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

August 28, 2023

Mr. Bob Coffey
Executive Vice President, Nuclear
Division and Chief Nuclear Officer
Florida Power & Light Company
Mail Stop: EX/JB
700 Universe Blvd
Juno Beach, FL 33408

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 273 AND 275 REGARDING REVISING LICENSING BASIS TO ADDRESS GENERIC SAFETY ISSUE 191 AND TO RESPOND TO GENERIC LETTER 2004-02 USING A RISK-INFORMED APPROACH (EPID L-2022-LLA-0106)

Dear Mr. Coffey:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment Nos. 273 and 275 to Renewed Facility Operating License Nos. DPR-24 and DPR-27, respectively, for the Point Beach Nuclear Plant, Units 1 and 2 (Point Beach). The amendments are in response to your application dated July 29, 2022, as supplemented by letter dated June 9, 2023.

The amendments revise the licensing basis described in the Point Beach Final Safety Analysis Report to allow the use of a risk-informed approach to address safety issues discussed in Generic Safety Issue (GSI)-191, “Assessment of Debris Accumulation on PWR [Pressurized-Water Reactor] Sump Performance,” and to respond to Generic Letter (GL) 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors.”

In addition to the license amendments, the Point Beach licensee, NextEra Energy Point Beach, LLC, requested an exemption pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) section 50.12, “Specific exemptions,” from certain requirements of 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors,” to allow the use of a risk-informed methodology instead of the traditional deterministic methodology to resolve the concerns associated with GSI-191 and to respond to GL 2004-02 for Point Beach. The approval of the exemption request is documented separately.

A copy of the related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's monthly *Federal Register* notice.

Sincerely,

/RA/

Scott P. Wall, Senior Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosures:
1. Amendment No. 273 to DPR-24
2. Amendment No. 275 to DPR-27
3. Safety Evaluation

cc: Listserv

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

NEXTERA ENERGY POINT BEACH, LLC

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 273
License No. DPR-24

1. The U.S. Nuclear Regulatory Commission (NRC, the Commission) has found that:

   A. The application for amendment by NextEra Energy Point Beach, LLC (the licensee), dated July 29, 2022, as supplemented by letter dated June 9, 2023, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by Amendment No. 273, Renewed Facility Operating License No. DPR-24 is hereby amended to authorize the revision to the Point Beach Final Safety Analysis Report as set forth in the application dated July 29, 2022, as supplemented by letter dated June 9, 2023, and as evaluated in the NRC staff's safety evaluation issued with this amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jeffrey A. Whited

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Date: 2023.08.28 15:21:03 -04'00'

Jeffrey A. Whited, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of issuance: August 28, 2023

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

NEXTERA ENERGY POINT BEACH, LLC

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 275
License No. DPR-27

1. The U.S. Nuclear Regulatory Commission (NRC, the Commission) has found that:

   A. The application for amendment by NextEra Energy Point Beach, LLC (the licensee), dated July 29, 2022, as supplemented by letter dated June 9, 2023, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by Amendment No. 275, Renewed Facility Operating License No. DPR-27 is hereby amended to authorize the revision to the Point Beach Final Safety Analysis Report as set forth in the application dated July 29, 2022, as supplemented by letter dated June 9, 2023, and as evaluated in the NRC staff's safety evaluation issued with this amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jeffrey A. Whited

Digitally signed by Jeffrey A. Whited
Date: 2023.08.28 15:21:21 -04'00'

Jeffrey A. Whited, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of issuance: August 28, 2023

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 273 AND 275 TO

RENEWED FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27, RESPECTIVELY

NEXTERA ENERGY POINT BEACH, LLC

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-266 AND 50-301

1.0 INTRODUCTION

By letter dated July 29, 2022 (Reference 1), as supplemented by letter dated June 9, 2023 (Reference 2), NextEra Energy Point Beach, LLC (NextEra, the licensee) submitted a license amendment request (LAR), exemption request, and an updated response to Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors" (Reference 3), for the Point Beach Nuclear Plant, Units 1 and 2 (Point Beach). The amendments would revise the licensing basis described in the Point Beach Final Safety Analysis Report (FSAR) (Reference 4) to allow the use of a risk-informed approach to address safety issues discussed in U.S. Nuclear Regulatory Commission (NRC, the Commission) Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR [Pressurized-Water Reactor] Sump Performance." The letter dated July 29, 2022, as supplemented, constitutes the licensee's final response to GL 2004-02. This safety evaluation (SE) reviews the LAR and the licensee's response to GL 2004-02.

In addition to the LAR, the licensee requested an exemption pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) section 50.12, "Specific exemptions," from certain requirements of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," to allow the use of a risk-informed methodology instead of the traditional deterministic methodology to resolve the concerns associated with GSI-191 and to respond to GL 2004-02 for Point Beach. The approval of the exemption request is documented separately and is being issued simultaneously with this amendment (Reference 5).

The supplemental letter dated June 9, 2023, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on October 4, 2022 (87 FR 60216).

1.1 Background

1.1.1 Challenges to Function of Safety Systems 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 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) coolant jet emanating from the broken reactor coolant system (RCS) 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.

1.1.2 Generic Safety Issue-191

In 1996, the NRC identified an issue associated with the effects of debris accumulation on PWR sump performance during design-basis accidents (DBAs).

Findings from research and industry operating experience raised questions concerning the adequacy of PWR recirculation 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 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.

1.1.3 GL 2004-02

As part of the actions to resolve GSI-191, in September 2004, the NRC issued GL 2004-02 to holders of operating licenses for PWRs. In GL 2004-02, the NRC staff requested that these 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 (HELB) inside containment, and, if appropriate, take additional actions to ensure system function. GL 2004-02 required, per 10 CFR 50.54(f), that these 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-10-0113, "Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance," dated December 23, 2010 (Reference 6), directed the NRC staff to consider a risk-informed approach for resolution of GSI-191. In 2012, the NRC staff developed three options to resolve GSI-191. These options were documented and proposed to the Commission in SECY-12-0093, "Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance," dated July 9, 2012 (Reference 7). The options are summarized as follows:

- Option 1 allows licensees to demonstrate compliance with 10 CFR 50.46 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 8), approving all three options for closure of GSI-191.

In a public meeting on May 18, 2021 (References 9 and 10), the licensee stated that it would pursue a full risk-informed resolution path (i.e., Option 2 of SECY-12-0093) to resolve GL 2004-02 and GSI-191 for Point Beach, based on the precedent of the South Texas Project, Units 1 and 2 (STP) pilot GSI-191 assessment (Reference 11).

1.2 Licensee's Approach

The licensee's risk-informed approach to evaluate the effects of debris on the sump strainer and pumps of the ECCS is documented in Enclosure 4 of the July 29, 2022, submittal. Effects referred to as "downstream effects" (including in-vessel effects) were addressed using methods in topical report (TR) WCAP-17788-NP, "Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090)," Volume 1, Revision 1 (Reference 12). The NRC staff found that the TR provided significant insights into the response of PWRs to the effects of debris that is transported to the vessel, but did not generically approve it. This SE documents the NRC staff's evaluation of the licensee's risk-informed approach to resolve GL 2004-02 at Point Beach.

The licensee's July 29, 2022, submittal is organized according to draft Regulatory Guide (RG) 1.229, "Risk-Informed Approach for Addressing the Effects of Debris on Post-Accident Long-Term Core Cooling" (Reference 13), addressing key principles of risk-informed integrated decision-making such as defense-in-depth (DID) and safety margins. The licensee's overall evaluation of risk attributable to debris for Point Beach is based on physical models that have been used in the past in similar GSI-191 risk-informed assessments and accepted by the NRC for GSI-191 resolution. The licensee provided a summary of the plant-specific conditions and models related to GSI-191, as well as a description of the risk quantification using a method similar to the one used in the STP pilot GSI-191 evaluation, relying on the Containment Accident Stochastic Analysis GSI Resolution and Evaluation (CASA Grande) software to compute debris amounts as a function of postulated break sizes and break orientations at multiple potential break locations, and loss of coolant break frequencies from NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," Volumes 1 and 2, dated April 2008 (Reference 14), and frequencies from other references used to define inputs to probabilistic risk analysis (PRA) models (Reference 15). The licensee considered limited outputs of the PRA model, such as the core damage frequency (CDF) and the large early release frequency (LERF) in its evaluation.

The licensee determined that most break scenarios were mitigated successfully when evaluated using deterministic methods. Any scenario that was not predicted to be mitigated was assumed to result in a core damage event. These failures, accounting for their frequency, were assumed to contribute to the change in plant risk and were compared against RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," acceptance guidelines (Reference 16). The licensee concluded that the change in risk is very small.

1.3 Method of NRC Staff Review

The purpose of the NRC staff's review summarized in this SE is to evaluate the licensee's assessment of the impact of debris on ECCS and CSS functions following postulated LOCAs at Point Beach. The NRC staff evaluated the LAR, conducted a regulatory audit, and performed confirmatory calculations in areas deemed appropriate by the NRC staff (Reference 17). The NRC staff issued a request for additional information (RAI) (Reference 18), and the licensee revised and updated its LAR by letter dated June 9, 2023, partially to address additional information requests and partially to address other issues identified by the licensee. 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 of this SE.

2.0 REGULATORY EVALUATION

2.1 Applicable Regulatory Requirements

The NRC staff assesses proposed remedial actions in accordance with the general standards for license amendments. Under 10 CFR 50.92(a), determinations of whether to grant an applied-for license amendment are to be guided by the considerations that govern the issuance of initial licenses or construction permits to the extent applicable and appropriate. Both the common standards for operating licenses and construction permits in 10 CFR 50.40(a), and those specifically for issuance of operating licenses in 10 CFR 50.57(a)(3), provide that there must be "reasonable assurance" that the activities at issue will not endanger the health and safety of the public.

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 exceeding 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 system and the ECCS 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 NRC requirements for technical specifications (TSs) are in 10 CFR 50.36, which states that TSs are to include items in, among others, the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls.

2.2 Applicable Regulatory Guides, Review Plans, and Guidance Documents

Volume 1 of Nuclear Energy Institute (NEI) 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," dated December 2004 (Reference 19), and Volume 2, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02," dated December 2004 (Reference 20), describe a method acceptable to the NRC staff, with limitations and conditions, for performing the evaluations requested by GL 2004-02. Taken together, NEI 04-07 and the associated NRC staff SE are often referred to as the guidance report/safety evaluation (GR/SE).

The industry developed the following additional 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 21) and the associated NRC SE (Reference 22).

- TR WCAP-16406-P-A, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," Revision 1, dated March 2008 (Reference 23) and the associated NRC SE (Reference 24).

- TR WCAP-17788-P, Volume 1 (Reference 12) and Volume 5, "Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090) – Autoclave Chemical Effects Testing for GSI-191 Long-Term Cooling" (Reference 25).

The reports listed above, subject to the limitations and conditions contained in the associated NRC SEs, describe methods acceptable to the NRC staff for performing the evaluations and analyses within the scope stated in those SEs (References 22 and 24).

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 26), 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 27). These documents provide guidance on the information to be submitted to the NRC for review.

RG 1.82, Revision 4, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," dated March 2012 (Reference 28), 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. Although the licensee used Revision 4 in its LAR, the NRC staff notes that Revision 5 was published in August 2022 (Reference 29). However, Revision 4 continues to be one acceptable way to meet the NRC's regulations. Accordingly, the NRC staff used Revision 4 during its review.

RG 1.174, Revision 3, provides guidance on the use of PRA findings and risk insights in support of licensee requests for changes to a plant's 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 decision-making.

RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, dated March 2009 (Reference 30), 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" (Reference 31). Revision 3 of RG 1.200 was issued in December 2020 (Reference 32). Revision 3 did not supersede Revision 2. The licensee chose to use Revision 2 of the RG as guidance for its LAR. The NRC staff notes that Revision 2 contains less flexibility than Revision 3 such that its use does not adversely affect the staff's conclusions in this SE. The ASME/ANS RA-Sa-2009 PRA Standard addresses PRAs for internal events and other hazards. RG 1.200 describes one acceptable approach for determining whether the PRA, in total, or the parts that are used to support an application, is acceptable for use in regulatory decision-making for light-water reactors (LWR).

General guidance for evaluating the technical basis for proposed risk-informed changes is provided in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," dated June 2007 (Reference 33).

2.3 Proposed FSAR Changes

In the LAR, the licensee provided proposed changes to the Point Beach FSAR. These changes describe the treatment of debris with respect to the operation of the ECCS and the CSS during sump recirculation. In implementing the proposed amendment, the licensee would update Sections 6 and 14 of the FSAR, and add an appendix, Appendix A.8, providing an overall description of the risk-informed evaluation.

3.0 TECHNICAL EVALUATION

There are no physical modifications needed or planned in support of the proposed changes at Point Beach. Operating procedures at Point Beach have actions that prevent and mitigate strainer blockage based on indications available to operators such as instrumentation to monitor core water levels, sump water levels, and containment temperatures. When the issue regarding the effects of debris on strainer performance was initially being addressed, the licensee replaced the original strainers with a design that has a much larger strainer area and improved filtering performance.

The NRC staff performed an integrated review of the proposed risk-informed approach, considering the five key principles of risk-informed decision-making set forth in RG 1.174, Revision 3.

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

The proposed change is to modify the Point Beach licensing basis analyses to show compliance with 10 CFR 50.46 considering the effects of debris using both deterministic and risk-informed methodologies.

NEI 04-07; RG 1.82, Revision 4; TR WCAP-17788-P; NRC Staff Review Guidance for In-Vessel Effects (Reference 34); and the SRP are the primary guidance documents used to show regulatory compliance with 10 CFR 50.46 considering the effects of debris using deterministic criteria. As described above, the proposed Point Beach method would use both deterministic and risk-informed criteria. Most of the break scenarios are shown to meet deterministic acceptance criteria. For scenarios where deterministic acceptance criteria are not satisfied, the licensee requested an exemption to 10 CFR 50.46. The requested exemption from 10 CFR 50.46(a)(1) was evaluated by the NRC staff against the criteria of 10 CFR 50.12, “Specific exemptions,” and found to be acceptable in the related exemption issuance (Reference 5). Therefore, a successful demonstration that all break scenarios are bounded by the deterministic criteria or fall within the bounds of the approved exemption demonstrates that regulations have been met.

The criteria to evaluate compliance with 10 CFR 50.46 using a risk-informed methodology are provided in the SRP section 19.2; RG 1.200, Revision 2; and RG 1.174, Revision 3. However, the NRC has historically interpreted and applied the current regulations in 10 CFR 50.46 as requiring a deterministic approach and bounding calculations to show compliance. Thus, the NRC’s longstanding practice may be regarded as a legally binding requirement from which an exemption is the appropriate means of granting dispensation from compliance. The licensee stated that, as allowed by SRM-SECY-12-0093, it chose to use a risk-informed method to resolve GSI-191 and to respond to GL 2004-02. Thus, in accordance with 10 CFR 50.12, the licensee requested an exemption from 10 CFR 50.46(a)(1) in Enclosure 1 of the July 29, 2022, submittal. The licensee concluded that for Point Beach the risk for the effects of debris is less than the threshold for Region III (very small changes) of RG 1.174, and no additional physical changes to the facility or changes to the operation of the facility were proposed.

The NRC staff separately determined that special circumstances exist to grant the proposed exemption and that granting the exemption would not result in a violation of the Atomic Energy Act of 1954, as amended. Therefore, since the NRC staff has granted the requested exemption, and the change explicitly relates to the exemption, the proposed change to use the risk-informed methodology meets the first key principle of RG 1.174.

3.2 Key Principle 2: The Proposed Change is Consistent with the DID Philosophy

Regulatory Position C.2.1.1, “Defense in Depth,” of RG 1.174, Revision 3, states that the DID philosophy consists of a number of considerations and consistency with the DID philosophy is maintained if the seven considerations addressed in sections 3.2.1 through 3.2.7 below are met.

In Enclosure 5 of the July 29, 2022, submittal, the licensee explicitly addressed DID and the seven considerations for Point Beach. Items associated with DID that were included in Enclosure 5 of the licensee’s analysis are evaluated in sections 3.2.1 through 3.2.7 of this SE.

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

The licensee highlighted physical and procedural changes such as the installation of new strainers with increased surface areas and a reduced opening size, the installation of flow diverters to prevent debris-laden fluid directly reaching the sumps, the implementation of the standard design change process that identifies potential impact to GL 2004-02 compliance by planned modifications, and program controls to ensure that the debris load limits are not exceeded.

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 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; 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 DID consideration evaluates if design features are substituted for programmatic activities to an extent that significantly reduces the reliability and availability of design features to perform their safety functions. The licensee identified that the proposed change would not adversely affect any of the programmatic activities already in place at Point Beach, such as the inservice inspection (ISI) program, plant personnel training, the RCS leakage detection program, and containment cleanliness inspection activities. The proposed change would not rely heavily on programmatic activities as compensatory measures or propose any new programmatic activities that could be heavily relied upon.

The NRC staff reviewed the licensee's description of programmatic activities and concludes that this DID consideration is met because the proposed change does not affect how safety functions are performed or 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 (for Example, No Risk Outliers)

The licensee highlighted the redundancy, independence, and diversity of the ECCS and containment heat removal equipment and asserted that the proposed change does not require any design change to these systems. Therefore, system redundancy, independence, and diversity are preserved. These systems were analyzed relative to their contribution to nuclear safety through the Point Beach plant-specific PRA (which meets industry PRA standards for risk-informed applications), accounting for a full range of LOCA events and uncertainties, and no risk outliers were identified.

The NRC staff reviewed the licensee's evaluation of this DID consideration and concludes that it is met because the risk-informed analysis is consistent with the assumptions in the safety analysis for Point Beach and does not significantly increase the expected frequency of challenges to the systems, 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 reductions in redundancy, independence, or diversity of systems resulting from postulated LOCAs do not cause a significant increase in risk, as evidenced by a margin to RG 1.174 risk acceptance guidelines. The licensee included sensitivity cases to assess uncertainty. Although some sensitivity cases yielded higher risk increases, those alternatives were considered as part of the uncertainty of the risk estimates and were controlled by different sets of assumptions. See section 3.4.9 of this SE for the discussion of this issue.

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 examined common-cause failure mechanisms in the context of GL 2004-02; specifically, the primary failure mechanism of concern is recirculation strainer clogging limiting adequate flow to any of the ECCS and CSS pumps. The defenses against potential strainer clogging are not changed by the risk-informed methodology because there are no design changes to the equipment or changes to emergency operating procedures (EOPs).

The NRC staff reviewed the evaluation of this DID consideration and concludes that it is met because the risk-informed evaluation does not introduce a new potential common-cause failure 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; and does not weaken or defeat an existing defense against a cause, event, or coupling factor. Even though the strainer blockage failure mechanism is not deterministically eliminated, the risk analysis shows that the risk is very small and that additional mitigative and DID measures exist.

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 noted that during recirculation, the post-LOCA fluid collecting in the containment sump pits is mobilized by the residual heat removal (RHR) pumps and recirculated back to the containment. The auxiliary building has dedicated filters in the ventilation system to limit offsite releases.

The licensee stated that containment is fully analyzed for design basis considerations considering a single failure that results in the loss of one air cooling train and a train of containment spray (CS) and that sufficient heat removal would occur to prevent containment pressure from exceeding its design limit. The licensee stated that the proposed licensing basis change would not alter the design or operating requirements of these systems and concluded that the independence of the barriers is maintained and not degraded by the change.

The licensee provided an additional evaluation of DID for the barriers to the release of radioactivity (page E5-10 of Enclosure 5 of the July 29, 2022, submittal). The licensee stated that the severe accident mitigation management guidelines (SAMGs) are designed to protect these barriers under conditions that warrant entry into the SAMGs. The licensee also provided a description of the programs that prevent and detect pipe breaks. The licensee discussed ASME Code requirements that are intended to prevent RCS pressure boundary failures and the leak detection program that can identify small leaks so that actions can be taken to address RCS leakage. The licensee also discussed reactor containment integrity. The containment is designed to accommodate the pressure from the most limiting RCS pipe breaks with margin while considering the most challenging single active failure of ECCS, CSS, and the containment cooling units during injection and the worst active or passive single failure during recirculation. The licensee stated that the methodology and acceptance criteria used to evaluate the acceptability of the containment are not affected by the proposed change and concluded that containment integrity is not affected. Some of these topics are discussed in greater detail in section 3.3 of this SE.

The NRC staff reviewed the licensee's evaluation of this DID consideration and concludes that it is met because implementation of the proposed 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 proposed 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 proposed change would not involve any additional operator actions or increase the complexity of any operator actions. The licensee concluded that the defenses that are already in place with respect to human errors would not be impacted by the proposed licensing basis change.

The NRC staff reviewed the licensee's evaluation of this DID 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, 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 stated that the proposed change would not involve any change to the physical design of the current plant equipment associated with GL 2004-02. The licensee stated that the acceptance criteria of 10 CFR 50.46 are not changed. The proposed change revises the licensing basis for acceptable containment emergency sump strainer design and performance in support of ECCS and CSS operation in recirculation mode following postulated LOCAs by demonstrating that the effect of debris on LTCC results in very small risk. The licensee concluded that the intent of the plant's design criteria is maintained.

The NRC staff reviewed the licensee's evaluation of this DID 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 that LTCC will be maintained following a LOCA and thus does not result in a reduction in the effectiveness of one or more layers of defense.

3.2.8 Additional DID Considerations - Detecting and Mitigating Adverse Conditions

The licensee provided additional information on DID measures in support of the response to GL 2004-02. In Enclosure 5 of the July 29, 2022, submittal, the licensee stated that adequate DID is maintained by ensuring the capability for operators to detect and mitigate inadequate flow through recirculation strainers and inadequate flow through the reactor core due to the potential impacts of debris blockage.

3.2.8.1 Prevention of Strainer Blockage

The licensee identified that the sump strainers are monitored for blockage and that actions are directed if blockage occurs as specified in Point Beach's EOPs and Emergency Contingency Actions (ECAs). The licensee stated that for smaller LOCAs, depletion of the refueling water storage tank (RWST) can be delayed by cooling the RCS to reduce break flow and therefore the injection flow rate. The licensee also described measures taken to control or reduce the debris source term inside containment (page E5-6 of Enclosure 5 of the July 29, 2022, submittal).

3.2.8.2 Detection of Strainer Blockage

The licensee stated that Point Beach has operational procedures to monitor operating parameters related to the RHR pump flow, discharge pressure, and amperage. These procedures allow control room personnel to properly diagnose the occurrence of cavitation as an indication of sump clogging (page E5-7 of Enclosure 5 of the July 29, 2022, submittal).

3.2.8.3 Mitigation of Strainer Blockage

The licensee highlighted that the Point Beach ECAs contain steps to reduce flow through the system and provide alternate injection flowpaths. The ECAs provide guidance for refilling the RWST to allow alternate injection flowpaths. The licensee also stated that temporarily terminating recirculation flow may prevent some debris to be dislodged from the strainer thus reducing the headloss across the strainer.

The licensee also noted the diverse and flexible coping strategies (FLEX) to maintain fuel cooling and containment integrity developed in response to NRC Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" (Reference 35). The licensee asserted that various modifications have been implemented such that non-emergency equipment can be credited during an event (page E5-7 of Enclosure 5 of the July 29, 2022, submittal).

3.2.8.4 Prevention of Inadequate Reactor Core Flow

The licensee stated that the actions discussed above for reducing flow through the strainers have a similar positive effect on reducing the potential for fuel damage. The licensee also discussed the capability to initiate simultaneous cold leg and upper plenum injection after the RHR pumps have been realigned to the sump. This provides an alternate flowpath for coolant to enter the core (page E5-8 of Enclosure 5 of the July 29, 2022, submittal).

3.2.8.5 Detection of Inadequate Reactor Core Flow

The licensee stated that there are multiple methods for detection of core blockage resulting in inadequate core cooling or RCS inventory. Core blockage would be indicated by an increase in core exit thermocouple temperature or a reduction in reactor water level as monitored by the reactor vessel level indications (RVLIS). The licensee noted that an additional method for detection of core blockage is monitoring containment radiation levels (page E5-8 of Enclosure 5 of the July 29, 2022, submittal).

3.2.8.6 Mitigation of Inadequate Reactor Core Flow

The licensee stated that multiple methods are available to mitigate inadequate core flow conditions. The Point Beach EOPs and ECAs include direction to reestablish cooling flow to the RCS, reduce RCS pressure, and restart reactor coolant pumps (RCPs) and open power operated relief valves. The licensee also stated that the SAMGs provide additional actions to protect fission product boundaries if the EOPs become ineffective. The SAMGs provide alternate injection means, including flooding the containment. FLEX equipment may also be used to mitigate inadequate core flow (page E5-9 of Enclosure 5 of the July 29, 2022, submittal).

3.2.8.7 NRC Staff Review of Additional DID

The NRC staff reviewed the licensee's additional DID actions and programs and concludes that the licensee's measures to prevent, detect, and mitigate adverse conditions (such as inadequate strainer flow or inadequate core flow), barriers for release of radioactivity, emergency plan actions, and SAMGs provide additional DID measures beyond the seven factors defined in RG 1.174, Revision 3.

3.2.9 NRC Staff Conclusion Regarding Key Principle 2: DID

The NRC staff finds that the philosophy of DID is maintained under the analysis described in Enclosure 5 of the July 29, 2022, submittal, because the licensee appropriately addressed each of the seven factors in section 2.1.1 of RG 1.174, Revision 3, and provided additional information, specifically for issues that have been identified as critical areas where DID can be beneficial.

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

RG 1.174 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., FSAR, supporting analyses) are met, or proposed revisions provide sufficient margin to account for analysis and data uncertainty.

The licensee examined the safety margins and stated that there are numerous conservatisms included in the risk-informed GL 2004-02 evaluation. The licensee described barriers for release of radioactivity in section 2.3 of Enclosure 5 of the July 29, 2022, submittal. The licensee cited the fuel cladding, RCS boundary, and containment as the barriers. The licensee summarized a list of conservatisms in table 3-1 of Enclosure 5 of the July 29, 2022, submittal related to several areas in the risk-informed analysis to show that the proposed approach maintains sufficient safety margins.

3.3.1 Barriers for Release of Radioactivity

The licensee concluded that the proposed change maintains sufficient DID for current barriers (cladding, RCS boundary, containment, and emergency plan actions) against the release of radioactivity. However, the NRC staff determined that some of the information provided by the licensee also contributes to margin in the analysis. Section 3.2 of this SE discusses the DID aspects of these barriers. However, the licensee provided information regarding measures taken to ensure that the RCS pressure boundary would not fail. These measures follow ASME Code and other requirements that ensure margin in the assumptions for failure of the RCS pressure boundary.

The licensee evaluated aspects contributing to safety margin such as the fuel cladding, emergency core cooling and long-term cooling, RCS pressure boundary, ISI program, RCS weld mitigation, RCS leakage detection, containment integrity, containment testing, and operator actions. The licensee highlighted the following aspects in sections 2.3 and 2.4 of Enclosure 5 of the July 29, 2022, submittal:

- Although the RCS pressure boundary is postulated to be failed for the GL 2004-02 sump risk-informed evaluation, the proposed change does not make any change to the previous analyses and testing programs that demonstrate the integrity of the RCS.

- The Point Beach ISI program plan provides verification that structural integrity of ASME Class 1, 2, and 3 piping components are within the limits specified in the ISI program, and verification that the structural integrity of the main feedwater piping is within the limits specified in the augmented ISI program.

- The leak detection program at Point Beach is capable of early identification of RCS leakage to provide time for appropriate operator action prior to a large break.

- The containment remains a low leakage barrier against the release of fission products for the duration of the postulated LOCAs. The containment cooling units and CSS are designed to reducing the containment pressure and temperature after a DBA including a loss of offsite power and a single failure.

- The proposed change to the licensing basis does not involve any changes to the emergency plan; the use of the risk-informed approach does not impose any additional operator actions or complexity.

During the regulatory audit, the NRC staff requested that the licensee provide additional information regarding DID for the RCS pressure boundary. By letter dated June 9, 2023, the licensee provided a supplement to the LAR and to address the NRC's questions as part of the audit activities. The NRC staff evaluated the response. The NRC staff conclusions are provided in the following evaluation.

The evaluation of DID for the RCS pressure boundary addresses whether the impact of the proposed licensing basis change (individually and cumulatively) is consistent with the DID philosophy, as outlined in RG 1.174. The evaluation also presents the measures available for preventing, detecting, and mitigating conditions that could challenge long-term core cooling due to strainer blockage and inadequate cooling flow to the reactor core. The measures discussed here contribute to both DID and safety margins in the analysis.

The licensee stated that the integrity of the RCS pressure boundary is assumed to be compromised for the GL 2004-02 sump performance evaluation. The licensee stated that the proposed licensing basis change would not modify the previous analyses or testing programs that demonstrate the integrity of the RCS. The licensee considers all Class 1 welds in the GSI-191 analysis.

To demonstrate DID, the licensee stated that Point Beach developed a plan to manage the risk of primary water stress corrosion cracking (PWSCC) degradation in Alloy 600 components and Alloy 82/182 welds that are used in Class 1 RCS piping as discussed below.

3.3.1.1 Inservice Inspection Program

In Section 2.3.2 of Enclosure 5 of the July 29, 2022, submittal (page E5-11), the licensee stated, in part, that the ISI program provides rules for examination and testing of ASME Class 1, 2, and 3 components and component supports. The licensee stated that the ISI program plan addresses those examinations and tests required by ASME Code, section XI, and Point Beach augmented ISI commitments. The integrity of the Class 1 welds, piping, and components are maintained at a high level of reliability through the inspection program. The licensee stated that the Point Beach ISI program also ensures that inspections are performed in accordance with the schedule requirements of the ASME Code, section XI.

In addition to the required ISI program as discussed above, the licensee stated that Point Beach developed a program plan to manage the risk of PWSCC degradation in Alloy 600 components and Alloy 82/182 welds. The plan is in accordance with 10 CFR 50.55a, "Codes and standards," ASME Code Cases N-722-1, "Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials," and N-770-2, "Alternative Requirements for 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," and NEI 03-08, "Guidance for the Management of Materials Issues," Revision 2 (Reference 36). The plan identifies all Alloy 600/82/182 locations, ranks the locations based on their risks of developing PWSCC, provides inspection requirements, and presents mitigation/replacement options. Periodic inspections of the Alloy 600 components and Alloy 82/182 welds are covered in the ISI program. In addition, the licensee stated that the reactor pressure vessel (RPV) head was replaced with an Alloy 690 RPV head and welded with Alloy 52/152 filler material and continues to be monitored by Code Case N-729-6, “Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1, Supplement 5.” The NRC staff notes that Alloy 690/52/152 materials are less susceptible to PWSCC than Alloy 600/82/182 materials.

In response to the NRC staff question concerning a program plan that was developed to manage the risk of PWSCC degradation in Alloy 600 components and Alloy 82/182 welds, the licensee submitted procedure NP 7.7.31, “Alloy 600 Management Program,” which describes the overall programmatic requirements followed at Point Beach for the development, control, and implementation of an Alloy 600 Management Program. In addition, the licensee provided a table summarizing welds/components that have been mitigated with Alloy 52/152 inlays/onlays, which is reproduced below:

Component	Strategy
RPV Heads	Replaced with Alloy 690 materials and Alloy 52/152 filler material. RPV head is continued to be monitored per ASME Code Case N-729.
RV Lower BMIs [bottom mounted instrumentation nozzles]	Continued monitoring per ASME Code Case N-722
Unit 2 steam generator (SG) Hot and Cold Leg nozzle and safe-end welds	Inlaid with Alloy 52/152 filler material during manufacture. Continued monitoring per Code Case N-722 and N-770. Approval of relief request 2-RR-11 allowed for extended inspection frequency due to inlay.
Unit 1 SG Bowl Drain	Alloy 82/182 weld and Alloy 600 nozzles were replaced with Alloy 690 nozzle and Alloy 52 filler material.

The NRC staff finds the licensee’s response regarding piping degradation to be acceptable because the licensee identified the components that were mitigated or replaced with Alloy 690 base material and Alloy 52/152 filler material. In addition, the licensee identified how these components and welds are being monitored. The NRC staff finds that the licensee will follow the inspection and monitoring requirements of ASME Code Cases N-722-1, N-729-6, and N-770-2 and that, therefore, the monitoring of the RCS piping material is acceptable.

3.3.1.2 Reactor Coolant System Leakage Detection Capabilities

The licensee stated that the leak detection program at Point Beach is capable of early identification of RCS leakage to allow time for appropriate operator actions to identify and address RCS leakage. The licensee further stated that the effectiveness of this program is not reduced by the proposed licensing basis change to the risk-informed approach for GL 2004-02. The leak detection program at Point Beach is capable of detecting early identified and unidentified RCS leakage and is implemented using the plant procedure OI-55, “Primary Leak Rate Calculation,” which is based upon the industry guidance developed to meet RG 1.45, Revision 1, “Guidance on Monitoring and Responding to Reactor Coolant System Leakage,” dated May 2008 (Reference 37). The plant’s RCS leakage detection system provides time for appropriate operator action to identify and address RCS leakage to reduce the probability of an RCS piping failure.

In response to the NRC staff's RAI concerning the RCS leakage detection program, the licensee stated that the leakage detection systems are designed in accordance with the requirements of 10 CFR part 50, appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 30, "Quality of reactor coolant pressure boundary," to provide a means of detecting and, to the extent practical, identifying the source of the reactor coolant leakage. The systems conform to RG 1.45. Main systems that monitor the environmental condition and the detection of leakage of the containment include the humidity detection instrumentation, sump level monitoring system, component cooling liquid monitoring system, the condenser air ejector gas radioactivity system, the steam generator liquid sample line monitor system, and the containment fan cooler condensate measuring system. In addition to the above systems, the humidity, temperature, pressure, and radioactive gas monitors provide indirect indication of leakage to the containment. Associated systems and components connected to the reactor coolant system have intersystem leakage monitoring devices.

The licensee stated that the leakage detection systems are qualified for all seismic events not requiring a shutdown. The airborne radioactivity monitoring system is qualified for a safe shutdown earthquake.

The limits for reactor coolant leakage are identified in the Point Beach TSs.

3.3.1.3 Tests and Inspections

The licensee stated that it conducts periodic testing of leakage detection systems to verify the operability and sensitivity of detector equipment at Point Beach. These tests include installation calibrations and alignments, periodic channel calibrations, functional tests, and channel checks.

The humidity detector and condensate measuring system are also periodically tested to ensure proper operation and verify sensitivity.

The NRC staff finds that the licensee's RCS leakage detection systems are consistent with the guidance of RG 1.45 and are tested and inspected periodically. Therefore, the RCS leakage detection systems are acceptable.

In response to the NRC staff's question regarding inspections of piping, the licensee stated that in addition to the required ASME Code, section XI, inspections, Point Beach staff perform containment closure walkdowns and inspections, system engineer walkdowns, and periodic containment entries at power. The licensee stated that plant procedure NP 7.7.22, "Service Water Inspection Program," involves additional internal inspections of service water piping that is accessible for repair or maintenance. As stated previously, identified and unidentified RCS leakage is also monitored on a daily basis by OI-55. The NRC staff finds the response to the audit question to be acceptable because the licensee is performing additional inspections for RCS leakage outside of the required inspections of ASME Code, section XI.

3.3.1.4 Reactor Coolant System Overpressure Protection

The licensee stated that at Point Beach, the RCS overpressure protection is provided by means of pressure relieving devices, as required by section III of the ASME Code. The RCS system is also protected from overpressure at low temperatures by the low temperature overpressure protection (LTOP) system. The NRC staff finds that the RCS piping is protected from overpressure using the relief and safety valves. In addition, the LTOP system has specific limits on the pressure in the RCS system as part of the DID effort to mitigate any potential for leakage or failure of the RCS piping.

3.3.2 Debris Generation

The licensee referred to approved guidance used in support of the debris generation analysis such as NEI 04-07 and the associated NRC SE. In table 3-1 of Enclosure 5 of the July 29, 2022, submittal, the licensee highlighted aspects related to conservatism such as:

- The debris generation analysis does not take credit for shielding within the zone-of-influence (ZOI) by equipment (e.g., steam generators, reactor coolant pumps) and large piping.

- Failure of 100 percent of the unqualified and degraded coatings inside containment as particulates is a very conservative assumption considering that much of the unqualified coating may not fail or may fail as chips.

- The ZOIs assumed for loop piping breaks were grouped by loop resulting in the overprediction of debris for some breaks.

3.3.3 Debris Transport

The licensee stated that debris transport analysis was performed in accordance with NRC-approved methods in NEI 04-07 and cited the following conservatisms in the risk-informed analysis and as discussed in table 3-1 of Enclosure 5 of the July 29, 2022, submittal:

- All fine debris is assumed to wash down to the sump pool elevation with no holdup on structures; however, some fine debris would be expected to be retained on walls and structures above the containment pool due to incomplete spray coverage and hold up on structures. This contributes to an overestimate in the amount of fine debris reaching the strainer.

- All fine debris in the pool is assumed to transport to the strainer surface. However, it is expected that debris will actually be trapped in stagnant pools. Therefore, the amount of fine debris reaching the strainer is overestimated.

- The erosion fraction of large and small pieces of fibrous debris was overestimated resulting in greater amounts of fine fiber predicted to reach the strainer.

- A conservative pool fill transport fraction of 15 percent was used even though calculations estimate that 70 percent of debris could be transported to inactive volumes and not reach the strainer.

- The unqualified coatings were assumed to fail after pool fill up, but before the start of recirculation resulting in a greater amount of particulate debris predicted to reach the strainer.

3.3.4 Chemical Effects

The licensee stated that the chemical effects analysis was performed in accordance with NRC-approved guidance in TR WCAP-16530-NP-A, which includes the following levels of conservatism and as discussed in table 3-1 of Enclosure 5 of the July 29, 2022, submittal:

- One hundred percent of chemical species in solution are assumed to precipitate after the solubility limit has been reached. Some breaks would not result in precipitate formation, and some would result in less precipitate than assumed in the analysis.

- The maximum pH was assumed for chemical release and the minimum pH was assumed for solubility. This results in an overprediction of precipitate amounts.

- All submerged aluminum is assumed to react with the sump fluid and all unsubmerged aluminum reacts with containment spray. Some of the aluminum would not be available to interact with the fluid, therefore, the amount of precipitate is overpredicted.

- The maximum amount of aluminum coatings on the pressurizer was assumed to be destroyed by every break postulated withing the pressurizer compartment and the aluminum from these breaks was assumed to be instantly released in the sump pool. This results in an overprediction of aluminum in the pool.

- All insulation debris was assumed to be available for reaction in the sump pool even though a significant amount of insulation would be held up above the pool. This contributes to an overprediction of the amount of aluminum available to precipitate.

3.3.5 Sump Pool Water Level

The sump pool water level contributes directly to the performance of the pumps taking suction from the pool by providing net positive suction head (NPSH) and strainer submergence. Increased submergence reduces the probability of deaeration of the fluid and flashing of the fluid as it passes through the debris bed. Both of these phenomena are detrimental to the performance of the pumps.

- The full volume of the pressurizer was treated as a holdup volume for large break LOCAs (LBLOCAs) above the top of the hot leg nozzle. The sump pool level was underestimated for these scenarios. Therefore, flashing, NPSH, and deaeration failures are overestimated for these scenarios.

- The RWST level was assumed to be at its minimum technical specification level. The RWST level is typically maintained above this minimum level. Therefore, sump level is underestimated resulting in flashing, NPSH, and deaeration failures being overestimated.

- Only fiber fines were used; larger pieces of fiber would have reduced the measured headloss.

- Fiber and particulate debris were collected on the strainer prior to the addition of chemical precipitates, which optimizes the headloss and the uniformity of debris beds.

Debris beds would be expected to be less uniform resulting in lower debris bed headloss.

- The headloss test did not credit near-field settling, which was accomplished by inducing turbulence in the test. The Point Beach system includes a 6-inch curb surrounding the strainers that would promote settlement.

- Metallic insulation was not included. Metallic insulation would disturb the formation of a uniform debris bed and lower the debris bed headloss. Paint chips would have a similar effect of disturbing the debris bed; however, particulates were conservatively employed as surrogate for paint chips in the testing.

- Strainer headloss tests were conducted at an approach velocity approximately 15 percent larger than the expected approach velocity, conservatively causing larger strainer headloss.

3.3.6 Strainer and Pump Failure Evaluations

Strainer testing guidance has been developed to ensure that headlosses 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 debris loads for all debris types, except fiber, that could be generated by any break.

In sections 3.f and 3.g of Enclosure 3 of the July 29, 2022, submittal, the licensee stated that strainer headloss testing was performed in 2015 and 2016 at Alden Research Laboratories (Alden) in accordance with the NRC guidance, "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Strainer Head Loss and Vortexing," dated March 2008 (Reference 38), which included several levels of conservatism, as listed in table 3-1 of Enclosure 5 of the July 29, 2022, submittal:

- A break is assumed to cause strainer failure if corresponding debris amounts are not bounded by strainer tests. The strainer tests did not detect conditions that would cause strainer failure. It is expected that many breaks would not cause strainer failure, especially when corresponding debris amounts are close to strainer tests although not bounded by strainer tests. The simplified approach of comparison of debris amounts to strainer tests results in an overprediction of the number of breaks that would cause strainer failure.

- Miscellaneous debris is assumed to transport to the strainer before all other debris types thus reducing the strainer area. Some miscellaneous debris would not transport or would transport later to the strainer. This assumption effectively results in lower allowable debris limits for the strainer and contributes to overpredicting the number of failures.

- Debris headloss was calculated using a rule-based approach that assumes a step change in headloss based on discrete bounding headloss values from limited strainer tests. This approach overpredicts the headloss for many break scenarios.

- Strainer failure was assumed in cases where the calculated structural margin of the strainer was exceeded. However, the strainer could likely withstand higher headlosses and remain functional.

- In the evaluation of the effect of gas voids on pump performance it was assumed that all voids that form transport to the pumps without compression. The pump NPSH was adjusted for gas voids using conservative RG 1.82 guidance. In practice, some voids would collect in the strainer and vent back to the containment and transported voids would compress thus reducing their volume.

- No credit was taken for containment accident pressure when calculating NPSH margin or voiding due to degasification. This results in a reduction in NPSH margin, partially due to the overprediction of void formation.

- Maximum sump temperatures were used for all break scenario evaluations. Sump temperatures would be lower for smaller breaks. Therefore, the amounts of chemicals and degasification are overpredicted.

3.3.7 Core Failure Evaluation

The core failures were evaluated using several conservative methods as listed in table 3-1 of Enclosure 5 of the July 29, 2022, submittal:

- Fiber penetration testing and the correlation used to predict penetration ignored the effect of particulates on the filtration that would occur in the debris bed. The penetration of fiber would be reduced by the combined effects of fiber and particulate. Therefore, the extent of fiber penetration is overpredicted in the analyses.

- During fiber penetration testing the number of strainer disks was reduced to increase spacing between adjacent disks to prevent the bridging of debris. Bridging is expected to occur at the plant strainer which could decrease penetration from the assumed amount. This testing approach causes overprediction in the extent of fiber penetration in the analyses.

- Fiber limits developed for core blockage are based on bounding tests and analyses. It is likely that significantly larger quantities of debris could accumulate in the reactor core without adversely affecting cooling. Therefore, core failures based on these limits would be overpredicted.

- All breaks were evaluated against the core fiber limits for a hot leg break, while cold leg breaks result in lower debris amounts transported to the core. Therefore, the corresponding analyses would overpredict core failures.

- All RHR pump flow to the upper plenum is assumed to reach the reactor core while some would actually spill out the broken hot leg. Only the amount of coolant needed to replace boil off would reach the core. Ignoring the boiloff balance overpredicts the amount of fiber reaching the core.

- Secondary side break scenarios that require recirculation are assumed to result in a failure. It is likely that secondary side breaks would not result in failures because of the low strainer flow rates and debris loads. This would result in increased failure predictions.

3.3.8 NRC Staff Conclusion Regarding Key Principle 3: Safety Margins

The NRC staff concludes that the licensee's evaluation of DID satisfactorily addresses the DID philosophy as outlined in RG 1.174 because the RCS piping is monitored for degradation, the RCS leakage detection systems monitor RCS leakage, and the LTOP ensures RCS piping will not be overpressurized. The NRC staff further concludes that the selected pipe break locations satisfy the guidance of GL 2004-02 and, therefore, are acceptable. The DID evaluation also provides information regarding the safety margins associated with the risk-informed analysis.

The NRC finds that the RCS piping considered 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 system will be able to detect leakage and the operator will take corrective actions in accordance with the requirements of Point Beach TSs. The NRC staff determined that the subject piping maintains DID 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, and GDC 31, "Fracture prevention of reactor coolant pressure boundary." Therefore, the NRC staff concludes that the use of a risk-informed approach would not result in any changes to the response requirements for plant personnel during a loss of coolant accident. Based on the above, 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 to significantly affect the containment sump performance.

The licensee considered NRC-approved guidance such as TR WCAP-16530-NP-A, the March 2008 NRC guidance on strainer headloss and vortexing, and NEI 04-07 to develop tests and analyses concerning debris generation, debris transport, chemical effects, and headloss testing. Some of the conservatisms highlighted by the licensee may not have contributed significantly to the risk values in the submittal individually, but when combined the margin is significant. The NRC staff concludes that the proposed approach maintains safety margins and that the licensee's evaluation included independent margins that help ensure 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 Point Beach, including the calculated total risk values (CDF and LERF) for each unit, and the licensee's risk-informed assessment of debris. A review of this information was necessary to determine whether the risk attributable to debris is small and consistent with the intent of the Commission's Safety Goal Policy Statement.

3.4.1 Acceptability of the Base PRA Model

Regulatory Position C.2.3 of RG 1.174, Revision 3, 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 requested change 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 Point Beach 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. In section 7 of Enclosure 4 of the July 29, 2022, submittal, the licensee stated that the Point Beach Internal Events (IE) PRA mostly satisfies Capability Category II requirements from the ASME/ANS PRA standard. The licensee also stated that the very few non-conforming aspects of the model (revealed by a peer review of the IE model) have no impact on the GSI-191 risk assessment. Therefore, the licensee concluded that the Point Beach IE PRA meets the requirements of RG 1.200, Revision 2, and, therefore, is acceptable to support the assessment of the risk of internal events associated with GL 2004-02.

The Point Beach IE PRA model was used in a very limited extent to support the risk-informed GSI-191 analysis. The use of the PRA was limited to:

- Determining the overall internal events, internal flooding, and internal fire events CDF and LERF.

- Providing probabilities for RHR and CS equipment configuration.

- Calculating the change in LERF (ΔLERF) using the conditional large early release probability associated with LOCA events requiring recirculation.

- Computing the bounding change in CDF (ΔCDF) and ΔLERF associated with secondary side breaks inside containment (SSBIs).

The NRC staff's review focused on the above uses of the licensee's PRA model to evaluate the risk of debris in containment. The NRC staff notes that the licensee did not use the PRA model, but separate external hazard analyses and bounding estimates of seismic effects to compute corresponding CDF and LERF contributions to the baseline or total CDF and LERF.

3.4.1.1 Scope of the Base PRA (Modes/Hazards)

Regulatory Position C.2.3.1 in RG 1.174, Revision 3, states, in part, 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," which are defined as 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, as described in SRM-SECY-04-0118, "Plan for the

Implementation of the Commission's Phased Approach to Probabilistic Risk Assessment Quality," dated October 6, 2004 (Reference 39), 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.

The licensee stated that the PRA is compliant with RG 1.200, Revision 2. The PRA was peer reviewed and determined to be appropriate for use for risk-informed applications. The few non-conforming aspects of the model with the ASME/ANS PRA standard were determined to have no impact to the GSI-191 risk assessment. The Point Beach PRA model was not modified to support the risk-informed GL 2004-02 evaluation.

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 considered 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 the debris ZOI is either approximately the same, or significantly higher, at full power RCS pressure and temperature, the flow rate required to cool the core is significantly reduced for low power or shutdown modes, and the pressure of LOCA water jets at full power would generate more debris; and (2) the use of the internal events PRA model is adequate because the risk contributions from other external hazards do not affect the evaluation of the risk attributable to debris.

3.4.1.2 Level of Detail of the Base PRA

Regulatory Position C.2.3.2 in RG 1.174, Revision 3, 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 stated that very few aspects of the PRA model do not conform to requirements of the ASME/ANS PRA standard; however, the NRC agrees that those aspects have no impact on the GL 2004-02 evaluation. The NRC staff reviewed the licensee's description of its base PRA and concludes that the level of detail in the licensee's base PRA is sufficient and consistent with its limited use to evaluate the risk attributable to debris from sump strainer and core blockage failures. The licensee-implemented peer reviews followed the ASME/ANS standards and NEI guidance and these reviews did not identify issues that would affect the risk-informed GL 2004-02 evaluation.

3.4.1.3 Base PRA Technical Elements

RG 1.200, Revision 2, 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 decision-making for LWRs. RG 1.200 endorses, with comments and qualifications, the use of the ASME/ANS RA-Sa-2009 PRA Standard; NEI 00-02, Revision 1, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance," dated May 2006 (Reference 40); and NEI 05-04, Revision 2, "Process for Performing Internal Events PRA Peer Review Using the ASME/ANS PRA Standard" (Reference 41).

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 RA-Sa-2009 PRA Standard provides technical supporting requirements in terms of three capability categories (CCs). The intent of the delineation of the CCs within the supporting requirements 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 NRC staff regards current good practice (i.e., CC-II of the ASME/ANS Standard) is adequate for most applications. Consistent with the guidance in RG 1.200 and RG 1.174 for this application of the Point Beach PRA to assess the risk associated with GL 2004-02-related phenomena, the NRC staff considered CC-II to be adequate.

The internal events PRA model was peer reviewed and assessed against RG 1.200, Revision 2, endorsed guidance consistent with NRC Regulatory Issue Summary (RIS) 2007-06, "Regulatory Guide 1.200 Implementation," dated March 22, 2007 (Reference 42). The review and closure of finding-level facts and observations (F&Os) were performed by an independent assessment team using the process documented in appendix X to NEI 05-04, NEI 07-12, and NEI 12-13, "Close-out of Facts and Observations" (F&Os) as accepted by the NRC in letter dated May 3, 2017 (Reference 43). The reviews also met the requirements of NEI 17-07, Revision 2, "Performance of PRA Peer Reviews Using the ASME/ANS PRA Standard," dated August 2019 (Reference 44). Only six F&Os remain open, which the licensee listed in table 7-2 of Enclosure 4 of the July 29, 2022, submittal. The licensee evaluated the impact of each of these open F&Os to the GSI-191 evaluation and concluded that there was no impact. The licensee stated that 97 percent of the supporting requirements in the internal events PRA model satisfy CC-II requirements of the PRA standard, and that the non-conforming aspects of the PRA model, as identified in the open F&Os, had no impact to the GSI-191 evaluation.

The licensee stated that the PRA scope and technical adequacy is met for this application because the ASME/ANS RA-Sa-2009 PRA Standard requirements are generally met at CC-II. The resolved findings and the basis for resolution are documented in the Point Beach PRA documentation and the F&O closure review reports.

Based on the licensee-implemented peer reviews following ASME/ANS standards and NEI guidance and the CC-II classification of the internal events PRA, the NRC staff concludes that the internal events PRA is adequate to support the risk-informed GSI-191 assessment.

3.4.1.4 Plant Representation

Regulatory Position C.2.3.4 in RG 1.174, Revision 3, states, in part, that:

> The PRA results used to support an application are derived from a base 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.

The NRC staff concludes that the licensee's internal events PRA model adequately represents the as-built and as-operated plant to the extent needed to support the GL 2004-02 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.

3.4.1.5 NRC Staff Conclusion Regarding the Base PRA Model

The NRC staff concludes that the Point Beach base PRA model used in support of the licensee's GL 2004-02 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 3, and RG 1.200, Revision 2.

3.4.2 Risk-Informed Approach for Addressing the Effects of Debris on LTCC

The licensee implemented simplified computations, using limited information from the internal events PRA, and combined those computations 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 relevant initiating events for which debris could adversely affect the CDF or LERF. This includes how relevant scenarios (i.e., an initiating event followed by a plant response 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 Enclosure 4 of the July 29, 2022, submittal, 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, not requiring sump recirculation, 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, Volume 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," dated March 2009 (Reference 45). Specifically, NUREG-1855, Volume 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 in the following subsections.

3.4.2.1.1 Initial Plantwide Screening

The initiating events considered in the licensee risk-informed analysis included those 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 (see section 3.0 of Enclosure 4 of the July 29, 2022, submittal).

The licensee 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. The licensee considered the following initiating internal events and excluded them from GSI-191 risk assessment because they do not generate debris inside containment:

- Steam generator tube rupture (SGTR)
- Interfacing systems LOCAs (ISLOCAs) that discharge outside containment
- Anticipated transients including inadvertent safety injection (SI), inadvertent or stuck-open power operated relief valves (PORVs) that discharge to the pressurizer relief tank (PRT), and loss of offsite power
- Secondary side breaks outside containment
- Initiating events due to loss of component cooling water, loss of service water, and loss of alternating current or direct current power

The licensee excluded the following hazards because they also do not generate debris inside containment:

- Internal and external flood
- Internal fires and fire-induced LOCAs
- Meteor impacts
- Aircraft impacts
- Turbine missiles
- Heavy load drops
- High winds

The licensee excluded initiating events based on (1) small equivalent break sizes, (2) reduced flows required to compensate the water inventory compared to equivalent flows in strainer tests, and (3) location of breaks away from significant insulation sources.

For seismic events, the licensee examined the Point Beach Individual Plant Examination for External Events (IPEEE) to assess the risk impact of seismic-induced LOCAs. The licensee excluded seismic-induced medium and large LOCAs based on a high confidence of low probability of failure of 0.3 g. The licensee concluded that seismic-induced small LOCAs were not a significant risk contributor. The licensee judged that seismic-induced LOCAs had a negligible risk contribution for ECCS strainer performance, based on low probability and not enough debris generated to challenge the sump strainer operability.

The licensee concluded that the only initiating events of relevance to the GSI-191 risk-informed assessment were (1) small, medium, and large LOCAs due to pipe breaks, failure of non-piping components, and water hammer and (2) SSBIs that cause a consequential LOCA (e.g., due to failure to terminate safety injection, loss of auxiliary feedwater, or a stuck open PORV) and require sump recirculation. The NRC staff reviewed the licensee's screening approach and concluded 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 GL 2004-02 phenomena.

3.4.2.1.2 Location-Specific Screening

For LOCA events, the effects of debris may be dependent on the location of the initiating event. Therefore, the licensee completed a location-specific analysis to identify accident sequences that could be adversely impacted by debris. The licensee performed a break location-specific evaluation for all ASME Class 1 piping welds in the RCS between the RPV and the first isolation valve (see section 3.2 of Enclosure 4 of the July 29, 2022, submittal). During the regulatory audit, the NRC staff requested clarification for the definition of "first isolation valve." In response to the NRC staff RAI (STSB-RAI-2), the licensee stated that the Class 1 boundary has two isolation valves in series that isolate the RCS from other systems. The first isolation valve was defined as the valve closest to the RCS side. The licensee further clarified that to fail, this valve would have to fail open in order for breaks downstream of the valve to cause a LOCA.

The licensee proposed that any LOCA break that generates and transports more debris to the strainer than debris loads shown to be acceptable by strainer tests could be assumed to result in strainer failure and core damage. Breaks that produce less than the amount of tested fiber reaching the strainer were considered to result in successful operation of the ECCS strainers. The licensee also examined the headloss from the calculated debris load to exclude failure from some other causes (for example deaeration, flashing, or mechanical collapse).

The acceptable debris loading was determined by testing (see section 3.4.8.3 of this SE). In order to determine the amount of debris generated from each potential break location, break size, and break orientation, the licensee developed a computer-aided design (CAD) model of containment (referred to as the BADGER model) and input generated debris amounts based on ZOI descriptions in the NARWHAL model. The NARWHAL model computed transported debris amounts loading the strainers, including generated chemical products such as sodium aluminum silicate, latent debris, and unqualified coatings. The model also accounted for transport processes such as blowdown, washdown, pool fill, and recirculation. The CAD model included the locations of each potential break location (welds in the RCS) and locations of debris sources that could be damaged by a LOCA jet.

Based on initial screening results, the licensee performed a quantitative analysis of LOCA RCS pipe breaks ranging from half-inch partial breaks to double-ended guillotine breaks (DEGBs) on every Class 1 ISI weld inside the first isolation valve. The licensee explicitly considered 378 and 341 ISI welds within the first isolation valve in its risk-informed analysis, in Units 1 and 2, respectively. The licensee concluded that breaks outside the first isolation valve are risk insignificant, due to the dependence on failure of that valve (e.g., failure to close, spurious opening, or leaking valve) for a LOCA to arise (see section 6.3 of Enclosure 4 of the July 29, 2022, submittal). Also, the licensee examined SSBIs (e.g., a large break in a main steam or feedwater line) using bounding analyses considering the internal events PRA model.

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. 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 adequate, because the licensee employed a systematic screening process using initial plantwide and location-specific screening approaches to identify relevant scenarios and eliminate scenarios that do not affect the GL 2004-02 risk assessment, in a manner consistent with state-of-practice approaches described in NUREG-1855, Revision 1. The licensee included all scenarios and initiating events relevant to the GL 2004-02 evaluation.

3.4.2.2 Initiating Event Frequencies

The licensee implemented a simplified computation of the ΔCDF using mean LOCA frequencies for small, medium, and LBLOCAs in the Initiating Event Data Sheets updated in 2015 by the Idaho National Laboratory (INL) (Reference 15) for the Base Case, and probabilities of strainer failure conditional on breaks of a given size computed using NARWHAL outputs. The LOCA frequencies in the INL document are based on geometric mean aggregation frequencies in table 7.19 of NUREG-1829, with a Bayesian correction to reflect no LOCA events in 797 reactor-years of operation of PWRs. The licensee provided additional information on the exceedance frequencies for small, medium, and large LOCA breaks in response to an NRC staff RAI (APLB-RAI-2).

The licensee did not use the internal events PRA model for the ΔCDF computation. For sensitivity/uncertainty and margin analyses, the licensee used LOCA frequencies in NUREG-1829, Volume 1, 40-year values, including geometric mean and arithmetic aggregations of expert elicited values.

For the calculation of small, medium, and large LOCA conditional failure probability, the LOCA frequencies were uniformly 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 location, operation conditions, and specific degradation mechanism (i.e., every break of the same size is assumed to occur with equal frequency).

The licensee used a log-linear interpolation scheme (i.e., linear interpolation between break sizes and logarithmic between frequencies) to estimate frequencies for specific large break sizes not explicitly listed in the INL Initiating Event Data Sheets or in the tables in NUREG-1829. The NRC staff considers the licensee’s use of log-linear interpolation of INL data acceptable and consistent with draft guidance.

The guidance in NUREG-1829 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. Point Beach Units 1 and 2 were initially licensed in 1970 and 1973, respectfully, with both licenses renewed in 2005. Point Beach has been operating for more than 40 years. For the sensitivity/uncertainty analyses, the licensee selected LOCA frequencies in NUREG-1829 corresponding to 40 years of operation, which the NRC staff considers acceptable. The NRC staff also finds acceptable defining the Base Case using the geometric mean aggregation values, which is consistent with the STP GSI-191 pilot assessment and the assessments of the risk-informed submittals for Vogtle Electric Generating Plant, Units 1 and 2 (Vogtle) (Reference 46), and Callaway Plant Unit No. 1 (Reference 47), and recommendations in NUREG-1829.

The licensee summarized ΔCDF results in table 5-4 of Enclosure 4 of the July 29, 2022, submittal. Those results were updated in response to an NRC staff RAI (NSG-RAI-1) to address an error in selecting a shorter duration for injection from the RWST in case of one failed train of the ECCS. Revised values of the ΔCDF and ΔLERF for Point Beach Units 1 and 2 increased by 3.6 percent and 7.4 percent, respectively, but did not change the licensee’s conclusion of very small risk.

The licensee also examined the effect of assuming every break to be a DEGB (as opposed to the baseline approach of considering breaks of variable size less than or equal to the pipe diameter) and concluded that the corresponding ΔCDF is greater than the baseline ΔCDF considering continuum break sizes but less than $1x10^{-6}$/year. The licensee's approach for evaluating the impact of the aggregation method and the continuum break and DEGB alternatives is consistent with the recommendation in NUREG-1829. The NRC staff reviewed the licensee's uncertainty and sensitivity analysis, with results in section 6 of Enclosure 4 of the July 29, 2022, submittal, and revised results in response to an NRC staff RAI (NSG-RAI-1) and concluded that the licensee identified and dispositioned key assumptions and sources of uncertainty in its systematic risk assessment consistent with the guidance in RG 1.174. The NRC staff's review of this topic is discussed in section 3.4.2.9 of this SE.

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:

- LOCA break frequencies were predominantly based on NUREG-1829, which is the most current source of information available.
- The licensee interpreted and used the NUREG-1829 data in a manner consistent with the guidance in NUREG-1829 and precedents addressing GL 2004-02.
- 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, continuum break and DEGB models.

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 considered system response to a LOCA break in the presence of debris. In the NARWHAL model, strainer testing was used to define debris amounts to postulate strainer failure. The Point Beach system includes two independent ECCS trains connected to two independent sumps with strainers. Each train includes an RHR pump, which supplies suction to safety injection (SI) and CS pumps. In addition to these pumps, each train of ECCS contains a safety injection accumulator (SIA).

The risk-informed analysis by the licensee considered different pump failure configurations and bounded all possible combinations by two cases: no pump failure case (i.e., all pumps, RHR, SI, and CS, assumed operational) and single train failure case (i.e., failed RHR, SI, and CS pumps in one train, and pumps assumed operational in the other train). Successful recirculation requires at least one RHR pump and one SI pump operational, so cases with failure of both RHR pumps and SI pumps were assumed to automatically cause core damage and were excluded from the risk-informed analysis to address GL 2004-02. The BADGER model was used to compute generated debris quantities for postulated breaks (which are independent of the pump configurations), and the NARWHAL model was used to compute transported debris, including chemical precipitates, to one or two strainers, depending on the pump configuration.

The licensee implemented independent computations to examine in-vessel debris buildup in Enclosure 3 of the July 29, 2022, submittal, with supplemental information in response to an NRC staff RAI (STSB-RAI-13). The licensee assumed two operational trains of ECCS. This pump and flow state tends to maximize fiber build-up in the vessel (more fiber would penetrate two operational strainers). The in-vessel mass balance computations were complementary to the NARWHAL computations, focusing only on cases with 550 pound mass (lbm) of fiber initially in the pool which would not cause strainer failure.

The licensee excluded other initiating events causing different scenarios (see section 3.4.2.1.1 of this SE).

NRC Staff Conclusion Regarding Scenario Development

The NRC staff evaluated the licensee’s scenario development process and results and concludes that the licensee adequately evaluated the relevant scenarios potentially causing strainer failure and in-vessel fiber buildup. The licensee considered models greatly simplifying the description of the system response. The licensee used a systematic process to identify germane operating components and states, and properly considered the period of performance in the risk-informed analysis. The NRC staff concluded that the licensee’s consideration of two bounding ECCS pump configurations for strainer debris bed buildup analysis and two functional strainers for in-vessel buildup analysis is adequate because it maximizes the probability of system failure.

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 failure processes below and did not identify additional failure modes (see section 4.1 of Enclosure 4 of the July 29, 2022, submittal). These failure modes are only those related to debris.

a. Excessive headloss at the strainer leads to loss of NPSH for adequate operation of the RHR pumps.

b. Excessive headloss at the strainer causes mechanical collapse of the strainer.

c. Debris prevents adequate flow to the strainer or prevents the strainer from attaining adequate submergence (strainer headloss exceeds half of the submerged strainer height when the strainer is partially submerged).

d. Excessive headloss at the strainer lowers the fluid pressure, causing release of dissolved gases (i.e., degassing) and void fractions in excess of pump limits.

e. Debris accumulation on the strainer exceeds amounts considered in strainer tests.

f. Upstream blockage.

g. Air entrainment due to vortexing.

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

i. In-vessel downstream effects: debris results in core blockage and decay heat is not adequately removed from the fuel, and debris buildup on cladding results in inadequate decay heat removal.

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

The licensee evaluated relevant failure modes in its risk-informed analysis in Enclosures 3 and 4 of the July 29, 2022, submittal. The licensee evaluated the failure modes a through e using the NARWHAL code, which computes debris loads and headloss on strainers considering strainer tests and dependencies on the temperature and flow rate. The licensee concluded that the only failure mode that could cause strainer failure was mode e (debris loads not bounded by strainer tests). The failure mode e (exceedance of debris amounts in strainer tests) must occur before any of the failure modes a, b, and d. Strainer tests were successful tests, not showing any indications of insufficient NPSH, mechanical collapse, or degassing and excessive void fractions. The licensee concluded that failure modes c and f through j would not occur, based on maintenance activities and bounding analyses. Specifically, failure mode c (partial strainer submergence during recirculation) would not occur as demonstrated by bounding computations for a full range of LOCA breaks (see response 3.g.1 in Enclosure 3 of the July 29, 2022, submittal); failure mode f (upstream blockage) would not occur based on maintenance activities and programmatic controls to reduce the latent debris burden (see response 3.i in Enclosure 3 of the July 29, 2022, submittal). Failure mode g (vortexing) was concluded not to occur based on bounding testing (see response 3.f.3 in Enclosure 3 of the July 29, 2022, submittal). Failure mode h (ex-vessel downstream effects) was concluded not to occur (see response 3.m in Enclosure 3 of the July 29, 2022, submittal) based on guidance in WCAP-16406-P-A, Revision 1. Failure modes i (in-vessel downstream effects) and j (boric acid precipitation) were concluded not to occur based on fiber penetration testing, bounding computations concluding that fiber accumulation inside the vessel per fuel assembly would not exceed acceptance limits (see response 3.n in Enclosure 3 of the July 29, 2022, submittal).

<u>NRC Staff Conclusion Regarding Failure Mode Identification</u>

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 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 used the Point Beach PRA model of record as the source of the base CDF and LERF values, to define the probability of pump configurations for two bounding cases (all pumps functional and one ECCS train failed). Also, the licensee used the internal events PRA for a bounding assessment of SSBIs. The licensee stated that the Point Beach PRA model was not modified to incorporate initiating events for the GL 2004-02 risk-informed analysis. The licensee performed the risk quantification outside the PRA model, and conservatively assumed specific equipment configurations.

NRC Staff Conclusion Regarding Changes to the Base PRA Model

The NRC staff reviewed the information provided by the licensee and concluded that the use of the Point Beach PRA model of record, without changes, is acceptable to provide supplementary information required by the risk-informed assessment implemented by the licensee.

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. 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 headloss tests that included amounts of debris that bounded most of the debris generated by LOCA initiating events, except debris potentially generated by large LOCA breaks. The individual debris amounts of each type of debris included in the testing were used to define fail/pass criteria for strainer performance. Using NARWHAL computations of generated and transported debris to the strainer for LOCA break sizes and break orientations at specific weld locations, if any type of debris was not bounded by the tested amounts, the specific LOCA break was considered to cause strainer failure and core damage and to contribute to the increase in core damage frequency (ΔCDF). Strainer failure criteria were established for low density fiberglass, mineral wool, calcium silicate (Cal-Sil), particulate, and chemical debris types. Table 3.f.5-1 of Enclosure 3 of the July 29, 2022, submittal contains the debris limits at the test scale. See section 3.4.2.8.3 of this SE for further discussion on this topic. In addition, the licensee also considered combinations of debris types in the definition of failure, such that LOCA breaks with computed debris amounts not bounded by any strainer tests were considered to cause strainer failure and core damage.

The licensee conducted a risk-informed debris generation evaluation that considered the sources of debris that may affect system performance. The risk-informed debris generation evaluation included thousands of debris generation cases based on postulated breaks at all welds that could result in a LOCA. The evaluation considered hundreds of welds with breaks of varying size and orientation at each weld. Any break that was computed to generate and transport more of any debris type than was included in the strainer tests was assumed to cause strainer failure and core damage.

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 and the method to address debris generation and ZOI in sections 3.a and 3.b of Enclosure 3 of the July 29, 2022, submittal. The licensee also considered other potential initiating events (debris generation locations). Some of these initiating events were excluded from the systematic risk assessment or were not explicitly considered in the break selection process, as discussed in section 3.4.2.1Scope of the Systematic Risk Assessment of this SE.

The licensee stated that the debris generation calculation was performed using the methodology of NEI 04-07 and the associated NRC SE. 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 also considered secondary side breaks inside containment (large main steam and feedwater line breaks). The secondary line breaks that required recirculation were all assumed to lead to scenario failures.

The licensee implemented a simplified risk-informed analysis relying on NARWHAL and BADGER software, considering a range of potential break sizes and orientations on welds in Class 1 piping. The BADGER model uses CAD models for Point Beach to identify weld locations and insulation and qualified coating distributions. Debris amounts are computed in the BADGER software based on a ZOI concept. Debris sources that are not break-dependent, such as latent debris and unqualified coatings debris, were evaluated in the NARWHAL software.

The licensee assembled three-dimensional CAD models of the Point Beach containment buildings, tracking the as-built insulation configuration and qualified coating distribution. The CAD model was used as input to the BADGER software to calculate debris quantities for each different weld, and for each break size and break orientation. BADGER implemented a ZOI concept to compute the debris amounts for each debris type. The ZOI represents the zone or volume in space where a two-phase jet from a HELB 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 material impacted by the jet. Higher system pressures result in increased ZOIs. Robust insulation materials have smaller ZOIs than fragile materials. The licensee considered each circumferential butt weld as a postulated break location, with partial breaks[1] of different size up to the pipe diameter. Breaks equal to the pipe diameter were considered DEGBs or single-ended guillotine breaks (SEGBs).[2] For each partial break size, the licensee considered different orientations of the hemispherical ZOI, in 45-degree angular increments, to evaluate a range of debris sources located around a break. Partial break sizes were evaluated at discrete values equal to 0.5, 2, 4, 6, 8, 10, 12, 14, 17, 20, 23, and 26 inches (see section 3.a.1 of Enclosure 3 of the July 29, 2022, submittal). The amounts of different types of debris were computed and compiled in a database generated by BADGER.

Strainer testing included bounding amounts of debris types for most break scenarios. The licensee listed the four breaks (i.e., a specific break location, size, and orientation) for each unit that produced the largest fiber and Cal-Sil amounts in table 3.a.3-1 of Enclosure 3 of the July 29, 2022, submittal. In the BADGER and NARWHAL analyses, the strainer fibrous debris amounts include ZOI-dependent (debris from insulation and qualified coatings) and ZOI-independent (latent debris and unqualified coatings) amounts, as well as chemical debris accounting for precipitation reactions. The ZOI-independent debris amounts are assumed to be the same for all breaks.

During the regulatory audit, the NRC staff asked whether figure 3.a.1-1 in Enclosure 3 of the July 29, 2022, submittal depicted all break locations. In response to the NRC staff's RAI (STSB-RAI-4), the licensee stated that some break locations were inadvertently left off of the

[1] The licensee defined a "partial break" as a break of diameter less than the pipe diameter. The ZOI was assumed to be of hemispherical shape for partial breaks and centered at the outside of the pipe circumference (see section 3.b.1 of Enclosure 3 of the July 29, 2022, submittal).

[2] A DEGB is a break of size equal to the pipe diameter, with a full pipe offset. The ZOI was assumed to be spherical and centered at the axis of the pipe at the break location (see section 3.b.1 in Enclosure 3 of the July 29, 2022, submittal). SEGBs were assumed to occur if a closed valve was within 10 pipe diameters, with a hemispherical ZOI centered at the axis of the pipe at the break location.

figure. The licensee stated that all welds within the first isolation valve were included in the analysis and provided an updated drawing as part of its response.

NRC Staff Conclusion Regarding Break Selection

The NRC staff concludes that the break selection evaluation is acceptable because the licensee evaluated all welds on ASME Code Class 1 pipes that can result in a LOCA. Although the NRC-approved NEI 04-07 guidance states 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 all breaks and are more likely break locations, consistent with recommendations in NUREG-1829.

The NRC staff concludes that the break selection process and criteria are acceptable because they identified 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 the licensee 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 its submittal the licensee:

- Described and provided the basis for the break selection criteria used in the evaluation.
- Examined the potential contribution of secondary line breaks inside containment (e.g., main steam lines and feedwater lines).
- Discussed the basis for reaching the conclusion that the break size(s) and locations chosen present the greatest challenge to post-accident sump performance.

The licensee provided a basis for the use of the break selection process in the overall evaluation of change in risk due to LOCAs. The licensee evaluated all Class 1 welds as potential break locations. Based on the above, the NRC staff concludes that the break selection methodology is acceptable to support estimates of risk.

3.4.2.6.2 Debris Generation and ZOI Submodels

The licensee defined the ZOIs for the following materials: NUKON™, low-density fiberglass (LDFG), Temp-Mat, mineral wool, Cal-Sil (including asbestos Cal-Sil), Transco reflective metal insulation (RMI), and Mirror RMI using NRC-approved guidance in NEI 04-07 (GR/SE). The licensee also used debris-specific ZOIs as allowed by the GR/SE. The licensee adopted debris size distributions for the various debris types based on testing. The licensee calculated the amounts of debris and debris size distributions that could be generated from each postulated break. To compute the probability of sump strainer failure, the licensee compared the computed debris amounts for each postulated break accumulating on the strainer to strainer tests. When a bounding debris test did not exist or when any debris type exceeded its test limit, the break was postulated to cause strainer and core failure.

The ZOIs for each material are summarized in table 3.b.1-1 of Enclosure 3 of the July 29, 2022, submittal. In the following discussion, the symbol D is used to represent the break size. The ZOI is assumed to be of hemispherical shape for partial breaks on circumferential welds or SEGBs, and of spherical shape for DEGBs on circumferential welds. The size of the ZOI is defined by

the radius (L) of the sphere or hemisphere, and expressed as multiples of the break size D. More robust materials have higher damage pressures and smaller ZOI radius.
NUKON™ and LDFG

For its risk-informed analysis, the licensee used a 17D ZOI radius for LDFG and NUKON™, which is consistent with the guidance in NEI 04-07. For LDFG and 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. 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. The licensee used a similar approach as was used for Vogtle, which was found to be acceptable by the NRC staff (Reference 46).

Temp-Mat

The licensee used a 11.7D ZOI for Temp-Mat insulation. This is consistent with the guidance in the GR/SE. The debris size distribution for Temp-Mat was also determined using the centroid method as discussed for LDFG and NUKON™, above, with zones that represent destruction test data for Temp-Mat.

Cal-Sil

The licensee used a ZOI of 6.4D for Cal-Sil and asbestos Cal-Sil insulation types. This ZOI is consistent with the guidance in the GR/SE. The centroid method was also used to determine the size distribution for Cal-Sil as fines, small pieces, and intact insulation. The centroid method appropriately reflected the size distribution of Cal-Sil during destruction testing.

Transco RMI

For Transco RMI, the licensee adopted the NRC-approved default value of 2D as prescribed in the GR/SE. The Transco RMI was assumed to fail as 75 percent small pieces and 25 percent large pieces.

Mirror RMI

The licensee used the NRC-approved ZOI default value of L/D equal to 28.6 to define the ZOI radius for Mirror RMI, as prescribed in the GR/SE. The Mirror RMI was assumed to fail as 75 percent small pieces and 25 percent large pieces, similar to the Transco RMI.

Mineral Wool

The licensee stated that the mineral wool at Point Beach was provided by Transco and encapsulated in stainless steel cassettes. For this insulation system the licensee used a 5.4D ZOI and assumed that 100 percent of the mineral wool was destroyed as fines. The ZOI size was based on a comparison with K-Wool insulation. K-Wool is an unjacketed mineral wool with a wire mesh reinforcement in a blanket. The licensee described the K-Wool insulation system and the testing that was performed to determine the ZOI. The licensee stated that the stainless-steel jacketing on the Point Beach mineral wool is more robust than that in the K-Wool destruction testing. The licensee concluded, based on the similarity between the insulating portion of the K-Wool and mineral wool, the more robust encapsulation for the Point Beach mineral wool, and the assumption that all of the insulation within the ZOI would be rendered into fines, that the 5.4D ZOI assigned to mineral wool is conservative and acceptable. In addition to the comparison, the licensee performed a sensitivity study to compare the mineral wool debris amount that would reach the strainer if a 17D ZOI and the NUKON™ debris size distribution were used. The sensitivity case included 10 percent erosion of the small and large pieces of NUKON™ into fines. The sensitivity case found that the using the 5.4D ZOI and assuming all fines resulted in three times more fine debris than using the NUKON™ assumptions. The licensee also stated that the assumptions used for mineral wool debris generation are similar to those used in the North Anna Power Station, Unit Nos. 1 and 2 (North Anna) and Surry Power Station, Unit Nos. 1 and 2 GL 04-02 submittals. This treatment was evaluated and accepted in an audit report for North Anna (Reference 48). The NRC discussed the treatment of mineral wool with the licensee during a regulatory audit to gain further insights into the basis for the assigned ZOI. The licensee stated that the cassettes that include the mineral wool at Point Beach are similar to those used at North Anna and they are both spot welded and riveted. Based on this description and the conservatism from treating all material within the ZOI as fines, the NRC staff determined that the licensee's debris generation analysis for mineral wool is acceptable.

The licensee provided the breaks that generated the most limiting fibrous and Cal-Sil amounts (see Tables 3.b.4-1 and 3.b.4-3) and the breaks with the most fibrous and Cal-Sil debris (see Tables 3.b.4-2 and 3.b.4-4) that do not fail any of the acceptance criteria for single train ECCS configuration for Units 1 and 2 (see Enclosure 3 of the July 29, 2022, submittal).

The licensee stated that the area of miscellaneous debris (signs, tags, placards, etc.) identified via walkdowns is 120 square feet (ft^2) for Unit 1 and 152 ft^2 for Unit 2. However, the licensee assumed a conservative surface area of 200 ft^2 for both units in its analyses.

Risk-Informed Analysis

The licensee followed the debris generation calculation methodology specified in NEI 04-07 and justified adopted deviations from the guidance. The licensee evaluated a full range of breaks instead of assessing only the limiting breaks, as recommended in NEI 04-07. All unisolable welds within the Class 1 ISI pressure boundary (i.e., welds inside the first isolation valve) were evaluated, including DEGBs, SEGBs, and partial breaks. In order to calculate thousands of break scenarios, the licensee considered a CAD model describing the insulation configuration, qualified coating distribution, and location of robust barriers within the containment, to automate computation of insulation and qualified coating amounts within the ZOI of each postulated break in the BADGER code. The licensee calculated debris amounts for breaks in each circumferential weld, ranging from ½ inch to the full pipe diameter, and considered a range of orientations for each break size in 45-degree increments. For DEGBs, a spherical ZOI was assumed centered at the axis of the pipe in the plane of the weld. For SEGBs, the ZOI was assumed hemispherical with a hemisphere axis parallel to the pipe axis. For the partial breaks, the ZOI was a hemisphere oriented normal to the pipe axis and centered at the edge of the pipe at the break location (see section 3.b.1 of Enclosure 3 of the July 29, 2022, submittal). The smallest break size computed by the licensee to potentially cause strainer failure was a DEGB of 11.188 inches in the baseline analysis as described in Enclosure 4 of the July 29, 2022, submittal (page E4-35). The licensee used the NARWHAL code to compute transported debris amounts

to the strainer, including latent debris, unqualified coatings, and chemical precipitates like sodium aluminum silicate (SAS).
NARWHAL code algorithms have been evaluated previously by the NRC staff as part of the Vogtle license amendment application related to GL 2004-02, supported by audits and independent calculations sponsored by the NRC to explore the adequacy of the BADGER and NARWHAL software to identify insulation and coating sources and compute debris amounts transported to the strainers. The NRC staff concluded that the use of NARWHAL in conjunction with a detailed CAD model in BADGER is a reliable approach to quantify potential debris amounts within the ZOI of a postulated break.

The licensee assumed that fixed amounts of unqualified and qualified but degraded coatings would detach immediately after the LOCA event and form particulate debris that could be transported to the strainer for any break. The licensee considered the presence of 200 pounds of latent debris, of which 15 percent was assumed in the form of fibrous debris (30 pounds) and 85 percent in the form of particulate (170 pounds). The estimated total amount of particulate from unqualified coatings was increased by 10 percent for use in the NARWHAL model. NARWHAL used 11.48 ft^3 for the unqualified coatings. For degraded qualified coatings the licensee assumed 3.26 ft^3, which included margin from the plant values. These values are provided in tables 3.h.1-5 and 6 of Enclosure 3 of the July 29, 2022, submittal (pages E3-120 and 121). These values were considered constant for all of the postulated breaks, independent of the radius of the ZOI.

The licensee also considered ZOI-dependent sources of particulates such as qualified inorganic zinc and qualified epoxy. By comparing the amounts of unqualified and degraded qualified coatings to the amounts of qualified coatings generated from the bounding breaks (in tables 3.b.4-1 through 4 of Enclosure 3 of the July 29, 2022, submittal) it is concluded that the dominant proportion of particulates from any break arise from ZOI-independent sources (i.e., unqualified or degraded coatings).

Strainer tests were conducted with particulates Cal-Sil and fiber that bounded most breaks. The licensee evaluated each postulated break size, location, and orientation for the evaluated welds to determine if the fibrous, Cal-Sil, particulate, or chemical debris generated and transported from that break exceeded the tested amounts of each debris type. Strainer failures were assumed to occur if the amount of any debris type transported to the strainer exceeded the tested amounts, or if no strainer test would bound[3] all debris types computed for a particular break.

The licensee programmed algorithms in the BADGER software to automate computation of debris amounts generated by each postulated break location at each size and orientation and used the NARWHAL software to compute debris transported to the strainers.

The NRC staff concludes that the licensee properly quantified amounts of debris that could be generated within the Point Beach containments by the postulated LOCA breaks. The analysis included ZOI-dependent (e.g., fibrous debris from different insulation types, Cal-Sil, and qualified coatings) and ZOI-independent (e.g., dust and latent debris, unqualified and degraded coatings), as well as SAS precipitates which depend on both ZOI-dependent (e.g., qualified

[3] A strainer test is considered bounding if each of the debris types is greater than the computed transported amounts by NARWHAL for a specific break. If the NARWHAL computed debris amount for at least one debris type is greater than the tested amount for a specific test, another test is sought as bounding test. If no bounding test is found, then the break is assumed to cause strainer failure and core damage.

aluminum coatings and mineral wool) and ZOI-independent (latent fibrous debris) debris sources. For the ZOI-dependent debris, the licensee computed debris amounts using BADGER, which relied on a CAD model capturing the location and distribution of insulation and debris sources within the containment. For each break (of specific location, size, orientation, and ZOI), BADGER used CAD model information to determine debris amounts for each material type. For each break, the CAD model clipped the ZOI to account for robust barriers. The NRC staff previously concluded that algorithms in the BADGER and NARWHAL codes for the computation of generated and transported debris were properly implemented. The licensee adequately considered random factors such as the break size and 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 postulated break is acceptable.

NRC Staff Conclusion Regarding Debris Generation and ZOI Submodels

The NRC staff notes that the licensee considered guidelines in NEI 04-07 to (1) define ZOIs; (2) account for robust barriers; (3) compute debris amounts of LDFG and fibrous insulation, Cal-Sil, and particulate sources such as qualified coatings; (4) compute debris size distributions; and (5) estimate debris amounts associated with latent fiber, latent particulate, and unqualified and damaged or degraded qualified coatings.

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 judgement, and review of the licensee's software outputs to perform the verifications. The Point Beach method to compute debris amounts relies on BADGER and NARWHAL software, which was examined in detail as part of the LAR for Vogtle to risk-inform its treatment of the effects of debris on LTCC. This approach allows the NRC staff to conclude, with a high level of confidence, that the calculations for debris generation were conducted and applied properly.

The NRC staff reviewed the licensee's evaluation against the NRC-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 a 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 strainer headloss tests. Any break that results in any type of debris reaching the strainer exceeding the amounts in the test was assumed to lead to strainer failure and contribute to the plant risk. To address combined debris types, if no strainer test was available that would bound all debris types loading the strainer for a specific break, it was assumed that the break would cause strainer failure and core damage. The licensee's methods are consistent with NRC 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 they identify 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 LAR is acceptable for use in an assessment or evaluation model of the effects of debris on the ECCS LTCC function, as required, in part, by 10 CFR 50.46.

The NRC staff concludes that the licensee provided sufficient information as requested by GL 2004-02 and further described in the revised content guide for GL 2004-02 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 materials with ZOIs not defined in the guidance, 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 destruction testing conducted to determine ZOIs.
- Quantified each debris type generated for each break location, size, and orientation evaluated.

3.4.2.6.3 Debris Characteristics

The licensee generally used the NRC staff's SE on NEI 04-07 to evaluate the debris characteristics.

The licensee listed the debris characteristics for the debris types listed below in table 3.c.1-1 of Enclosure 3 of the July 29, 2022, submittal (page E3-23). Since testing was used to determine the headlosses associated with debris, the microscopic characteristics are not important to the risk-informed evaluation. On the other hand, debris densities are important inputs to the evaluation.

NUKON™ and LDFG

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

Mineral Wool

For mineral wool, which is encapsulated in stainless steel cassettes, the licensee assumed a ZOI of 5.4D. All mineral wool material within the ZOI was assumed to be destroyed to form fine fibrous debris. The licensee's treatment of this material is discussed in detail in section 3.4.2.6.2 of this SE.

Temp-Mat

The licensee used a 11.7D ZOI for Temp-Mat insulation. This is consistent with the guidance in the GR/SE. The debris size distribution for Temp-Mat was also determined using the centroid method as discussed in section 3.4.2.6.2 of this SE, with zones that represent destruction test data for Temp-Mat.

RMI

For RMI, the licensee used the GR/SE guidance to determine the size distributions. RMI was not included in the headloss testing.

Cal-Sil and Asbestos Cal-Sil

Cal-Sil and Asbestos Cal-Sil were assumed to have a ZOI of 6.4D, which is consistent with the NEI 04-07 guidance. The licensee then further analyzed the ZOI using the centroid methodology discussed in section 3.4.2.6.2 of this SE. The centroid distance was used to determine the size distribution of these materials predicted damaged within the ZOI.

The licensee provided the as-fabricated and material densities of the fibrous materials and the RMI. Because headloss testing (not theoretical calculations) was used to establish the headlosses caused by the debris, the licensee did not provide microscopic material characteristics that were not important to the evaluation.

<u>NRC Staff Conclusion Regarding Debris Characteristics</u>

The NRC staff concludes that the debris characteristics were defined per the applicable guidance with the exception of mineral wool. For mineral wool, the licensee assumed a ZOI and destruction properties that are conservative as discussed in section 3.4.2.6.2 of this SE. As previously discussed, the NRC staff found the treatment of mineral wool acceptable.

The NRC staff concludes that the licensee 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, because the licensee:

- Identified debris characteristics using an approved methodology and default values where available.
- For the mineral wool insulation system that is not defined in the guidance, the licensee chose a ZOI and destruction properties that conservatively bounded the effects of the debris.
- For RMI which was not included in headloss testing, it has been shown that the material does not result in increased headloss if transported to the strainer. The NRC staff also noted that for most scenarios a small amount of RMI could be mobilized but it would remain on the floor near the strainer.

3.4.2.6.4 Latent Debris

The licensee followed the guidance in the GR/SE to evaluate latent debris. A bounding value of 150 lbm of latent debris was assumed in the analysis, with the recommended 15 percent being latent fiber. The remaining 85 percent of the latent debris was assumed to be particulate debris.

The licensee sampled containment to determine the actual amount of latent debris present, following accepted guidance. The licensee performed sampling that determined that the amounts of latent debris in containment are approximately 62 lbm for Unit 1 and 55 lbm for Unit 2. Based on this, the amount of latent debris was conservatively assumed to be 150 lbm in each containment.

The licensee provided the assumed characteristics for the fibrous and particulate latent debris and stated that the characteristics are consistent with the GR/SE.

The licensee stated that a total surface area of 200 ft^2 of miscellaneous debris (tags and labels) was assumed to transport to the strainer with 25 percent overlap, per NRC guidance. The actual surface area of miscellaneous debris was determined via walkdowns, which identified 120 ft^2 in Unit 1 and 152 ft^2 in Unit 2. Therefore, the amount of miscellaneous debris assumed in the analysis is conservative.

NRC Staff Conclusion Regarding Latent Debris

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

- Identified the amounts of latent debris and miscellaneous debris in containment using an approved methodology.
- Used conservative values in the headloss analysis compared to the actual values in containment.
- Identified debris characteristics using an approved methodology and default values.

3.4.2.6.5 Coatings

The licensee stated that the qualified and unqualified coatings debris within the ZOI were treated in accordance with the guidance in the GR/SE. The licensee defined its DBA qualified coatings systems used in the containment and provided the coating systems and manufacturers of those systems considered to be qualified. Tables 3.h.1-1 and 3.h.1-2 of Enclosure 3 of the July 29, 2022, submittal listed these systems, including pertinent information, for Units 1 and 2, respectively.

The licensee stated that all other coatings were assumed to be unqualified or actively delaminating qualified (ADQ) epoxy. The licensee listed the generic types of unqualified coatings and their amounts for each unit in tables 3.h.1-3 and 3.h.1-4 of Enclosure 3 of the July 29, 2022, submittal. The licensee also provided a table that listed the amounts of each type of unqualified coating assumed in the NARWHAL model in table 3.h.5 of Enclosure 3 of the July 29, 2022, submittal. The values in the table correspond to Unit 2 values plus a 10 percent margin.

The licensee discussed the assumptions used for coatings transport. The licensee assumed that the unqualified coatings fail as 10-micron particulates in lower containment, and those outside the ZOI enter the pool after the pool fill phase and transport 100 percent to the strainer. The licensee stated that settling is not credited for any particulate debris. The licensee also assumed that the qualified coatings in the ZOI fail as 10-micron particulate and transport 95 to 100 percent to the strainer, depending on the break location. The licensee provided chip sizes for ADQ epoxy that could fail as chips. The licensee used 3 categories of particulate debris in its analysis: (i) coating particulates, (ii) coating particulates and chips, and (iii) coating particulates and latent particulates. The NRC staff asked a question during the audit to gain a better understanding of the way these particulate debris groups were represented in testing and the analysis. In response to the NRC staff RAI (STSB-RAI-4), the licensee provided additional information regarding the way in which particulate and coating debris types were combined for the analysis. The licensee provided the logic for the debris grouping by explaining that the particulate debris increases headloss by filling voids in the debris bed. The smaller particulate debris has a greater impact on headloss. Therefore, it is acceptable to allow larger particulates to be represented by smaller particulates in the analysis. The NRC staff considered this response and concluded that it accurately represents both theoretical and empirical information regarding the impact of particulate debris on headloss. The NRC staff concluded that the debris groupings used in the analysis are acceptable.

The licensee stated that headloss testing used silica flour, with a median size distribution of 13.5 microns, as a surrogate for qualified coatings, unqualified coatings, and ADQ epoxy particulate. Pressure washed paint chips with a nominal size of approximately 0.125 inches were used to model the flat fine ADQ epoxy chips. Paint chips were used as a surrogate for the small ADQ epoxy chips. Additional detail on the amount and size of chips added to the headloss testing was included in section 3.f.4 of Enclosure 3 of the July 29, 2022, submittal.

The licensee based the ZOI sizes for qualified coatings on NRC-accepted jet impingement testing and staff guidance for reviewing the coatings evaluation. The licensee assumed the ZOI for epoxy topcoat to be 4D. No exposed inorganic zinc coatings were identified as installed in the Point Beach containments. The licensee determined the qualified coating debris amounts by using a three-dimensional model of containment that modeled the orientation of coated surfaces to postulated break jets. The licensee assumed that the coated areas within the appropriate ZOIs fail and calculated the debris amounts using the dry film thicknesses and densities for the coating system on the impacted surface.

The licensee assumed that all unqualified coatings in containment fail regardless of location. The quantities of unqualified coatings were calculated similarly to the qualified coatings. As discussed above, tables of unqualified coating amounts were provided for each unit. The licensee stated that the unqualified coating amounts are based on logs maintained at the plant.

The licensee discussed ADQ epoxy coatings, which are coatings that are initially installed as a qualified system but have physically degraded over time. Table 3.h.1-6 of Enclosure 3 of the July 29, 2022, submittal provides the size ranges and mass percentages of each size range for the ADQ coatings. The licensee assumed that ADQ epoxy coatings in the containment fail for every postulated break. The licensee used the information regarding the ADQ epoxy coating sizes for further evaluation during the transport and headloss evaluations. The licensee stated that flat large chips and curled chips were not included in the strainer testing because it was observed that they would not transport to the strainer, even with agitation added to promote transport.

The licensee stated that the plant conducts coating condition assessments at each refueling outage (RFO) to ensure that the coatings debris source term remains bounded by the design basis assumptions. The inspections are visual and are performed on all accessible coated surfaces in containment. If the visual examination identifies degradation, additional inspection and testing is performed to determine whether the coating system remains qualified and if any corrective actions are required. The licensee stated that the inspections and supplemental inspections are performed in accordance with current industry guidance. Non-conforming conditions are added to the corrective action program for evaluation and resolution. The licensee stated that after each RFO, a report is issued that summarizes the condition of the coatings in containment and contains metrics that can be used to evaluate the debris source term. The report also contains operating margin compared to the coating amounts used in the strainer evaluation.

NRC Staff Conclusion Regarding Coatings

The licensee performed its evaluation in accordance with NRC-approved guidance. The guidance includes the GR/SE and subsequent guidance based on TR WCAP-16568-P, "Jet Impingement Testing to Determine the Zone of Influence (ZOI) for DBA Qualified/Acceptable Coatings," dated June 2006 (Reference 49), and subsequent NRC review guidance, "Revised Guidance Regarding Coatings Zone of Influence for Review of Final Licensee Responses Generic Letter 2004-02, 'Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors'" (Reference 50), for coatings evaluations. The debris generation amounts were determined using appropriate ZOIs and the debris volume was properly preserved for testing by correcting for density differences between the coatings and the test surrogates. The transport metrics used for the coatings debris were also based on the approved guidance or plant-specific testing.

The licensee maintains a coatings condition assessment program that appropriately monitors, tests, and repairs coatings as required.

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

- Identified the amounts of coatings materials in containment that can become debris using approved methodologies.

- Used appropriate characteristics for the coatings as applied in the transport and headloss analyses.

- Identified an appropriate coatings surrogate for testing and used conversions to ensure that the volume of debris is preserved for the headloss tests.

- Identified a coatings assessment program to provide ongoing inspection and repair of qualified coatings systems.

3.4.2.6.6 Containment Material Control

The licensee stated that containment cleanliness controls had been implemented as described in the plant's response to NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors" (Reference 51). Containment cleanliness control programs exist to enhance containment cleanliness in Modes 1 through 4, for power entries, and for unit restart following outages. The licensee stated that the programs follow the guidance of NEI 02-01, "Condition Assessment Guidelines: Debris Sources Inside PWR Containments," Revision 1 (Reference 52), and minimize miscellaneous debris sources. A thorough inspection is performed at the end of each outage to ensure that unanalyzed debris sources have been removed. The licensee stated that implementation of these procedures also satisfies Point Beach TS SR 3.5.2.6, which requires a visual inspection of the strainers for debris and damage. The licensee also stated that foreign material controls are in place to prevent the introduction of foreign materials that could affect strainer performance.

The licensee described programmatic controls used to ensure that design changes inside containment do not result in unanalyzed debris sources. Materials added to containment are assessed for potential debris generation including that from insulation, coatings, and exposed aluminum. The licensee developed an engineering specification that defines the design basis for insulation debris amounts for the strainer.

The licensee stated that procedures are in place to control maintenance activities and temporary modifications that may affect the debris source term. Guidance for design changes is also applied to temporary modifications.

The licensee replaced the mineral wool insulation on the pressurizers, and fibrous insulation on the RCPs, with RMI. The licensee also stated that fibrous insulation on the Point Beach, Unit 2 main RCS loop piping was replaced with RMI. No additional insulation changes were described in the licensee's submittal, nor were any planned for the future. The licensee stated that there are no actions planned to reduce the debris source term at the sumps.

The licensee also cited that debris interceptors had been installed in the Point Beach, Unit 1 containment, but that credit was not taken for these to reduce the transport of debris to the strainers. The licensee stated that it is likely that the interceptors would result in some reduction of transport.

<u>NRC Staff Conclusion Regarding Containment Material Control</u>

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

- Identified the programs in place to reduce the introduction of debris during maintenance and modification activities, including temporary modifications.
- Described significant actions taken to reduce the debris source term that may occur at the sumps.

<u>NRC Staff Conclusion Regarding Debris Source Term Submodel</u>

Each of the aspects of the debris source term was evaluated. The NRC staff concluded that the debris source term submodel, including break selection, debris generation and ZOI, debris characteristics, latent debris, coatings, and containment material control were adequately addressed. 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 the debris source term submodel (including break selection, debris generation and ZOI, debris characteristics, latent debris, coatings, and containment material control) is acceptable because it identifies a number of postulated LOCAs of sufficiently differing properties 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 licensee's submittal is acceptable for use in an assessment or evaluation model of the effects of debris on the ECCS LTCC function, as required, in part, by 10 CFR 50.46.

The NRC staff concludes that the licensee 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.

3.4.2.7 Debris Transport Submodel

3.4.2.7.1 Strainer Transport

The licensee stated that the transport evaluation was based on the methods in the GR/SE. The blowdown, washdown, pool fill, and recirculation phases were modeled in the evaluation. The licensee used a transport logic tree methodology, to account for blowdown, washdown, pool fill, and recirculation, in the NARWHAL software to calculate the debris amounts reaching the strainer for each break. Erosion of larger pieces of debris was included in the evaluation.

The licensee provided the bounding debris transport fraction values for each phase of the transport evaluation for various break locations. For blowdown the fractions minimized the amount of debris retained in the break compartments, and for washdown the bounding values maximized the washdown to lower containment. The transport fractions were provided in tables 3.e.6-1 through 3.e.6-15 of Enclosure 3 of the July 29, 2022, submittal.

For pool fill transport, the licensee stated that although the transport to inactive volumes was calculated to be over 70 percent for both units, the transport value was limited to 15 percent as directed by the GR/SE.

For the recirculation phase, the licensee used Flow-3D (computational fluid dynamics (CFD) software) to perform transport simulations. Although the licensee stated that Stokes' Law was used for determining the settling velocity of fine particulates, settling of fines was not credited. The recirculation transport analysis compared transport metrics for the different debris types and sizes, determined via testing, to the local pool hydraulic conditions. If the velocity or turbulent kinetic energy in any specific area of the pool exceeded a metric, the debris in that region was considered to transport. The licensee stated that for the recirculation phase of the transport analysis all particulate fines and fiber fines transport 100 percent and do not settle. The licensee analyzed four cases of recirculation transport for each unit, a break in each of the two loops with a case run for each strainer in service. The maximum flow rates and minimum pool levels were used to maximize the amount of transported debris.

The licensee assigned an erosion fraction of 10 percent to larger pieces of fiber in the pool and 1 percent for fiber held up on gratings. The pool erosion fraction based on 30-day erosion testing and the erosion for fiber held on gratings is justified by the Drywell Debris Transport Study. The 10 percent erosion fraction for fiber fines deviates from the recommended guidance in NEI 04-07 (which recommends a 90 percent erosion fraction for fiberglass debris). The licensee provided that the 10 percent erosion fraction was based on generic 30-day erosion testing. The NRC staff reviewed the testing that was performed by ALION in 2010 (Reference 53). In LDFG erosion testing conducted by ALION, it was determined that small and large pieces of fiber in the sump pool eroded at a rate below 10 percent. The NRC staff reviewed and developed conclusions regarding this report that are documented in a letter dated June 30, 2010 (Reference 54). The NRC staff concluded that plants that could demonstrate that 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 stated that the generic testing is applicable to Point Beach. The licensee also assigned an erosion value of 17 percent for small pieces of Cal-Sil insulation settled in the pool. Erosion of

Cal-Sil was assumed for both the transportable and non-transportable pieces to maximize the amount of fine Cal-Sil reaching the strainer. An erosion fraction of 1 percent was used for fibrous debris and 17 percent for Cal-Sil held up above the pool. These erosion fractions are consistent with NRC staff guidance.

The licensee stated that debris interceptors are installed in the Unit 1 containment inside the secondary shield wall. These barriers were not credited for the capture of debris in the transport analysis. For the recirculation evaluation the debris interceptors were assumed to be blocked resulting in diverted flow through open flowpaths, increasing the flow velocity and transport rates.

The licensee stated that the overall transport fractions were determined by incorporating the different phases of transport for each break location, which were summarized in tables 3.e.6-16 through 3.e.6-25 of Enclosure 3 of the July 29, 2022, submittal.

The licensee used the maximum transport fractions for all break locations evaluated in the analysis. During the regulatory audit, and by conducting independent examination of the NARWHAL results, the NRC staff reviewed the transport calculations and logic trees and confirmed that the calculations were performed correctly, consistent with the transport fractions in tables 3.e.6-16 to 3.e.6-25 of Enclosure 3 of the July 29, 2022, submittal.

The debris transport amounts for scenarios that resulted in the greatest amounts of fiber and Cal-Sil and the greatest amounts of fiber and Cal-Sil that did not fail any acceptance criteria were provided in tables 3.e.6-27 through 3.e.6-30 of Enclosure 3 of the July 29, 2022, submittal.

3.4.2.7.2 In-Vessel Transport

The licensee calculated the amount of debris that could reach the reactor core after penetrating the strainer and transporting through the ECCS. The fiber penetration rate as a function of strainer loads was characterized by the licensee using plant-specific testing for Point Beach in 2014. The testing used a scaled strainer and included scaled amounts of debris based on the ratio of plant to the test strainer areas. Empirical functions for fiber penetration rates and fiber shedding as a function of strainer fiber loads were used in the calculation of in-vessel debris amounts for various ECCS pump configuration and pump flow rate scenarios. The licensee provided detailed fiber penetration and shedding functions in response to an NRC staff RAI (STSB-RAI-13). During testing the licensee used conservative methods to ensure that the penetration was maximized. For example, the spacing between strainer disks was increased to ensure that fiber did not bridge between the disks which would provide additional filtration and less penetration. Only fiber fines were used in the testing and the licensee performed three tests using different mixtures of the plant specific fiber types. The testing was conducted using plant specific attributes where possible. For example, the water chemistry was prototypical for the plant as was the approach velocity during the testing. Efforts were taken to ensure that all fiber reached the strainer or was accounted for by subtracting the settled amount from the amount introduced when calculating the percentage of penetration. The test result demonstrated that the NUKON™-only test resulted in the greatest penetration, so those results were used to develop empirical fiber penetration equations for Point Beach.

The licensee performed the in-vessel debris loading calculation for different scenarios with different pump configurations and flow rates to ensure that the maximum amount of debris reaching the core over time was determined. The analysis found that the safeguards condition with no spray pumps available and a reduced RHR flow rate with two RHR pumps running resulted in the largest amount of debris reaching the vessel, of the potential plant-specific scenarios within the licensing basis (see table 3.n.1-2 of Enclosure 3 of the July 29, 2022, submittal). The NRC staff noted that for all the cases with two RHR pumps running, the penetration results did not vary significantly. The NRC staff discussed the methodology with the licensee during the regulatory audit because it was not clear that the licensee had assumed the maximum amount of fibrous debris that could be transported to the strainer. The licensee stated that the fiber amount considered in the computations was 550 lbm, but NRC review of the NARWHAL output files suggested that different initial fiber masses could be considered. During the technical audit the NRC staff discussed this issue with the licensee and the licensee stated that volumes of fibrous debris types in NARWHAL were provided in LDFG equivalent volumes. The licensee considered 550 lbm in the in-vessel fiber penetration analysis as an upper bound of the fiber mass transported to strainers not causing strainer failure. The NRC staff examined the NARWHAL outputs considering this information and found that the transported fiber mass not causing strainer failure was indeed bounded by 550 lbm in both units for all postulated LOCA breaks.

During the regulatory audit, the NRC staff considered the empirical equations to compute fiber penetration through the strainer for different pump configuration and flow scenarios, as well as an initial fiber mass of 550 lbm, and independently computed practically identical in-vessel fiber loads to values reported in table 3.n.1-2 of Enclosure 3 of the July 29, 2022, submittal. The licensee's calculations predicted a maximum in-vessel fiber amount of 85.48 grams per fuel assembly (g/FA).

The results of the transport evaluation are important inputs for the strainer and in-vessel evaluations that are further evaluated in this SE.

NRC Staff Conclusions Regarding the Debris Transport Submodel

The licensee's approach to evaluating debris transport was consistent with the NEI 04-07 guidance and its associated NRC staff SE, and the licensee provided information requested in the content guide for GL 2004-02. For the in-vessel evaluation, the licensee's approach followed the NRC-approved guidance from TR WCAP-17788-P.

The NRC staff reviewed the licensee's transport evaluation against the NRC-accepted guidance in NEI 04-07 and verified the consistency of the computed debris amounts. 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. For the in-vessel analysis, the NRC staff verified the licensee's analyses with independent calculations. 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 licensee's submittal is acceptable for use in an assessment or evaluation model of the effects of debris on the ECCS LTCC function, as required, in part, by 10 CFR 50.46.

The NRC staff concludes that the licensee 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:

- Used approved and accepted guidance to perform the majority of the calculations.

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

- Provided a summary of, and supporting basis for, credit taken for reduction of debris amounts.

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

- Provided the limiting amount of fibrous debris that could accumulate in the reactor core.

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 of this SE, may have on the operation of equipment important to LTCC. This section examines the operation of the ECCS strainer, 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 in containment such that it may not reach the sump pool.

For this section, all descriptions attributed to the licensee's submittal are taken from the July 29, 2022, submittal, and the June 9, 2023, supplement. The majority of the information is from sections 3f, 3g, 3j, 3k, 3l, 3m, 3n, and 3o of Enclosure 3 of the July 29, 2022, submittal.

3.4.2.8.1 Upstream Effects

The licensee evaluated the containment for the potential for blockage or impedance of the transport of water to the sump and stated that the following areas were considered as part of the evaluation:

- Refueling Canal
- Steam Generators
- Annulus in Lower Containment
- Reactor Cavity (reactor cavity breaks only)
- Containment Spray Washdown

The licensee evaluated the layout of the structures and equipment and determined that significant blockage would not occur. The licensee used its containment CAD model and reviewed the transport CFD model to ensure that areas of the containment would not become blocked with debris and significant water volumes would not be held up during recirculation.

The licensee identified a potential for blockage of the refueling canal drain and installed a strainer designed to ensure that the drain will not become blocked and that flow of water to the sump from the refueling canal will be unimpeded.

The licensee evaluated the containment for structures that could prevent or delay the flow of water to the sumps. Areas that could retain water were identified. The licensee noted that the lower containment is not compartmentalized and that the steam generator compartments have significant open area between potential break locations and the containment floor. The licensee stated that at the base of each steam generator in Unit 1 there are five passages that communicate with the annulus and three of these have debris interceptors. The two open passages are adequate to allow necessary flow if the interceptors become blocked. Unit 2 does not have debris interceptors installed.

Some areas of the containment, like the reactor cavity, were identified as water holdup volumes. Other areas of the containment will not hold up water from reaching the sump. The holdup is accounted for in the sump level calculations. For the reactor cavity, a 16-inch diameter hole was bored in each unit to allow the two sumps to communicate with each other and eliminate a chokepoint and holdup of water for breaks within the reactor cavity.

The licensee stated that containment spray washdown has a clear path to the sump area. The water will pass through unobstructed grated areas and open stairways.

The licensee stated that the pressurizer cubicle was evaluated for holdup and a holdup of 313 ft^3 was assigned.

The licensee stated that the refueling pool in each unit has a single 4-inch diameter drain in the floor. There are strainers installed over the drains to prevent large pieces of debris from blocking them. The strainers have 200 one-inch diameter holes and are constructed of 10-inch and 6-inch pipe. As part of the modification, the licensee stated that an evaluation was performed that showed water will flow freely through the refueling canal drain. In addition, testing for Turkey Point Nuclear Generating Unit Nos. 3 and 4 demonstrated that the strainer would not become blocked by post-accident debris and that the design containment spray flow rates would not mobilize debris to the extent that the strainer would be blocked. The licensee stated that even if debris collected at the bottom of the strainer that adequate flow to meet assumptions regarding sump level would be maintained.

NRC Staff Conclusion Regarding Upstream Effects

The NRC staff reviewed the licensee's evaluation against the NRC-accepted guidance in the GR/SE 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 the licensee 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 fields 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 described the original strainers and the new strainers that replaced them at Point Beach. The original strainers had an effective area of about 21 ft^2 if fully submerged and had 1/8-inch diameter penetrations. The licensee estimated that the submerged area of the strainers would have been about 11 ft^2 at the start of recirculation. The original strainers were contained within individual trash racks.

The new strainers have an area of about 1,905 ft^2 per train. There are two trains per unit. The new strainers incorporate a uniform flow design and were supplied by Performance Contracting, Inc. Figures 3.j.1-1 and 3.j.1-2 of Enclosure 3 of the July 29, 2022, submittal provide the general arrangement of the strainers. The design flow rate of the strainers is 2,200 gallons per minute (gpm) resulting in an approach velocity of 0.0026 feet per second (ft/sec). When originally installed, each train consisted of 11 modules, but three modules per train were added to increase the strainer area. There are two redundant strainer arrays per unit installed on the containment floor. The strainer arrays are about 3 inches above the containment floor and will be fully submerged before the swapover to sump recirculation. The strainers are constructed of support structures covered by perforated plate. The filtration surface holes are 0.066 inches in diameter. Because the strainers are ruggedly constructed, trash racks were not installed during the modification. The strainers are constructed of stainless steel. The strainer modules are hydraulically connected to each other and to the containment outlets which supply fluid to the RHR pumps. The licensee stated that no further modifications were required to install the strainers.

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 submittal. Based on its review, the NRC staff concludes that 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 because the licensee:

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

- Described modifications necessitated by the sump strainer installation.

3.4.2.8.3 Headloss and Vortexing

The licensee stated that the headloss and vortexing evaluations were revised based on updated testing completed in 2015 and 2016. The NARWHAL analysis does not include headloss calculations, but instead uses a headloss determined from lookup tables based on headlosses that occurred at various debris loads during the testing, with corrections to account for different temperatures and approach velocities.

The licensee stated that for the small break LOCA (SBLOCA) and LBLOCAs at the top of the pressurizer the strainers are submerged with greater than 2 inches of water above the top of the strainers at the time of swapover to recirculation. For the other LBLOCA cases, the strainers are submerged by more than 8 inches of water at swapover. The minimum submergence occurs at swapover, and submergence increases due to continued injection from the RWST. Submergence increases by about one foot due to continued injection from the CSS before the RWST is exhausted. Details of the strainer submergence were provided in section 3.g.1 of

Enclosure 3 of the July 29, 2022, submittal. The values discussed here are based on limiting hand calculations. The NARWHAL calculations used more refined values that account for plant/time specific fluid volumes, flow rates, and temperatures.

During testing the strainer was monitored for vortex formation under clean conditions, with the strainer submerged by 2 inches and the strainer approach velocity at its design value. No vortex formation was observed for this case. The licensee stated that they observed vortex formation twice during testing. During the Full Debris Load Test 1 (FDL1) a vortex was observed when reducing the fluid level prior to the fourth debris addition. At the time the vortex was observed the submergence was about 3 inches. The vortex was eliminated by increasing the submergence to about 4 inches. During the Full Debris Load Test 2 (FDL2) a vortex was observed after the final flow sweep and prior to drain down. In all cases the strainer approach velocity was at the design value or greater. For the second vortex observation the submergence was also less than 4 inches. The licensee stated that it is reasonable to conclude that vortex formation will not occur for the limiting breaks because the minimum submergence for these scenarios is more than double that when the vortexes were observed during testing. Additionally, the licensee stated that for all breaks, the submergence will increase to levels greater than those at which vortex formation was observed before a significant amount of debris will arrive at the strainer.

The licensee stated that strainer testing was conducted at Alden Research Laboratory. Two FDL tests, and confirmatory test (CR), and a thin bed (TB) test were performed. FDL1 targeted a low Cal-Sil, high fiber debris load. FDL2 was performed at a higher Cal-Sil to fiber ratio and the CR targeted the highest Cal-Sil to fiber ratio postulated. The TB test was a typical high particulate to fiber ratio test. The test description provides an overview of the important aspects of the testing and indicates that NRC staff review guidance in RG 1.82, Revision 4, was followed. Specifically, the licensee stated that the “Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Strainer Head Loss and Vortexing,” dated March 2008 (Reference 38), was followed. The test used prototypical plant strainer modules and the test scaling accounted for area reduction due to miscellaneous debris. The licensee stated that debris amounts in the testing were derived from the amounts predicted to reach the strainer based on the debris generation and transport analyses. The testing followed a test for success strategy and ultimately established debris limits for each debris type. These limits were used to determine whether each break could be successfully mitigated considering the deterministic criteria for strainer and ECCS performance. For breaks with computed debris amounts not bounded by the strainer tests, it was postulated that strainer failure and core damage occurred in the risk-informed analyses.

The licensee provided a description of the tests including the debris surrogates and amounts used in the testing. Temp-Mat was not used in the testing since 95 percent of the breaks do not produce Temp-Mat debris. The fibrous materials used in testing were NUKON™ and mineral wool. Particulate debris was Cal-Sil, Asbestos Cal-Sil, qualified and unqualified coatings, degraded epoxy qualified coatings, and latent debris. The debris and surrogates and the preparation of the materials used in testing were described in detail. The licensee stated that no debris settlement occurred during testing. The test setup had sufficient turbulence to ensure that all debris transported to the strainer surface without disturbing the debris bed on the strainer.

For each test the licensee provided the amounts of each debris type used including the debris generated directly from materials in containment, and chemical debris that is predicted to form over time as materials dissolve into the sump pool and later precipitate. The licensee also provided plots that depict the debris additions, flows, and temperatures over the course of the test along with the associated headlosses. The test plots were divided into two figures with one depicting the addition of conventional debris and the other showing the effects of chemical debris addition. Each plot also noted the peak headloss during the test for both conventional and chemical debris. The licensee stated that these peak headlosses were used to derive the headloss lookup tables in NARWHAL. The added chemical headloss was derived by subtracting the peak chemical headloss from the headloss just prior to the initial chemical addition for each test.

The FDL2 test headloss was limiting for both conventional and chemical debris. The conventional headloss was stated to be 1.592 pounds per square inch differential (psid) at 123 degrees Fahrenheit (°F) and a test flow rate of 165.5 gpm. The FDL2 chemical headloss was 2.142 psid at 99.9 °F and a flow rate of 162.7 gpm. FDL2 also had the greatest increase in headloss when chemicals were added and the headloss increased to the final peak headloss of 2.142 psid from a pre-chemical steady state headloss of 1.208 psid at 164.4 gpm and 120.8 °F. The results of all of the tests were used to develop the headloss lookup tables in NARWHAL.

The licensee provided the debris limit failure criteria in table 3.f.5-1 of Enclosure 3 of the July 29, 2022, submittal. The criteria were provided at the test scale, to be multiplied by the test scaling factor to determine the plant-scale debris limits. Table 3.f.10-1 of Enclosure 3 of the July 29, 2022, submittal provides the conventional debris headloss lookup table with the debris loads also at the test scale. All four of the tests are included in the table in increasing order of peak conventional debris headloss. To find a bounding test in NARWHAL for a break, the search starts with the first row (row with the lowest peak conventional headloss). If any debris type amount for the postulated break in the first row is exceeded, the next row is checked. This search is repeated until the final row. If the debris generated by the break and transported to the strainer is not bounded by any tests, the break is considered to cause strainer failure and core damage. For chemical effects, table 3.f.10-2 of Enclosure 3 of the July 29, 2022, submittal provides the lookup values. Any break that results in chemical debris loads greater than 15.65 lbm at test scale is considered to cause strainer failure.

The licensee corrected the headloss values from test conditions to plant flows and temperatures for each time step in the NARWHAL calculation based on the flow sweeps performed during the four tests. During the regulatory audit the NRC staff requested the licensee to provide additional information regarding the scaling of the headloss values. In response to this NRC staff RAI (STSB-RAI-5 and 6), the licensee provided details of the scaling method. The response stated that the methodology was the same as that used in the Vogtle GL 2004-02 submittal (Reference 55), and accounted for the predicted plant conditions by using the headloss test flow sweeps to determine the flow characteristics through the debris bed. The NARWHAL software calculated a scaling factor that accounted for the debris bed flow characteristics, the flow rate, and the temperature of the water at the considered plant conditions. The NRC staff concluded that the scaling was conducted appropriately.

The licensee provided the criteria used to determine strainer failure. The criteria considered were debris limits, strainer structural margin, unsubmerged strainer limits, void fraction limits, flashing failure limits, and pump NPSH limits. The only criteria that resulted in scenario failures in the risk-informed analysis were debris limits, as these were exceeded before other failure criteria were reached. It is noted that the structural differential pressure criterion was originally stated to be 10 ft. However, the NRC staff requested additional information regarding the debris loading used in the structural analysis because it appeared that the debris loading used in the analysis did not bound the greatest potential debris amount that may arrive at the strainer as computed by BADGER and NARWHAL. In its response to this NRC staff RAI (ESEB-RAI-1), the licensee revised its debris load for the strainer and repeated the structural evaluation. The licensee determined that the structural differential pressure limit needed to be reduced to 7 ft. However, this change did not result in additional strainer failures due to exceeding the structural limit. This issue is also discussed in section 3.4.2.8.4 of this SE.

The licensee provided conservatisms associated with the testing in section 3.f.8 of Enclosure 3 of the July 29, 2022, submittal. The licensee stated that the approach velocity used during testing was greater than expected to occur in the plant due to assuming a larger than expected sacrificial area for miscellaneous debris, along with a higher strainer flow rate than expected based on conservative hydraulic flow calculations. In addition, the minimum submergence used during testing was significantly lower than expected after a short time after the initiation of recirculation. These two conservatisms provide margins in the vortex evaluation. For the headloss evaluation the licensee stated that the higher approach velocity along with the use of a greater source term for latent debris would result in greater headloss than would occur in the plant. The licensee also stated that the temperature corrections for headloss were based on the limiting data set from the flow sweeps conducted during all the tests. The licensee also stated that using the NARWHAL strainer test lookup table causes the next peak headloss to be adopted as soon as the prior lookup debris amount is exceeded. Some breaks would actually cause lower headlosses than those computed by the NARWHAL bounding strainer test search algorithm.

The NRC staff recognizes that the assumptions and test practices used by the licensee provide margins that help to ensure that the test results are bounding of the conditions in the plant. In some cases, the margins add significant conservatism.

The licensee stated that the clean strainer headloss (CSHL) (strainer headloss with no debris) was calculated by the strainer vendor using test data from prototypical strainers. The test results were corrected to the plant strainer dimensions and were determined using a flow rate higher than the plant flow rate. The CSHL was stated to be 0.56 ft at 212 °F and a flow rate of 2,200 gpm. The NRC staff asked whether the CSHL was included in the headloss calculations in the risk-informed analysis, particularly for the flashing evaluation during the regulatory audit. In its response to this RAI (STSB-RAI-8), the licensee responded that the NARWHAL flashing calculations used the total strainer headloss which includes CSHL.

The licensee provided a description of the flashing evaluation performed in the NARWHAL software. A flashing failure was assumed to occur if the pressure downstream of the strainer was calculated to be less than the vapor pressure of the fluid at the sump temperature using a conservative containment pressure curve. For temperatures below 212 °F, the containment pressure was assumed to be 14.7 pounds per square inch absolute (psia). For higher temperatures, containment pressure was assumed to be the vapor pressure of the sump fluid. For the flashing evaluation, the licensee credited 2 psi of containment accident pressure for the first 200 minutes of the accident. The NRC staff discussed this credit during the regulatory audit to determine whether the margin above the amount of pressure credited to suppress flashing was adequate. In its response to an NRC staff RAI (STSB-RAI-8), the licensee provided details regarding the available margin in the analysis. The licensee stated that the analysis used a model that was biased to maximize sump temperature and minimize containment pressure. At the times when 2 psi of pressure is credited, the minimum containment pressure remains at least 6 psi above the value used in the analysis. The minimum margin occurs at the start of the scenario and increases with time. The NRC staff reviewed the response and concluded that the credit for the use of containment accident pressure in the flashing analysis contained significant margin and was appropriately limited to a relatively short time during the event recovery.

The licensee evaluated the degasification of fluid as it passes through the strainer and debris bed. For the degasification evaluation no containment pressure was credited. The licensee estimated degasification using the midpoint of the strainer for strainer submergence. The licensee assumed that any gasses liberated due to the pressure drop across the strainer transported to the pump suction without compression or reabsorption. Any break that exceeded 2 percent void fraction at the pump suction was considered to cause strainer failure. For values less than 2 percent the required NPSH was adjusted to account for the voids.

NRC Staff Conclusion Regarding Headloss and Vortexing

The NRC staff reviewed the licensee's evaluation against the NRC-accepted guidance and concludes that the licensee has appropriately determined the headloss across the sump strainer for the debris load tested. The licensee has shown that the potential for formation of a vortex at the strainer does not exist under the plant-specific conditions at Point Beach. The licensee has demonstrated that the strainer will perform acceptably under postulated LOCA conditions, limited by the amount of debris represented in the testing. Therefore, the NRC staff concludes that the licensee's evaluation of headloss and vortexing is acceptable.

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

- Provided the minimum submergence of the strainer under SBLOCA and LBLOCA conditions and noted that recirculation is not within the design basis for a SBLOCA.

- 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 headloss testing for the strainer, including chemical effects, and provided bases for key assumptions.

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

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

- Provided the basis for the strainer design maximum headloss.

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

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

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

- Showed that the sump is fully submerged for all accident scenarios.

- Stated that near-field settling was not credited for the headloss testing.

- Used flow sweep results from testing to scale the results of the headloss tests to actual plant conditions.

- Stated that a small amount of the available containment accident pressure was credited in evaluating whether flashing can occur across the strainer surface and summarized the methodology used to determine the available containment pressure.

3.4.2.8.4 Sump Structural Analysis

The NRC staff review is based on documentation provided by the licensee in the July 29, 2022, submittal and the June 9, 2023, supplement.

The Point Beach structural evaluations were performed using manual calculations and finite element analysis by employing GTSTRUDL and ANSYS software. The licensee performed detail evaluations using applicable rules from American National Standards Institute (ANSI)/ASME B31.1, “Power Piping,” 1998 Edition through 1999 Addenda. The support structures for the strainers were analyzed using:

- American Institute of Steel Construction (AISC) 9th Edition and supplemented by ANSI/AISC N690-1994, “Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities,”

- Structural Engineering Institute (SEI) / American Society of Civil Engineers (ASCE) Standard SEI/ASCE 8-02, “Specification for the Design of Cold-Formed Stainless Steel Structural Members,” and

- American Welding Society (AWS) D1.6, “Structural Welding Code – Stainless Steel”.

Since ANSI/ASME B31.1 does not provide any design guidance, the perforated plates and core tube end cover plate stiffeners were stated to have been modeled in accordance with appendix A, article A-8000 of the ASME Boiler and Pressure Vessel Code.

The licensee provided the load combinations for the design of the strainer, the core tube, strainer pressure retaining plates, strainer structural components, debris interceptors, piping, and piping support structural components. The combinations include consideration of dead load, live load (not applicable during operation), weight of debris, differential pressure, and seismic loads including earthquake induced sloshing.

During the audit, the NRC staff noted that on page E3-134 of Enclosure 3 of the July 29, 2022, submittal, the total debris load per module is stated to be 100 lbm per module. The NRC staff concluded that the assumed total strainer debris amount for the structural analysis is 1,400 lbm since each strainer train has 14 modules. In tables 3.e.6-28 and 3.e.6-30 of Enclosure 3 of the July 29, 2022, submittal (pages E3-67 and 68), the worst debris quantities that do not fail any acceptance criteria are tabulated. It appears that these debris amounts would result in a total debris load of more than 1,400 lbm. Accordingly, the NRC staff issued an RAI (ESEB-RAI-1) that states, in part, “Clarify if 1400 lbm is the limiting debris load on the strainer. If it is not the limiting debris load, update the structural analysis to account for the limiting debris load.…

Update any other responses or evaluations that may be impacted by the change in the structural analysis."

The licensee agreed with the NRC staff's observation that the debris loads in tables 3.e.6-28 and 3.e.6-30 of Enclosure 3 of the July 29, 2022, submittal exceed the analyzed 1,400 lbm debris mass in the strainer structural evaluation. The licensee reevaluated the strainer using an increased debris load of 3,500 lbm on one strainer, which bounds the maximum transported debris mass for all breaks for Unit 1 (2,972 lbm) and Unit 2 (3,271 lbm). The 3,500 lbm bounding debris load is consistent with the independent calculation performed by the NRC staff based on NARWHAL outputs provided by the licensee. The updated evaluation reduced the maximum allowable differential pressure of the strainer from 11.5 ft to 7 ft. The maximum headloss that occurred during strainer testing is less than 7 ft. Therefore, in the NARWHAL model the strainer would be considered to have failed due to excessive debris loading for any case approaching 7 ft of headloss based on the debris load strainer test limits.

The licensee provided the design interaction ratios for the sump strainer components which show, that at all locations, the calculated stress is less than the allowable stress. In its response to ESEB-RAI-1, the licensee evaluated structural components whose interaction ratios would likely be above 1.0 with the updated debris loads. The licensee concluded that the new interaction ratios are still below 1.0 or not adversely impacted due to the differential pressure decrease, and concluded that the components remain qualified.

To address the issue of potential dynamic effects due to a HELB, the licensee stated that the operation of the sump would only be required following a LOCA. Thus, other dynamic effects due to HELB associated with feedwater or main steam piping were not evaluated. The licensee also utilized the relief associated with leak before break (LBB) analyses to eliminate the primary RCS piping, accumulator discharge lines, pressurizer surge line, and high-pressure residual heat removal connections to the RCS. Furthermore, the licensee stated that the replacement screens have been located outside of the reactor coolant loop compartments. These compartment walls are stated to be "thick walled" which would provide protection from potential missiles and dynamic effects.

The licensee stated that a backflushing strategy is not credited in the analysis.

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 deterministically 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 because 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 and demonstrated that code allowable stresses are not exceeded.

- Demonstrated that dynamic effects such as pipe whip, jet impingement, and missile impacts associated with HELBs are not applicable due to the use of LBB methodology.

- Stated that a backflushing strategy is not used at Point Beach.

3.4.2.8.5 Net Positive Suction Head

The licensee performed the NPSH margin evaluation in accordance with the guidance in the GR/SE. The NARWHAL software was used to calculate NPSH margins for each time step of the evolution of a LOCA caused by a pipe break. The calculation considered changes in parameters like sump water level, water temperature, and strainer headloss to make the NPSH margin determination.

The licensee considered the strainer, ECCS, and CS pump flow rates, sump temperatures, and the containment water levels used in its NPSH analysis. The water levels provided are based on hand calculations that contain conservatisms to establish the minimum values for the vortexing evaluation and chemical precipitate hand calculations. The NARWHAL software calculates a water level for each time step in the analysis. The NARWHAL calculation retains some conservatisms, but not all of those included in the hand calculations. The licensee provided the assumptions used in the water level calculation in section 3.g.2 of Enclosure 3 of the July 29, 2022, submittal.

The flow rate used in the analysis is 2,100 gpm which is rounded up from a maximum rate from a hydraulic calculation and is also higher than a procedural limit of 2,000 gpm. The CS and SI pumps take suction from the RHR pump discharge, so the RHR pump is the only one taking suction through the strainer.

The NPSH for a time step is calculated at the maximum calculated temperature within that time step. The sump temperatures used in the NARWHAL analyses are based on the double-ended pump suction break with minimum safeguards to maximize the temperatures. For the NPSH calculation, at temperatures greater than 212 °F the containment pressure is set equal to the vapor pressure at the analyzed temperature.

The licensee stated that the required NPSH values for the pumps were determined by the pump performance curves provided by the vendor and that these values were determined using the Standards of the Hydraulic Institute. The NARWHAL model used a constant NPSH required value determined using a conservative hand calculation. The NPSH required values were corrected for void fractions predicted to be present at the pump suctions under some plant-specific conditions. The licensee stated that the bounding pump NPSH margin was analyzed via a hand calculation, as-built piping information, and PROTO-FLO software. The losses from the piping were calculated considering Darcy's flow and the component headlosses were calculated from standard industry handbooks. NARWHAL uses the piping losses from the limiting case.

The licensee described the response of the system to LBLOCAs and SBLOCAs. The response includes injection and recirculation modes. For a LBLOCA, the RCS depressurizes and the reactor trips, and safety injection is initiated via a safety injection actuation (SIAS) signal. The SIAS signal activates the SI pumps, RHR pumps, and charging pumps, and all the injection valves open. This is the injection mode. All the pumps take suction from the RWST during injection. The ECCS pumps (charging, SI, and RHR) inject to the RCS (cold legs and reactor upper plenum). When the containment pressure reaches the containment spray actuation system setpoint, the CS pumps inject via spray headers to the containment. The CS pumps also take suction from the RWST during injection. When the RWST level reaches a low setpoint, the RHR pumps are realigned to recirculation mode, taking suction from the sump instead of the

RWST. During swapover, the RCS injection and CS are switched to single train operation with the redundant train in standby. During swapover, the CS pumps are realigned to take suction from the RHR pump discharge. The SI pumps are also aligned to take suction from the RHR pump discharge instead of the RWST but are not restarted until after the CS pumps are secured. After about 2 hours following the switchover of the CSS to recirculation, the CS pump is stopped and an SI pump is started to inject to the RCS cold legs to mitigate the potential for boric acid precipitation. The RHR pump continues to inject to the upper plenum.

For the SBLOCA, the licensee stated that RHR (low head) injection into the core is not required at the beginning of the event. The RHR pumps are isolated after it is determined that a SBLOCA has occurred, and the SI pumps remain in service during the injection phase. The atmospheric dump valves are opened to reduce RCS pressure to allow the low head injection within about 6 to 7 hours. The licensee stated that the risk quantification evaluated many equipment configurations and was not limited to the worst single failure. The licensee screened all potential configurations of the RHR, CS, and SI pumps and identified two bounding pump configurations for detailed analysis in NARWHAL. Those pump configurations are (1) no equipment failure and (2) single train failure. The licensee stated that section 4.3.3 of Enclosure 4 of the July 29, 2022, submittal provided the details of the screening process. See section 3.4.2.3 of this SE for the NRC staff evaluation of the screening of pump configurations.

The licensee provided an overview of the methods used to calculate the sump level. As discussed above, these methods were used for the bounding hand calculation which determined levels at the start of RHR recirculation and the end of CS injection. The NARWHAL calculations used inputs appropriate for each time step of the analysis. The sources of inventory along with the mass of water credited from each source were provided. The processes that hold up or remove inventory and decrease the pool level were provided, along with the corresponding reductions in volume. The licensee also provided a summary of the assumptions including those intended to provide margin in the pool level calculation. The licensee also provided a list of structures and components that will displace water resulting in a higher pool level. These structures and components were included in the CAD model that was used to assist in determining the sump level. Minor components were not credited for increasing the level.

The licensee stated that no containment accident pressure was credited for the NPSH evaluation. For sump temperatures equal to or less than 212 °F, a containment pressure of 14.7 psia was assumed. For sump temperatures greater than 212 °F, the containment pressure was assumed equal to the vapor pressure of the fluid.

The licensee provided the NPSH margin results for the limiting fiber case for Unit 1 and the limiting Cal-Sil case for Unit 2. Since only the RHR pumps take suction from the containment sump, the results were provided only for the RHR pumps. Tables 3.g.16-1 and 3.g.16-2 of Enclosure 3 of the July 29, 2022, submittal provide the NARWHAL results for a range of temperatures between 120 °F and 242 °F. The highest temperatures result in the limiting NPSH margins. The lowest margin for Unit 1 is 2.69 ft and the lowest margin for Unit 2 is 2.18 ft.

<u>NRC Staff Conclusion Regarding Net Positive Suction Head</u>

The NRC staff reviewed the licensee’s NPSH evaluation against the NRC-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 the RHR pumps, running in recirculation mode, for all cases that are not considered to result in an increase in plant risk. Therefore, the NRC staff concludes that the licensee's evaluation of NPSH is acceptable.

The NRC staff concludes that the licensee has provided sufficient information as requested by GL 2004-02 and further described in the revised content guide for GL 2004-02 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 for the pumps.

- Described how friction and other flow losses are calculated.

- Described the system response scenarios for LBLOCAs and SBLOCAs.

- Described the operational status for each ECCS and CSS pump before and after the initiation of recirculation.

- Described the limiting 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 how the volumes associated with empty spray pipe, water droplets, condensation and holdup on horizontal and vertical surfaces were accounted for in pool level calculations.

- Provided assumptions (and their bases) for equipment credited to displace water resulting in higher pool level.

- Provided assumptions (and their bases) as to the water sources that are credited to provide pool volume, and the volume 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 that 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 objective of this chemical effects section is to evaluate chemical precipitate effects on the Point Beach sump strainer headloss. The evaluation of chemical effects on the reactor vessel is contained in section 3.4.2.8.8 of this SE.

The overall chemical effects evaluation methodology for Point Beach included:

- Quantification of chemical precipitates using the methodology in WCAP-16530-NP-A

- Introduction of pre-mixed chemical precipitate solution into the strainer test fluid after the formation of a stable particulate and fibrous debris bed

- Application of an aluminum solubility correlation to determine the maximum precipitation temperature at which to apply the chemical effects related strainer headloss

Point Beach specific ECCS sump strainer headloss testing was performed at the Alden Research Laboratory. The test strainer consisted of one prototypical 10-disk strainer module with a flow controlled suction pipe passing through the center (core tube) such that the flow rate through each strainer assembly is uniform. The total area of the test strainer was 136 ft^2, which resulted in a strainer approach velocity of 0.00267 ft/sec. The test flume construction and piping encourage debris transport to the strainer. Both thin bed and full debris load headloss tests were performed by first adding particulate and fibrous debris (NUKON™ fiberglass and mineral wool insulation) to build a representative post-LOCA containment building debris bed on the strainer. Details of the test debris preparation are provided in section 3.4.8.3 of this SE. Once headloss from the particulate and fibrous debris bed had stabilized, pre-mixed chemical precipitate solution was added to the test flume and testing continued until a maximum headloss value had been established. Two full debris load headloss tests and one thin-bed debris headloss test were performed.

Chemical precipitates were prepared according to the WCAP-16530-NP-A instructions. After preparation, precipitate settlement testing was performed to verify that the precipitate settlement met the WCAP-16530-NP-A acceptance criterion. Settlement testing verifies that the chemical precipitate settles in a representative manner. The sodium aluminum silicate precipitate solution was added to the test flume in two batches. A total of 187 gallons and 160 gallons of sodium aluminum silicate precipitate solution were added to the full debris load headloss Test 1 and Test 2, respectively. The thin bed headloss test also added 160 gallons of sodium aluminum silicate precipitate solution. Testing continued until peak headloss had been verified according to the test plan. The headloss tests demonstrated that there was adequate pump NPSH margin for the fiber and chemical effects break size limits assumed in the Point Beach evaluation. Any Point Beach pipe break that transports either more particulate or fibers than was tested (after scaling to the plant strainer surface area), is conservatively assumed to be a strainer failure in the risk quantification. Likewise, any pipe break that results in a greater chemical precipitate quantity amount than was tested (after scaling) is conservatively assumed to be a strainer failure in the risk quantification.

As part of the risk informed evaluation of strainer debris load, the licensee quantified chemical effects at each postulated break location. The amount of chemical precipitate was determined using the WCAP-16530-NP-A methodology with a break-specific quantity of LOCA generated debris. The NRC staff has previously reviewed and approved WCAP-16530-NP-A as one acceptable method to calculate the amount of chemical precipitate and to prepare precipitates for strainer testing. The amount of calculated chemical precipitate for each break was maximized by the licensee by applying conservative plant specific inputs for pH, temperature, and aluminum quantity. The NARWHAL base case used a 10.5 pH to determine chemical release during the time sodium hydroxide solution is added to the containment spray solution during the ECCS injection stage. The sump and recirculation spray assumed a 9.5 pH to determine the chemical release from aluminum and other plant materials.

During the regulatory audit, the NRC staff discussed with the licensee staff the CS pH as a function of time relative to the 10.5 pH value assumed in the base case calculation. In preparation for the audit discussions, the licensee detected an error in the inputs to the NARWHAL analysis. The CS injection phase pH was applied up to 23.8 minutes after the start of the accident, when the RHR pumps switch to recirculation mode. The case considering one functional train of the ECCS holds the pH constant at 10.5 for 64.1 minutes, until the CS recirculation begins. The NRC staff asked the licensee to provide the results from a sensitivity study to evaluate the effects of a higher pH for a longer period. By applying the maximum spray pH for 184 minutes after the accident (much longer than the 64.1 minutes value) the revised risk calculation showed a base case ΔCDF of 2.280×10^{-8}/year for Unit 1 and ΔCDF of 3.944×10^{-8}/year for Unit 2. The NRC staff also asked the licensee to evaluate the sensitivity of a CS pump trip relative to the spray pH, chemical precipitate formation, and the risk quantification. A single CS train failure at the initiation of the accident had the most impact on the duration of the injection phase. The Unit 1 sensitivity case resulted in a ΔCDF of 2.299×10^{-8}/year for Unit 1 and a ΔCDF of 4.020×10^{-8}/year for Unit 2. The sensitivity case ΔCDF results also include the pH error correction previously discussed.

The Point Beach chemical effects approach relies on an aluminum solubility correlation to determine the maximum chemical precipitation temperature for each break. If precipitation is not predicted for a given pipe break before 24 hours, the analysis assumes precipitation occurs at 24 hours. The aluminum solubility correlation used by Point Beach was developed by Argonne National Laboratory (ANL) based on laboratory testing (Reference 56). As a conservative assumption, the licensee assumed a constant 8.25 sump pool pH for aluminum solubility calculation purposes. Assuming a 9.5 pH for aluminum dissolution while simultaneously assuming an 8.25 pH for aluminum solubility provides conservatism that accounts for uncertainty in the ANL aluminum solubility correlation. After development of the ANL solubility correlation, additional chemical effects testing was performed by the PWR Owners Group and documented in WCAP-17788-P, Volume 5. An aluminum precipitation boundary, as a function of pH and temperature, was developed from the WCAP-17788-P test results. For the pH value that the licensee used to determine aluminum solubility, the ANL solubility equation was found to predict precipitation sooner (i.e., was more conservative) than the WCAP-17788-P precipitation boundary function.

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

The NRC staff finds the overall Point Beach chemical effects sump strainer evaluation acceptable for the following reasons. The base model WCAP-16530-NP-A approach was used to determine the amount of precipitate for each break. The licensee made conservative assumptions related to pH, temperature, and aluminum quantity to maximize the amount of predicted precipitate. Strainer chemical effects testing was performed with sodium aluminum silicate prepared according to the WCAP-16530-NP-A instructions and precipitate settling testing met the requirements. WCAP-16530-NP-A was previously reviewed and approved by the NRC staff as an acceptable method to evaluate plant specific chemical effects. The licensee also credited aluminum solubility in its chemical effects evaluation using the ANL equation that was shown to be conservative for the pH assumed in the Point Beach solubility calculations. In addition, the licensee simultaneously analytically assumed that the sump pool was 9.5 pH to determine aluminum corrosion and 8.25 pH to determine aluminum solubility. These conservative pH assumptions account for the uncertainty associated with an aluminum solubility correlation.

The NRC staff concludes that the licensee 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 for the assumed pipe breaks 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 headloss results, or deposit downstream of the sump screen to the extent that LTCC is unacceptably impeded.

- Those pipe breaks that produce greater amounts of chemical precipitates than was found to be acceptable by strainer headloss testing are conservatively assumed to cause strainer failure in the risk quantification.

3.4.2.8.7 Downstream Effects – Components and Systems

The licensee stated that debris effects on components downstream of the sump screen were addressed using NRC-approved methods. The licensee used TR WCAP-16406-P-A, Revision 1, for the analysis including the limitations and conditions provided in the NRC staff SE. The licensee took no exceptions to this methodology. The evaluation determined that no modifications were necessary for components or instrumentation.

The licensee stated that it was assumed that the maximum amount of particulate debris would arrive at and penetrate the strainer. For the erosive wear evaluation, the assumed large debris was depleted over time as recommended by WCAP-16404-P-A. For blockage evaluations, the debris was assumed to be 10 percent larger than the ECCS strainer hole size. The fiber amount used in the evaluation was assumed to be equal to 100 g/FA which is greater than the actual plant value of 86 g/FA (see section 3.4.2.7.2 of this SE for an evaluation of the actual value). The licensee summarized the methodology used to identify the components that required evaluation. The licensee considered the systems and components that perform or support a safety function for ECCS and containment heat removal.

The licensee determined the initial debris concentrations using the methods described in chapter 5 of WCAP-16406-P-A. The concentrations were calculated to be 22.5 parts per million (ppm) for fibrous debris and 1,969.4 ppm for particulate debris.

The licensee stated that both trains of ECCS and CSS were reviewed to ensure that all potential flowpaths that could be affected by debris were evaluated. The licensee evaluated erosive wear, abrasion, and potential blockage that could affect the recirculation mode of ECCS for a mission time of 720 hours. The mission time of the CSS was assumed to be 6 hours. The components evaluated included valves, spray nozzles, orifices, heat exchanger tubes, instrument tubing, and

the ECCS and CSS pumps. All the components passed the acceptance criteria in TR WCAP-16406-P-A, Revision 1.
The licensee stated that no design or operational changes are being implemented to manage downstream effects.

NRC Staff Conclusion Regarding Downstream Effects – Components and Systems

The NRC staff reviewed the evaluation methods and results and finds that the licensee followed the NRC-accepted guidance contained in TR WCAP-16406-P-A, Revision 1, including its associated NRC staff SE. 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 the required mission time after a LOCA.

The NRC staff concludes that the licensee 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 stated that the NRC-approved methods were used for the evaluation without exception.

- Provided a summary and the conclusions of the downstream effects evaluations.

- Stated that no design or operational changes are required as a result of the downstream evaluations.

3.4.2.8.8 Downstream Effects – In-vessel

The licensee stated that it assessed the in-vessel effects using the methodology from TR WCAP-17788-P and followed the latest NRC staff review guidance for in-vessel effects. The NRC staff performed a thorough review of TR WCAP-17788-P and also performed its own analyses and confirmatory calculations to gain insights into the relevant phenomena. The NRC staff determined that TR WCAP-17788-P provides significant insights into the response of PWRs to the effects of debris that transports to the vessel but did not approve it. The NRC staff guidance provides additional information on the staff's views of TR WCAP-17788-P. The amount of fiber assumed to transport to the core was based on strainer penetration test results. The transport of fiber to the core is discussed in section 3.4.2.7.2 of this SE. The maximum amount of fiber calculated to reach the reactor was 85.48 g/FA. The in-vessel evaluation assumed a fibrous debris source term (total on both strainers) of 550 lbm, which bounded the transported fibrous debris of breaks that would not cause strainer failure as computed by BADGER and NARWHAL. The licensee stated that post-accident LTCC will not be challenged by the accumulation of debris within the reactor core for all postulated LOCAs.

The licensee stated that both Point Beach units are 2-loop Westinghouse plants. The Point Beach design is different than most PWRs in that the ECCS injects into the upper plenum of the reactor vessel. Debris bed formation on the top of the core would be subject to disruption from boiling. The NRC guidance for these plants recognizes the advantages of this ECCS design related to the effects of debris. For these plants, the NRC review guidance requires the following to be demonstrated to ensure that in-vessel effects will not adversely affect LTCC:

- Confirm that boric acid precipitation (BAP) mitigation measures are taken prior to 24 hours.

- Confirm that the maximum combined amount of fiber that may arrive at the core inlet and heated core for the hot leg break are below the WCAP-17788 fiber limit.

The licensee stated that the calculated maximum in-vessel fiber load is lower than the WCAP-17788 limit for Westinghouse 2-loop plants. The licensee also stated that measures are taken to prevent BAP two hours after the start of full recirculation, which begins 1.4 to 2.6 hours after the start of the event. Therefore, BAP mitigation measures are taken about 3 to 5 hours after the start of the event and this meets the NRC guidance. The licensee concluded that, since the deterministic in-vessel evaluation shows that Point Beach meets the NRC review guidance, in-vessel failures are not predicted to occur and do not contribute to the risk quantification.

The NRC staff reviewed the information provided by the licensee and determined that the licensee had provided adequate information to demonstrate that cooling of the fuel will be maintained and LTCC will not be affected by the effects of debris passing downstream of the strainer at Point Beach.

<u>NRC Staff Conclusion Regarding Downstream Effects – In-vessel</u>

The NRC staff reviewed the licensee's in-vessel evaluation against the NRC-accepted guidance for the topic. The NRC staff concludes that the licensee has appropriately evaluated the ability of the ECCS to ensure LTCC considering the potential for buildup of debris at the core inlet and inside the reactor vessel. Therefore, the NRC staff concludes that the licensee's evaluation of in-vessel downstream effects is acceptable.

The NRC staff concludes that the licensee has provided sufficient information as requested by NRC staff review guidance for in-vessel effects and further described in TR WCAP-17788-P concerning in-vessel effects, because the licensee demonstrated that:

- The ECCS is realigned to prevent BAP within 24 hours of the start of the event.

- The total amount of fiber reaching the core will not exceed the total in-vessel fiber limit defined by WCAP-17788.

<u>NRC Staff Conclusion Regarding Impact of Debris Submodel</u>

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, headloss and vortexing, sump structural analysis, NPSH, chemical effects, downstream effects – components and systems, and downstream effects – in-vessel were adequately addressed. Based on the evaluations for each of these subsections, the NRC staff concludes that the licensee's impact of debris evaluation is acceptable.

3.4.2.8.9 Submodel Integration

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

The licensee used an integrated model called NARWHAL to calculate conditional probabilities of system failure for each of the two pump state scenarios (one or two functional trains of the ECCS). The model calculated strainer parameters (e.g., buildup of fiber, SAS, particulates, headloss, extent of unsubmerged strainer) 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 postulated break at pipe weld locations. NARWHAL also used a significant number of user-generated inputs to define quantities such as debris transport, dissolution rates and chemical precipitation, strainer headloss, 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, 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 that lead to system failure (strainer failure and core damage) 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 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 for by BADGER), which are ZOI independent (e.g., latent debris and unqualified coatings) and assumed present for all breaks. NARWHAL also computes chemical precipitates considering dissolution rates in debris sources with aluminum. NARWHAL is used to compute the debris distribution as a function of time in the pool, and on the strainers. 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 or pump). To establish whether failure occurs, NARWHAL computes, at each time step, debris buildup at strainers, strainer headloss, 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 headloss 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. Strainer failure is postulated to occur if debris limits established by the ranges of testing are exceeded. In the detailed computations of the two pump configuration scenarios, debris limit exceedance and debris not bounded by strainer tests accounted for all instances of strainer failure.

Failures due to debris accumulating in the core were deterministically shown not to result in a loss of core cooling for all breaks. Therefore, in-vessel failures were not modeled in NARWHAL, and did not result in any increase in core damage frequency.

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 pump configuration analyzed. The main outputs to compute the ΔCDF are the strainer CFP for the three break ranges. The computation of the ΔCDF and ΔLERF was done outside of NARWHAL and the Point Beach PRA.

The NRC staff verified that the licensee's 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 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, and identifying the consistency of failure conditions. The verifications also included independent computations of the CFP for large breaks and verifying that the in-vessel fiber accumulation would not cause core damage. The approach allowed the NRC staff to conclude, with a high level of confidence, that the calculations for debris generation were conducted and applied properly and are, therefore, acceptable.

NRC Staff Conclusion Regarding Submodel Integration

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 licensee's submittal is acceptable for use in an assessment or evaluation model of the effects of debris on the ECCS LTCC function, as required, in part, by 10 CFR 50.46.

3.4.2.9 Systematic Risk Assessment

RG 1.174 states that a licensee may use the decisionmaking principle that proposed increases in risk are small and are consistent with the intent of the NRC's Safety Goal Policy Statement. In Enclosure 4 of the July 29, 2022, submittal, the licensee described the risk-informed basis, including the systematic risk assessment. The licensee accounted for potential failure modes addressed in section 3.4.2.8 of this SE. In the NARWHAL computations, various breaks generated and transported more debris to the strainer than in headloss strainer tests. These breaks were assumed to cause strainer failure and core damage. No other physical failure mode was recorded in the NARWHAL runs. By counting the number of breaks causing strainer failure and the frequency of breaks, the licensee quantified the ΔCDF.

The licensee screened break locations and GL 2004-02 relevant scenarios, which are evaluated in section 3.4.2.1 of this SE. The licensee concluded that the only breaks contributing to the ΔCDF are breaks in Class 1 welds. Debris generating models are evaluated in section 3.4.2.6 of this SE. The licensee considered LOCA break frequencies from NUREG-1829 and related sources to estimate ΔCDF and ΔLERF; the frequency selection approach is evaluated in section 3.4.2.2 of this SE. LOCA break frequencies from an approach considering geometric mean aggregation of expert elicited frequencies, and 40-year plant life frequencies, were used by the licensee in the Baseline computations.

The licensee reported Baseline ΔCDF and ΔLERF values in table 5-4 of Enclosure 4 of the July 29, 2022, submittal. Those values were updated in response to an NRC staff RAI (NCSG-RAI-1) addressing erroneous consideration of a short time for high pH conditions during the injection phase. The updated ΔCDF is equal to 2.280×10^{-8}/year and 3.944×10^{-8}/year for Point Beach, Units 1 and 2, respectively, both of which values are in the RG 1.174 risk Region III. To estimate the ΔLERF, the licensee used the internal events PRA to compute the conditional large early release probability (CLERP), based on a large LOCA accident, equal to

2.33×10^{-3} for both units.[4] Accordingly, the licensee estimated the Baseline ΔLERF to equal 5.311×10^{-11}/year and 9.189×10^{-11}/year for Point Beach, Units 1 and 2, respectively; which values are also in the RG 1.174 risk Region III.

The licensee evaluated the impact of key assumptions and sources of uncertainty in the systematic risk assessment in section 6 of Enclosure 4 of the July 29, 2022, submittal. The licensee examined non-consensus inputs such as the pool volume, flow rate, and input LOCA frequencies, and concluded that those inputs have a negligible effect on the ΔCDF, except for the input LOCA frequency which effect is major, and presented those results in tables 6-2 and 6-3 of Enclosure 4 of the July 29, 2022, submittal and corresponding updated tables in response to NCSG-RAI-1 provided in the June 9, 2023, supplement. The licensee also examined model uncertainty, and considered different models such as assuming that every break is a DEGB, alternative approaches to partition the large LOCA break range, different timesteps of dynamic NARWHAL simulations, and selecting alternative input LOCA frequencies for example based on the NUREG-1829 arithmetic mean aggregation of expert elicited frequencies. Results were summarized in figure 6-1 of Enclosure 4 of the July 29, 2022, submittal and the corresponding updated figure in response to NCSG-RAI-1. The most significant change to the ΔCDF arose by considering the NUREG-1829 arithmetic mean LOCA frequency, followed by the DEGB-only model. In both cases, the ΔCDF is still well within the RG 1.174 risk Region III.

As previously stated, the licensee examined changes in the ΔCDF magnitude when considering arithmetic mean aggregation LOCA break frequencies. The sensitivity analysis revealed that the ΔCDF estimate increased by a factor of 6 with respect to the baseline ΔCDF (the baseline ΔCDF is approximately based on geometric mean aggregation LOCA break frequencies). The NRC staff notes that the licensee identified features of the analysis in support of DID (section 3.2 of this SE) and safety margins (section 3.3 of this SE) which would ensure that inputs, assumptions, and conclusions of the LAR remain valid under uncertainties of the analysis. The NRC staff concluded that the sensitivity analysis does not change the license's conclusion of very low risk (that is, the change lies in the RG 1.174 risk Region III) because conservatisms exist in the licensee's assessment that can offset the impact of single or combined uncertainties, including uncertainties in LOCA break frequencies. The NRC staff views the aggregation method of the LOCA break frequencies as a key assumption and source of uncertainty for the systematic risk assessment. The NRC staff is not generically endorsing any specific aggregation method of LOCA expert-elicited frequencies.

The NRC staff concluded that the licensee addressed dominant uncertainties in sensitivity analyses. The licensee's approach to compute the ΔCDF only includes a few factors. Other factors, such as uncertainty in fiber generated and transported, were addressed for example by using guidance with safety margin in the ZOI and consideration of bounding transport fractions. Other failure modes and secondary break sources were properly screened and excluded. For example, the licensee concluded that fiber buildup in the vessel would not compromise heat dissipation of the core, considering reasonable sources of uncertainty in the analysis (see section 3.n of Enclosure 3 of the July 29, 2022, submittal and response to STSB-RAI-13 in the June 9, 2023, supplement).

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[4] Section 4.8.3 of Enclosure 4 of the July 29, 2022, submittal states that the CLERP is 1.93×10^{-3} for both units; however, the ratio ΔLERF/ΔCDF computed using values in table 5-4 indicates that CLERP is 2.33×10^{-3}. The value CLERP = 2.33×10^{-3} is consistent with entries in table 5-4 and the updated table 5-4 of the June 9, 2023, supplement.

The NRC staff performed verification calculations, considering data provided during an audit. The staff verified, for example, the CFP in tables 5-1 and 5-2 of Enclosure 4 of the July 29, 2022, submittal by counting the total number of breaks causing strainer failure and considering exceedance LOCA frequencies in table 3-1 of Enclosure 4 of the July 29, 2022, submittal. The NRC staff also reproduced the ΔCDF in tables 5-4 and 5-5 of Enclosure 4 of the July 29, 2022, submittal using the CFP, LOCA frequencies in table 3-1 of Enclosure 4 of the July 29, 2022, submittal, and equipment configuration probabilities (one train or two trains of ECCS) in table 4-4 of Enclosure 4 of the July 29, 2022, submittal. The NRC staff performed independent in-vessel mass balance computations, considering information in response to STSB-RAI-13 and derived results practically identical to the licensee results in table 3.n.1-2 of Enclosure 3 of the July 29, 2022, submittal. Based on the above, the NRC staff developed sufficient confidence that the licensee properly computed the ΔCDF and ΔLERF.

<u>NRC Staff Conclusion Regarding Systematic Risk Assessment</u>

The NRC staff evaluated the systematic risk assessment methodology and concluded that it was acceptable because the inputs and assumptions (e.g., initiating event frequencies for critical welds) were derived using state-of-practice data and approaches, scenarios that affect the GL 2004-02 risk assessment were adequately identified and included in the risk evaluation, elements of the risk evaluation were developed in a systematic and acceptable manner, and key assumptions were appropriately considered and described. The NRC staff verified selected computations in support of the ΔCDF. Therefore, the NRC staff concludes that the licensee used a verifiable and robust methodology to calculate the risk attributable to debris.

The licensee properly considered sources of uncertainty in the computation of the ΔCDF and ΔLERF and concluded that a dominant source of uncertainty is the input LOCA frequencies. The NRC staff found that the conclusion that ΔCDF and ΔLERF belong in the RG 1.174 risk Region III is acceptable, based on the licensee's baseline computations, as well as the licensee's factors contributing to safety margins evaluated in section 3.3 of this SE, and factors contributing to DID evaluated in section 3.2 of this SE.

3.4.3 NRC Staff Conclusion Regarding Key Principle 4: Risk Assessment

The licensee used a method to quantify the ΔCDF and ΔLERF, outside of the PRA model, 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 sensitives for each key assumption employing non-consensus approaches. The increase in risk is very small and in accordance with the risk Region III acceptance guidelines defined by RG 1.174. Therefore, key principle 4 of integrated risk-informed decision-making is met.

3.5 <u>Key Principle 5: The Impact of the Proposed Change Should Be Monitored Using Performance Measurement Strategies</u>

RG 1.174, Regulatory Position C.3, "Element 3: Define Implementation and Monitoring Program," states, in part, that:

> The primary goal of Element 3 is to ensure that no unexpected adverse safety degradation occurs because of the change(s) to the licensing basis. The [NRC] staff's principal concern is the possibility that the aggregate impact of changes that affect a large class of SSCs could lead to an unacceptable increase in the

> number of failures from unanticipated degradation, including possible increases in common-cause mechanisms. Therefore, an implementation and monitoring plan should be developed to ensure that the engineering evaluation conducted to examine the impact of the proposed changes continues to reflect the actual reliability and availability of the SSCs evaluated. This ensures that the conclusions drawn from the evaluation remain valid.

In section 9 of Enclosure 4 and in Enclosure 5 of the July 29, 2022, submittal, the licensee stated that it has implemented programs and procedures to evaluate and control potential sources of debris in containment, including TS SRs that require visual inspections of all accessible areas of the containment to check for loose debris, and each containment sump to check for debris. The licensee stated that its design change control procedure includes provisions for managing potential debris sources such as insulation, qualified coatings, exposed aluminum or zinc, and potential effects of post-LOCA debris on recirculation flow paths and downstream components. The licensee's change process explicitly requires that changes that involve any work or activity inside the containment be evaluated for the potential to affect the following:

- High or moderate line break analysis
- Add or remove equipment inside containment
- Quantity of aluminum or zinc inside containment
- Quantity or type of coatings inside containment
- Change or addition of thermal insulation
- Addition of materials that could affect sump performance or cause equipment degradation, affect the design, performance, or operation of pumps

The licensee stated that a 10 CFR 50.59, "Changes, tests, and experiments," screening or evaluation is required to be completed for all design changes, which ensures that new insulation material that may differ from the initial design is evaluated for concerns related to the effects of debris. It also stated that it has implemented procedures to ensure that Service Level 1 protective coatings used inside containment are maintained in compliance with applicable regulatory requirements. The licensee noted that the 10 CFR 50.65 program, also known as the Maintenance Rule program, includes performance monitoring of functions associated with ECCS and CSS. The inclusion of ECCS and CSS into the Maintenance Rule program and the assessment of acceptable system performance provide continued assurance of the availability for performance of the required functions.

The licensee also stated that periodic updates to the risk-informed analysis will be performed to capture the effects of changes that may affect the results of the analysis. The review and updates will include plant changes, procedure changes, or new information regarding the risk-informed analysis. The licensee stated that as PRA inputs related to reliability data, unavailability data, initiating event frequency data, human reliability data, and other similar inputs are reviewed, the base PRA model for Point Beach will be updated to be consistent with the as-built, as-operated plant. The base PRA model will be updated within two refueling cycles to capture the effects of any plant changes, procedure changes, or new information on the risk-informed analysis and to confirm that the acceptance criteria are still maintained (see section 11 of Enclosure 4 of the July 29, 2022, submittal).

Procedures for operators to detect and respond to sump blockage issues related to GL 2004-02 have been developed and incorporated into the plant emergency operating procedures and emergency contingency actions.

The Point Beach quality assurance (QA) program is implemented and controlled in accordance with the Quality Assurance Topical Report (QATR) and is applicable to SSCs to an extent consistent with their importance to safety. The QA program complies with the requirements of 10 CFR Part 50, Appendix B and other program commitments as appropriate.

The NRC staff reviewed the licensee's information and concludes that the licensee's monitoring program is acceptable because it is consistent with the guidance in RG 1.174, Regulatory Position C.3.

3.5.1 NRC Staff Conclusion Regarding Key Principle 5: Performance Monitoring

The Point Beach Maintenance Rule program includes performance monitoring of functions associated with ECCS and CSS, including sump recirculation. Technical Requirements implemented by Point Beach procedures require visual inspections of all accessible areas of the containment to check for loose debris, and each containment sump to check for debris. Licensed operators are trained on indications of and actions in response to sump blockage issues related to GL 2004-02, and performance is evaluated during training scenarios designed to simulate plant responses. The effect of changes incorporated into the at-power PRA model of record are periodically assessed to ensure that the results of the analysis used to close GL 2004-02 remain within the aggregate baseline acceptance criteria in RG 1.174. Therefore, key principle 5 of performance monitoring is met.

3.6 Technical Evaluation Conclusion

The proposed licensing basis changes referenced in section 2.3 of this SE describe how the effects of debris are evaluated and how these effects are incorporated into other calculations like NPSH margin for the pumps taking suction from the ECCS sumps. The NRC staff concluded that the description of the key methods, identified in the FSAR markup, used in the risk-informed evaluation were acceptable. Any change in these methods is to be evaluated by the licensee to determine whether a departure from an approved methodology as described in the FSAR would arise.

The FSAR markup accurately describes the various aspects of the risk-informed analysis and provides references to important guidance documents and calculations in a new section A.8. Sections 6 and 14 of the FSAR were revised to refer to the information in the new section. Table A.8-1 of Enclosure 2 of the July 29, 2022, submittal (Enclosure 2, attachment 1, page A.8-9 of the FSAR markup) contains information regarding the analyzed debris limits for various debris types included in the design basis headloss testing for Point Beach. The NRC staff concluded that the proposed FSAR changes are acceptable.

The licensee's submittal contains information regarding reporting and corrective actions associated with the changes made via the LAR. Section 12 of Enclosure 4 of the July 29, 2022, submittal describes the criteria for reporting in the event that the ΔCDF and ΔLERF associated with the GL 2004-02 risk-informed analyses are found to increase.

Related to the debris amounts contained in table A.8-1 of Enclosure 2 of the July 29, 2022, submittal, the submittal contained information regarding how compliance with the risk-informed analysis would be maintained in attachment 1 to Enclosure 4 of the July 29, 2022, submittal. The information was provided for information only, but the NRC considers the licensee's methods for compliance determination to be important. During the regulatory audit, the NRC staff discussed the methodology that will be used to determine compliance with the regulations and operability associated with the debris effects on the ECCS performance during LTCC. In response to an NRC staff RAI (STSB-RAI-11), the licensee clarified that the debris limits in table A.8-1 are based on a single strainer. In addition, the licensee stated that chemical debris limits are not included in this table because they are calculated based on other debris in containment. The licensee provided the actions taken to track the amount of aluminum in containment because aluminum is the major contributor to chemical debris. The NRC staff also requested clarification on the assumptions for debris margin in tables 1-1 and 1-2 in Enclosure 4 of the July 29, 2022, submittal. In its response to this RAI (STSB-RAI-12), the licensee stated that the margins are based on debris accumulation on a single strainer. Additionally, the licensee provided figures in the June 9, 2023, supplement that illustrate how the margins are calculated for one-train and two-train operation. Based on the clarifications, the NRC staff concluded that the margins provided in the LAR are acceptable.

As discussed in this SE, the NRC staff reviewed the July 29, 2022, submittal and the June 9, 2023, supplement. The NRC staff finds that the information provided by the licensee demonstrates that there is reasonable assurance that debris will not adversely affect LTCC at Point Beach. The NRC staff's conclusions are described in detail in the previous sections of this SE. The NRC staff finds that the licensee adequately addressed each technical area of GL 2004-02 using deterministic methods. For a limited set of scenarios that do not meet the acceptance criteria using the accepted deterministic methods, the licensee demonstrated that the cumulative risk of these scenarios is very small. To complete its evaluation of these low-risk scenarios, the licensee demonstrated that the five key principles of RG 1.174, Revision 3, were met. Therefore, the NRC staff concludes that the licensee's risk-informed methodology for assessing the effects of debris on LTCC at Point Beach (including submodels and integration of those submodels) demonstrates that the requirements of 10 CFR 50.46 for LTCC will not be adversely affected by debris.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, on July 31, 2023, the Wisconsin State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR part 20 or change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding in the *Federal Register* on October 4, 2022 (87 FR 60216) that the amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

7.0 REFERENCES

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SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 273 AND 275 REGARDING REVISING LICENSING BASIS TO ADDRESS GENERIC SAFETY ISSUE 191 AND TO RESPOND TO GENERIC LETTER 2004-02 USING A RISK-INFORMED APPROACH (EPID L-2022-LLA-0106) DATED AUGUST 28, 2023

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