calculations. The size intervals decrease with increasing break size because the larger failures are much more likely to result in a challenge to LTCC. The licensee stated that breaks of 12 inches and below do not generate enough debris to cause failures based on NARWHAL analysis results. Most breaks that cause failures are much larger. The licensee performed sensitivity studies to show that its choice of break size ranges resulted in adequate debris generation calculations and that further refinements to this parameter would not significantly affect computations of strainer failure probability and ΔCDF. The NRC staff performed an audit of the BADGER and NARWHAL software in May of 2016 (Reference 23). During the audit, the staff discussed how break size ranges and break orientations were determined. The licensee's technical report included a sensitivity study for the modeling of break size discretization that showed low sensitivity of the conditional strainer failure probability on the selected discretization. Independent calculations were performed by the NRC (Reference 22) to ensure the licensee's characterization of debris amounts was well characterized by the broad discretization of break size range, as well as break orientations, and to ensure it is unlikely that any major debris source was overlooked by the discretization considered by the licensee. For DEGBs, a spherical ZOI was assumed centered at the center of the pipe in the plane of the weld. For the partial breaks, a hemispherical ZOI was assumed with a direction normal to the pipe axis and centered at the edge of the pipe.

In order to validate the debris generation models, the licensee performed comparisons of the CAD model/BADGER results with calculations that had been performed for an earlier limiting break analysis. The results of the comparison were acceptable. The licensee also used laser scans of the containment to ensure that the CAD model (which was generated from drawings and physical measurements) accurately reflected piping, component, and insulation locations.

For coatings, the BADGER calculations assumed the thickest possible coating thickness from installation records or the plant specifications. This assumption was intended to ensure that the amounts of coating debris calculated were conservative.

The licensee identified a total of 413 welds located on pipes distributed throughout the containment as potential LOCA break locations. As described in Section 3.4.6.1, "Break Selection," isolable welds were excluded from the evaluation.

The licensee did not consider it appropriate to credit unqualified coatings as remaining in place in post-LOCA conditions, and assumed unqualified coatings as debris. The amounts of latent debris and unqualified coating debris generated were modeled by the licensee as independent of the size of the postulated LOCA break.

The licensee evaluated each break size, location, and orientation to determine if the debris generated and transported from that break resulted in a strainer or in-vessel failure. Strainer failures were assumed to occur if any type of debris, generated and transported, exceeded the tested amount, flashing at the strainer occurred, excessive deaeration of the fluid occurred, head loss exceeded the strainer structural margin, or RHR pump NPSH margin was lost. In reality, the need to evaluate for NPSH margin failures was minimal because the strainer head loss did not exceed the limits at the maximum tested amount of debris. In general, the strainer

failures were due to exceeding a debris limit, and not as a result of exceeding a head loss limit. An exception could be cases where deaeration of the fluid resulted in increasing the pump NPSH required value to the point where NPSH available was insufficient to maintain margin. The NRC staff verified that the predominant failure mode was debris limit exceedance. In the cases where a full RHR-CS train was assumed out of service, other failure modes were exhibited, but for very few breaks. However, for those same breaks, the debris limits were also exceeded. There were no cases where other failure modes alone, other than debris limit exceedance, would result in a scenario failure for the strainers. The debris limits exceeded were fiber limits or calcium phosphate limits. The licensee programmed algorithms in the BADGER model to automate computation of debris amounts generated by each postulated break location at each size and orientation. The BADGER results were post-processed into a file suitable for use by NARWHAL. The post-processed file includes debris masses, size distributions, and debris volumes.

For low-density fiber glass (e.g., NUKON™), the licensee used a centroid model to estimate the debris size distribution (amounts of fiber fines, small fiber, large fiber, and intact fiber blankets). The centroid is the average distance from the break to the multiple insulation locations within the ZOI. Closer to the break, the debris produced is mainly fiber fines and small fiber pieces. Further from the break, the debris is mostly large pieces of fiber. Some insulation is considered to remain intact (within its protective cover) if far from the break, even if within the ZOI. The centroid methodology is a new method for estimating fibrous size distributions. Fractions or percentages for a four-category fiber size distribution (e.g., 10 percent intact fiber blankets, 20 percent large fiber pieces, 30 percent small fiber pieces, and 40 percent fiber fines) were defined by the licensee as functions of the centroid distance. The fractions were selected to be consistent with the NEI 04-07 guidance.

Because each strainer independently supplies its individual pump, the debris amounts predicted to transport to the strainer are based on the amount of debris in the sump pool that is available for transport and the relative flow through each pump. High-likelihood pump operating states were identified (e.g., all pumps are operational, or one or two pumps are out-of-service, one train out-of-service) and NARWHAL was executed with assumptions consistent with the pump state scenario. NARWHAL output conditional probabilities of strainer failure for the different pump operating states, which were then input to the PRA for the calculation of the ΔCDF. The licensee also addressed low-likelihood states with simplified or bounding calculations and demonstrated that those states would not significantly contribute to the ΔCDF.

The NRC staff asked RAI 19 (Reference 7) to ensure that the flow rate from breaks in the "out" locations would not result in LOCA flow rates. The licensee stated that although the locations are, or would be, isolated by check valves, the valves might leak at greater than the makeup flow rate. However, the licensee justified that the "out" locations were evaluated adequately.

The NRC staff concludes that the licensee properly quantified amounts of debris that could be generated within the Vogtle containments by the postulated LOCA breaks and the secondary breaks. Some debris types evaluated in the technical report were considered to be bounding values, and the amounts were the same for all breaks. For example, latent debris and unqualified coatings were assumed to have the same source term for all breaks. The debris generation model included detailed information necessary to calculate amounts of fiber, fire barrier, and qualified coatings debris within any break ZOI. The licensee computed debris amounts using BADGER. BADGER used CAD information to determine debris amounts for each break size of interest. The CAD model clipped the ZOI to account for robust barriers. The licensee also manually estimated the debris amounts for some breaks and compared these to

BADGER model results. There was satisfactory agreement between the two methods. The NRC staff concludes that algorithms in the BADGER code for the computation of debris were implemented properly and verified by the licensee. The licensee adequately considered random factors such as the jet orientation and identified debris amounts to compare to strainer tests. The NRC staff concludes that the licensee's methodology to calculate debris loads for each evaluated scenario is acceptable.

The NRC staff verified independently the ZOIs used for the secondary line breaks by comparing them with previous calculations for breaks in fluid systems at similar conditions. The ZOI sizes may appear to be nonintuitive based on the source pressures of the feed and steam lines. That is, the feedwater line pressure is higher, but two of the ZOIs calculated are slightly smaller than the steam ZOIs calculated for a lower pressure source. This is caused by the difference between the jet originating from a steam source or a subcooled liquid source. The NRC staff also evaluated the locations assumed for the secondary breaks and concludes that these locations were postulated in accordance with NRC staff guidance (Reference 32) and is, therefore, acceptable.

Use of a very small centroid distance in the analysis could under-predict the amount of small and fine fiber debris. Based on further staff review and a revision to the methodology, the NRC staff was able to conclude that the method provides an adequate estimation of size distribution for fibrous debris. The centroid method results in size distributions that are comparable to a similar methodology used by Indian Point Nuclear Generation and reviewed by the NRC staff as part of an audit of its GL 2004-02 closure ("Audit Report – Indian Point Nuclear Generating Unit Nos. 2 and 3 – Report on Result of Staff Audit of Corrective Actions to Address Generic Letter 2004-02," dated July 29, 2008 (Reference 57). The size distribution calculated using the centroid method is consistent with Appendix II of the NRC staff SE on NEI 04-07 and a calculation by Alion Science and Technology concerning insulation debris size distribution (Reference 58) for debris characterization. Both the SE and the proprietary report are based on industry testing that exposed various insulation types to air jets. The calculation has been used by several licensees to refine fibrous debris generation amounts, and its use has been accepted by the NRC staff.

<u>NRC Staff Conclusion Regarding Break Selection Debris Generation and ZOI Submodels</u>

The NRC staff notes that the licensee adopted guidelines in the NEI 04-07 report to: (1) define ZOIs; (2) account for robust barriers; (3) compute debris amounts of low-density fiber glass (LDFG), fire barrier material, and qualified coatings; (4) compute debris size distribution for LDFG into four categories; and (5) estimate debris amounts associated with latent fiber, latent particulate, and unqualified coatings. The NRC staff concludes the approach to compute the LDFG size distribution using the centroid methodology is acceptable.

The NRC staff verified that the licensee's debris generation calculations were performed accurately and used acceptable assumptions. The NRC staff used a combination of confirmatory calculations, engineering review, and review of the licensee's software outputs to perform the independent verifications.

The NRC staff reviewed the licensee's evaluation against the NRC staff-accepted guidance and concludes that the licensee adequately determined for each postulated break location, size, and orientation, the zone within which debris would be generated by the two-phase jet. The NRC staff also concludes that the amount and characteristics of debris predicted to be generated are acceptable. The licensee calculated amounts for all types of debris and compared these values

to the amounts of debris included in the 2009 strainer head loss test. Any break that results in any type of debris reaching the strainer exceeding the amounts in the test is assumed to lead to strainer failure and contribute to plant risk. Additionally, if the debris amount causes head loss at the strainer that results in another failure mode, or debris amounts in the vessel to exceed limits, the scenario would be considered to contribute to plant risk. However, the licensee demonstrated that failure modes other than debris limits would not contribute to plant risk. The licensee's methods are consistent with staff guidance. Therefore, the NRC staff concludes that the licensee's evaluation of the ZOI and debris generation is acceptable. The amounts of debris from each postulated break scenario were determined appropriately.

The NRC staff concludes that debris generation and ZOI analysis and methodology are acceptable because it identifies a number of postulated LOCAs of differing properties sufficient to provide assurance that the most severe postulated LOCAs are calculated. Also, the NRC staff concludes that the debris generation and ZOI submodel described in the technical report is acceptable for use in the plant-specific assessment or evaluation model of the effects of debris on long-term cooling of ECCS at Vogtle, as required, in part, by 10 CFR 50.46.

The NRC staff concludes that SNC has provided sufficient information as requested by GL 2004-02 (Reference 14) and further described in the revised content guide for GL 2004-02 (Reference 31) concerning the debris generation and ZOI, because the licensee:

- Described the methodology used to determine the ZOIs for generating debris. Identified which debris analyses used approved methodology default values. For debris with ZOIs not defined in the guidance report, or if using other than default values, discussed methods used to determine ZOI and the basis for each.

- Provided destruction ZOIs and the basis for the ZOIs for each applicable debris constituent.

- Identified if destruction testing was conducted to determine ZOIs. If such testing was not submitted previously to the NRC for review or information, described the test procedure and results with reference to the test reports.

- Provided the quantity of each debris type generated for each break location evaluated.

- Provided total surface area of all signs, placards, tags, tape, and similar miscellaneous materials in containment.

3.4.2.6.3 Debris Characteristics

The licensee included the following debris types in scope for its evaluation of debris characteristics: NUKON™, Interam E-50 fire barrier, and coatings. See Table 1 below for the as-fabricated densities, microscopic densities, and dimensions for the debris types as assumed in the Vogtle analysis.

Table 1: Debris Densities and Dimensions

Debris Material	As-Fabricated Density (lbm/ft³)	Microscopic Density (lbm/ft³)	Characteristic Diameter (μm)
Fibrous (Fine) Material Characteristics			
NUKON™	2.4	159	7
Fire Barrier Fiber 30% of total mass	54.3 (bulk)	175	1.5 50% fine, 50% small
Particulate Debris Characteristics			
Fire Barrier Particles 70% of total mass	54.3 (bulk)	151	10 100% fine particles
Qualified Coatings	N/A	IOZ – 208 Epoxy – 107	10 100% fine particles
Unqualified and Degraded Coatings	N/A	IOZ – 208 Epoxy – 109 Alkyd – 122	10 100% fine particles

The licensee used Nukon fiber, green silicon carbide powder, Interam E-54A fire barrier, and silica sand in the 2009 head loss testing. These materials were used to represent the fibrous and particulate debris constituents that may be generated during a LOCA. Nukon was used to represent Nukon and latent fiber. The Interam was used to represent the plant fire barrier material. Silica sand was used to represent latent particulate debris. Green silicon carbide was used to represent coatings.

The licensee did not attempt to use the microscopic debris characteristics to calculate head loss behavior. Therefore, the major debris physical characteristics important for head loss are the fiber density, which is used in the calculation of the mass of fiber arriving at the strainer, and the size distribution of the fiber after being damaged by a LOCA jet. Latent debris was added as fine fibers and an appropriate size distribution of particulates. Other important physical characteristics are the density and size distribution of the coatings.

For its risk-informed analysis, the licensee used a 17D ZOI for NUKON™, which is consistent with the NEI 04-07 guidance. The licensee then analyzed the 17D ZOI using the centroid methodology discussed in Section 3.4.2.6.2, "Debris Generation and Zone of Influence." The centroid distance was used to determine the size distribution of the NUKON™ that was damaged within the ZOI.

The licensee determined the debris characteristics of fire barrier material by using information from the material safety data sheet and performing tests to assess the material's resistance to damage from a high pressure water jet. Water jet test results were compared against similar tests for Temp-Mat (a type of insulation). As a result of the testing, the licensee assigned a ZOI of 11.7D to the material. This ZOI is the same as the NRC-approved ZOI for Temp-Mat. Within this ZOI, the material was assumed to fail as 50 percent fines and 50 percent small pieces. The licensee stated that the characterization of the fire barrier was conservative because it was observed to be significantly more resistant to fragmentation than Temp-Mat.

The NRC staff reviewed the licensee's evaluation of debris characteristics and determined that the information provided was consistent with the NEI 04-07 guidance and the associated NRC staff SE. The methodology used to determine fibrous debris sizes is also consistent with the NEI 04-07 guidance and is, therefore, acceptable.

The licensee's methodology for determining the characteristics for Interam fire barrier material was evaluated by the NRC staff and found to be acceptable based on high pressure water jet testing.

NRC Staff Conclusion Regarding Debris Characteristics

The NRC staff reviewed the licensee's evaluation against the NRC staff-accepted guidance and concludes that the licensee has appropriately characterized the debris for use in determining the transportability of debris and its contribution to sump strainer head loss. In addition, the debris surrogates used in the testing appropriately represented the debris that can be generated, or may be preexisting, in the plant. Therefore, the NRC staff concludes that the licensee's evaluation of NUKON™ and Interam E-50 fire barrier and coatings debris characteristics is consistent with the NRC SE on NEI 04-07, and, therefore, is acceptable.

The NRC staff concludes that SNC has provided sufficient information as requested by GL 2004-02 and further described in the revised content guide for GL 2004-02 concerning the debris characteristics. In the technical report, the licensee:

- Provided the assumed size distribution for each type of debris.
- Provided bulk densities (i.e., including voids between the fibers/particles) and material densities (i.e., the density of the microscopic fibers/particles themselves) for fibrous and particulate debris.
- Provided assumed specific surface areas for fibrous and particulate debris.
- Provided the technical basis for any debris characterization assumptions that deviate from NRC-approved guidance.

3.4.2.6.4 Latent Debris

The licensee stated that it evaluated latent debris in a manner consistent with NEI 04-07 (Reference 26). The licensee determined the total source term through the collection of debris samples from multiple locations in containment. The evaluation for latent debris divided containment into categories representing areas where debris would collect in similar amounts, and the licensee collected a minimum of three samples from each. Prior to collecting samples, the licensee surveyed containment through a series of walkdowns to locate desirable sample locations. Areas that exhibited unusually large concentrations of dirt and dust were sampled to add conservatism to the evaluation. The licensee also surveyed and documented foreign materials and other debris sources.

The licensee stated that it calculated the total amount of latent debris to be 60 lbm based on walkdown data. However, the licensee assumed 200 lbm in the debris generation calculation to conservatively bound the 60 lbm of actual latent debris to ensure ample operating margin.

The licensee assumed that the latent debris was 85 percent particulate and 15 percent fiber. Table 2 lists the latent fiber and particulate constituents and their material characteristics.

Table 2: Latent Fiber and Particulate Constituents

Latent Fiber and Particulate Constituents				
	Latent Debris (lbm)	Bulk Density (lbm/cubic foot (ft^3))	Microscopic Density (lbm/ft^3)	Characteristic Size (μm)
Particulate (85 percent)	170	-	168.6	17.3
Fiber (15 percent)	30	2.4	93.6	5.5

The licensee identified 2 ft^2 of miscellaneous material during its containment inspections. However, the NARWHAL analysis assumed that there was 50 ft^2 of miscellaneous material (considered to reduce the active strainer area). Use of the assumption of a greater amount of miscellaneous debris potentially overestimates strainer failure probability. The area blocked by miscellaneous debris is allocated to strainers based on which pumps are running during the scenario being evaluated.

The NRC staff reviewed the licensee's response and concludes that the approach is consistent with the guidance specified in NEI 04-07, including the NRC SE on that document. The licensee used default values for latent debris amounts, even though sampling of the containment found a lesser amount of debris.

NRC Staff Conclusion Regarding Latent Debris

The NRC staff reviewed the licensee's evaluation against the NRC staff-accepted guidance in NEI 04-07 and concludes that the licensee has appropriately identified the amounts and types of latent debris existing within the containment so that its potential impact on sump screen head loss and plant risk can be evaluated. Therefore, the NRC staff concludes that the licensee's evaluation of latent debris is acceptable.

The NRC staff concludes that SNC has provided sufficient information as requested by GL 2004-02 and further described in the revised content guide for GL 2004-02 concerning latent, because the licensee:

- Provided the methodology used to estimate quantity and composition of latent debris.
- Provided the basis for assumptions used in the evaluation.
- Provided results of the latent debris evaluation, including amount of latent debris types and physical data for latent debris as requested for other debris.
- Provided amount of sacrificial strainer surface area allotted to miscellaneous latent debris.

3.4.2.6.5 Coatings

The licensee's analysis assumed a spherical ZOI of 4D for qualified coatings based on WCAP-16568-P (Reference 59). The licensee stated that all of the coatings destroyed within the ZOI are assumed to fail as fine particulate. In addition, 100 percent of the unqualified coatings inside containment is assumed to fail as fine particulate. Silicon carbide particulate was used as a surrogate for coatings debris in head loss testing.

The NRC noted that the coatings source term applied to head loss testing was underestimated due to an improper density correction. In RAIs 24 and 35 (Reference 7), the NRC staff also raised questions regarding the transport of coatings. The licensee stated the unqualified coatings were originally modeled to transport 100 percent to the strainer, although the submittal incorrectly showed some lower transport values in the transport section. The licensee stated that values less than 100 percent in the initial submittal were in error. The licensee implemented a proper density correction for the coatings. However, this, along with the assumption of 100 percent transport of the unqualified coatings, would have resulted in many scenarios exceeding the tested value for coatings. To show that the tested coating values bound the majority of break scenarios, the licensee changed the transport assumption so that 10 percent of the coatings in upper containment transport when CSs are not actuated. This assumption is allowed by NRC staff guidance. For the no spray cases, unqualified coatings in upper containment were assumed to transport to the strainer at a rate of 10 percent, while those in lower containment were assumed to transport at a rate of 100 percent. For cases where sprays are modeled to actuate, 100 percent transport of unqualified coating is assumed.

The licensee stated that coating condition assessments are conducted every outage. As localized areas of degraded coatings are identified, they are evaluated and scheduled for repair as necessary. These periodic condition assessments and repairs minimize the amount of coatings that may be susceptible to detachment from the substrate in the event of a LOCA.

<u>NRC Staff Conclusion Regarding Coatings</u>

The licensee provided information such that the NRC staff has concluded that coatings have been addressed conservatively or prototypically. Analysis and testing were performed in a manner consistent with NRC staff review guidance (References 31 and 32). Head loss testing surrogate materials were representative of the size and shape of the actual plant coatings debris. The licensee's coatings assessment program will identify and mitigate any degraded coatings prior to them becoming a significant debris source. The program prevents coating failures from challenging the margins in the strainer analysis. Therefore, the NRC staff concludes that the coatings evaluation described in the technical report enclosed with the letter dated July 10, 2018, is acceptable.

The NRC staff concludes that SNC has provided sufficient information as requested by GL 2004-02 and further described in the revised content guide for GL 2004-02 concerning coatings (i.e., coatings evaluation), because the licensee:

- Provided a summary of type(s) of coating systems used in containment.
- Described and provided bases for assumptions made in post-LOCA paint debris transport analysis.
- Discussed suction strainer head loss testing performed as it relates to both qualified and unqualified coatings and what surrogate material was used to simulate coatings debris.
- Provided bases for the choice of surrogates.
- Described and provided bases for coatings debris generation assumptions.

- Described what debris characteristics were assumed.

- Described any ongoing containment coating condition assessment program.

3.4.2.6.6 Containment Material Control

The licensee stated that it has "Containment Exit Inspection" and "Containment Entry" procedures to ensure no loose debris is present in the containment that could be transported to the containment sump and cause restriction of pump suctions during LOCA conditions. The licensee stated that the "Containment Exit Inspection" contains an extensive checklist detailing all areas of containment that must be inspected for cleanliness prior to plant startup after each outage. The licensee's "Foreign Material Exclusion Program" establishes the administrative controls and personnel responsibilities for the foreign material exclusion program.

The licensee stated that the design input process was enhanced by adding screening guidelines and considerations for the design input process. The screening reviews the impact of a proposed change on the design basis for the response to GL 2004-02. The licensee stated that the following specific areas are addressed:

- Insulation inside containment
- Fire barrier material inside containment
- Coatings 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 strainers)

The licensee stated that inclusion in the design input process ensures all design changes consider these attributes during the design process.

The licensee stated that maintenance activities, including temporary changes, are subject to the provisions of (a)(4) of 10 CFR 50.65, as well as Vogtle's TSs. The licensee also stated that SNC fleet procedures provide guidance, and further guidance is available in the temporary configuration change procedure.

The licensee removed all of the Min-K insulation located inside the steam generator compartments in Units 1 and 2 containments during refueling outages 1R13 (Fall 2006) and 2R12 (spring 2007), respectively.

The licensee stated that although no specific actions were taken to modify or improve the containment coatings program, enhancements were made to the screening guidelines and considerations for the design input process to ensure all design changes consider GL 2004-02 attributes during the design process, as discussed earlier in this section.

The lack of control within containment of sources of calcium and aluminum are potential contributors to chemical effects. In response to RAI 31 (Reference 7), the licensee stated that its design control procedures require evaluation of the impact of the changes of materials in containment. Additionally, the containment aluminum inventory is tracked in a calculation. The NRC staff concluded that the materials are controlled adequately.

The NRC staff reviewed the licensee's evaluation and concludes that the licensee has significant design and operational measures in place to control or reduce the plant debris source term. Therefore, the staff concludes that the licensee's evaluation of containment material control is acceptable.

NRC Staff Conclusion Regarding Debris Source Term Submodel

Each of the aspects of the debris source term area has been evaluated above. The NRC staff concluded that the debris source term submodel, including break selection, debris generation and zone of influence, debris characteristics, latent debris, coatings, and containment material control were addressed adequately. Based on the evaluations for each of these subsections, the NRC staff concludes that the debris source term evaluation is acceptable.

The NRC staff concludes that debris source term submodel (incl., break selection, debris generation and zone of influence, debris characteristics, latent debris, coatings, and containment material control) is acceptable because it identifies a number of postulated LOCAs of differing properties sufficient to provide assurance that the most severe postulated LOCAs are calculated. Also, the NRC staff concludes that the debris source term submodel described in the technical report is acceptable for use in an assessment or evaluation model of the effects of debris on long-term cooling of ECCS, as required, in part, by 10 CFR 50.46.

The NRC staff concludes that SNC has provided sufficient information as requested by GL 2004-02 and further described in the revised content guide for GL 2004-02 concerning the debris source term, because licensee:

- Described recent or planned insulation change-outs in the containment which will reduce the debris burden at the sump strainers.
- Described actions taken to modify existing insulation (e.g., jacketing or banding) to reduce the debris burden at the sump strainers.
- Described modifications to equipment or systems conducted to reduce the debris burden at the sump strainers.
- Described actions taken to modify or improve the containment coatings program.

3.4.2.7 Debris Transport Submodel

For this section, all descriptions attributed to the licensee's submittal are taken from the letter dated April 21, 2017, unless otherwise noted. The majority of the information is from Section 3.e (Debris Transport) of Enclosure 5. Unless otherwise noted, the RAIs discussed in this section were transmitted to the licensee by NRC letter dated January 11, 2018 (Reference 60). The licensee's responses were transmitted to the NRC by letter dated February 12, 2018 (Reference 7).

The licensee performed debris generation and transport calculations to determine the amounts and types of debris that could arrive at the emergency core cooling system (ECCS) and CS strainers. These amounts are used to determine when failures of the strainer may occur by comparing the transported values to the debris amounts included in the strainer test. Cases that result in a debris amount of any type exceeding the amount included in the test are considered to result in a strainer failure and add to plant risk. If the transported amounts do not exceed the specific debris amounts included in the test, the licensee assigned a head loss value to the debris amount based on testing. The head loss was then used to calculate NPSH margins and strainer structural margins. If these margins were negative, the scenario would be assumed to result in core damage and counted toward plant risk. However, the analyses the licensee developed did not result in any case of negative margins, with a few exceptions related to scenarios with a full RHR-CS train out of service. The transport calculation also determines the amount of fiber that may pass through the strainer and collect at the reactor vessel inlet or in the reactor core.

A nuance to the Vogtle design and analysis is that each pump has its own strainer. The debris is assumed to transport to each strainer based on the flow rate of the associated pump for the scenario being evaluated. The CS strainers have a lower flow rate than the RHR strainers, but the CS strainers are smaller. However, the ratio of flow to strainer area is greater for the RHR pumps; thus, they reach debris limits faster than the CS strainers. The CS strainers are assumed to have equal flow rates to each other as are the RHR strainers. Accordingly, for scenarios where two trains are operating, if one CS or RHR strainer is calculated to fail due to debris effects, the other one will fail at the same time because of the symmetry assumed in debris deposition. The CS pumps are assumed to start only on hot-leg breaks greater than 15 inches. Washdown transport is increased for cases where the CS pumps start, but the total amount of debris is split between the CS and RHR pumps instead of it all going to the RHR pumps. These and other assumptions and methods used, and the large number of breaks evaluated, make the transport calculation complex, even though accepted NRC guidance (Reference 27) was used to develop the transport model.

The licensee considered three types of debris sources: (1) debris directly generated by postulated breaks in the RCS, (2) latent debris already present in the containment structure prior to any break, and (3) protective coatings used inside the containment that could become debris. The licensee reduced the source debris by estimating the amount that could be trapped in inaccessible locations or settled out during transport to the strainer. The licensee also estimated the amount of debris arriving at the strainer due to erosion of larger debris (that was assumed to settle prior to reaching the strainer) into fine pieces during transport. The licensee also assumed that small and large fibrous debris predicted to reach the strainer eroded into fine pieces in the same proportion as debris settled in the pool.

The licensee used debris transport calculations for two separate purposes. The first was to calculate the amount of debris that reaches the strainer from each postulated break scenario. The second was to calculate the amount of fiber that could transport downstream of the strainer. Debris that penetrates the strainer can transport to the core and other downstream components and may affect the ability of systems to cool the core.

The four modes of transport are blowdown (transport of debris by the break jet), washdown (vertical transport by CSs and break flow), pool fill-up (horizontal transport to active and inactive areas of the sump pool), and recirculation (horizontal transport in the active portions of the sump pool by recirculation flow). The licensee analyzed the specific effect of each of the four modes of transport for each type of debris generated. The licensee applied the logic tree approach

recommended by NEI 04-07 to determine transport fractions for each type of debris determined from the debris generation calculation. In order to calculate the many scenarios evaluated as part of the licensee's risk-informed approach, the logic trees were implemented in the NARWHAL model.

For the 2009 testing, the licensee assumed all particulate debris as fines prior to transport; fines are the most readily transported particulate size. The licensee included small and fine fiber in its testing as well.

In Enclosure 5, page E5-33 of the letter dated April 21, 2017, the licensee provided the basic methodology used for the transport analysis. The licensee also provided a diagram showing the significant parts of the computational fluid dynamics (CFD) model used to evaluate recirculation transport. The diagram highlights the sump mass sinks, mass sources, and various direct runoff and spray regions.

The licensee used the containment CAD model to determine break locations and to evaluate breaks of various sizes and orientations. The licensee used Flow-3D®, a commercial software package, for the CFD modeling for the recirculation phase of transport. The key CFD modeling attributes/considerations included computational mesh, modeling of CS flows, break flow, and emergency sump flows. Turbulence modeling, steady-state metrics, and debris transport metrics were also included. The licensee performed a graphical determination of debris transport fractions and determined the percentage of each type of debris that could be expected to transport through the containment pool to the strainers. The licensee provided example plots showing the turbulent kinetic energy and velocity magnitude in the pool to determine in which areas specific types of debris would be transported. The licensee also provided figures and discussions as an example of how the transport analysis was performed for a generic small debris type. This example was illustrative of how all debris types were evaluated with respect to transport.

The licensee described how the potential for erosion of larger pieces of material into fines was evaluated. Based on the results of the Drywell Debris Transport Study (DDTS) (NUREG/CR-6369) testing (Reference 61), a 1 percent erosion factor was applied for small and large piece fiber debris held up in upper containment. This is consistent with the approach described in Appendix VI of the staff SE on NEI 04-07 (Reference 26).

The erosion mechanism for debris in the pool is somewhat different than what was tested in the DDTS. Assumptions concerning the erosion of small and large pieces of fiber used in the low-density fiberglass erosion testing conducted by Alion Science and Technology are described in the test report that document the results and analysis of the erosion tests (Reference 62). The NRC staff reviewed and developed conclusions regarding this report that are documented in a letter dated June 30, 2010 (Reference 63). The NRC staff concluded that plants that could demonstrate the testing was conducted under conditions that represented or bounded their plant could assume a 30-day erosion value of 10 percent for fiber settled in the sump pool. The licensee provided the information necessary to show that the specified erosion rate is applicable to Vogtle.

The licensee stated the following with respect to the transport evaluation:

- Debris interceptors are not credited in the debris transport analyses.
- Debris settling is not credited for fine debris in the debris transport analyses.

The licensee provided tables of transport fractions that were developed based on an example scenario being evaluated. Tables for blowdown, washdown, and recirculation were provided. The blowdown fractions for break generated fine debris were based on the ratio of the volumes of upper and lower containment. All sizes of debris blown to lower containment were assumed to be deposited directly in the sump pool (no holdup). The licensee stated that all debris that transports to lower containment during blowdown or washdown is assumed to be in the pool. The NRC staff concludes that this is conservative because it results in all debris having the potential to transport to the strainers.

The distribution of small and large debris due to blowdown was calculated based on guidance in Appendix VI of the staff SE on NEI 04-07. The licensee stated that the transport evaluation included simplifications. The fine debris was assumed to transport, during blowdown, to upper and lower containment based on their relative volumes. Fine debris was assumed not to be held up during blowdown, even though a significant amount could be trapped on surfaces due to impaction. The licensee stated that all fine debris calculated to transport to lower containment was placed directly in the sump pool and that a significant amount of this debris would actually be trapped on structures above the pool. The licensee stated that small and large debris was also assumed to transport to upper and lower containment in proportion to these containment volumes. However, the licensee considered the break location (compartment) and reduced the amount of small and large debris that transported to upper containment by assuming inertial capture of some of the debris. Debris assumed to be captured in its break compartment was assumed to be in the pool and available for recirculation transport. One exception is that for breaks in the pressurizer compartment, debris trapped in the compartment was assumed to be in upper containment at the end of blowdown instead of the sump pool. The licensee stated that the methodology used for the Vogtle blowdown transport was consistent with the approach used for the volunteer plant in Appendix VI of the staff SE on NEI 04-07. The licensee's simplifications were similar to those used to estimate transport in Appendix VI of the staff SE on NEI 04-07. The methodology includes adequate conservatism to offset uncertainties introduced by the simplifications. The NRC staff concludes that the methodology used for the blowdown analysis was consistent with staff guidance.

The NRC staff noted that the overall latent debris transport fractions in Tables 3.e.6-7 through 3.e.6-14 were not consistent. The licensee responded that the original methodology assumed that 75 percent of the latent debris was resident in upper containment, which resulted in reduced latent debris transport when CS is not initiated. The licensee stated that the NARWHAL model actually assumed full transport for latent debris, independent of its initial location. Although it is estimated that 75 percent of the latent debris is in upper containment, the transport model assumes that all of the debris is washed down to the containment floor regardless of the status of CS. The licensee could have assumed that only 10 percent of the latent debris in upper containment washes down when sprays are not initiated. The assumption that all latent debris washes down is conservative.

The washdown evaluation was significantly affected by the assumptions associated with the operating state of the CS. The CS was assumed to operate only for hot-leg breaks greater than 15 inches. For other breaks, CS was assumed to remain off, and thus, a significant amount of debris remained in upper containment and was not washed into the pool. This assumption was based on a realistic thermal hydraulics analysis that determined that the CS initiation setpoint would not be reached for any cold-leg breaks or any hot-leg breaks less than 15 inches. The licensee performed a sensitivity study to show that even if sprays were initiated for all medium and large breaks, the added transport would not significantly increase risk. The initiation of the CS by an operator for small breaks could have a significant impact on debris transport. The

licensee stated that EOPs do not direct operators to start CS following an SBLOCA and that it would be unlikely for operators to start the sprays because the release in fission products is anticipated to be minimal for these scenarios. The licensee also stated that even if an SBLOCA were to occur and the sprays were initiated, the debris amounts would be so small that the strainer operation would not be affected.

The licensee assumed that a small amount (i.e., 2 percent) of debris was held up in the elevator cavity during pool fill and that an additional small amount of debris was washed directly to the strainers. Unqualified coatings were not credited to be held up because they would likely fail after the pool fill phase of transport.

The licensee ran seven cases for recirculation transport. The results varied, depending on the number of pumps in service, whether CS was operating, and the break location. Cases 3 and 5 were not used because their transport fractions were bounded by Cases 2 and 1, respectively. The licensee stated that the difference between Cases 1 and 5 was the water level, but noted that the transport results for both cases were identical. Cases 2 and 3 were different breaks in the annulus. Case 2 resulted in higher transport fractions and was used for all annulus breaks. The licensee stated that the debris locations at the start of recirculation were determined based on the compartment in which the break occurred for the debris remaining in lower containment. The debris blown to upper containment was assumed to be near the locations where it could be washed down. The licensee provided figures to show the debris distribution in the pool at the start of recirculation and areas where the debris was assumed to wash down. The NRC staff reviewed the figures and found the distribution reasonable. The licensee provided a table that clarified which CFD cases were applied to each scenario (break location, number of trains operating, sprays on or off). The licensee also clarified that CFD steady state conditions were used for the recirculation transport evaluation.

During its initial review, the NRC questioned why some recirculation cases resulted in different amounts of debris reaching the RHR and CS pumps, considering that the debris was assumed to transport to the strainers based solely on the pump flow and assuming homogeneous debris mixing in the pool. The question arose because, in part, tables for overall debris transport in the submittal stated that different fractions of debris were predicted to arrive at strainers with identical flow rates for some break scenarios. During the audit from October 24, 2017 through March 26, 2018 (Reference 21), the staff learned that the tables in the submittal represented an intermediate step in the transport evaluation and that once the total debris transport amount for all strainers was determined, the total was split between strainers based on flow rate and flow history.

For recirculation cases, the licensee considered breaks in the annulus, the reactor cavity, a steam generator compartment near to the strainers, and a steam generator compartment distant from the strainers. The cases also considered whether sprays were operating and the number of ECCS trains operating.

The licensee originally assumed that 100 percent of unqualified coatings transported to the strainers. As discussed in Section 3.4.2.6.5, "Coatings," the source term added to the head loss testing was underestimated due to an improper density correction. The NRC staff also had questions regarding the transport of coatings because the description in the transport section of the submittal did not match information provided in Tables 3.e.6-7 through 3.e.6-14. IN response to RAI 24, the licensee stated the unqualified coatings were originally modeled to transport 100 percent to the strainer, although the submittal incorrectly showed some lower transport values in tables. The licensee stated that values in tables less than 100 percent in the

initial submittal were in error. The licensee implemented a proper density correction for the coatings. However, this, along with the assumption of 100 percent transport of the unqualified coatings, would have resulted in many scenarios exceeding the tested value for coatings. To show that the tested coating values bound the majority of break scenarios, the licensee changed the transport assumption so that only 10 percent of the coatings in upper containment transport of sprays are not actuated. This assumption is consistent with NEI 04-07, which allows a 10 percent value for fine debris washed down by condensation with no spray actuation. For scenarios where sprays actuate, the licensee assumed 100 percent transport of coatings. The model for coatings transport is in accordance with staff guidance (Reference 32). However, washdown sensitivity studies were conducted to assure that CS status does not significantly affect plant risk. In the letter dated July 10, 2018, the licensee provided the results of sensitivity studies for washdown and re-performed other studies using updated assumptions regarding coatings transport. The effect on the sensitivity studies and on overall risk was found to be insignificant.

The licensee also calculated the load-dependent rate of fiber passing through the sump strainers for each of the potential LOCA events based on the calculated rates of fine fiber arrival at the strainer. The licensee provided the methodology used to calculate the amount of fiber that could pass downstream of the strainer. The licensee used fiber penetration testing to develop a model of fiber penetration through the strainer over time. The testing was performed using a prototypical strainer module under prototypical plant flow conditions. The test program varied the number of disks in the strainer to determine the effects of this parameter on penetration. The velocity through the strainer surface was controlled by adjusting the flow rate to the strainer area. The debris batch sizes were also adjusted to account for variations in the strainer area. The debris amounts added during testing were split into batches. The initial batches were relatively small to ensure that fiber concentration was not so high that penetration would be inhibited by early formation of a filtering bed. Batch sizes were later increased.

During penetration testing, the licensee varied the flow velocity through the strainer, the water chemistry, and the number of strainer disks. Ten penetration tests were run. The licensee used the penetration values from the testing to develop a mass balance model. The model calculates the amount of fiber at various locations throughout the system at each time step. The model was implemented in NARWHAL. For example, the model tracks the amount of fiber in the pool, deposited on the strainer, and in the reactor vessel. The model accounts for flows through the ECCS and CSS and estimates the amount of fiber in each flow path. The model ultimately calculates the amount of fiber that reaches the core on a time-dependent basis. The licensee performed sensitivity studies to determine the effect of penetration fractions on risk. One study varied the base-case penetration fraction by +/- 25 percent; the study found that the effect on risk was negligible. The sensitivity study results were nonintuitive because risk was increased both by increasing and decreasing the penetration fraction. The licensee explained that this is possible because increasing the penetration fraction increased in-core loading (leading to core blockage), while decreasing penetration increased strainer loading (increasing the probability of strainer failure). More debris in either location (core or strainer) can result in different failure causes, increasing the plant risk. The licensee also performed a parametric sensitivity study that included cases with zero penetration. The licensee considered simultaneous variation of other parameters. In all cases, the change in risk for the zero penetration cases was insignificant.

The licensee stated that the penetration model is based on the amount of fine fiber that arrives at the strainer and the testing included only fine fiber. The mass of fiber that bypassed the strainer and collected in filter bags was used to develop a model of the amount of fiber that

penetrates the strainer immediately upon arrival (referred to as prompt penetration) and that which is slowly released from the bed (referred to as shedding) and passes downstream of the strainer. The empirical equations of the model were based on nine separate tests and predict higher penetration amounts than any single test. The licensee provided a plot to illustrate that the cumulative fiber penetration is overestimated with respect to the measured bypass from the tests. The licensee provided sensitivity analyses, including a case of zero penetration (case that would increase the conditional strainer failure probability) and concluded that variations to the extent of penetration only marginally affected risk estimates.

The NRC staff reviewed the penetration test methodology, test results, and application of the results in the mass balance model. The staff found that the test methodology would overestimate the amount of fiber that could penetrate the strainer. For the penetration testing, the licensee varied the test conditions to determine how the input parameters affected penetration. The licensee used parameter values, resulting in maximum penetration to develop the penetration model. The mass balance model was reviewed in detail by the NRC staff during an audit in October 2017. The NRC staff created an independent fiber penetration model based on an independent interpretation of the test results and concluded that the licensee acceptably represented the test data in the NARWHAL model. The NRC staff observed that the NARWHAL methodology provided reasonable predictions of the amounts of fiber that can reach the core. The NRC staff considered the different scenarios and assumptions used by the licensee and verified that the model produced reasonable results. Therefore, the NRC staff concludes that the licensee's calculations for fiber penetration through the strainer and transport to the reactor core were performed acceptably.

The licensee provided the maximum amounts of fibrous debris predicted to reach the core following a hot-leg and cold-leg break on a per fuel assembly basis. The licensee stated that the amounts of fiber that reach the core are calculated to be less than the assumed acceptance criteria.

The licensee used debris transport calculations to compare predicted debris loads against debris amounts that were used in the 2009 strainer head loss testing. Debris transport amounts were calculated for each weld break scenario. If, for any weld break scenario, debris type is calculated to exceed the tested amount of that debris, the strainer is considered to fail and contribute to plant risk. Other parameters that affect strainer and ECCS pump performance are also calculated based on debris amounts (along with other parameters) to determine if ECCS performance may be affected. If NPSH margins become negative, structural margins are exceeded, flashing across the strainer occurs, or excess void fractions are predicted to reach the pumps, and the strainer is also considered to fail in the NARWHAL model.

The NRC staff noted that the fiber penetration model was developed to bias penetration towards high values, thus decreasing the amount of fiber on the strainer. The staff also noted that strainer failures contribute to increases in plant risk to a greater extent than in-vessel failures. Based on the above, the staff determined that using a model with lower penetration could result in increased risk by increasing strainer failures. As described above, sensitivity studies were performed to show that the penetration model did not significantly affect the overall risk.

The NRC staff finds the licensee's methodology for calculating the transport amounts to individual strainers based on CFD simulations and summing the total amounts of debris mobilized is acceptable. The NRC staff also finds the use of the transport fractions based on the CFD simulation as inputs to NARWHAL, and allowing NARWHAL to independently compute strainer debris buildup amounts based on strainer flow rates and uniform debris mixing in the

pool to be acceptable. The staff concludes that this simplifying step in the model was acceptable because the pool fill portion of the process would mix the debris in the pool due to high pool velocities and turbulence. Also, the prediction of the locations of debris following blowdown has significant uncertainty. This uncertainty, along with the mixing due to pool fill, makes the prediction of debris locations highly uncertain, if not impossible. The licensee stated that equal collection of debris on RHR strainers leading to simultaneous failure is conservative, compared to cases of uneven debris distribution. The NRC staff finds this conclusion is acceptable, given the licensee analyses demonstrating that one functional RHR strainer is sufficient to prevent core damage.

The NRC staff noted that one break location that resulted in a failure of the strainer generated and transported less fiber than other breaks that did not result in failures. The licensee stated that although the break did not exceed the fiber limit, the scenario resulted in a calculation of calcium phosphate chemical precipitate greater than the tested amount. The break in question resulted in a greater amount of non-transportable fibrous debris that remained in the pool but not on the strainer. The debris in the pool contributed to chemical effects but not strainer fiber loading. The NRC staff concludes that this is acceptable because it reflects the expected physical behavior of the system.

The NRC staff evaluated the licensee's debris transport calculations by comparing the licensee's methodology with appropriate guidance and comparing the licensee's assumptions with their intended purpose. The NRC staff used calculations to verify that the licensee's transport analyses were performed accurately and used acceptable assumptions. The licensee provided combined overall transport fractions that indicate reasonable reduction in risk due to limited transport of debris types (transported fractions in NARWHAL ranged from 10 to 28 percent for fiber, particles, and qualified coatings in the case where the CSS remains off, and 100 percent for fine fibrous and particulate debris types. In the case of small Nukon fiber, 63 percent was transported for cases where the CSS is activated for very large hot-leg breaks. Using these overall transport fractions, the NRC staff confirmed consistency of debris amounts computed using BADGER with debris amounts tracked in NARWHAL (Reference 22). The NRC staff noted an imbalance in the amounts of fire barrier fiber, and the licensee addressed the mismatch in a revised version of NARWHAL. The NRC staff used a combination of confirmatory calculations and engineering reviews to perform the verifications that allowed the NRC staff to conclude that the calculations for transport and debris penetration were conducted and applied properly.

NRC Staff Conclusion Regarding Debris Transport Submodel

The licensee's approach to evaluating debris transport was based on NEI 04-07 guidance (Reference 26) and the accompanying NRC staff SE, and the licensee provided information requested in the Content Guide (Reference 31).

The NRC staff reviewed the licensee's transport evaluation against the NRC staff accepted guidance in NEI 04-07, performed confirmatory calculations for debris penetration, and verified the consistency of debris amounts computed by BADGER and tracked in NARWHAL. The NRC staff concludes that the licensee appropriately estimated the fraction of debris that would transport from debris sources within containment to the ECCS strainers and the amount of fibrous debris that may penetrate the strainers and transport to the reactor core. Therefore, the NRC staff concludes that the licensee's evaluation of debris transport is acceptable.

The NRC staff concludes that the debris transport submodel described in the technical report is acceptable for use in an assessment or evaluation model of the effects of debris on long-term cooling of ECCS, as required, in part, by 10 CFR 50.46.

The NRC staff concludes that SNC has provided sufficient information as requested by GL 2004-02 and further described in the revised content guide for GL 2004-02 concerning debris transport, because the licensee:

- Described the methodology used to analyze debris transport during the blowdown, washdown, pool-fill-up, and recirculation phases of an accident.

- Provided the technical basis for assumptions and methods used in the analysis that deviate from the approved guidance.

- Identified any computational fluid dynamics codes used to compute debris transport fractions during recirculation and summarize the methodology, modeling assumptions, and results.

- Provided a summary of, and supporting basis for, any credit taken for debris interceptors.

- Stated whether fine debris was assumed to settle and provide basis for any settling credited.

- Provided the calculated debris transport fractions and the total quantities of each type of debris transported to the strainers.

3.4.2.8 Impact of Debris Submodel

This section evaluates the potential effects that the debris, as described in Section 3.4.2.6, "Debris Source Term," may have on operation of equipment important to LTCC. This section looks at the operation of the ECCS strainers, the ECCS and CSS pumps, and other equipment downstream of the strainer, including the fuel and vessel. This section also evaluates the potential for the holdup of water and debris in containment such that it may not reach the sump pool.

For this section, all descriptions attributed to the licensee's technical report are taken from the licensee's letters dated April 21, 2017; February 12, 2018; and January 2, 2018, unless otherwise noted. The majority of the information is from Sections 3.f, 3.g, 3.j, 3.k, 3.l, 3.m, 3.n, and 3.o of Enclosure 5 of the licensee's letter dated April 21, 2017.

3.4.2.8.1 Upstream Effects

The licensee considered the flowpaths in the refueling cavity, inside the secondary shield wall, and those associated with CS washdown as part of its evaluation to determine potential choke points for flow upstream of the sump. The licensee stated that for the refueling cavity, evaluations of containment, along with review of the CFD model, indicated no significant areas would become blocked with debris and hold up water during the sump recirculation phase. The licensee stated that the flow path from a break inside the secondary shield wall to the sump strainers is primarily through two labyrinth-like walkways in the shield wall, which provide a large clear flow path from inside the shield wall to the strainer location. The licensee stated that CS

washdown has a clear path to the containment sump area and that large sections of the floor on each level in containment are covered with grating that allows the water to pass. The licensee stated that for CS flow to the sump, a complete evaluation of containment, along with review of the CFD model, indicated no significant areas would become blocked with debris and hold up water during the sump recirculation phase.

The licensee stated that no measures are necessary to mitigate potential choke points. The licensee stated that there are no curbs or debris interceptors that provide water volume holdup in the containments.

The licensee stated that the refueling cavity drains via two horizontal 12 inch pipes. During refueling, these drains are secured by installing flanges, which are then removed prior to entry into Mode 4. The limiting break with respect to the potential for upstream flow blockage occurs under the operating deck and inside the secondary shield wall. The break would require large debris to travel a torturous path to transport above the operating deck and land in the refueling cavity. Large debris would have to travel through the RCP access ports or the steam generator and pressurizer cubicles. The licensee stated that the RCS access ports are covered with grating, and there are significant structural elements that would prevent any large pieces of debris from entering upper containment. The same is true for the path through the steam generator and pressurizer cubicles, where each cubicle contains several levels of grating and significant structural elements that would make large pieces of debris entering upper containment via these paths highly unlikely. The licensee thus stated that the possibility of clogging the refueling cavity drains is minimized.

Breaks other than those under the operating deck and inside the secondary shield wall could result in the transport of debris to the refueling cavity that might block the drains. The licensee stated that the refueling cavity has two 12-inch drains that are unlikely to become blocked due to their large diameter. The licensee also stated that there are numerous levels of grating that would prevent the transport of large debris pieces to the refueling cavity. The only piping that does not have significant grating above it is at the top of the pressurizer. The licensee stated that the insulation in the vicinity of this break is on the pressurizer and below the break. Therefore, the debris would be blown downward, and minimal large debris would transport out of the compartment. The licensee provided a figure that illustrated the grating in containment that is credited to prevent large piece transport to the refueling cavity. The NRC staff concluded, based on the size of the drain lines, the grating between almost all break locations and the refueling cavity, and the relative position of the piping at the top of the pressurizer and any debris sources, that blockage of the refueling canal drains will not occur.

The licensee stated that drains into the lower reactor cavity could become blocked. For breaks outside of the reactor cavity, there is no detrimental impact of this blockage, as it would inhibit loss of water from the active ECCS sump to an inactive area beneath the vessel. The licensee stated that the flooding analysis assumes this area floods during the event. For breaks inside the reactor cavity, there is also no detrimental impact of this blockage, since the majority of the flow from the break would travel to the ECCS sump through the hot-leg and cold-leg penetrations.

NRC Staff Conclusion Regarding Upstream Effects

The NRC staff reviewed the licensee's evaluation against the NRC staff-accepted guidance (Reference 32) and concludes that the licensee has appropriately evaluated the flow paths upstream of the containment sump for holdup of inventory that could reduce flow to the sump and possibly starve the pumps that take suction from the sump. Therefore, the NRC staff concludes that the licensee's evaluation of upstream effects is acceptable.

The NRC staff concludes that SNC has provided sufficient information as requested by GL 2004-02 and further described in the revised content guide for GL 2004-02 concerning upstream effects, because the licensee:

- Summarized the evaluation of the flow paths from the postulated break locations and containment spray washdown to identify potential choke points in the flow field upstream of the sump.

- Summarized measures taken to mitigate potential choke points.

- Summarized the evaluation of water holdup at installed curbs and/or debris interceptors.

- Described how potential blockage of reactor cavity and refueling cavity drains has been evaluated, including likelihood of blockage and amount of expected holdup.

3.4.2.8.2 Screen Modification Package

The licensee stated that 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 separate strainers, one for each RHR pump and one for each CS pump. Each of the two RHR strainer assemblies provide 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 licensee stated that it will modify the RHR strainers 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. The licensee stated that the new strainer design does not involve backflushing or any other active approach.

The licensee stated that all of the analyses shown in the letter dated April 21, 2017, were performed for the modified strainer configuration. The licensee also stated that it is revising operating procedures, in addition to the planned physical modification, to ensure that the RHR strainers are completely submerged for an increased number of postulated LOCA scenarios.

The licensee stated that the following modifications were, or will be implemented, as part of the GL 2004-02 modifications:

- Installed new and replaced existing ECCS flow orifices to allow new ECCS throttle valve settings.
- Removed cage assembly vortex suppressors in the sumps.

- Replaced and relocated temperature elements for the Units 1 and 2 RHR sumps.
- Rerouted two conduit interferences at the Unit 2 RHR sump Train A screen through an area outside of the sump screen envelope.
- Relocated/rerouted three electrical interferences for the new Unit 2 CS sump Train A screen through an area outside of the sump screen envelope.
- Will reduce height of RHR strainers by removing two disks from each stack.

NRC Staff Conclusion Regarding Screen Modification Package

The NRC staff reviewed the design changes made by the licensee in response to GL 2004-02. The information from the design was appropriately included in the licensee's submittals. Based on its review, the NRC staff finds the licensee has provided sufficient information, as requested by GL 2004-02, and used appropriate inputs for its evaluation of LTCC, considering the effects of debris. The licensee:

- Provided a description of the major features of the sump screen design modification.

- Provided a list of any modifications, such as reroute of piping and other components, relocation of supports, addition of whip restraints and missile shields, etc., necessitated by the sump strainer modifications.

3.4.2.8.3 Head Loss and Vortexing

In 2009, the licensee performed a strainer head loss test, with chemicals, using NRC-approved guidance for testing (Reference 32). The licensee provided the head loss test methodology and results in its technical report. The testing resulted in head loss values that allowed the licensee to demonstrate that ECCS and CSS performance would not be adversely affected by the debris that might be generated and transported during a LOCA, as long as that debris amount is bounded by the amounts used in the test. The strainer test provides acceptance criteria for the amount of debris that can arrive at the strainer and allow the strainer to perform its design function. Any break or initiating event that generates and transports more debris of any type than that included in the testing is considered to result in failure and contribute to plant risk. Breaks that generate less than the debris limits are evaluated against head loss values, also determined by the test program, to determine if other failure modes may occur. The other failure modes are NPSH margin, the occurrence of flashing at the strainer, excessive void fraction at the pump suctions (caused by degasification due to the pressure drop through the debris bed), and structural failure. If any of these modes result in a failure, the scenario is also assumed to contribute to plant risk.

Although the test was performed using acceptable methods, the amount of some debris types included in the test was much less than what could be generated and transported during a worst case LOCA. The reason for this is that the licensee assumed a 7D ZOI for fibrous debris generation based on insulation destruction testing that was later found to be non-conservative by the NRC staff. The licensee then reverted its fibrous debris ZOI to the value approved by the NRC staff (17D). Using the larger ZOI meant that some very large breaks could result in fibrous debris amounts not bounded by the 2009 test.

The licensee's evaluation of head loss and vortexing was performed in accordance with NEI 04-07 guidance, including the NRC staff SE (Reference 25) and review guidance, dated

March 28, 2008 (Reference 32). Information received from the licensee in the letter dated April 21, 2017, included the information requested in the Content Guide (Reference 31).

The level of water in the sump with respect to the strainer elevation is an important parameter in the strainer evaluation. The relative water level can affect NPSH available, vortexing, flashing, and degasification of the fluid. In cases where the strainer is not fully submerged, NRC guidance recommends specific methods of assessing strainer operation. In order to determine the minimum strainer submergence, the licensee calculated the minimum water level for various break locations and sizes. The minimum strainer submergence is 1.2 inches for LBLOCAs not inside the reactor cavity. For breaks in the reactor cavity when CS is initiated, the strainers may not be fully submerged at the start of recirculation. However, once the switchover to recirculation is complete (CS is switched to sump recirculation), the RHR strainers are submerged by 4.2 inches. The licensee stated that the longest a strainer would be operating and not fully submerged was 11 minutes. The licensee evaluated the RHR strainers for partial submergence, using approved methods, and determined that operation would be satisfactory until they are fully submerged. For SBLOCAs, the minimum submergence for the RHR strainers is 9 inches. The CS strainer submergence is always greater than the RHR strainer submergence.

The licensee provided the assumptions and results of its head loss evaluation. The licensee's assumptions include:

- The sump fluid is saturated at the surface of the pool and no credit is given for subcooling. The NRC staff notes that this is true for NPSH determination, but the licensee credits some accident pressure to suppress flashing across the strainer and deaeration of the fluid. The licensee demonstrated that the accident pressure credited would be available using conservative calculations.

- The head loss and vortexing evaluation is based on the modified, reduced RHR strainer height (16-disk vs.18-disk).

- Head loss is linearly proportional to dynamic viscosity.

- The test strainer parameters are scaled using a scaling factor defined as ratio of the surface area of the scale strainer to the surface area of the full-sized strainer.

- The test head loss was measured with relatively cold water; therefore, the head loss was corrected by the ratio of warm water to cold water viscosity to predict the head loss at post-LOCA water temperatures.

- Small corrections for head loss were made to correlate test data to plant flow conditions.

- Testing was conducted using the NRC staff review guidance of March 2008 for head loss testing (Reference 32).

The licensee stated that the results of the evaluation show neither vortexing or air ingestion or significant voiding will occur. The licensee tested the strainer for vortex formation for both clean and debris laden conditions. The tests were conducted at conservative flow rates and submergence values. No evidence of vortex formation was observed for the clean strainer condition until the water level was reduced to just below the top of the strainer. This clean

strainer test was conducted at 2.5 times the design flow rate. For debris laden tests, during the thin-bed test, air ingestion did not occur until the water level was 0.25 inches below the top of the strainer. For the full-load test, intermittent vortex formation was observed with the level about 3 inches above the top of the strainer, and sustained vortices were observed when the level was reduced to 2.25 inches above the top of the strainer. The licensee stated that vortex formation is not a concern for any analyzed break scenario. For most scenarios, the strainer is submerged by greater than 3 inches before debris could arrive. For other scenarios, the water level was more than 4 inches above the strainer before significant debris could arrive at the strainer. The licensee credited some containment pressure to show that flashing would not occur.

There is a potential for vortex formation to occur due to the collection of debris on top of the strainer building dams and preventing water from fully covering the strainer area. The licensee stated that the issue was resolved by the change in strainer height and the current analysis. The licensee stated that the tested debris load represents the maximum debris that can collect on the strainer, while allowing the strainer to continue to function. The strainer was tested at the maximum debris load for vortex formation, and the results were found to be acceptable. The NRC staff concludes that the testing conducted bounded the limiting condition for the strainer with respect to vortex formation.

The licensee performed head loss testing at an Alion test facility in 2009 using a strainer module smaller than (fewer disks), but otherwise identical to, those installed in Vogtle containments. Four tests were performed: a vortex test, a thin-bed test, and two full-load tests. The second full-load test was a confirmatory test for the first full-load test. The test apparatus included a flume or tank, pumps, instrumentation, piping, chemical mixing tanks, a heater, a chiller, and a strainer module to create prototypic plant conditions. The test strainer had a surface area of 65.67 ft^2. The flow rate and debris quantities were scaled based on the strainer surface area. The test flow rate was maintained at about 400 gallons per minute (gpm). This resulted in a velocity slightly higher than the strainer design approach velocity of 0.0122 ft/second. The test velocity was about 0.0136 ft/second. The test temperature was controlled at about 80 degrees Fahrenheit (°F).

Miscellaneous debris such as tags and labels were not included in the testing. However, the risk-informed analysis assumed 50 ft^2 even though a walkdown identified only 2 ft^2 in the Vogtle containments. In the test, the licensee used NUKON™ as a surrogate fibrous debris, including latent fibers. Green silicon powder was used as a surrogate for coating particulates. Other particulates included in the testing were Interam E-54A (fire barrier material) and a dirt/dust mixture made of silica sand to represent latent particulate debris. Chemicals used during the testing were Calcium Phosphate ($Ca_3(PO_4)_2$) and Sodium Aluminum Silicate ($NaAlSi_3O_8$).

The licensee provided information regarding the methodology used to perform the test, including debris preparation and addition, and other activities performed during testing sequence. The licensee also provided photographs of some steps during the test, and of the strainer with debris loads following the test.

The testing determined the test facility clean strainer head loss (CSHL) by recording pressure drop readings at the target flow rate. The CSHL value was measured at about 0.03 psi. This value was subtracted from the head losses measured during the test to determine the debris head loss.

For the analysis, the licensee stated that it added the CSHL calculated for the plant strainer to the debris head loss values from the test. The total CSHL for the plant strainer is determined by including head losses from strainer components, internal strainer flow losses, and the head loss associated with the sump pit below the strainer modules, including loss coefficients. The plant CSHL was calculated assuming pump runout flow of 4,500 gpm and a temperature of 120 °F. The flow rate is conservative because it is higher than any flow that will occur during a LOCA response at Vogtle. The temperature is expected to be greater than 120 °F for most cases. If the sump temperature cools to less than 120 °F, the CSHL would be slightly higher, but at lower temperatures, slight increases in differential pressure are not significant. The licensee stated that the CSHL was calculated for the original RHR strainers that had 18 disks. The strainers are being reduced to 16 disks. The licensee concluded that the CSHL for the 18-disk strainer was acceptable for use for the smaller strainer because most of the flow is through the disks closest to the discharge of the strainer, and a conservative flow rate was used for the CSHL calculation.

The licensee measured the head loss in three tests: a thin-bed test, a full-load test, and a repeat full-load test. Based on the results of these tests, the head loss was calculated for different amounts and combinations of debris. These head losses were applied based on predictions of debris transport amounts and assumptions regarding chemical effects, and were corrected for flow and temperature. The applied head loss also included extrapolation to the 30-day mission time to account for slight increases in head loss that were still ongoing at the end of the tests. The extrapolation was conservatively added at all times after 7.5 hours. For the partially submerged strainer, the 32.04 ft^3 (in the model below) is scaled to the effective strainer area based on the sump level. If the available strainer area is 90 percent of the total area, the 32.04 ft^3 would also be reduced to 90 percent of the value. The licensee also provided a figure that illustrated how the various significant changes in head loss were modeled in NARWHAL. The analysis of the model follows.

- If there is any debris on the strainer, but it is less than 0.45 inches theoretical thickness, the thin-bed head loss is applied.

- If the debris bed has a theoretical thickness of greater than 0.45 inches, the calcium phosphate head loss is added to the total.

- If the fibrous debris amount is greater than 32.04 ft^3 (0.57 inches theoretical thickness) per RHR strainer, the full-load head loss is applied in place of the thin-bed head loss, and appropriate chemical effects adders are applied.

- If the amount of aluminum in solution reaches the calculated saturation limit, or at 24 hours, whichever occurs first, the sodium aluminum silicate (SAS) head loss is added to the total.

- The 30-day extrapolation value is added to the head loss at 7.5 hours.

- The head loss is corrected for flow and temperature (difference from test flow velocity and temperature) at each time step. The head loss correction used is the greater of that derived from the thin-bed or full-load test for the conditions that exist at the time step being evaluated.

- When there is flow through the strainer, the CSHL is added at every time step.

For the thin-bed test, all particulate debris was added first. The particulate term added to the thin-bed test was the same as that added to the full-load tests and was intended to bound the potential particulate debris load from all breaks.

For the thin-bed test, after the particulate was added, fine fiber batches were added in 1/8 inch increments. Five fiber batches were added. Head loss did not increase significantly until the fifth batch was added. This is accepted as a sign that a thin debris bed, one with sufficient fiber to filter particulate debris, has formed on the strainer. Based on the results, the licensee concluded that chemical precipitates would not contribute significantly to debris beds of less than 0.45 inches theoretical thickness. The licensee added chemical debris to the thin-bed test after all of the particulate and fiber had been added and the head loss had stabilized. Three batches of calcium phosphate and three batches of SAS were added. These batches represented the maximum chemical term calculated for Vogtle. The thin-bed test resulted in a non-chemical head loss of 0.625 ft, a calcium phosphate head loss of 1.65 ft, and a final head loss (including SAS) of 2.56 ft.

The licensee performed a "full-load" test that was based on fibrous debris amounts generated by an assumed ZOI smaller than that approved by the NRC. Therefore, the full-load test included the full particulate load (with some discrepancy described above) and the entire fiber load predicted for the reduced ZOI. This source term provides the criterion for debris types above which the strainers are assumed to fail. The full-load test was conducted differently than the thin-bed test. For the full-load test, the particulate and fibrous debris were mixed in homogeneous batches and added incrementally to the test tank. There were four equal batches of fiber/particulate added to the test followed by three batches of calcium phosphate and 3 batches of SAS. The full-load test resulted in a conventional debris head loss of 5.46 ft, a calcium phosphate head loss of 6.57 ft, and a final head loss of 11.81 ft.

The full-load test had a significantly higher head loss than the thin-bed test. Therefore, a repeat of the full-load test was performed to confirm the results. The test was performed using the same procedure as the first full-load test. The confirmatory test resulted in a conventional head loss of 3.50 ft, a calcium phosphate head loss of 5.75 ft, and a final head loss of 8.99 ft.

The licensee performed flow sweeps at the end of each test to characterize the flow through the debris beds. This information was used to make flow and temperature corrections to the test data in order to predict head loss at plant conditions. The licensee used the more conservative full-load test to define head losses in the model.

The NRC staff noted that the full-load test did not appear to have a stable head loss after the addition of calcium phosphate but prior to the addition of SAS. Staff guidance (Reference 32) is that head loss should be relatively stable prior to the addition of subsequent debris batches, especially if the previous batch represents a criterion that is used in the analysis. For example, in this case, the licensee uses the calcium phosphate head loss value until it is predicted that SAS will precipitate. The licensee provided justification that the test results did not cause the analysis to be non-conservatively affected. The licensee stated that a conservative extrapolation constant was added to the calcium phosphate head loss value used in the analysis prior to the predicted formation of aluminum precipitates. The licensee also stated that the increase in head loss at the time of the addition of the aluminum precipitates in the test was small. The licensee concluded that the head loss values used in the analysis were acceptable.

The NRC staff noted that the confirmatory test resulted in a larger increase in head loss due to calcium phosphate than the full-load test, even though its head loss values were lower overall.

The licensee justified the use of full-load test calcium phosphate addition to head loss instead of that from the confirmatory test. The licensee stated that even though the calcium phosphate head loss was greater during the confirmatory test, the absolute value of head loss was greater for the full-load test at all debris loads. Therefore, the licensee concluded that using the full-load head loss values for all inputs to the analysis was conservative. The licensee also performed a sensitivity study to evaluate variability in head loss by increasing the full-load head loss values for conventional and aluminum debris by 25 percent and the calcium phosphate head loss to the value from the confirmatory test. The study found the change in risk to be negligible.

The NRC staff finds that the licensee's responses to RAIs 4 and 5 are acceptable based on the conservative extrapolation constant applied to the head loss and the use of the higher full-load test values for its head loss model. The staff also understands that a sensitivity study was performed to evaluate the risk associated with using a higher calcium phosphate head loss and that the result showed a negligible increase in risk.

The licensee evaluated flashing and deaeration that may occur as fluid undergoes pressure changes as it passes through the debris bed. The licensee credited 3.5 psi of accident pressure, and this was sufficient to preclude predictions of flashing across the strainer. The licensee calculated containment pressures and sump temperatures using minimum safeguards design-basis calculations. Minimum safeguards is the condition with fewest number of ESF pumps operating as allowed in the design basis after accounting for a single failure. These calculations maximize both pressure and temperature in containment. The licensee performed a sensitivity study that showed that the maximum temperature case is more conservative than the cooler case when all safeguards equipment is operating. The licensee's calculations determined that there is always more than 3.5 psi margin (about 6 psi minimum) for flashing across the strainer, and that the low margin exists only for a short period of time. As soon as the sump temperature begins to decrease below saturation, the margin to flashing increases significantly. To confirm the design-basis calculation, the licensee performed a realistic containment analysis. The realistic analysis showed a minimum of about 9.3 psi to flashing for the cold-leg DEGB and 8.7 psi for the hot-leg DEGB.

The licensee applied the 3.5 psi accident pressure credit when calculating degasification of the fluid as it passes through the debris bed. The licensee stated that the acceptable void fraction was to be less than 2 percent, or the scenario was considered to be a failure. Additionally, the NPSH required was increased, in accordance with staff guidance (Reference 32), to account for any void fraction less than 2 percent at the pump suction. The increase in NPSH required results in a decrease in NPSH margin. In its response to RAI 16 regarding the head loss model and discussed above, the licensee also stated that the NARWHAL model had been changed to credit accident pressure for only the first 2.5 hours of the event for both degasification and flashing calculation. The use of containment pressure for a limited time is conservative, compared to a more extended credit, and is therefore, acceptable to the NRC staff.
The licensee stated that accident pressure was not credited for NPSH margin calculations.

The licensee calculated total strainer head losses for various plant conditions (e.g., sump temperature and debris load) by correcting the test head loss values for flow and temperature to account for each analyzed plant condition and then adding the CSHL. Head loss calculations are performed for each time step by the NARWHAL software.

The licensee used extrapolation to calculate the head loss at the end of the 30-day mission time from the conditions present at the end of the testing. The head loss was fit to a linear function of the logarithm of time and shifted upward to bound the head loss data. An extrapolated head

loss was computed for the thin-bed test and both of the full-load tests. The extrapolated head loss from the first full-load test was greater than the other tests; thus, it was used for all of the calculations performed by NARWHAL.

The NRC staff reviewed the licensee's evaluation and found that the testing and evaluation of test results were conducted in accordance with NRC staff accepted practices (Reference 32). The NRC staff concludes that the maximum tested debris head loss represents a value that is bounding for the Vogtle strainers for the debris load tested. This conclusion is reasonable because the test was performed using conservative inputs and test methods that have been accepted by the NRC staff. The staff also finds that the extrapolation of the head loss to the strainer mission time and the corrections for head loss and flow were performed acceptably because methods used followed staff guidance (Reference 32).

The Vogtle test program was performed assuming that a single train of ECCS was operating. Instead of using the limiting debris load from the test, the licensee assumed that the transported debris source term would be split among the strainers modeled to be operating, depending on the scenario being evaluated (i.e., assumed operational pumps). The NRC staff finds that this is an acceptable use of the test data because the modeled head loss and the acceptance criteria are based on the amount of debris that was deposited on the strainer during testing. Also, the tests were conducted at the highest flow rate experienced by any strainer; thus, applying the resulting head loss to strainers with lower flow rates is conservative. The NRC staff also considered the particulate and chemical debris loading on the strainers and determined that the test program, which included a thin-bed test and two full-load tests, adequately represented the potential debris loads as evaluated in the analysis. Therefore, the NRC staff finds the licensee's use of the test data acceptable.

The testing was performed on strainer modules with 7 vertically stacked disks. The plant currently has 18 disk RHR strainers installed. The test data was applied to the planned configuration of RHR strainers with 16 vertically stacked disks. The CS strainers have 14 disks. The NRC staff concluded that the head loss results from a 7-disk test strainer would be conservative, compared to a taller strainer (more disks), because the debris tends to concentrate at the lower disks closer to the pump suction, until the debris differential pressure causes the flow to move to disks further from the pump suction. Eventually, the debris bed becomes relatively uniform, but is skewed until adequate debris amounts arrive. This skewed debris distribution would be more pronounced on a strainer with more disks so that head losses, at least at lower debris loads, would be lower on the larger strainer in the plant.

The NRC staff reviewed the scaling with respect to the testing as it applies to the RHR and CS strainers and found the scaling to be adequate. The scaling to the planned 16-disk RHR strainer was accurate. The staff confirmed that the velocity through the CS strainer is lower than that of the RHR strainer, and that the RHR strainer would collect more debris per surface area based on the design flow rates for the strainers.

The NRC staff verified independently that the licensee's inputs for its head loss and vortexing analyses were determined accurately (Reference 22). The NRC staff used a combination of confirmatory calculations, engineering review, and exercising of the licensee's software to perform verifications of strainer head loss testing, time-dependent fiber penetration, and scaling of head loss test results to the RHR and CS strainers. This approach enables the staff to conclude that a significant portion of the inputs to the head loss testing was conducted properly.

The NRC staff concludes that the licensee's evaluation of the potential for vortexing was sufficient. The licensee identified, for some breaks, that the strainer would not be fully submerged when it was placed into service. The licensee stated that the strainer would quickly become fully submerged, and that prior to full submergence, the potential for strainer failure was addressed using NRC staff guidance for unsubmerged strainers (Reference 32). The staff concluded that the licensee's evaluation was reasonable because significant head loss would not occur before the strainer became fully submerged. For submerged strainers, the NRC staff concludes that the licensee's testing adequately demonstrated that vortex formation would not occur under the operating conditions expected following a LOCA. For vortex tests with no debris loads, the velocity through the strainers was much higher than it would be in the plant, and is, therefore, conservative.

The NRC staff noted that the licensee calculated CSHL assuming the majority of flow is through the lower disks. Normally, the CSHL is calculated assuming equal flow through all disks to simulate a debris laden strainer. Based on the strainer design, the staff concludes that the alternate calculation would not result in a significant difference in CSHL for the Vogtle strainers. The staff also noted that the RHR and CS systems have suffcient NPSH margins. The staff also considered that flashing is calculated based on the limiting condition (lowest submergence) at the top of the strainer so that CSHL is not a factor. The licensee also stated that deaeration of the sump fluid did not contribute to risk increase because any failures caused by deaeration were already failures due to debris limits being exceeded. Based on the above, the NRC staff finds the CSHL calculation is acceptable.

The NRC staff reviewed the licensee's credit for containment accident pressure (CAP) to prevent flashing and reduce the potential for degasification of the fluid as it passes through a debris bed. The staff reviewed the design basis and realistic containment analyses and concludes that both showed significant margin above the credit used by the licensee in the strainer analysis. The licensee's evaluation of NPSH did not credit any pressure above the vapor pressure of water. The licensee correctly adjusted NPSH required values to account for void fraction at the pump suction. The licensee applies the credit for accident pressure for a limited time at the beginning of the event response. Therefore, the staff concludes that these portions of the evaluation are acceptable.

The NRC staff reviewed the test methods and assumptions used in the analysis and concluded that they are consistent with staff guidance (Reference 32). The staff also reviewed the ways in which the licensee extrapolated the test results and applied the data and verified that the usage was in accordance with staff guidance.

The licensee's use of 0.45 inches as the theoretical bed thickness below which chemical effects would not increase head loss across the debris bed on the strainer was a concern. During other tests, the staff has observed additional head losses from chemicals with thinner theoretical bed thicknesses. The licensee stated that the debris tended to collect on the bottom disks of the strainer so that there was a significantly skewed distribution of debris until a significant amount of fiber was added to the test. The licensee also referred to the head loss results, which show that head loss did not increase until the fiber batch after the batch that resulted in a 0.45-inch bed was added. Finally, the licensee pointed to a sensitivity study that found even if chemical effects were assumed to add to head loss with a 0-inch theoretical bed, risk to the plant remained very low. Based on the above, the NRC staff concluded that the use of 0.45 inches as the theoretical bed thickness for the Vogtle specific condition is acceptable.

NRC staff guidance (Reference 32) indicates that debris should be added so that less transportable debris will not inhibit more transportable debris from reaching the strainer. The staff noted that for the full-load tests, small fiber pieces were added with the other fine debris. Generally, small debris is less transportable than fines and might inhibit transport of the smaller particles. However, the licensee's test program was designed to ensure that all debris was transported to the strainer. Transport was verified during the testing. Also, each batch included more than two times the mass of fine fiber, compared to small pieces of fiber. Therefore, the staff concludes that the addition of the fine and small debris in homogeneous batches is acceptable. The thin-bed test used only fine fibrous debris.

The licensee chose the tests with the highest head losses as inputs to the head loss analysis. The staff finds the licensee's assumption that the arrival at the strainer of any amount of any type of debris greater than the amount included in the test to be conservative, because even at the loads tested, there is significant NPSH margin. Therefore, larger debris amounts could be accommodated before encroaching on NPSH margins. The NRC staff determined that most scenarios that are considered to fail occur due to exceeding debris limits and not due to NPSH margin encroachment. This indicates that there is margin in the methodology.

The Vogtle evaluation calculated a debris source term arriving at the strainer for each of the potential initiating events that could reasonably lead to sump recirculation. The 2009 head loss test provides an acceptance criterion for the amount of each debris type that can arrive at the strainer before it is considered to fail. The evaluation is valid for each of the four strainers that supply recirculating fluid to the RHR and CS pumps, because it is based on the amount of debris predicted to transport to each strainer and each strainer is evaluated separately. Therefore, the NRC staff concludes that the use of the Vogtle 2009 test for determination of the debris acceptance criteria is acceptable for all potential pump operating combinations.

<u>NRC Staff Conclusion Regarding Head Loss and Vortexing</u>

The NRC staff's conclusions regarding head loss and vortexing are based on RHR strainers with 16 disks. The NRC staff reviewed the licensee's evaluation against the staff-accepted guidance and concludes that the licensee has appropriately determined the head loss across the sump strainer for the debris load tested. The licensee has shown that the potential for formation of a vortex at the proposed strainers does not exist under the planned modified plant-specific conditions at Vogtle. The licensee has demonstrated that the strainer will perform acceptably under postulated LOCA conditions, limited by the amount of debris represented in the 2009 test. Therefore, the NRC staff concludes that the licensee's evaluation of head loss and vortexing is acceptable.

The NRC staff concludes that SNC has provided sufficient information as requested by GL 2004-02 and further described in the revised content guide for GL 2004-02 concerning head loss and vortexing, because the licensee:

- Provided the minimum submergence of the strainer under SBLOCA and LBLOCA conditions.

- Provided a summary of the methodology, assumptions and results of the vortexing evaluation and bases for key assumptions.

- Provided a summary of the methodology, assumptions, and results of prototypical head loss testing for the strainer, including chemical effects. Provide bases for key assumptions.

- Addressed the ability of the design to accommodate the maximum volume of debris that is predicted to arrive at the screen.

- Addressed the ability of the screen to resist the formation of a "thin bed" or to accommodate partial thin bed formation.

- Provided the basis for the strainer design maximum head loss.

- Described significant margins and conservatisms used in the head loss and vortexing calculations.

- Provided a summary of the methodology, assumptions, bases for the assumptions, and results for the clean strainer head loss calculation.

- Provided a summary of the methodology, assumptions, bases for the assumptions, and results for the debris head loss analysis.

- Stated whether the sump is partially submerged or vented (i.e., lacks a complete water seal over its entire surface) for any accident scenarios and describe what failure criteria in addition to loss of net positive suction head (NPSH) margin were applied to address potential inability to pass the required flow through the strainer.

- Stated whether near-field settling was credited for the head-loss testing and, if so, provide a description of the scaling analysis used to justify near-field credit.

- State whether temperature/viscosity was used to scale the results of the head loss tests to actual plant conditions. If scaling was used, provide the basis for concluding that boreholes or other differential-pressure induced effects did not affect the morphology of the test debris bed.

- Stated whether containment accident pressure was credited in evaluating whether flashing would occur across the strainer surface, and if so, summarize the methodology used to determine the available containment pressure.

3.4.2.8.4 <u>Sump Structural Analysis</u>

In Enclosure 5, Section 3.k, "Sump Structural Analysis," of the letter dated April 21, 2017, the licensee stated finite element models were developed and analyzed using ANSYS computer software to structurally qualify the replacement sump strainers. Using ANSYS, the models were subjected to bounding loading combinations consisting of dead weight (weight in air and water were considered), debris weight, the crush pressure resulting from debris blockage, design-basis earthquake loads, and thermal effects of the system at 250 °F. The maximum-induced stresses for each component were then evaluated against the applicable allowable stress value from the identified code. The design code of record used in the qualification of the strainer and associated components is the ASME Code Section III, Subsections NC and ND, 1989 Edition. The licensee outlined the material properties that were used in the structural

analyses and noted that the properties were taken from the ASME Code. The licensee also indicated that stainless steel (SS304) was used for all portions of the replacement strainers except for the tie rods, which were made of a different type of stainless steel (SS410).

The licensee stated that the crush pressure of the strainer is 4.46 psi (10.1 ft) at the design temperature. The strainer head loss, based on testing, was calculated to be less than 10.1 ft for all temperatures greater than 140 °F. Below this temperature, the head loss may increase to higher values due to the precipitation of chemicals and the increased viscosity of water. The licensee stated that the head loss for the strainer could increase to 14 ft at 120 °F. In the letter dated May 23, 2018, the licensee explained that the original head loss limit was an assumed value based on what was believed to be adequate during the original strainer design. When a risk-informed methodology was pursued, the stress model was revisited to determine if a higher crush pressure could be justified. A finite element analysis model was run on the limiting plate to frame weld location, which determined that the previous crush pressure based on this location was overly conservative, and that the crush pressure for this location was 30.47 psi. Therefore, the second most limiting location (plate to finger weld) was used, along with the ASME Code Section III allowable stresses and the methodology previously described, to determine the max allowable crush pressure of 10.7 psi for an 18-disk strainer. This value is reduced to 10.39 psi (24 ft) for a 16-disk strainer and aligns with the information provided in Section 3.f.7 of the submittal. The staff reviewed the licensee's response and finds it acceptable because it sufficiently identifies the limiting strainer crush pressure and explains the process used to calculate the value. The approach is in accordance with the ASME Code and NEI 04-07 guidance (Reference 26), and the resulting value of 24 ft bounds all the postulated head loss values.

Utilizing the ANSYS analyses that provided the strainer component and weld responses under the various loading combinations, the licensee compared the results to the ASME Code allowable stress values. The licensee tabulated the resultant design margins showing the ratio of the ASME Code allowable vs. calculated max stress (stress ratio) for the strainer components and welds under review to demonstrate their structural integrity. The ratios for all of the components were acceptable. The licensee addressed the change from an 18-disk to a 16-disk strainer and stated that the evaluation remains valid for the smaller strainer.

Regarding potential loadings associated with a high-energy line break, the licensee noted that there is only one line outside the steam generator compartments with analyzed breaks. This is a 2-inch line that is approximately 32 ft above the strainer. The line is located on the pressurizer cubicle wall, where no unsecured items would be located.

The NRC staff reviewed the licensee's submittal and finds the analyzed loads and load combinations acceptable because they are consistent with the considerations in the staff-approved guidance in NEI 04-07. The resultant stresses were compared to allowable stresses from the ASME Code, which is the licensee's existing code of record and provides staff-approved acceptance criteria. All of the resultant stresses were below the allowable ASME Code stress limits. The use of the ASME Code and the NEI 04-07 guidance provides assurance that the sump strainer structural design used appropriate design inputs and load combinations.

The NRC staff reviewed the licensee's technical report and noted that the analysis for the original strainer was applicable to the modified strainer based on similar operating conditions and a favorable configuration for seismic loading for the modified strainer. Additionally, the licensee provided the stress ratios (ASME Code limits/calculated maximum stress) for all the

strainer components. All ratios were above one, and are, therefore, acceptable (Tables 3.k.2-1 – 3.k.2-4, and 3.k.2-6 of the licensee's letter dated April 21, 2017). The licensee also provided the worst case interaction ratios for the anchor bolts (Table 3.k.2-5), which were less than one. Interaction ratios are the inverse of stress ratios, and acceptable values should be less than one. The tables demonstrating stress ratio values above one and interaction ratios less than one based on acceptable load combinations and acceptable allowable stresses, the NRC staff concludes that the sump strainer assemblies will remain structurally adequate under the expected loading conditions.

The NRC staff reviewed the licensee's technical report and verified that the referenced line is the only high-energy line in the area of the strainer. Due to the distance from the strainer and the lack of credible missiles near the line, no high-energy line break evaluation is necessary for this line. Based on the above, the staff finds that the licensee's assessment of dynamic effects for the sump structural analysis is acceptable.

The staff notes that backflushing is not credited in the new strainer design; thus, no structural analysis was necessary to address reverse flow.

NRC Staff Conclusion Regarding Sump Structural Analysis

The NRC staff concludes that the sump strainer is structurally acceptable for the assumed design-basis loads for which it is qualified. The NRC staff finds that the licensee has provided the information requested in item k (Sump Structural Analysis) of the NRC's Revised Content Guide for GL 2004-02 Supplemental Responses (Reference 31). The licensee:

- Summarized the design inputs, design codes, loads, and load combinations utilized for the sump strainer structural analysis.
- Summarized the structural qualification results and design margins for the various components of the sump strainer structural assembly.
- As applicable, summarized the evaluations performed for dynamic effects such as pipe whip, jet impingement, and missile impacts associated with high-energy line breaks (as applicable).

3.4.2.8.5 Net Positive Suction Head

The licensee stated the pump flow rates for each operating train are 3,700 gpm for RHR, 425 gpm for SI, 150 gpm for charging, and 2,600 gpm for CSS. The licensee further stated that the design flow rate for the RHR pumps is 3,000 gpm, and that using a higher flow rate was conservative with respect to recirculation timing, flashing calculations, and head loss calculations. The licensee stated that 3,700 gpm is consistent with the design-basis single train NPSH calculation flow rate. The charging and SI pumps piggyback or take suction from the RHR pumps. Each RHR and CSS pump has its own independent suction strainer. The sump flow rate for an LBLOCA is 3,700 gpm per strainer for the RHR strainers and 2,600 gpm for the CSS strainers. Sump temperature for the analysis was evaluated for the range of temperatures based on the design-basis LOCA temperatures with minimum safeguards. The temperatures were calculated for a power uprate LAR for a double-ended guillotine recirculation pump suction break. The maximum temperature is about 250 °F, and the minimum approaches 100 °F toward the end of the 30-day mission time.

The licensee calculated sump levels for several break scenarios. For breaks within the reactor cavity that cause CS actuation, the strainers may not be submerged at the start of recirculation. The minimum level calculated for this scenario when RHR is placed in recirculation mode is 1.384 ft below the top of the strainers and corresponds to a pool height of 3.054 ft. However, the licensee stated that by the time switchover is complete (CS is placed in recirculation mode), the strainers are submerged by 0.35 ft corresponding to a pool height of 4.788 ft. The increase in level is caused by continued injection of RWST inventory by the CSS. For reactor cavity breaks where CSS does not actuate, the strainers are submerged by over 0.5 ft when recirculation is initiated. The strainers are fully submerged at the start of recirculation for all cases other than the reactor nozzle break with CS. The sump levels referenced here are based on a hand calculation.

The containment water level in NARWHAL is calculated for each time step in the calculation using a correlation developed by the licensee using a three-dimensional CAD model. The correlation accounts for the volume of the lower containment, including water displaced by solid objects. The correlation also accounts for water temperature. Water sources and water that is held up and may not contribute to the containment pool are also included in the model. Holdups include the reactor cavity, RCS, CS pump discharge piping, water in transit from the CSS, steam held up in the atmosphere, and smaller volumes such as the containment sumps, floor drains, and elevator pit. The SBLOCA case results in a lower water level because it is assumed that the RCS is refilled with water and the accumulators do not discharge. The licensee stated that smaller equipment located in lower containment is not modeled as displacing water in the water level calculation, thus adding some small conservatism to the calculation.

NPSH margin is calculated for the pumps at each time step. Each scenario has a different sump level trend that is applied to the NPSH calculation, as applicable. The sump level trends are calculated based on conservative inputs as discussed below.

The licensee used the following assumptions when calculating sump and pump flow rates, sump temperature, and containment water level:

- RHR pump flow rates are assumed to be 3,700 gpm even though they would be lower. The value is consistent with the value used in the design-basis NPSH calculations.

- Variation in SI and charging pump flow is assumed to have no effect on RHR flow.

- CSS pump flow rate is based on the design flow rate.

- Sump temperature from the containment analysis maximizes the sump temperature by using the minimum safeguards design-basis case.

- Containment water level used conservative input values for pool contributions and accounted for holdup in the containment, filling of empty piping, water in transit, and steam holdup. For SBLOCAs and secondary side breaks, the accumulators were assumed not to inject. The reactor cavity was assumed to fill before level begins to rise.

- The minimum TS RWST water level was assumed for the sump level calculation.

The licensee stated that the basis for the required NPSH values came from the test curves supplied by the pump vendor. The licensee also stated that NPSH margins were calculated using the NARWHAL software. The NARWHAL software calculated margin for each time step using NPSH available, NPSH required, sump water level, strainer head loss, sump temperature, and pump flow rates. The NPSH available values include the piping frictional losses. The piping losses were calculated using standard industry calculations and losses taken from standard industry handbooks. Strainer head loss was calculated for each time step by combining CSHL with debris bed losses calculated and corrected.

The licensee described the LBLOCA system response scenarios. The RHR, SI, and charging pumps start automatically and take suction from the RWST. The pumps inject to the RCS cold-legs. If containment pressure reaches the CSS actuation setpoint, the CSS pumps also start with the RWST as their suction source. When RCS pressure decreases to about 600 psia, the accumulators begin to inject into the RCS loops. When the RWST reaches the low-low level setpoint, the RHR pump suction valves to the sump automatically open. The switchover for CSS begins when the RWST reaches the empty setpoint. The switchover to recirculation is completed when all valves taking suction from the RWST are closed manually. If CSS is not actuated, the ECCS pumps follow the logic described above. After switchover, the charging and SI pumps take suction from the RHR pump discharge.

The licensee stated that about 7.5 hours after the initiation of the event, the ECCS lineup is modified for simultaneous hot and cold-leg injection. The RHR and SI pumps are aligned to inject to the hot-legs, while the charging pumps continue to inject to the cold-legs.

For a small SBLOCA, the RCS pressure may remain above a pressure that would allow the accumulators to inject. Also, it is unlikely that CSS will actuate for a SBLOCA. If the break is small enough, the plant may be able to be shut down before the RWST is depleted. In that case, recirculation would not be required.

In the NARWHAL analysis, CS was only assumed to actuate for hot-leg breaks greater than 15 inches. This assumption is important because it significantly affects the transport of debris. The assumption has competing effects for transport. If the sprays actuate, all fine debris is washed down from upper containment. If sprays do not actuate, only 10 percent of the fine debris is washed down to the sump. However, if the CSS actuates, the debris is split between the CSS and RHR strainers. The licensee conducted sensitivity studies to determine the effect of the CS actuation assumption on risk.

The licensee's evaluation assumes that all ECCS and CSS pumps are available at recirculation. The licensee stated that its risk-informed analysis considered the possible equipment combinations that could be available at the start of a LOCA scenario. The pump combinations were not limited by a single failure. The likelihood of being in any given condition was determined by using the Vogtle probabilistic risk assessment (PRA) model. The licensee's evaluation assumed that all random equipment failures occur at the beginning of recirculation.

Although some CAP was credited in the head loss and vortexing analysis to prevent flashing, the licensee does not credit CAP in Vogtle's analysis of NPSH. For the NPSH calculation, for sump temperatures greater than saturation, the containment pressure is assumed to be equal to the saturation pressure of the fluid. As temperature decreases below saturation, NPSH margins increase due to subcooling. The discussion also provides a context for the margins related to containment pressure and sump temperature that are likely available for NPSH calculations, but not credited.

The RHR pump NPSH margin was reported by the licensee for varying sump temperatures (see Table 3 below). The strainer head loss in this table includes the debris bed head loss and the CSHL. More detail on the head loss model is presented in Section 3.4.2.8.3, "Head Loss and Vortexing." For sump temperatures above about 212 °F, the NPSH available considered that the containment pressure was equal to the vapor pressure. For sump temperatures below 212 °F, the containment pressure was taken as 14.7 psia. Vogtle actually used a slightly lower pressure and temperature corresponding to the minimum allowable containment pressure. As can be seen from the table, the NPSH margins for the RHR pumps are large.

Table 3: Limiting NPSH Margin vs. Sump Temperature
(Table 3.g.16-1 in Enclosure 5 of the licensee's letter dated July 10, 2018)

Pool Temperature (°F)	NPSH Margin Before Subtracting Strainer Head Loss (ft-H_2O)	Strainer Head Loss (ft-H_2O)	Net NPSH Margin After Subtracting Strainer Head Loss (ft-H_2O)
212	23.119	5.515[a]	17.6
205	27.255	5.544[a]	21.7
195	32.441	5.589[a]	26.9
165	44.863	5.729[a]	39.1
153	45.657	9.006[b]	36.7
140	47.339	9.128[b]	38.2
133	44.076	14.359[c]	29.7
120	45.861	14.578[c]	31.3

[a] This includes clean strainer and conventional and chemical debris (calcium phosphate) head losses.
[b] This includes clean strainer, conventional and chemical debris (calcium phosphate) head losses, and extrapolation constant.
[c] This includes clean strainer, conventional and chemical debris (calcium phosphate and SAS) head losses, and extrapolation constant.

The licensee stated that the CS pumps are expected to have greater NPSH margins than the RHR pumps due to lower flow rates and lower debris loads. The NRC staff recognized that NPSH margins are dependent on several factors other than those discussed by the licensee. The NRC staff asked RAI 36, which requested that the licensee provide additional justification that the CS pumps have greater NPSH margins than the RHR pumps. The licensee stated that the NARWHAL software evaluated the NPSH margins for the CS pumps, as well as the RHR pumps. The model includes specific inputs relevant to NPSH for the CS pumps. The licensee stated that the NARWHAL model consistently showed greater NPSH margins for the CS pumps. Based on the licensee's response to RAI 36, the NRC staff finds that the treatment of NPSH margins (i.e., that the NPSH margins for the RHR pumps are lower, and therefore, bound the CS pump margins) is acceptable. RAI 36 and RAI 29, discussed below, are the only RAIs asked for the NPSH area.

The NRC staff concludes that the flow rates used for the pumps are acceptable, and in the case of the RHR pumps, conservatively high, and that the NPSH required values were determined using acceptable methods. The 3,700 gpm flow rate used for the RHR pumps is consistent with the flow used for the design-basis single train flow rate and higher than the dual train flow rate,

which is the flow expected for the majority of scenarios. The staff finds that using the design-basis minimum safeguards sump temperature curve is acceptable because it results in the highest sump temperatures, which is conservative for NPSH calculations.

The NRC staff reviewed the licensee's assumptions for sump level and concludes that the methods used are consistent with NEI 04-07 (Reference 26). The licensee used conservative mass sources and holdups when calculating sump level. The licensee provided a comparison between the NARWHAL sump level calculations and the hand calculations, which demonstrated that the results are consistent. The licensee noted that the NARWHAL calculation takes into account the actual break elevation, while the hand calculation assumes the break is at the highest RCS elevation. This affects the amount of RCS holdup in the calculation. Additionally, the hand calculation assumes a pool temperature of 100 °F, while NARWHAL uses a time-dependent pool temperature. This affects pool level by accounting for the changing density of the fluid. The NRC staff concludes that the hand and NARWHAL calculated sump levels were consistent, and use of the NARWHAL calculated levels is acceptable for the analysis. The NRC staff concludes that the sump levels used in the NPSH calculation are sufficient to demonstrate the minimum levels that will be present in the sump at a specific time for a specific break scenario. In most cases, the levels would be expected to be higher due to additional mass (above the TS minimum) in the RWST and other less significant conservatisms in the calculations.

The assumption that the CSS actuates only for large hot-leg breaks has an effect on strainer head loss, containment temperature, sump temperature, and flows in the containment. Because the licensee assumes the maximum head loss for the debris bed based on testing, and assumes that at all temperatures above saturation, the pressure in containment equals the saturation pressure of the sump fluid, the effect of the CSS operation on temperature and pressure is inconsequential with respect to NPSH. Therefore, the NRC staff concludes that the assumption regarding CSS actuation does not affect the NPSH calculations for the risk-informed analysis.

The licensee did not credit CAP for the NPSH calculations. This is in accordance with staff guidance in RG 1.82 (Reference 34), and is, therefore, acceptable.

The NRC staff reviewed the NPSH margin calculations presented in Enclosure 5 of the licensee's letter dated April 21, 2017, and concludes that the methods used to arrive at the margin values are acceptable. The licensee used staff guidance in the margin assumptions and calculations. Based on the margins, there are few NPSH-related failures that contribute to the plant risk. Most failures are due to exceeding tested debris amounts. It is likely, in many cases, that the added debris would not result in head loss significant enough to result in a loss of NPSH margin.

NRC Staff Conclusion Regarding Net Positive Suction Head

The NRC staff reviewed the licensee's NPSH evaluation against the NRC staff-accepted guidance and concludes that the licensee has appropriately validated that the plant design provides adequate margin between the NPSH available and the NPSH required for each pump taking suction from the recirculation sump. Therefore, the NRC staff concludes that the licensee's evaluation of NPSH is acceptable.

The NRC staff concludes that SNC has provided sufficient information as requested by GL 2004-02 (Reference 27) and further described in the revised content guide for GL 2004-02 (Reference 31) concerning NPSH, because the licensee:

- Provided applicable pump flow rates, the total recirculation sump flow rate, sump temperature(s), and minimum containment water level.
- Described the assumptions used in the calculations for the above parameters and the sources/bases of the assumptions.
- Provided the basis for the required NPSH values.
- Described how friction and other flow losses are accounted.
- Described the system response scenarios for LBLOCA and SBLOCAs.
- Described the operational status for each ECCS and CSS pump before and after the initiation of recirculation.
- Described the single failure assumptions relevant to pump operation and sump performance.
- Described how the containment sump water level is determined.
- Provided assumptions that are included in the analysis to ensure a minimum (conservative) water level is used in determining NPSH margin.
- Described whether and how the following volumes have been accounted for in pool level calculations: empty spray pipe, water droplets, condensation and holdup on horizontal and vertical surfaces.
- Provided assumptions (and their bases) as to what equipment will displace water resulting in higher pool level.
- Provided assumptions (and their bases) as to what water sources provide pool volume and how much volume is from each source.
- Provided description of the calculation of containment accident pressure used in determining the available NPSH.
- Provided assumptions made which minimize the containment accident pressure and maximize the sump water temperature.
- Specified whether the containment accident pressure is set at the vapor pressure corresponding to the sump liquid temperature.
- Provided the NPSH margin results for pumps taking suction from the sump in recirculation mode.

3.4.2.8.6 Chemical Effects

The licensee evaluated chemical effects as part of the head loss testing that was performed in 2009 at the Alion Science and Technology Hydraulics Laboratory in Warrenville, Illinois (Reference 58). Vogtle's overall chemical effects evaluation methodology included the following features:

- Plant-specific chemical precipitate loading was calculated using the WCAP-16530-NP-A, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191" (Reference 28), base methodology with significant modification to the aluminum release rate by crediting phosphate passivation of aluminum.

- Strainer head loss testing with initial addition of conventional (i.e., fibrous and particulate) debris followed by the addition of pre-mixed WCAP-16530-NP-A precipitates.

- Time-based analysis of head loss that considered aluminum solubility to determine when to account for head loss caused by SAS precipitate additions during strainer testing.

The licensee performed three tank head loss tests for Vogtle that included chemical precipitates: (1) a thin-bed test, (2) a full debris load test, and (3) a confirmatory full debris load test. During strainer testing, the debris addition was sequenced so that all fibrous and particulate debris was added, and the head loss stabilization criterion was satisfied before chemical precipitates were added. Chemical precipitates were pre-mixed and placed in holding tanks prior to addition to the test loop. After the precipitates were prepared, 1-hour settling tests were performed to ensure the chemical precipitates met the settlement acceptance criteria contained in WCAP-16530-NP-A. Chemical precipitates were added to the test tank in batches, starting with calcium phosphate and ending with SAS. The initial full-load test produced the highest head loss at each debris addition stage of the three licensee's tests, and was, therefore, selected to develop the relative head loss contributions from conventional debris, calcium phosphate precipitate, and SAS precipitate.

As part of the risk-informed analysis, Vogtle evaluated breaks for all Class 1 weld locations on the primary RCS piping upstream of the first isolation valve. The amount of chemical precipitate was calculated for each of these breaks using the break-specific debris generation quantity. Other plant-specific inputs needed to calculate the amount of chemical precipitate, such as pH, temperature, aluminum amount, and spray time, were selected to maximize the amount of predicted 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.

Chemical head loss was not applied until a uniform 0.45-inch theoretical bed thickness was formed on the strainer. Calcium release is determined using the WCAP-16530-NP-A release rates. Calcium phosphate is assumed to precipitate immediately. The licensee used an aluminum release rate that was developed based on testing performed at the University of New Mexico in a trisodium phosphate buffered environment (Reference 63). The licensee assumed 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) (Reference 64). For each pipe break that does

not reach the calculated aluminum solubility limit before 24 hours, NARWHAL conservatively assumes all aluminum in solution precipitates at 24 hours.

The licensee performed a chemical effects evaluation using the WCAP-16530-NP-A method with refinement for aluminum corrosion (aluminum passivation by phosphate) and with credit for aluminum solubility at elevated temperatures.

The licensee performed sump strainer testing and simulated chemical effects by adding pre-mixed precipitate at the Alion Science and Technology Laboratory. The NRC staff visited the vendor test facilities multiple times to observe testing, and is, therefore, very familiar with the licensee's test and evaluation methods for strainer testing ((Reference 65), (Reference 66) (Reference 67), (Reference 68), (Reference 69), and (Reference 70)). The licensee's strainer test added SAS precipitates to represent all aluminum-based precipitates. Use of either aluminum oxyhydroxide or SAS precipitates is acceptable to the NRC staff, as previously discussed in the SE for WCAP-16530-NP, page 16 (Reference 71).

Since the NRC staff has previously approved the WCAP-16530-NP-A base methodology for evaluation of chemical effects, the staff's evaluation of the Vogtle approach focused on the unique features (e.g., aluminum passivation) from the licensee's chemical effects evaluation. Since benchmarking the licensee's aluminum release with aluminum release from WCAP-17788-P (Reference 72), autoclave testing with similar environments showed that aluminum was under-predicted in some cases. Although the NRC staff recognizes that phosphate from the trisodium phosphate buffer will cause passivation of aluminum surfaces, thereby reducing aluminum release, the licensee applied the aluminum passivation function beyond the range of the testing that was used to develop it. Another key refinement in the licensee's approach related to aluminum solubility credit. The licensee assumed that aluminum-based precipitates would not form until the earlier of 24 hours or the dissolved aluminum reaching a solubility limit. The NRC staff questioned how using a more conservative bounding aluminum solubility limit discussed in the Howe, et al., paper (Reference 73), instead of the ANL solubility equations, would affect the Vogtle chemical effects evaluation.

The licensee addressed the concerns about under-prediction of aluminum release in some of the benchmarked autoclave tests by providing a summary of conservative assumptions in the overall chemical effects approach. For example, when calculating aluminum release, the licensee assumes the post-LOCA sump pool pH is 7.8. When determining aluminum solubility, the licensee assumes the sump pool pH is 7.0. The licensee determined that the best estimate maximum sump pool pH is 7.42, which occurs for SBLOCAs since the injected volume of borated water is conservatively minimized. For LBLOCAs with maximized injected volumes of borated water, the best estimate containment sump pool pH is 7.16. The licensee performed spreadsheet calculations using the same aluminum release methodology as in NARWHAL to determine the amount of aluminum released at the higher 7.8 pH value assumed in the Vogtle plant-specific calculation, as compared to a 7.42 pH. The use of the higher pH results in a 230 percent increase in aluminum relative to the 7.42 pH. Applying this 230 percent increase to the benchmarked autoclave tests bounds all the autoclave testing results except for one data point that was determined by the PWROG to be an outlier.

Additional conservative assumptions were included in the chemical effects submodel. These assumptions included:

- A double-ended pump suction LOCA with minimum safeguards temperature profile is used to determine chemical release, which promotes greater aluminum release.

- NARWHAL analytically combines a maximum sump pool pH of 7.8 for aluminum release with a less than design-basis minimum 7.0 pH for solubility in a way not physically possible, thereby bounding potential pH profile variations.
- Unsubmerged aluminum is treated as fully submerged or fully wetted in the CS solution.
- The amount of calcium phosphate that is assumed to form and cause head loss was not observed in representative autoclave testing.
- No credit is taken for aluminum that remains soluble after precipitation is predicted to occur.

The staff considers the additional conservative assumptions made by the licensee to be an essential part of the submodel (i.e., evaluation). The staff found the licensee's responses acceptable since additional conservative assumptions adopted by the licensee were shown to bound the uncertainty in the use of aluminum release equations that in some cases under-predicted aluminum release in the benchmarked autoclave testing. The staff concludes that the licensee's overall methodology for determining the amount of precipitate is acceptable since the calcium phosphate precipitate quantity is determined using previously-approved WCAP-16530-NP-A, and the amount of aluminum released that is available to precipitate is acceptable as discussed above.

With regard to using the bounding aluminum saturation line from the Howe et al., paper instead of the ANL equation, the licensee stated that the bounding maximum precipitation temperature would increase from approximately 137 °F to 159 °F. Since the allowable strainer head loss in this temperature range is limited by structural margin, and is, therefore, constant, the licensee indicated that no additional failures related to head loss would occur with the most conservative aluminum solubility assumption. The licensee also noted that the ANL equation that was used in NARWHAL is conservative within the initial 24 hours after a LOCA. The licensee provided multiple examples from proprietary document WCAP-17788-P, Volume 5, where the ANL equation predicted precipitation during autoclave testing with trisodium phosphate buffer, but none was observed. The NRC staff has reviewed the autoclave data and determined that it provides valid data for this comparison. The staff also notes that after 24 hours, the licensee assumes all dissolved aluminum precipitates if not already predicted by the solubility limit.

The staff notes the bounding Howe et al., aluminum saturation line was developed using results from longer term tests that went well beyond the 24-hour period during which the licensee evaluates aluminum solubility. The staff also reviewed the results from the autoclave testing in WCAP-17788-P. The staff concludes that the ANL solubility equation used by the licensee is conservative for the pH range and temperature range of interest, and is, therefore, acceptable to the staff.

During the October 2017 audit, the NRC staff questioned the licensee about those cases where sodium aluminum silicate (SAS) does not precipitate until 24 hours. In particular, the staff was interested in how SAS accumulates on the CS and RHR sump strainers. Following the audit, in the letter dated July 10, 2018, the licensee responded that the CS pumps run through the last 1-minute time step from 1,440 to 1,441 minutes. SAS precipitation occurs at the beginning of that time step. The 1-minute overlap allows the transport of a negligible amount of precipitate to the CS strainers. This does not affect the RHR strainer head loss, since NARWHAL assumes

the full SAS head loss if any precipitate arrives and a filtering bed of fiber is present. Since the quantity of SAS that transports to the RHR strainer can impact the number of breaks assumed to go to failure due to exceeding the tested SAS quantity, the licensee performed an evaluation with the CS pumps shutoff immediately before SAS precipitation was assumed and showed that the CFP was unaffected by the change. The staff finds the licensee's evaluation results acceptable since NARWHAL assumed timing of turning off the CS pumps relative to SAS precipitation does not reduce the strainer CFP.

The licensee also performed sensitivity analysis on various pieces of their risk-informed analysis to determine how changes to various chemical effects components would change ΔCDF. In the RAI responses, the licensee discussed three additional chemical sensitivity cases, compared to the base Vogtle NARWHAL case. Case 1 used more conservative models for aluminum release and aluminum solubility. Case 1 also assumed best estimate maximum pH and a best estimate temperature profile to 620 minutes followed by design-basis temperatures. Case 2 eliminated calcium phosphate debris limit failures, while keeping the calcium phosphate head loss contribution. Case 3 assumed the more conservative aluminum release and aluminum solubility but only precipitated the aluminum amount above the solubility limit. These three cases showed that the changes either produced a decrease in ΔCDF (Cases 1 and 2) or only slightly increased ΔCDF from 2.47×10^{-8} to 2.49×10^{-8} (Case 3).

<u>NRC Staff Conclusion Regarding Chemical Effects</u>

The NRC staff concludes that Vogtle's evaluation of chemical effects is acceptable. The licensee's method for determining the quantity of chemical precipitate and the licensee's evaluation of how the precipitate will affect the ECCS strainer head loss are acceptable to the NRC staff. This staff notes that this evaluation of Vogtle chemical effects does not include the reactor vessel.

The NRC staff concludes that SNC has provided sufficient information as requested by GL 2004-02 and further described in the revised content guide for GL 2004-02 concerning chemical effects, because the licensee:

- Provided a summary of evaluation results that showed that chemical precipitates formed in the post-LOCA containment environment, either by themselves or combined with debris, do not deposit at the sump screen to the extent that an unacceptable head loss results, or deposit downstream of the sump screen to the extent that long-term core cooling is impeded unacceptably.

3.4.2.8.7 <u>Downstream Effects-Components and Systems</u>

The licensee stated that the ex-vessel downstream effects evaluations did not use any unapproved methods or take any exceptions to the NRC-approved methods. The licensee stated that debris effects on components downstream of the sump screen was addressed using staff-approved methods. WCAP-16406-P-A, Revision 1 (Reference 74), was used for the analysis. The licensee stated that each unit has two sets of screens – RHR and CS emergency sump screens. The licensee stated that the adequacy of the sump screens' mesh spacing or strainer hole size is conservatively addressed by assuming that the maximum amount of particulate transported to the strainers passes through the strainers. The licensee also stated that the evaluation used a quantity of fiber debris that passes through the strainers (100 grams per fuel assembly (g/FA)), which is greater than the maximum total reactor vessel fiber load

amount calculated for a hot-leg break. The NRC staff noted that the hot-leg break results in more fiber transport to the core than a cold-leg break, and is, therefore, limiting.

The licensee stated that the Unit 1 strainers were inspected after installation and found to conform to design specifications and that no adverse gaps or breaches were found on the screen surface. The Unit 2 strainers were inspected upon installation, and deficiencies in the fabrication of the CS sump screens were discovered. Specifically, 124 holes had greater than the nominal specified sump screen hole diameter of 0.09375 inches (3/32 inches). The licensee stated that since no holes greater than 0.25-inch diameter were found, ingestion of debris will not cause plugging of downstream CS components because the smallest component diameter is 0.375 inches.

The licensee developed initial debris concentrations using the assumptions and methodology described in Chapter 5 of WCAP-16406-P-A, Revision 1. For conservatism, the licensee assumed that the maximum amount of particulate transported to the strainer would pass through the strainer. The licensee determined that the total maximum initial debris concentration was 919.17 parts per million (ppm), with fiber debris contributing 11.61 ppm, and particulate and coating debris contributing 907.56 ppm.

The licensee reviewed both trains of the RHR system, SI system, component cooling system, and CSS to ensure all of the flowpaths and components impacted by the debris passing through the sump screens were considered.

Based on use of the approved methodology, the licensee stated that the following emergency core cooling system (ECCS) and containment spray system (CSS) components evaluated for Vogtle can accommodate sump bypass particles without blockage: throttle valves, pipes, valves, instrumentation, orifices, heat exchangers, and spray nozzles.

According to the criteria established in WCAP-16406-P-A, the wear impact on four throttle valves was identified as "Evaluate" and was analyzed further for additional opening to address erosion concerns. Based on the additional analysis, the licensee determined that the valves passed the acceptance criteria.

For pumps, the effects of debris ingestion through the sump screen were evaluated for hydraulic performance, mechanical shaft seal assembly performance, and mechanical performance (vibration) of the pumps. The hydraulic and mechanical performances of the ECCS and CSS pumps were determined to be unaffected by the recirculating sump debris. The mechanical shaft seal assembly performance evaluation resulted in one action item suggesting replacement of the RHR pumps' carbon/graphite backup seal bushings with a more wear-resistant material, such as bronze. However, because Vogtle has an engineered safety feature atmospheric filtration system in its auxiliary building, this action was not required per WCAP-16406-P-A.

<u>NRC Staff Conclusion Regarding Downstream Effects Components and Systems</u>

The NRC staff reviewed the evaluation methods and results and the licensee's conclusions, and concludes that the licensee followed the NRC staff-accepted guidance contained in TR WCAP-16406-P-A, Revision 1, including the NRC SE for that document. The NRC staff concludes that the licensee performed an adequate downstream effects evaluation of components and systems and that the components are capable of performing their safety-related design functions for their required mission times after a LOCA.

The NRC staff concludes that SNC has provided sufficient information as requested by GL 2004-02 and further described in the revised content guide for GL 2004-02 concerning downstream effects components and systems, because the licensee:

- Summarized the application of NRC-approved methods, and indicated where NRC-approved methods were not used or exceptions were taken, and provided sufficient technical evaluation of those areas.
- Provided a summary and conclusions of downstream evaluations.
- As applicable, provided a summary of design or operational changes made as a result of downstream evaluations.

NRC Staff Conclusion Regarding Impact of Debris Submodel

Each of the aspects of the impact of debris area has been evaluated above. The NRC staff concludes that the sub-areas of upstream effects, screen modification package, head loss and vortexing, sump structural analysis, NPSH, chemical effects, and downstream effects – components and systems were addressed adequately. The NRC staff finds that the licensee's impact of debris evaluation is acceptable with the exception of the downstream effects-fuel and vessel submodel (i.e., in-vessel evaluation).

3.4.2.9 Submodel Integration

This section provides an overview of how the submodels discussed above are combined to obtain the final results of the risk analysis.

The licensee used an integrated model called NARWHAL to calculate conditional probabilities of system (strainer or core) failure for each pump state scenario. The model calculated strainer parameters (e.g., buildup of fiber, calcium phosphate, SAS, particulates, head loss, extent of unsubmerged strainer), and core fiber buildup parameters (e.g., accumulated fiber) for each time step and compared those parameters to different failure criteria. The NARWHAL model used as inputs post-processed outputs from a program called BADGER, which was used to compute the amounts of each type of debris generated for each break scenario. NARWHAL also used a significant number of user-generated inputs to define quantities such as debris transport, dissolution rates and chemical precipitation parameters, strainer head loss, debris penetration, flow rates, etc. The major components of the NARWHAL model are a water balance model, a chemical product formation and precipitation model, and a debris mass balance model using pump flow rates to define rates of debris buildup on strainers and in the core, as well as transport and erosion fractions, to define amounts of transportable debris. For each break and orientation, NARWHAL keeps track of whether conditions were established at strainers or in the core leading to system failure during the simulation time.

The debris generation model is implemented in a separate, independent code named BADGER. BADGER includes a 3-dimensional description of the distribution of debris sources, robust barriers, and weld locations. For each weld location, BADGER varies the break size and orientation, and outputs debris amounts for each postulated break. The BADGER outputs are post-processed to define a database of debris amounts for each of the weld locations, break sizes, and orientations. The break and debris amount database is input to NARWHAL for the computation of the conditional strainer failure and core failure probability. NARWHAL applies blowdown, washdown, pool fill, and recirculation transport fractions, as well as erosion fractions, to define debris amounts that transport to the strainers. NARWHAL accounts for additional

debris sources (besides those accounted by BADGER), which are ZOI independent (e.g., latent debris and unqualified coatings) and assumed present for all breaks. NARWHAL is used to compute the debris distribution as a function of time in the pool, on the strainers, and in the core. Based on debris amounts and other physical variables such as temperature, pressure, flow rates, and the pool level, NARWHAL computes whether a specific break would lead to a failure state (failure of the strainer, pump, or core). To establish whether failure occurs, NARWHAL computes, at each time step, debris buildup at strainers, fiber buildup in the core, strainer head loss, NPSH margins, strainer deaeration rates, strainer structural margin, and local pressures (to determine whether flashing may occur). Failure is defined based on comparison of specific quantities (e.g., mechanical collapse of the strainer would occur if the head loss exceeds the strainer structural margin). The NARWHAL model also calculates whether a partially submerged strainer would pass adequate flow to the pumps and whether the pump void fraction would exceed an operational limit. Core failure is postulated to occur when fiber buildup in the core exceeds in-vessel limits. In addition, strainer failure is postulated to occur if debris limits established by the ranges of testing are exceeded. In the detailed computations of the various pump state scenarios, debris limit exceedance accounted for all instances of strainer failure.

NARWHAL computes conditional failure probability (CFP) based on counts of breaks (different breaks are defined by location, size, and orientation triplets) causing failure. The CFPs were split up into three break size ranges or categories: small (break < 2 inches), medium (2 inches ≤ break < 6 inches), and large break categories (break > 6 inches). NARWHAL retains separate counts of different failure mechanisms and has the capability of computing the CFP for each high-likelihood configuration analyzed. The main outputs for use in the PRA model are the strainer CFP (strainer failure by any failure mode but excluding breaks causing core blockage) and the core-blockage CFP (independent of whether the same break could cause strainer failure) for the three break ranges. The CFP is split into strainer failure and core blockage as inputs to the PRA because core blockage and strainer failure due to debris are independent items in the Vogtle PRA.

The NRC staff verified that the licensee's calculations were performed accurately and used acceptable assumptions (Reference 22). The NRC used a combination of confirmatory calculations, engineering review, and review of the licensee's software outputs to perform the verifications. The verifications included balancing debris amounts output by BADGER with debris amounts tracked in NARWHAL, exploring trends in debris amounts vs. break size and orientation for numerous welds, identifying the consistency of failure conditions, examining dynamic outputs (e.g., head loss vs. time plots), and consistency of changes and inflections with the modeled phenomenology. The verifications also included independent computations of the CFP for large breaks. The approach allows the staff to conclude, with a high level of confidence, that the calculations for debris generation were conducted and applied properly, and are, therefore, acceptable.

<u>NRC Staff Conclusion Regarding Submodel Integration</u>

The NRC staff concluded that the licensee's submodel integration was acceptable based on its review of the methodology and the NARWHAL results. The NRC staff concludes that the approach for integrating submodels described in the technical report is acceptable for use in an assessment or evaluation model of the effects of debris on long-term cooling of ECCS, as required, in part, by 10 CFR 50.46.

3.4.2.10 Systematic Risk Assessment

RG 1.174, Revision 2 (Reference 20), states that the licensee may use its risk assessment to address the risk-informed decision-making principle that proposed increases in risk are small and consistent with the intent of the NRC's Safety Goal Policy Statement.

In Enclosures 1 and 3 of the letter dated April 21, 2017, the licensee described its approach used to quantify the risk impact (ΔCDF and ΔLERF) attributable to debris. This risk was defined as the difference in risk between the as-built, as-operated plant (with debris) and a hypothetical plant with no risk from debris.

The licensee first screened out or determined the bounding risk attributable to debris for all plant operating modes, except full-power operation, and for all initiating events, except small-, medium-, and large break LOCAs (including seismically-induced LOCAs), and SSBIs. A location-specific screening process further refined the scope to a discrete set of breaks: breaks in ISI welds in the unisolable portion of the Class 1 pressure boundary (i.e., inside the first isolation valve) and main steam line break (MSLB) and feedwater line break (FWLB) side breaks inside containment (SSBIs).

The conditional probability of strainer failure was computed based on comparison of the amount and type of debris generated and transported by each break to test amounts. For the purposes of risk quantification, any break predicted to generate and transport debris in excess of the tested value was assumed to fail the strainer. The licensee used NARWHAL to quantify the CFP of the strainers. In the letter dated April 21, 2017, the licensee stated NARWHAL accounts for the possibility of strainer failure due to physical processes such as strainer head loss (arising from debris buildup) in excess of the NPSH margin and strainer head loss in excess of structural margin, as well as pump failure due to activating the ECCS when the strainer is partially submerged, causing gas voids downstream of the strainers in excess of void fraction limits. However, the only failure mode that affected the risk attributable to debris was strainer debris buildup beyond tested debris limits because this failure mode occurred before any other failure criteria were reached. Breaks predicted to produce debris below the testing limits were assumed not to fail (i.e., CFP = 0).

A total of 413 inservice inspection (ISI) unisolable welds were explicitly included in the analysis. The licensee excluded welds downstream of the first isolation valve from the analysis because those welds are, in general, located in small-diameter pipes and because of the low likelihood of the isolation valve failing to close, spurious opening of the valve, or a large valve leak (Reference 7). The NRC staff concludes that exclusion of welds downstream of the first isolation valve is acceptable because those welds are located in small-diameter pipes (i.e., the largest diameter pipe is 10.5 inches), and breaks in those welds would produce limited debris amounts, well bounded by debris amounts generated by breaks considered in the analysis.

The next step in the licensee's methodology was to estimate the weld-specific break frequency for each of the 413 welds considered in the analysis. In order to perform this step, the licensee made the following key assumptions:

1. The frequency of a LOCA at a given location is only a function of the break size (e.g., all 7-inch breaks have the same frequency, all 8-inch breaks have the same frequency, etc.). This approach is sometimes referred to as the top-down approach because it starts with a plant-wide ("top level") LOCA frequency and allocates it uniformly to the possible break locations according to break size. The licensee also

considered alternative frequency allocation schemes based on alternative damage mechanisms using sensitivity analyses.

2. A complete break of a given size in one pipe is exactly as likely as a partial break of the same size in a larger pipe. This assumption is referred to as the "continuum break" approach. The licensee also considered a "DEGB-only" assumption in which only complete DEGBs were evaluated using sensitivity analyses.

3. NUREG-1829 (Reference 54) frequencies were combined using a geometric mean aggregation scheme. The licensee also considered an arithmetic mean aggregation scheme using sensitivity analyses.

4. The frequency for LBLOCAs (> 6 inches) is discretized with break intervals from 6 to 15 inches, 15 to 25 inches, and 25 inches and larger with frequency allocated individually within each interval. The licensee considered alternative intervals using sensitivity analyses.

Other assumptions used by the licensee in its analyses are explicitly listed in Enclosure 3, Section 13.2, of the letter dated April 21, 2017. Those assumptions are related to the physical characteristics of the system, the debris generation process, and establishment of strainer failure conditions that are described elsewhere in this SE. Uncertainty is introduced into the PRA results if there is no consensus about which model most appropriately represents the particular aspect of the plant being modeled. The licensee recognized Key Assumptions 1 to 4 to be associated with "non-consensus models" and requiring additional uncertainty evaluation per NUREG-1855, Revision 1 (Reference 47)[3].

With respect to the licensee's choice of a top-down allocation scheme (i.e., Key Assumption 1 listed above), the NRC staff notes that there is a lack of consensus regarding how to apportion plantwide LOCA frequencies to individual weld locations. NUREG-1829 (Reference 54) contains only qualitative statements about similarly sized welds in different locations having different expected rupture frequencies due to degradation mechanisms (e.g., hot-leg vs. cold-leg). Similarly, the NRC staff notes that there is a lack of consensus on how to compare the likelihood of a complete break of a given size to a partial break of the same size (i.e., Key Assumption 2 listed above). NUREG-1829 states that the expert panel was, in general, of the opinion that a complete break of a given size is more likely than a partial break of the same size; however, the panel did not offer quantitative information to support this opinion. NUREG-1829 does not endorse a specific aggregation scheme (i.e.., Key Assumption 3 listed above), but states, "The purposes and context of the application must be considered when determining the appropriateness of any set of elicitation results." NUREG-1829 also does not endorse a specific method for discretizing specific break intervals. The licensee used NUREG-1829 LOCA frequencies (25 years fleet average operation tables) except for small breaks.

The licensee postulated a range of break sizes, varying the break orientation in 45-degree increments around the pipe circumference. For each break, the licensee used the BADGER and NARWHAL software to compute debris amounts that could be transported to the strainer and compared those amounts to test limits. The licensee defined the strainer CFP as the number of breaks associated with failure conditions divided by the total number of examined

[3] RG 1.174, Revision 2, defines a "consensus" model as one that the NRC has utilized or has accepted for the specific risk-informed application under consideration or that has a publicly-available basis, has been peer reviewed, and has been widely adopted by the appropriate stakeholder group.

breaks within a specific break size interval. This ratio implicitly assumes that all breaks of the same size on different welds are equally likely. The CFP for each break size interval included a factor equal to the conditional probability for a break to belong to the break interval, given the occurrence of an LBLOCA (break greater than 6 inches). This conditional probability equals the difference in exceedance frequency of the break interval ends (interpolated from NUREG-1829 LOCA frequencies) divided by the frequency of breaks exceeding 6 inches (i.e., LBLOCAs). The total CFP is the sum of the individual CFP for each break interval. The licensee also examined alternative qualitative strategies to distribute LOCA frequencies unevenly among different welds, considering different degradation mechanisms for different welds, and presented those results as sensitivity analyses.

The licensee identified different pump states or configurations and computed the strainer CFP for each state using the NARWHAL software program. The high-likelihood scenarios examined were: (1) no equipment failure (0.915), (2) two CS pumps failed (0.0531), (3) one RHR pump failed (0.0146), (4) one CS pump failed (0.0126), and (5) one RHR and one CS pumps failed (0.0039). The numbers in parenthesis are the conditional state probabilities provided by the licensee from its PRA model (values referred to as functional failure probabilities or FFP; see Table 3, Enclosure 3 of the licensee's letter dated April 21, 2017).

For each configuration *i*, the licensee computed the corresponding CFP_i. The value CFP_i was input to the PRA for each LOCA break category (i.e., small, medium, and large). Since CFP strainers were determined to not fail for SBLOCAs and MBLOCAs for high-likelihood configurations, the licensee concluded that only LBLOCAs could cause strainer failure and core damage. Each unit at Vogtle includes two independent trains, trains A and B, of the ECCS (each train with a CS and RHR pump and corresponding individual strainers). As stated in the letter dated January 9, 2018, the licensee assumes symmetric transport of debris to both RHR strainers. Since the PRA success criteria only requires operation of a single RHR train for LBLOCAs, the licensee determined the symmetric transport assumption to yield conservative risk values because it maximizes the potential for both strainers to fail.

The licensee relied on the PRA model for the computation of the total ΔCDF and ΔLERF for high-likelihood configurations with CFPs calculated in NARWHAL as inputs to GSI-191-related PRA logic.

The licensee also examined low-likelihood configurations such as failed NSCW and failed AC power. The licensee implemented a simplified approach to compute the ΔCDF associated to low-likelihood LBLOCA configurations, using the following equation:

$$\Delta\mathrm{CDF} = F(x \geq 6\ \mathrm{in}) \sum_{i=1}^{n} FFP_i\, CFP_i \quad (1)$$

The symbol $F(x \geq 6\ \mathrm{in})$ is the LBLOCA exceedance frequency (i.e., frequency of the initiating event for large LOCA breaks based on NUREG-1829); FFP_i is the functional failure probability of the low-likelihood configuration *i* (based on the PRA model), and CFP_i is the strainer CFP. The licensee determined CFP_i for low-likelihood configurations by identifying an equivalent high-likelihood configuration value (e.g., failure of the NSCW is equivalent to failure of one train − one RHR and one CS pump). If there was no equivalent high-likelihood state, the licensee assumed that the strainer would fail ($CFP_i = 1$). The licensee assumed that only LBLOCAs would cause strainer failure associated with eventual core damage. Table 4 provides total risk impact attributable to debris at Vogtle.

Table 4: Vogtle Total Risk Impact Attributable to Debris
(Table 1-1 in Enclosure 1 of the licensee's letter dated April 21, 2017)

Case	ΔCDF per reactor year (/ry)	ΔLERF (/ry)
High-likelihood LOCA configurations	2.32×10^{-8}	3.10×10^{-11}
Low-likelihood LOCA configurations	1.41×10^{-9}	4.09×10^{-12}
Seismically-induced LOCAs	1.50×10^{-9}	1.50×10^{-10}
SSBIs	1.39×10^{-9}	8.25×10^{-11}
Total	$\mathbf{2.75 \times 10^{-8}}$	$\mathbf{2.68 \times 10^{-10}}$

In the table above, the licensee computed the ΔLERF using its PRA model except for the low-likelihood LOCA configurations. As stated in the May 23, 2018, letter for that configuration, the licensee used an average CLERP, given core damage equal to 2.9×10^{-3}, derived on the basis of the base-case model of record (i.e., the ratio of LERF to CDF), including all relevant initiating events that could cause core damage and large early releases (not exclusively GSI-191 LOCA events).

3.4.2.10.1 NRC Staff Confirmatory Calculations

The NRC staff performed confirmatory calculations, focusing on the high-likelihood equipment configurations, to examine if the estimated risk impact attributable to debris is within the RG 1.174, Revision 2, acceptance guidelines (Reference 20) even with the conservative assumption that any break larger than the minimum break would cause strainer failure and core damage. The confirmatory calculations also assess the sensitivity of change-in-risk estimates to CFP with different modeling assumptions (e.g., alternative frequency allocation to LOCA breaks). These calculations assumed core damage (i.e., CFP = 1) for the smallest break the licensee identified with strainer failure conditions and all breaks equal to or larger than this break regardless of location.

Table 5 includes NRC staff estimates. As described in the letter dated February 12, 2018, the licensee reported a minimum break size associated with strainer failure equal to 12 inches for the configuration that one train is out of service (one functional RHR pump and one functional CS pump). For the configuration one RHR pump failed, it is expected that the minimum break causing strainer failure would also be close to 12 inches. For all of the other configurations when at least two RHR pumps are functional, it is expected that the minimum break size would be larger than 12 inches. Thus, in the conservative computations for each of these pump configurations, it was assumed that any break larger than the minimum break would cause strainer failure and core damage. Several alternative values of the minimum break were considered (e.g., 13, 15, 17 inches) for the configurations with at least two functional RHR pumps. While minimum break sizes larger than 17 inches are plausible for configurations with at least two functional RHR pumps, break frequencies used to calculate ΔCDF for larger break sizes would be smaller, and therefore, would be less conservative in this simplified calculation. Therefore, Table 5 reflects break sizes between 12 inches and 17 inches.

The ΔCDF associated with each alternative minimum break was estimated as follows: the column labeled minimum break is the assumed minimum break size causing strainer failure for each of the indicated configurations. The NRC staff used semi-log interpolation of NUREG-1829 LOCA frequencies to determine the plantwide initiating event exceedance

frequencies (columns labeled f_{GM} and f_{AM}). Semi-log interpolation is acceptable as explained in Section 3.4.2.2, "Initiating Event Frequencies." Interpolated LOCA frequencies from NUREG-1829, Table 7.1 (geometric mean aggregation), and from Table 7.11 (arithmetic mean aggregation), are included in the columns labeled f_{GM} and f_{AM}. The functional failure probabilities in the column labeled *FFP* are the CFPs of the equipment configuration computed by the licensee as reported in Table 3-3 of Enclosure 3 of the letter dated April 21, 2017. The total ΔCDF was computed by adding the products $FFP \times f_{GM}$ or $FFP \times f_{AM}$.

Table 5: Summary of NRC Staff Conservative Confirmatory Calculation

Equipment configuration	**Minimum break size (inches)**	**Geometric mean f_{GM} (min break) /yr**	**Arithmetic mean f_{AM} (min break) /yr**	**Functional failure prob. *FFP***	**$FFP \times f_{GM}$ (min break) /yr**	**$FFP \times f_{AM}$ (min break) /yr**
No equipment failure	12 – 17	2.18×10^{-7} – 5.44×10^{-8}	2.06×10^{-6} – 9.89×10^{-7}	0.915	7.08×10^{-8} – 1.99×10^{-7}	9.04×10^{-7} – 1.88×10^{-6}
Two CS pumps failed	12 – 17	2.18×10^{-7} – 7.74×10^{-8}	2.06×10^{-6} – 9.89×10^{-7}	0.0531	4.11×10^{-9} – 1.16×10^{-8}	5.25×10^{-8} – 1.09×10^{-7}
One RHR pump failed	12	2.18×10^{-7}	2.06×10^{-6}	1.46×10^{-2}	3.18×10^{-9}	3.00×10^{-8}
One CS pump failed	12 – 17	2.18×10^{-7} – 6.12×10^{-8}	2.06×10^{-6} – 9.89×10^{-7}	1.26×10^{-2}	9.75×10^{-10} – 2.74×10^{-9}	1.25×10^{-8} – 2.59×10^{-8}
One train (one RHR and one CS pumps) failed	12	2.18×10^{-7}	2.06×10^{-6}	3.90×10^{-3}	8.49×10^{-10}	8.02×10^{-9}
				Total ΔCDF	**7.99×10^{-8} – 2.18×10^{-7}**	**1.01×10^{-6} – 2.05×10^{-6}**
				Licensee	**2.32×10^{-8}**	

The estimate of ΔCDF using the geometric mean aggregation NUREG-1829 LOCA frequencies ranges from 7.99×10^{-8} to 2.18×10^{-7} /yr for the high-likelihood configurations. The corresponding licensee's estimate is 2.32×10^{-8} /yr (Enclosure 1, Table 1-1, to the letter dated April 21, 2017), which is lower than the NRC estimates because the minimum break sizes were larger (and less frequent) than the NRC assumed values, and because only a fraction of the breaks above the minimum break cause strainer failure in the detailed computations by the licensee. Using the arithmetic mean aggregation NUREG-1829 LOCA frequencies, the conservative ΔCDF ranges from 1.01×10^{-6} to 2.05×10^{-6} /yr. As stated in Table 3-17 of Enclosure 3 of the letter dated April 21, 2017, the licensee estimated the ΔCDF (arithmetic mean aggregation) is 5.28×10^{-7} /yr. The estimates are conservative and indicate a small to very small increase in risk.

Using the average CLERP equal to 2.9×10^{-3}, the conservative ΔLERF ranges from 2.32×10^{-10} to 6.31×10^{-10} /yr for the geometric mean aggregation, and from 2.92×10^{-9} to 5.96×10^{-9} /yr for the arithmetic mean aggregation, for the high-likelihood configurations. Both of these estimates are conservative and consistent with the licensee's conclusion of a very small increase in risk.

The staff's confirmatory calculations indicate that the estimated risk impact attributable to debris for this assessment is within the RG 1.174, Revision 2 (Reference 20), acceptance guidelines, even with the conservative assumption that any break larger than the minimum break would cause strainer failure and core damage. Alternative bounding estimates of the ΔCDF available elsewhere support the same conclusion by the NRC staff (Reference 22).

NRC Staff Conclusion Regarding the Systematic Risk Assessment

The NRC staff reviewed the licensee's GSI-191 systematic risk assessment methodology and concludes that it is acceptable because inputs and assumptions (e.g., initiating event frequencies for critical welds) were derived using state-of-practice data or approaches, adequate changes were made in the base PRA to establish the cause-effect relationship, scenarios that affect the GSI-191 risk assessment were adequately identified and included in the risk evaluation, various elements of the risk evaluation (e.g., development of CFPs using the BADGER and NARWHAL software) were developed in a systematic and acceptable manner, and key assumptions were appropriately considered. Therefore, the methodology described in the technical report can be used to calculate the risk attributable to debris using an acceptable PRA.

3.4.2.11 Sensitivity and Uncertainty Analyses

RG 1.174, Revision 2, states, in part, "The licensee should appropriately consider uncertainty in the analysis and interpretation of findings." Consistent with RG 1.174, Revision 2, comparisons to the risk acceptance guidelines should be made with appropriate consideration of the uncertainties involved.

RG 1.174, Revision 2, refers to the uncertainty framework described in NUREG-1855, Revision 1 (Reference 47), for acceptable approaches to addressing uncertainty. Section 7 of NUREG-1855, Revision 1, states that the goal of an uncertainty analysis is examining challenges posed by uncertain elements of the risk-informed analysis to the conclusion that ΔCDF and ΔLERF estimates are within acceptance guidelines defined in RG 1.174, Revision 2.

In the letters dated April 21, 2017; January 9, 2017; and July 10, 2018, the licensee provided information regarding its uncertainty analysis and results. Consistent with NUREG-1855 guidance, the licensee addressed parametric, model, and completeness uncertainty by performing techniques such as direct quantitative analysis, bounding approaches, screening techniques, sensitivity studies, and use of approved consensus deterministic methods.

In assessing parametric, model, and completeness uncertainties, the licensee examined sources of uncertainty that could affect the decision being made (i.e., whether the RG 1.174 risk acceptance guidelines are met). For example, the uncertainties associated with initiating event frequency (plantwide and location-specific), debris generation, debris transport, head loss at the strainer, chemical effects, strainer penetration, downstream effects (in-vessel and ex-vessel), and calculation of the baseline CDF and LERF (for comparison to risk acceptance guidelines) were examined. The sensitivity studies cover a range of uncertainties in the assumptions related to physical characteristics and attributes of the system (e.g., pump flow rates, pressure and temperature, sump pH), debris amounts (e.g., uncertainty in ZOI and latent debris), and failure criteria (e.g., debris limits for fiber and particulates). The licensee identified the parameters with the greatest effect on ΔCDF, including the hot-leg break fiber fine limit for core blockage, pump flow rates, strainer debris limits, and amount of debris generated from each initiating break. The licensee explained nonintuitive trends in its sensitivity results in the letter

dated January 9, 2018. The licensee indicated that nonintuitive results are expected from competing physical factors and processes modeled in the NARWHAL code. The NRC staff evaluated the licensee's discussion related to nonintuitive results and concludes it is acceptable.

3.4.2.11.1 Parametric Sensitivity and Uncertainty Analysis

The licensee performed parametric sensitivity analysis to identify which inputs have the greatest impact on the risk quantification results. The licensee stated its analysis included the process of identifying input variables to evaluate; selecting minimum, nominal, and maximum values for each variable; quantifying risk in terms of ΔCDF as a common output that can be compared for each sensitivity; and using the ΔCDF results to rank the sensitivity of each input variable. The licensee provided its approach for determining the minimum, nominal, and maximum values for each of the parametric sensitivity inputs (e.g., change value by a percentage of nominal value, use design limits, use minimum or maximum range value, 5^{th} or 95^{th} percentiles).

The inputs and results of the licensee's parametric sensitivity analysis were provided in Tables 3.11 and 3.12 and Figure 3-9 of Enclosure 3 of the letter dated July 10, 2018. The licensee determined that the parameters that have the most significant effect on ΔCDF are the in-vessel fiber limit for hot-leg breaks, the LOCA frequency values, and pump flow rates.

The licensee performed a parametric uncertainty analysis to quantify the overall uncertainties associated with the input parameters. Therefore, the effect of simultaneous variations in multiple input parameters was considered, but none of the inputs were shifted beyond the bounds of realistic plant-specific conditions.

The parametric uncertainties were quantified using a series of sensitivities with a bounding set of input parameters with respect to strainer and core failures. In cases where the bounding direction for a given input parameter (e.g., pool volume/level) could not be determined, both the minimum and maximum values were considered. Table 3-13 of Enclosure 3 of the licensee's letter dated July 10, 2018 shows the worst case conditions for strainer failures; Table 3-14 shows the worst case conditions for core failures.

The head losses measured at the tested debris quantities allow for significant margins for strainer acceptance criteria such as NPSH and strainer structural criteria. The NRC staff evaluated the licensee's plant-specific testing to verify that conservative inputs were used to develop the strainer head loss testing results. The NRC staff concludes that the inputs are suffcient, and the testing results provide assurance that uncertainty has been addressed and that the head loss associated with the debris load in the testing bounds any head loss that would occur in the plant for the same debris load. The NRC staff finds that the analysis is conservative in that the strainer will likely withstand debris loads beyond the tested amounts if not failed during the test.

The licensee equated exceedance of debris limits with strainer failure. If any single debris type amount calculated to reach the strainer was greater than the tested value for the debris type, the scenario was considered to end with core damage. This is conservative because it is likely that small or potentially even large increases in debris amounts arriving at the strainer would not cause a failure of the strainer or ECCS. There are large margins between the maximum strainer test head losses and the NPSH and structural failure criteria. In addition, strainer failure may not result in core damage in every case because the licensee has procedures to detect and mitigate strainer blockage, even during scenarios where ECCS and CSS performance is challenged by debris.

The licensee did not credit containment pressure for its NPSH calculations. However, it did credit pressure of a limited magnitude and duration to suppress flashing across the strainer and degasification of the fluid as it passes through the debris bed. The licensee demonstrated that only a fraction of the available pressure was credited, using both design basis and realistic calculations. Because the licensee calculated significant margins to flashing and deaeration using design basis and realistic calculations, the NRC staff has confidence that uncertainties associated with the calculations will not significantly affect the risk results of the evaluation. The licensee's parametric sensitivity analysis that varied the amount of pressure credited for the analysis supports the conclusion that the change in risk is not significantly affected by the value of pressure credited.

The licensee's parameter sensitivity and uncertainty analyses included variation in the amounts of debris, debris acceptance criteria, head loss assumed to occur from debris loads, flow rates through the strainers, duration of flow through the CS strainers, as well as many other parameters. These studies indicate that reasonably expected variations in the parameters analyzed, as well as changing inputs to values outside of their reasonably expected range, do not affect the decision related to very small risk attributable to strainer-related failures.

The NRC staff concludes that the licensee appropriately identified and evaluated sources of parameter uncertainty because uncertainty analyses were performed in accordance with guidance provided in NUREG-1855, Revision 1.

3.4.2.11.2 Model Uncertainty Analysis

To address model uncertainty associated with the use of non-consensus models, the licensee calculated the change in ΔCDF between the NARWHAL base-case and alternative models. The licensee's results of these sensitivity analyses are provided in Table 3-17 of Enclosure 3 of the licensee's letter dated July 10, 2018. The licensee examined the effects of using different break models (continuum vs. DEGB), aggregation methods for LOCA frequencies (geometric vs. arithmetic mean aggregation), LOCA frequency allocation to individual welds (uniform allocation vs. hybrid allocation based on weld degradation mechanism probability weighting), LBLOCA size range discretization, chemical product generation models, and NARWHAL time step sizes (2, 3, 4, 5, and 15 min). While selection of the arithmetic aggregation method resulted in the largest increase in ΔCDF estimates compared to other sensitivity cases, all cases evaluated resulted in ΔCDF estimates less than 1×10^{-6} /yr.

The licensee used accepted methods and consensus models for many aspects of the evaluation. These methods and models, when applied to deterministic evaluations, are expected to result in outcomes that are conservative for a plant, given its plant-specific conditions. The discussion below includes additional details regarding the licensee's use of consensus and non-consensus models in its analysis.

The NRC staff determined that the licensee conducted strainer testing using realistic or conservative inputs and assumptions based on consensus models. The test inputs were developed, and the test was performed using NRC staff guidance (References 27 and 34). Use of test inputs consistent with NRC staff guidance ensures conservative results such that the test will result in a head loss that is bounding for the plant-specific conditions.

The evaluation of the transport of debris is discussed in Section 3.4.2.7, "Debris Transport." The licensee used assumptions in the development of these inputs that were consistent with accepted staff guidance (Reference 27) and that are considered to consist of consensus models. The staff guidance identifies conditions that would be most challenging to the strainer, considering plant-specific inputs.

LOCA frequency allocation is a key model uncertainty in the analysis; however, the licensee used NUREG-1829 (Reference 54) as the source of LOCA frequency information and presented results based on several different sets of assumptions (e.g., "top-down" LOCA frequency allocation with values from the Vogtle PRA vs. NUREG-1829 arithmetic mean values and examination of various hybrid allocation methodologies). Additionally, in the letter dated February 12, 2018, the licensee stated that welds on pipes are proper representatives of LOCA break sites. The licensee considered non-pipe-break LOCAs (e.g., valve bodies and manways) and determined that pipe weld locations bounded non-pipe locations. Therefore, the licensee used the plantwide LOCA frequencies from NUREG-1829, which includes both pipe and non-pipe locations, for the pipe-break LOCAs. An additional assumption is that pipe-break LOCA events only occur at welds; this is consistent with the guidance in NUREG-1829.

The licensee performed model uncertainty evaluations that studied the effect of assumptions regarding the fiber bed thickness required to cause chemical precipitates to contribute to head loss.

The licensee performed a chemical effects evaluation using the WCAP-16530-NP-A (Reference 28) method with refinements for aluminum passivation by phosphate and credit for solubility of aluminum at elevated temperatures. The licensee performed sump strainer testing and simulated chemical effects by adding pre-mixed precipitate at the Alion Science and Technology Laboratory. The licensee also performed plant-specific testing at the University of New Mexico to evaluate aluminum passivation in a trisodium phosphate buffered solution. Although the licensee modified the WCAP-16530-NP-A evaluation by crediting aluminum solubility and aluminum passivation, its evaluation is acceptable since the method for determining the quantity of chemical precipitate and the evaluation of how the precipitate will affect the ECCS strainer head loss are acceptable to the NRC staff. The licensee evaluated three additional chemical sensitivity cases, compared to the base Vogtle NARWHAL case. These cases either evaluated more conservative models for aluminum release and aluminum solubility, or eliminated calcium phosphate debris limit failures, while keeping the calcium phosphate head loss contribution, or assumed the more conservative aluminum release and aluminum solubility but only precipitated the aluminum amount above the solubility limit.

The NRC staff concludes that the licensee appropriately identified and evaluated sources of model uncertainty because uncertainty analyses were performed in accordance with guidance provided in NUREG-1855, Revision 1.

3.4.2.11.3 Completeness Uncertainty

The licensee provided its qualitative assessment of completeness uncertainty in Enclosure 1, Section 5.3, of the letter dated July 10, 2018. The licensee stated that it determined completeness uncertainty to be low because its risk assessment was of the appropriate scope and depth, and areas not explicitly evaluated were determined to have a low potential for any significant risk impact. The licensee noted that all known GSI-191 phenomena and debris failure mechanisms were evaluated either in a bounding manner for phenomena not explicitly included in the Vogtle risk model or in a reasonably conservative manner for phenomena that were included in the risk model. The licensee described that NRC and industry testing has elucidated many potential effects of GSI-191 phenomena, including aspects related to insulation and coatings destruction from break jets; unqualified coatings failure; blowdown and washdown debris transport; containment pool settling, tumbling, and lift-over-curb debris transport; debris erosion; chemical release, solubility, and precipitation; strainer head loss, vortexing, and penetration; ex-vessel component wear; and in-vessel core blockage and boron precipitation. The licensee concluded that, given the extensive research performed, it is unlikely that there are unidentified phenomena that would significantly increase the risk of GSI-191-related failures.

The NRC staff reviewed the licensee's assessment of completeness uncertainty and concludes it is acceptable because the licensee addressed known and unknown phenomena associated with GSI-191 in its risk assessment and addressed uncertainties related to those phenomena in a comprehensive and consistent manner with guidance in NUREG-1855.

NRC Staff Conclusion Regarding Sensitivity and Uncertainty Analyses

The licensee stated that because all of the cases that were evaluated for model uncertainty and parametric uncertainty resulted in a ΔCDF less than 1×10^{-6}, it can be concluded with high confidence that the risk associated with GSI-191 is very low as defined by the acceptance guidelines in RG 1.174. The NRC staff reviewed the basis for this determination and concludes that the licensee has adequately identified parameter, modeling, and completeness uncertainties that may affect risk assessment results; performed adequate analyses to understand the impact of these uncertainties (and related assumptions);, and demonstrated that the risk attributable to debris is very small (all of the ΔCDF estimates using alternative assumptions were less than 1×10^{-6}), as defined in RG 1.174, Revision 2, because it appropriately applied techniques described in NUREG-1855 for assessing uncertainty.

3.4.3 NRC Staff Conclusion Regarding Key Principle 4: The Increase in Risk

Principle 4 in RG 1.174, Revision 2, states that any increase in risk associated with a proposed change should be small (i.e., within the risk acceptance guidelines). The risk acceptance guidelines are presented in Figures 4 and 5 of RG 1.174, Revision 2, for core damage frequency (CDF) and large early release frequency (LERF), respectively. Note that those figures use the mean values of CDF, ΔCDF, LERF, and ΔLERF.

As stated in Enclosure 1, Table 1-3, of the letter dated April 21, 2017, the CDF for Vogtle Units 1 and 2, is 4.39×10^{-5} /yr and 5.05×10^{-5} /yr, respectively. The LERF for Units 1 and 2 is 1.73×10^{-6} /yr and 1.90×10^{-6} /yr, respectively. From RG 1.174, Revision 2, Figure 4, this base

CDF means that an increase in CDF of 10^{-5} /yr or less is considered "small," and an increase of 10^{-6} /yr or less is "very small." Similarly, from Figure 5, the base LERF means that an increase in LERF of 10^{-6} /yr or less is considered "small," and an increase of 10^{-7} /yr or less is "very small."

The licensee's mean-value estimates for the increase in CDF attributable to debris provided in the letter dated April 21, 2017, Enclosure 1, Table 1-1, and the corresponding conclusion from RG 1.174, Revision 2, Figures 4 and 5, are shown in the following table, which also includes the NRC staff's conservative calculation:

Table 6: Total Risk Attributable to Debris for Vogtle

Description	ΔCDF (/ry)	ΔLERF (/ry)	Increase
Total risk increase associated with GSI-191 (GM aggregation)	2.75×10^{-8}	2.68×10^{-10}	Very small
Total risk increase associated with GSI-191 (AM aggregation)	5.28×10^{-7}	1.53×10^{-9}	Very small
NRC conservative (GM aggregation)	2.18×10^{-7}	6.32×10^{-10}	Very small
NRC conservative (AM aggregation)	2.06×10^{-6}	5.97×10^{-9}	Small

GM: geometric mean
AM: arithmetic mean
ry: reactor year

The NRC staff finds that the base PRA risk and various estimates of the increase in CDF associated with debris, including consideration of uncertainty, meet the acceptance guidelines in RG 1.174, Revision 2. The acceptance guidelines for LERF (Figure 5 of RG 1.174) are an order of magnitude lower than CDF for both base LERF and ΔLERF. In estimating the ΔLERF, an average CLERP equal to 2.9×10^{-3} (RAI 38) was used for all entries in Table 6, except the first row. The estimated increase in LERF is "very small" for all the cases shown in the above table. Therefore, the NRC staff concludes that the licensee adequately demonstrated that the increase in risk attributable to debris is acceptably small because:

- The licensee used a PRA of the appropriate scope, level of detail, and technical elements and plant representation.

- The risk-informed approach used by the licensee to address the effects of debris on LTCC is acceptable.

- Alternative assumptions were considered as sensitivities for each key assumption employing non-consensus approaches.

- The increase in risk meets the risk acceptance guidelines, as defined in RG 1.174, Revision 2.

3.5 Key Principle 5: The Impact of the Proposed Change Should be Monitored Using Performance Measurement Strategies

The NRC staff did not consider Principle 5 because Principle 5 was outside the scope of the technical report review.

4.0 PROGRAMMATIC ASPECTS RELIED UPON BY NRC STAFF FOR THIS REVIEW

4.1 Quality Assurance

RG 1.174, Revision 2, Section C.5, "Quality Assurance," provides the NRC staff's position on quality assurance requirements for risk-informed changes to the licensing basis. Specifically, RG 1.174, Revision 2 (Reference 20), states, in part,:

> As stated in Section 2 of this guide [RG 1.174], the quality of the engineering analyses conducted should justify proposed LB [licensing basis] changes will be appropriate for the nature of the change. In this regard, it is expected that for traditional engineering analyses (e.g., deterministic engineering calculations), existing provisions for quality assurance (e.g., Appendix B to 10 CFR Part 50, for safety-related SSCs) will apply and provide the appropriate quality needed. Likewise, when a risk assessment of the plant is used to provide insights into the decision-making process, the PRA is to have been subject to quality control.

RG 1.174, Revision 2, further states that when PRA information is used to enhance or modify activities affecting the safety-related functions of SSCs, four pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 should be met:

- Use personnel qualified for the analysis.
- Use procedures that ensure control of documentation, including revisions, and provide for independent review, verification, or checking of calculations and information used in the analyses. (An independent peer review or certification program can be used as an important element in this process.)
- Provide documentation and maintain records in accordance with the guidelines Section 6 of RG 1.174, Revision 2.
- Use procedures that ensure that appropriate attention and corrective actions are taken if assumptions, analyses, or information used in previous decisionmaking are changed (e.g., licensee voluntary action) or determined to be in error.

In the letters dated April 21, 2017, and January 9, 2018, the licensee described the quality assurance program that was applied to the risk-informed assessment of debris. The licensee stated that the NARWHAL software was developed by ENERCON under its 10 CFR Part 50 Appendix B Quality Assurance (QA) program and that the status, adequacy, and effectiveness of ENERCON's QA program is regularly assessed by internal audits and audits conducted by external stakeholders. The licensee noted that ENERCON successfully passed a joint nuclear utility audit conducted under the auspices of the Nuclear Procurement Issues Committee (NUPIC) in June 2016. The NUPIC audit scope included ENERCON's software development processes and resulted in ENERCON being added by NUPIC as an approved supplier of

safety-related software. The licensee stated that NARWHAL was developed as a safety-related item and provided a list of software documentation.

The licensee summarized aspects of the NARWHAL validation and verification process and discussed features enabling users to identify software anomalies and for developers to subsequently determine whether an anomaly is a software error. Software errors and resolutions in terms of their effects on the application were provided and discussed in the licensee's letter dated January 9, 2018. During the October 2017 audit, the NRC staff examined user manuals, validation and verification documentation, and spreadsheets that included the NARWHAL outputs for seven pump state scenarios. The licensee also provided the outputs (a database of breaks and debris amounts) from the BADGER debris generation software for NRC review. The NRC staff reviewed the documents and the data provided and discovered some minor anomalies with the results. None of the identified issues had a significant effect on risk.

Based on review of the information provided by the licensee, the NRC staff finds the QA program used for the risk-informed approach to be acceptable because it meets the regulatory position in RG 1.174, Revision 2, Section C.5. Specifically, the NRC staff concludes that the licensee's QA approach is adequate because it included the use of qualified personnel, the availability and maintenance of extensive documentation related to the NARWHAL software and the PRA model, and the existence of procedures to address analyses impacts in case of model changes or errors.

5.0 LICENSING BASIS AND CORRECTIVE ACTIONS

5.1 Licensing Basis

As requested in GL 2004-02, SNC provided a description of licensing basis changes for VEGP. The licensee stated that it plans to submit licensing actions to change the current deterministic methodology with the risk-informed methodology described in the technical report. The NRC staff concludes that SNC has provided sufficient information as requested by GL 2004-02 (Reference 14) and further described in the revised content guide for GL 2004-02 (Reference 31) concerning licensing basis.

5.2 General Description of Corrective Actions

In Enclosure 5 of the technical report, the licensee, as requested in GL 2004-02, provided a list of 25 completed and 3 planned actions, analyses, and modifications in response to GL 2004-02. The NRC staff concludes that SNC has provided sufficient information as requested by GL 2004-02 and further described in the revised content guide for GL 2004-02 concerning general description of and schedule for corrective actions.

6.0 LIMITATIONS AND CONDITIONS

The NRC is limiting its acceptance of the technical report enclosed with the licensee's letter dated July 10, 2018, to plant-specific licensing applications for Vogtle. Approval for use of the proposed risk-informed assessment methodology for assessing debris accumulation at Vogtle (incl., submodels and integration of the submodels) described in technical report enclosed with the letter dated July 10, 2018, in future licensing actions is contingent upon the satisfaction of the following conditions and limitations:

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.

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.

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

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.

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

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.

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.

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.

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.

7.0 SUMMARY AND CONCLUSIONS

The NRC staff concludes that the SNC's proposed risk-informed assessment methodology for assessing debris accumulation at Vogtle as described in technical report enclosed with the letter dated July 10, 2018, is acceptable because it satisfies the following key principles of risk-informed decisionmaking as delineated in RG 1.174, Revision 2:

- The proposed change is consistent with defense-in-depth philosophy, as discussed in Section 3.2 of this SE.
- The proposed change maintains sufficient safety margins, as discussed in Section 3.3 of this SE.
- The increases in risk resulting from the proposed change are small and consistent with the Commission's Safety Goal Policy Statement, as discussed in Section 3.4 of this SE.

The NRC staff concludes that the SNC's proposed risk-informed assessment methodology for assessing the effects of debris on long-term core cooling at Vogtle (incl., submodels and integration of the submodels) described in the technical report enclosed with the letter dated July 10, 2018, is an acceptable evaluation model, as required by 10 CFR 50.46, for use in licensing actions concerning long-term core cooling provided that the limitations and conditions listed in Section 5.0 of this safety evaluation report are met.

With the exception of downstream effects – fuel and vessel and licensing basis, the NRC staff concludes that SNC has provided the information requested by GL 2004-02 and further delineated in the revised content guide for GL 2004-02.

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SUBJECT: 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) DATED SEPTEMBER 30, 2019

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