FORT CALHOUN STATION

PILOT PLANT AUDIT REPORT

ANALYSES REQUIRED FOR THE RESPONSE TO GENERIC LETTER 2004-02 AND GSI-191 RESOLUTION

Table of Contents

Acronym List

Alion	Alion Science and Technology
CFD	computational fluid dynamics
CDI	Continuum Dynamics, Inc.
CS	containment spray
ECCS	emergency core cooling system
FCS	Fort Calhoun Station
GE	GE Energy
GL	generic letter
GR	Guidance Report
GSI	Generic Safety Issue
HPSI	high-pressure safety injection
ICM	interim compensatory measure
LBLOCA	large break loss of coolant accident
LOCA	loss-of-coolant accident
OPPD	Omaha Public Power District
NEI	Nuclear Energy Institute
NPSH	net positive suction head
NPSHA	net positive suction head available
NPSHR	net positive suction head required
NRC	Nuclear Regulatory Commission
RMI	reflective metal insulation
SBLOCA	small break loss of coolant accident
SE	Safety Evaluation
ZOI	zone of influence

1.0 BACKGROUND

1.1 Introduction

During the 2004 Sump Performance Workshop in December 2004, the Nuclear Energy Institute (NEI) proposed that the NRC conduct pilot plant reviews of licensee submittals in response to Generic Letter (GL) 2004-02 [1], “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accident at Pressurized-Water Reactors.” After the workshop, representatives of several utilities contacted the NRC to express interest in volunteering for the program. The pilot plant review is an effort between the industry and the NRC to effectively evaluate the implementation of the NEI’s sump evaluation methodology on a plant-specific basis at selected pilot plants. The review is intended to enable the NRC staff to (1) identify and resolve potential issues that arise when employing the approved methodology, (2) improve the review process, and (3) focus the audit process planned to examine industry-wide implementation of Generic Letter (GL) 2004-02.

The pilot review effort was intended to yield benefits to both the NRC and industry. For the NRC these included:

> Lessons learned during the pilots will guide the agency in determining the NRC staff and contractor resources needed for the future reviews, audits, and/or inspections. The lessons learned will also allow the NRC staff to focus the audits to minimize the impact on NRC staff and licensee resources.
>
> The NRC staff can identify early during the resolution of Generic Safety Issue (GSI)-191 [9], issues that need to be further addressed and clarified in the safety evaluation.
>
> Feedback from the lessons learned could help the NRC staff enhance research and testing programs and inspection activities.

Benefits envisioned for the industry include:

> Feedback from the lessons learned will enable the industry to make high-quality submittals related to GL 2004-02. High-quality submittals will need less NRC staff resources to review and take regulatory actions.
>
> NRC staff clarifications regarding GL 2004-02, if needed, early during the resolution of GSI-191.
>
> Reduced need for audits because the NRC staff will become familiar with the approach taken for the resolution of GSI-191.
>
> Fee waiver for NRC review of license amendment requests related to GSI-191 for pilot plants.
>
> Enable the industry to focus and prioritize the open items impacting the GSI-191 resolution.

Omaha Public Power District (OPPD), the licensee, volunteered Fort Calhoun Station (FCS) to be a pilot plant. The FCS licensee supplied documentation relating to its proposed analyses and planned changes. On July 14, 2005, the NRC commenced the audit with a meeting between the NRC staff and the licensee with its vendors to go over the supplied documents and references.

The following NRC staff, licensee and contractors, and NRC consultants were major participants in the pilot audit:

Name	Affiliation
Industry	
Michael Friedman	OPPD
Joe Gasper	OPPD
Larry Fleischer	GE
Andrew Kaufman	CDI
Stephen Moen	GE
Peter Mast	Alion
Phil Roberts	GE
Jan Bostelman	Alion
W. R. Peebles	Sargent & Lundy
Alan Bilanin	CDI
NRC	
David Solorio	NRC
Mark Kowal	NRC
Ralph Architzel	NRC
Thomas Hafera	NRC
Hanry Wagage	NRC
Shanlai Lu	NRC
John Lehning	NRC
Steven Unikewicz	NRC
Ruth Reyes-Maldonado	NRC
John Hannon	NRC
Alan Wang	NRC
NRC Contractors	
Clint Shaffer	ARES Corporation
Bruce Letellier	Los Alamos National Laboratory
Dave DeCroix	Los Alamos National Laboratory
Luke Bartlein	Los Alamos National Laboratory

Following this meeting the NRC staff continued in-depth review of the FCS documentation. Several telephone conferences were held to clarify the methods and calculations and to communicate NRC concerns and questions. These included February 22, 2005, to plan for the audit, and followup telephone conferences on April 20, 21, and 27, May 4 and May 23, and June 6, 15, and 16, 2005.

FCS sponsored head loss testing performed by GE Energy (GE) at the facilities of Continuum Dynamics, Inc. (CDI) in Ewing, NJ. On August 28-September 1, 2005, the NRC staff and its contractor visited the head loss test facility as part of the pilot audit.

Based on the status of the licensee's design and engineering effort, and considering available staff resources, several areas that were examined to some extent during the previous pilot audit at Crystal River Unit 3 [70] were not included within the scope of the FCS pilot audit. The NRC staff did not review the chemical effects area as part of the FCS pilot audit effort. The NRC and licensee staffs, however, participated in a phone call to discuss the FCS GL 2002-04 response related to chemical effects. The NRC only performed a limited review of upstream effects, latent debris and downstream effects.

Prior to the audit on March 29-30, 2005 several NRC staff visited Fort Calhoun Station to obtain information in advance of the pilot audit. During this site visit the staff examined the current sump and screen, and containment layout related to debris generation and transport. The licensee's plans for future modifications and vendor selection were discussed. The staff also visually observed the accessible containment coatings which appeared to be in good condition. No large areas of new degradation were noted, nor was any widespread areas of flaking or peeling paint noted. There were some areas where the original top coat had been scraped away leaving only the primer, but flaking and peeling at the edges was not visually apparent. Coating rework/repair was apparent in some significant areas, especially in the lowest level of containment (within the flood zone), in upper portions of containment and on piping. During this visit documentation detailing coatings walkdowns, pictures and maintenance performed over the last few years was made available to the staff. The NRC staff also discussed flaking paint on the reactor vessel, which was a primary reason for the visit. The affected area was localized, contained in the reactor vessel annulus, insulation covers the coating and a torturous path exists in order for the coating to reach the sump. FCS was taking steps to remove the loose flaking coating on and below the vessel. The NRC staff did not otherwise review the coatings area as part of the FCS pilot audit effort.

On September 27, 2005, the audit team held a telephone conference near the end of the audit to go over issues identified during the pilot audit. During the course of the audit the team identified issues related to the licensee's implementation and plans that need to be assessed as part of the licensee's closure of GL 2004-02. These are discussed throughout the audit report and were communicated to the licensee during the audit meetings and telephone conferences. The NRC staff stated that they expected that these issues would be considered and addressed by the licensee during its implementation of changes associated with GSI-191. Additionally, the team noted that selected issues would be followed during NRC staff review of the GL 2002-04 responses. The final discussion was held in a meeting on September 30, 2005 at NRC Offices.

1.2 Bulletin 2003-01 Response

The OPPD FCS response to Bulletin 2003-01 [10], "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-water Reactors," documented in OPPD letters dated August 8, 2003, and June 11, 2004, were responsive to and met the intent of Bulletin 2003-01. The licensee addressed the six interim compensatory measure (ICM) categories of Bulletin 2003-01 as follows:

ICM Category #1: Operator Training on Indications of and Responses to Sump Clogging

The licensee provided additional operator training on the identification of degraded sump conditions during a loss of coolant accident (LOCA) (e.g. through monitoring for pump cavitation, erratic pump discharge flow, erratic pump current, etc.), and developed procedural guidance and training for responding to an emergency sump clogging event. This procedural guidance included, but was not limited to:

(1) when appropriate, the establishment of a “limiting water volume” which fills the containment to at least the top of the reactor coolant system hot legs to allow for long term vessel and core cooling; and

(2) during long-term emergency sump recirculation, switching back and forth between drawing suction from the containment sump and the safety injection refueling water storage tank to allow time for debris settling from the sump screens.

ICM Category #2: Procedural Modifications, If Appropriate, That Would Delay the Switch over to Containment Sump Recirculation (e.g., shutting down redundant pumps that are not necessary to provide required flows to cool the containment and reactor core, and operating the containment building spray system intermittently).

The licensee decided, when appropriate for prevailing plant conditions, to secure one or two containment spray pumps prior to switch over to sump recirculation, to secure one high pressure safety injection (HPSI) pump prior to switch over to sump recirculation, and to provide more aggressive cooldown and depressurization following a small break loss of coolant accident (SBLOCA) to potentially enter shutdown cooling mode directly without entering emergency sump recirculation mode.

ICM Category #3: Ensuring That Alternative Water Sources are Available to Refill the Refueling Water Storage Tank or Otherwise Provide Inventory to Inject Into the Reactor Core and Spray Into the Containment Atmosphere.

The licensee developed procedures for commencing safety injection refueling water storage tank refill directly upon switch over to sump recirculation (rather than waiting for a sump clogging event to occur), and to inject more than one volume from a refilled safety injection refueling water storage tank or bypass the safety injection refueling water storage tank if necessary to cool the core during a sump clogging event.

ICM Category #4: More Aggressive Containment Cleaning and Increased Foreign Material Controls.

The licensee provided more aggressive containment cleaning and foreign materials/debris controls for outages and containment entries at power.

ICM Category #5: Ensuring Containment Drainage Paths are Unblocked

The licensee performs walkdowns of the containment during refueling outages and verifies that drainage paths are unblocked, with Condition Reports generated as appropriate.

ICM Category #6: Ensuring Sump Screens are Free of Adverse Gaps and Breaches

The licensee verifies that containment sump screens are free of adverse gaps and breaches during refueling outages.

1.3 Generic Letter 2004-02 90-Day Response

OPPD submitted their 90-day response on March 4, 2004 in accordance with the schedule in (GL) 2004-02. The licensee stated in their 90-day response they will perform their sump performance evaluations using methods intended to conform to NEI Guidance Report (GR) 04-07 [71], "Pressurized Water Reactor Sump Performance Evaluation Methodology," Volume 1 - "Pressurized Water Reactor Sump Performance Evaluation Methodology," and Volume 2 - "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC GL 2004-02, Revision 0, December 6, 2004" [2]. The licensee will use computational fluid dynamics modeling to perform recirculation transport assessments. The licensee also stated that the chemical precipitation effects of debris accumulation will not be addressed until current NRC testing is completed, the data has been appropriately evaluated with respect to plant-specific conditions, and an approved methodology for application of chemical effects is established.

The scheduled completion date for the licensee's evaluation was stated to be September 1, 2005, with the final head loss analysis not completed until December 31, 2007.

The licensee reported that they completed the FCS containment walkdown surveillance during the Fall 2003 refueling outage.

After reviewing OPPD's response, the NRC staff issued a request for additional information on June 3, 2005, concerning the licensee's approach to addressing chemical effects. The request for additional information explained that the delay in addressing chemical effects was contrary to the NRC staff's position that there are sufficient bases to address sump vulnerability to chemical effects and that the September response would be incomplete if chemical effects were not addressed. The licensee was requested to discuss their plans and schedule for evaluating chemical effects.

In a letter dated August 1, 2005, the licensee responded to the request for additional information stating that their initial approach would be to utilize margin (bump-up factor) specifically allocated for chemical effects to address the impact of chemical effects. The bump-up factor would be validated by testing and/or evaluation data and adjusted as necessary.

The licensee's September 2005 response to GL 2004-02 was not reviewed during the pilot audit.

2.0 DESCRIPTION OF PLANNED CHANGES

The new containment sump suction strainer design proposed by the licensee has not been finalized. However, the planned design consists of an array of large passive stacked disk strainers with a total area of approximately 2800 ft^2 [51]. From the licensee's August 31, 2005 Generic Letter 2004-02 response:

> OPPD plans to replace existing sump strainers and install new strainers to assure compliance with applicable regulatory requirements. The existing screen area is approximately 28 square feet per train or 56 square feet total. The total screen area of the replacement and new strainers has not yet been finalized but is expected to be 2800 square feet, or greater. The screens for the strainers are planned to have round openings with a proposed maximum diameter of 3/32 inch. The design of these strainers will minimize head loss and bypass due to the small hole size.
>
> Evaluation of Other Potential Modifications - OPPD plans to replace the Steam Generators and Pressurizer at FCS during the 2006 [refueling outage]. At that time, the Calcium Silicate (with and without asbestos) and Tempmat (high density fiberglass) insulation on these vessels are planned to be replaced with Reflective Metal Insulation (RMI). The portions of the interconnecting piping insulation are also planned to be replaced with low density fiberglass. Other modifications currently planned or under consideration include installation of new drain caps for the Reactor Pressure Vessel (RPV) cavity drain and refueling cavity drain, RPV flange seal ring spacers and removing the autostart feature from one of the three containment spray pumps.

3.0 BASELINE EVALUATION AND ANALYTICAL REFINEMENTS

3.1 Break Selection

The objective of the break selection process is to identify the break size and location that presents the greatest challenge to post-accident sump performance. Sections 3.3 and 4.2.1 of the NEI GR [71] and the associated staff safety evaluation (SE) approving the methodology [2] provide the criteria to be considered in the overall break selection process in order to identify the limiting break. In general, the criterion used to define the most challenging break is the estimated head loss across the sump screen. Therefore, all phases of the accident scenario must be considered for each postulated break location, including debris generation, debris transport, debris accumulation and sump screen head loss. Two attributes of break selection which are emphasized in the approved evaluation methodology and can contribute to head loss are (1) the maximum amount of debris transported to the screen, and (2) the worst combinations of debris mixes that are transported to the screen. Additionally, the approved methodology states that breaks should be considered in each high-pressure system that relies on recirculation, including secondary side system piping.

Section 4.2.1 of the SE discusses a proposed refinement which would allow considering only break locations which are consistent with Branch Technical Position MEB 3-1, “Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment,” and Standard Review Plan (NUREG-0800) Section 3.6.2, “Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping.” The proposed application of Branch Technical Position MEB 3-1 for pressurized water reactor sump analyses was intended to focus attention on high-stress and fatigue break locations such as at the terminal ends of a piping system and intermediate pipe ruptures at locations of high stress. However, as discussed in Section 4.2.1 of the SE, the staff rejected the application of this proposed refinement for pressurized water reactor sump analyses.

FCS Calculation No. FC06985 [49], “Fort Calhoun Station Debris Generation Post LOCA,” [49], documents the assumptions and methodology the licensee applied as part of the overall break selection process, and to determine the limiting break for FCS.

Alion Science and Technology (Alion) Calculation No. ALION-REP-OPPD2522-002, “Fort Calhoun Station LOCA Pool CFD Transport Analysis,” [66] provides assumptions and methods applied for debris transport calculations. This calculation supports and informs the limiting break selection process. The licensee did not provide head loss calculations for the staff’s review.

NRC Staff Evaluation:

The NRC staff reviewed the licensee’s overall break selection process and the methodology applied to identify the limiting break. Specifically, the staff reviewed Calculation FC06985 [49] against the approved methodology documented in Sections 3.3 and 4.2.1 of the staff’s SE [2]. The staff concluded that the licensee’s break selection evaluation is partially complete with an approach that appears reasonable and that the evaluation was generally performed in a manner consistent with the approved SE methodology. Deviations from the NRC staff approved methodology were judged by the staff to be acceptable based on the technical basis provided by the licensee (with 2 issues identified). A detailed discussion is provided here.

The NRC staff’s review found that the licensee evaluated a number of break locations and piping systems, and considered breaks in each high-pressure system that relies on recirculation to mitigate the event. As a minimum, the following break locations were considered:

Break No. 1 - Breaks in the reactor coolant system with the largest potential for debris.

Break No. 2 - Large breaks with two or more different types of debris.

Break No. 3 - Breaks with the most direct path to the sump.

Break No. 4 - Large breaks with the largest potential particulate debris to insulation ratio by weight.

Break No. 5 - Breaks that generate a “thin-bed” - high particulate with 1/8" fiber bed.

This spectrum of breaks is consistent with that recommended in the approved methodology and is also consistent with NRC staff regulatory position 1.3.2.3 of Regulatory Guide 1.82, Revision 3 [16].

The licensee considered breaks in the primary reactor coolant system piping, secondary system piping (main steam and feedwater), and other high energy line break piping systems having the potential to rely on emergency core cooling system (ECCS) sump recirculation. The licensee reviewed accident analysis scenarios, emergency operating procedures and the FCS Updated Safety Analyses Report [63] to determine which accidents and piping systems may require sump recirculation. The licensee concluded that large break loss of coolant accidents (LBLOCAs) and certain SBLOCA's would require sump recirculation. For SBLOCA's, only piping 3-inches in diameter and larger was considered because breaks sizes smaller than this would not require sump recirculation. The NRC staff finds this to be reasonable. The approved sump evaluation methodology states that breaks less than 2-inches in diameter need not be considered (discussed in Section 3.3.4.1 of the NRC staff's SE).

The licensee also evaluated a 14-inch diameter pipe break in the reactor coolant system for debris generation, consistent with the alternate evaluation methodology in the NRC staff's GSI-191 SE (Section 6). The licensee performed this alternate break evaluation as a contingency should it be needed; however, the licensee is not crediting this alternate break size at this time. The NRC staff did not review the details of this 14-inch break calculation because it is bounded by the LBLOCA analysis and the licensee is not applying this methodology at this time.

With respect to secondary side piping system breaks, the licensee's evaluation concluded that these breaks do not rely on sump recirculation. The licensee's conclusions are based on a review of the FCS Updated Safety Analyses Report accident analysis scenarios. The licensee provided technical justification to demonstrate that ECCS sump recirculation is not needed for these secondary side piping system breaks for either long term decay heat removal or containment heat removal functions. The NRC staff reviewed the FCS Updated Safety Analyses Report [63] Sections 6 and 14, and verified that a LBLOCA and certain SBLOCA's are the only breaks scenarios which may require sump recirculation.

The licensee identified and evaluated three primary breaks as being potentially limiting:

- reactor coolant system hot-leg double ended guillotine break in steam generator bay A (high fiber scenario)
- reactor coolant system break at discharge of reactor coolant pump RC-3D in steam generator bay B (high particulate scenario)
- 3" spray control valve piping from the reactor coolant system to the spray control valves (break with the most direct path to sump)

The licensee determined that these large reactor coolant system breaks in the steam generator bays generate the largest amount of debris, and also the worst combination of debris. The licensee evaluated the accident scenarios for the LBLOCA cases considering debris generation, debris transport and debris accumulation. The licensee had not yet completed the sump screen head loss analyses. The licensee will be installing new insulation on the replacement steam generators and pressurizer in the upcoming 2006 refueling outage and is interested in the effects of selected insulation replacement on the ECCS sump screen sizing. As such, the licensee evaluated two debris load cases for the large break scenarios.

Section 3.3.5 of the NRC staff's SE describes a systematic approach to the break selection process which includes beginning the evaluation at an initial location along a pipe, generally a terminal end, and stepping along in equal increments (5-ft increments) considering breaks at each sequential location. The NRC staff expected this type of approach to be documented in the licensees' calculations, however, the FCS break selection process did not apply such a systematic approach. The licensee stated in their response to an NRC staff request for additional information that the concept of equal increments is only a reminder to be systematic and thorough, and the NRC staff agrees, as this was stated in the NRC staff's SE. For the limiting LBLOCA cases, the exact location of a break along the length of the piping has very little impact on most debris sources because the size of the zone of influence (ZOI) applied captures the entire compartment. For calcium silicate destruction, however, the licensee did apply such a systematic approach because a smaller ZOI was used (5.5 length/diameter). For calcium silicate, the licensee evaluated a ZOI sphere based on the size of the hot leg piping and moved this ZOI along the reactor coolant system hot and cold legs within the steam generator bays to determine the limiting location. Based on the application of the systematic approach for calcium silicate and the magnitude of the ZOI applied for the other debris sources, which capture the entire compartment, the NRC staff agrees that performing the analysis by considering 5-ft increments is not necessary.

The licensee also addressed breaks that could generate a "thin-bed." Many possible high energy line breaks at FCS can be postulated where a small quantity of fibrous debris is generated and transported to the sump along with pool transport and washdown of particulate debris potentially resulting in the thin-bed effect. Rather than analyzing specific high energy line breaks, the licensee stated that the thin-bed effect will be specifically addressed in the FCS head-loss calculations. However, as discussed further in the Head Loss section of this audit report, the licensee did not provide these head loss calculations to the NRC staff for review during the audit.

Furthermore, the licensee did not provide head loss calculations for any of the breaks considered. Head loss calculations are needed to determine the limiting break location (all phases of accident analysis must be considered). This item is also discussed in the Head Loss section of this audit report. The NRC staff notes that the head loss calculations have not been provided or reviewed as part of this audit.

In accordance with the NRC staff's SE, the licensee did not apply the proposed refinement of Section 4.2.1 of the SE which would allow considering only break locations which are consistent with Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," and Standard Review Plan (NUREG-0800) Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping." The NRC staff finds this to be appropriate.

In conclusion, based on the above discussions, the NRC staff finds that the licensee's evaluation of break selection is partially complete with an approach that appears reasonable. The evaluation was generally performed in a manner consistent with the approved SE methodology. Deviations from the NRC staff approved methodology were judged by the NRC staff to be acceptable based on the technical basis provided by the licensee (with head loss remaining to be considered). The licensee has not provided the NRC staff with any head loss calculations for the breaks analyzed, and the licensee has not provided the NRC staff with any of the head loss calculations to demonstrate that the thin-bed effect can be accommodated by the sump screen design.

The criterion used to define the most challenging break is the estimated head loss across the sump screen. Therefore, all phases of the accident scenario must be considered for each postulated break location, including debris generation, debris transport, debris accumulation and sump screen head loss. As such, should the licensee need to revise other phases of their analyses (e.g., debris transport - transport fractions), then they should also reassess the limiting break location to verify that it remains limiting.

3.2 Debris Generation/Zone of Influence

The objective of the debris generation/zone of influence (ZOI) process is to determine, for each postulated break location, the zone within which the break jet forces will be sufficient to damage materials and create debris, the amount of debris generated by the break jet forces and the need to determine the characteristics of the debris. Sections 3.4 and 4.2.2 of the NEI GR [71] and the approved NRC staff evaluation methodology [2] provide the methodology to be considered in the ZOI and debris generation process. In general, the baseline methodology for ZOI is based on the American National Standards Institute ANSI/ANS 58.2 1988 standard [17]. The baseline methodology incorporates ZOI's based on material damage pressures and the corresponding volume-equivalent spherical ZOI radii. Debris generation is then calculated based on the amount of materials within the ZOI. Other sections of the GR and SE provide guidance on particle size distribution and characterization of the debris types.

Section 4.2.2 of the SE discusses proposed refinements which would allow application of debris-specific spherical destruction zones (ZOI's), and direct jet impingement modeling. As discussed in Section 4.2.2 of the SE, the NRC staff accepted the application of these proposed refinements for pressurized water reactor sump analyses.

FCS Calculation No. FC06985 [49], "Fort Calhoun Station Debris Generation Post LOCA," [65], documents the assumptions and methodology the licensee applied to determine the ZOI and debris generated for each postulated break.

NRC Staff Evaluation:

The NRC staff reviewed the licensee's ZOI and debris generation evaluations and the methodology applied. Specifically, the NRC staff reviewed FCS Calculation No. FC06985 [49] against the approved methodology documented in Sections 3.4 and 4.2.2 of the staff's SE. The NRC staff concluded that the licensee's evaluation is partially complete with an approach that appears reasonable, with several incomplete items regarding ZOI and debris generation. The evaluation was performed in a manner consistent with the approved SE methodology. A detailed discussion is provided below.

The first step in evaluating the debris generated following a high energy line break is to determine the appropriate ZOI for each high energy line break considered. Once the ZOI is established, potential debris sources within the ZOI can be identified and the quantity of each debris source can be calculated. The types and locations of potential debris sources (insulations, coatings, dirt/dust, fire barrier materials, etc.) can be identified using plant-specific drawings, specifications, walkdown reports or other such reference materials.

The NRC staff's review concluded that the licensee correctly applied NRC staff approved methodology to determine the ZOI to be used for debris generation. The licensee applied material-specific damage pressures and corresponding ZOI radius/break diameter ratios as shown in Table 3-2 of the NRC staff SE, with one exception. Table 2-1 of FCS Calculation No. FC06985 [49] lists the ZOI values used by the licensee. The only values in this table that are relevant to the debris types in FCS are coatings, Calcium Silicate, TempMat and Nukon®. The NRC staff's review identified that the licensee's calculation is not clear with respect to ZOI application and requested that the licensee clarify the methodology being applied and the ZOI values being used for each debris material applicable for FCS. In their request for additional information response, the licensee stated that they will revise the debris generation calculation to clearly state the debris generation and ZOI methodologies. Additionally, the licensee stated in a request for additional information response that they will also revise the calculation to incorporate consistent treatment of pipe diameter for ZOI calculations.

The ZOI exception involves application of a coatings ZOI value of 4.0 radius/break diameter ratio. The NRC staff SE recommended a value of 10.0 for coatings. A coatings ZOI of 4.0 is based on an unverified FCS assumption, pending further industry testing. At this time, the licensee is not crediting the 4.0 length/diameter coatings ZOI and their analyses and head loss testing are based on a 10 length/diameter coatings ZOI. Although the NRC staff did not review the coatings area as part of this pilot audit, the NRC staff notes that technical justification for using a 4.0 length/diameter ZOI is necessary, should the licensee decide to apply this value.

The NRC staff review identified that the Transco Products Inc. reflective metal insulation (RMI) debris size distribution is not discussed in the debris generation calculation. The licensee's request for additional information response stated that this is developed and discussed in the debris transport calculation, and states that the quantity of RMI foils that arrive at the sump screen is overly conservative as the ZOI of the RMI was assumed to be that for Nukon® (17 length/diameter). Table 2-1 of the FCS debris generation calculation shows Transco Products Inc. RMI as 1.3 length/diameter. The NRC staff agrees that application of the Nukon® 17 length/diameter is very conservative and is acceptable. However, should the licensee choose to revise this analysis in the future (possibly to remove some of this conservatism), the NRC staff notes that the approved length/diameter value for Transco Products Inc. RMI in the SE Table 3-2 is 2.0, and this value is based on the RMI material tested by the Boiling Water Reactor Owners' Group. In this case, the licensee should confirm that the SE value of 2.0 is applicable for the FCS RMI.

The FCS walkdown report divided the containment into 17 separate walkdown areas that were defined by elevation and system. The NRC staff requested additional information regarding partial area applications. For example, if the ZOI for a particular break did not encompass an entire area, was partial destruction within an area considered, or was all material within the area boundary assumed to be destroyed? If partial destruction was assumed, the request for additional information requested the licensee to discuss how the debris generation volumes were determined. The licensee's request for additional information response stated that if the entire compartment's debris inventory was not used, then the detailed calculations are described in the text of the debris generation calculation and shown in Attachment B. The NRC staff was looking for a qualitative discussion on the method applied to determine partial area debris volumes. Attachment B provides the final result for insulation data on specific pipes in certain areas. The NRC staff was not able to reach conclusions by reviewing Attachment B, as it is difficult to for the NRC staff to determine whether or not pipes and insulation materials should be included within a certain ZOI by reviewing Attachment B. The licensee may consider providing such a qualitative description of the methodology applied in the calculation.

The licensee also credited walls as robust barriers in its ZOI evaluation. Section 3.4.2.3 of the NRC staff SE states that, "[f]or the baseline analysis, the NRC staff position is that licensees should center the spherical ZOI at the location of the break. Where the sphere extends beyond robust barriers, such as walls, or encompasses large components, such as tanks and steam generators, the extended volume can be truncated. This truncation should be conservatively determined with a goal of +0/-25 percent accuracy, and only large obstructions should be considered. The shadow surfaces of components should be included in this analysis and not truncated, as debris generation tests clearly demonstrate damage to shadowed surfaces of components."

In conclusion, based on the above discussions, the NRC staff finds that the licensee's evaluation of debris generation/zone of influence is partially complete with an approach that appears reasonable. The evaluation was generally performed in a manner consistent with the approved SE methodology. Deviations from the NRC staff approved methodology were judged by the NRC staff to be acceptable based on the technical basis provided by the licensee (with several issues identified). The licensee stated in a request for additional information response that they will revise the debris generation calculation to clearly state the debris generation and ZOI methodologies applied and incorporate consistent treatment of pipe diameter for ZOI calculations. Additionally, the licensee should include a qualitative description of the methodology applied regarding partial walkdown area applications in the calculation.

3.3 Debris Characteristics

The debris source inventory for insulation materials in the FCS containment is obtained from the Debris Source Inventory Walkdown Master Spreadsheets and from the proposed configurations for the steam generator replacement project. The types, quantities and locations of insulation debris within the containment are documented in these spreadsheets. These spreadsheets were included as Attachment A in FCS Calculation No. FC06985 [49], "Fort Calhoun Station Debris Generation Post LOCA". The insulation debris sources within the containment are obtained from the FCS confirmatory walkdowns performed in the 2003 refueling outage. The results of the walkdowns identified the following types of insulation within the containment:

- Nukon® Fiberglass
- Reflective Metallic Insulation (RMI)
- Calcium Silicate & Calcium Silicate with Asbestos
- Foam Rubber

- Temp-Mat
- Cerafiber®
- HD Supertemp
- Fiberglass (low density filter media)

The licensee incorporated several debris characteristics assumptions which are documented in FCS calculation No. FC06985 [49]. These are summarized here:

- The RMI insulation used for the steam generator and pressurizer replacement will be 5-inch, 15 foil Transco Products Inc. RMI. Only the equipment insulation was assumed to be replaced, no lines connected to the equipment. The licensee stated that this is conservative because it maximizes the fibrous debris quantity, the debris type that most significantly challenges the sump head loss. The licensee also stated that this assumption will be updated post-modification to reflect the as-built configuration.

- For JPS GLASS's Temp-Mat (non woven glass fiber) it is assumed that the size distribution is no worse than Nukon®. The licensee states that this is conservative because the destruction pressure for Temp-Mat is higher than Nukon®

- The source for the entire filter media fiberglass source is Flanders Brand Nuclear Grade High Efficiency Particulate Air Filter media. This is assumed to be low density fiberglass.

- The pressurizer currently has two layers of Temp-Mat. It is assumed that the RMI replacement will only consist of one layer on the hemi head. The licensee also stated that this assumption will be updated post-modification to reflect the as-built configuration.

- Calcium Silicate with asbestos is assumed to behave the same as Calcium Silicate without asbestos in regards to destruction pressure. The licensee states that this is an unverified assumption.

NRC Staff Evaluation:

The NRC staff reviewed the licensee's debris characteristics evaluations and the methodology applied. Specifically, the NRC staff reviewed FCS Calculation No. FC06985 [49] against the approved methodology documented in Sections 3.4.3 and 4.2.2.2 of the NRC staff's SE. The NRC staff concluded that the licensee's evaluation is partially complete with an approach that appears reasonable, with five open items regarding debris characteristics. A detailed discussion is provided here.

The NRC staff's review found that the licensee's debris size distribution assumptions are consistent with the recommended values in the NRC staff's SE [2]. The size distribution of debris created in the ZOI is classified into two categories: small fine pieces and large pieces. For Calcium Silicate & Calcium Silicate with Asbestos, 100% small fines and 0% large pieces are assumed, consistent with the NRC staff SE. For Temp-Mat, the licensee assumed that the size distribution is the same as for Nukon® and that this is conservative because Temp-Mat has a higher destruction pressure. This assumption is consistent with the NRC staff's SE, which recommends 60% small fines and 40% large pieces. For Cerafiber®, the licensee applies a value of 100% small fines and 0% large pieces, consistent with that for generic fiberglass in the NRC staff's SE. This is conservative because it results in 100% transport to the sump screen. For HD Supertemp and fiberglass (low density filter media), the licensee determined that this material does not exist within the ZOI, and therefore is not considered in the head loss analysis. For foam rubber insulation, the licensee concluded that this material does not need to be considered in the head loss analysis. Foam rubber insulation will float in water, and the sump screen is submerged and well below the water line during sump recirculation. Therefore, foam rubber insulation will not impact sump screen head loss.

With respect to the Nukon® size distribution applied in FCS analyses, the NRC staff identified that the FCS calculation is not consistent, as one area showed that NRC staff's SE values are used, while a different area stated that the SE values are not applied. The licensee stated in a request for additional information response that the SE values for Nukon® are applied in the FCS calculations. The licensee agreed to revise FCS Calculation No. FC06985 [49] to reflect this. The SE values for Nukon® are 60% small fines and 40% large pieces.

RMI size distribution is not discussed in the FCS debris generation calculation (FC06985 [49]). The licensee responded to a NRC staff request for additional information on this topic by stating that this is developed and discussed in the FCS transport calculation, and states that the basis of the debris size distribution is a conservative application of the distribution provided in NUREG/CR-6808 [18]. Additionally, the licensee states that the quantity of RMI foils that reach the sump screen is conservative because the ZOI applied for the RMI was assumed to be that of Nukon® (see Section 3.2, Debris Generation/Zone of Influence, for NRC staff discussion of RMI ZOI). The licensee stated that the RMI debris size distribution discussion would be relocated to the FCS debris generation calculation (FC06985 [49]).

Section 3.3 of FCS debris generation calculation (FC06985 [49]) lists debris generation and characteristics assumptions which include notes stating that the calculation will be updated post-modification to reflect the sump screen as-built configuration. The NRC staff requested additional information regarding the administrative controls in place at FCS that ensure that these assumptions and the calculations will indeed be updated once the screen design modification is finalized. The licensee's request for additional information response stated that as-built requirements are tracked through PED-QP-3.8 as part of the modification process, and that all insulation removal and replacement done inside of containment during the 2006 refueling outage is being done under EC 32583. The as-built configuration will be captured through the modification process and the close-out of this EC package.

The licensee's evaluation assumes that calcium silicate insulation with asbestos behaves the same as calcium silicate insulation without asbestos. This is presented as an unverified assumption in the FCS debris generation calculation (FC06985 [49]). The NRC staff requested additional information regarding this assumption because debris generation, transport and head loss may differ considerably between different calcium silicate materials. The licensee did provide the NRC staff with a scanning electron microscope analysis of the FCS calcium silicate materials and also compared these to other calcium silicate vendor products. The licensee provided this information to the NRC staff during the conclusion of the NRC staff's audit review, so the NRC staff did not have any interactions with the licensee regarding this analysis. Given more time for this pilot audit, the NRC staff would have pursued this question with the licensee.

The licensee is assuming that it is conservative to apply Nukon® parameters and size distribution for THERMAL-WRAP® insulation. In response to a NRC staff request for additional information, the licensee stated that these materials are essentially identical, and stated, “This evidence will be provided in a follow-up correspondence.” At the time of the conclusion of this pilot audit, this information has not been provided to the NRC staff. It should be noted that THERMAL-WRAP® is not listed on page 25 or in Tables 4.3-1 and 4.3-1 as a current FCS insulation debris source term because THERMAL-WRAP® is being considered as a replacement insulation for the steam generator and pressurizer replacement project. The NRC staff expects to receive this information and may review this as part of the GL closeout process.

The NRC staff also identified an issue with Temp-Mat insulation material based on the fact that high density fibrous insulation debris does not sink as readily as does low density debris. This concern is illustrated in NUREG/CR-2982, Table 4.1 where mineral wool shreds did not sink in 10 days in 120°F water. If Temp-Mat were to remain buoyant for even tens of minutes, it could easily float transport to the recirculation sumps and subsequently sink directly onto the screens during the 30 day mission time. The NRC staff requested additional information from the licensee regarding this issue. This issue is being treated and addressed by the NRC staff as a transport issue, and this is discussed in the Transport Section of this report.

Based on the above discussions, the NRC staff finds that the licensee’s evaluation of debris characteristics is partially complete with an approach that appears reasonable. The evaluation was generally performed in a manner consistent with the approved SE methodology. Deviations from the NRC staff approved methodology were judged by the NRC staff to be acceptable based on the technical basis provided by the licensee (with 5 open items). The licensee agreed to revise FCS Calculation No. FC06985 [49] to accurately reflect the Nukon® size distribution applied, the licensee stated that the RMI debris size distribution discussion would be relocated to the FCS debris generation calculation (FC06985 [49]), given more time for this audit, the NRC staff would have pursued questions regarding the licensee’s calcium silicate scanning electron microscope analysis, the NRC staff expects to receive information regarding Nukon® vs. THERMAL-WRAP® material and may review this as part of the GL closeout process, and the NRC staff identified an issue regarding float-transport of Temp-Mat insulation material (addressed in the Transport Section of this report).

3.4 Latent Debris

The objective of the latent debris evaluation process is to provide a reasonable approximation of the amount and types of latent debris existing within the containment, and its potential impact on sump screen head loss. Section 3.5 of the NEI GR [71] and the approved staff Safety Evaluation [2] provide the methodology to be considered for evaluation of latent debris. In general, the GR outlined the following five generic activities to quantify and characterize latent debris inside containment: (1) Estimate horizontal and vertical surface area; (2) Evaluate resident debris buildup; (3) Define debris characteristics; (4) Determine fractional surface area susceptible to debris buildup; and (5) Calculate total quantity and composition of debris. The SE provided alternate guidance for sampling techniques and analysis to allow licensee’s to more accurately determine the impact of latent debris on sump-screen performance.

Section 4.2.3 of the SE did not provide any additional refinements to the methodology for evaluation of Latent Debris.

The FCS Debris Generation Post-LOCA evaluation performed by Alion [65] section 4.5 and request for additional information response's dated September 8, 2005[66], provided the information regarding the assumptions and methodology the licensee applied to determine the amount, type, and impact on sump screen head loss from latent debris.

NRC Staff Evaluation:

A limited review was performed by the NRC staff to assess the licensee's latent debris evaluations and the methodology applied against the approved methodology documented in Section 3.5 of the SE. Documentation provided by FCS indicated that the initial evaluation for latent debris was performed in a manner significantly different than the SE approved methodology, and appeared to lack the detail and rigor that would be required to provide a reasonable evaluation. The NRC staff issued requests for additional information to the licensee to provide additional information to allow a more detailed assessment. In the response to the request for additional information, the licensee outlined a follow up evaluation that was performed during a later refueling outage. This follow-up evaluation and analysis provided a much more rigorous evaluation, and is similar to the methodology outlined in the SE.

Additionally, the licensee applied the use of a more conservative value as the design input for the ECCS sump screen. The NRC staff has concluded that this evaluation by the licensee is considered acceptable based on the follow up evaluation and the margins of conservatism applied in the final evaluation of sump screen performance.

3.5 Debris Transport

Debris transport analysis estimates the fraction of debris that would be transported from debris sources within containment to the sump suction strainers. Debris transport would occur through four major modes:

6. blowdown transport, which is the vertical and horizontal transport of debris throughout containment by the break jet,
7. washdown spray transport, which is the downward transport of debris by the containment sprays and break flow,
8. pool fill transport, which is the horizontal transport of debris by break flow and containment spray flow to areas that may be active or inactive during recirculation flow, and
9. recirculation transport, which is the horizontal transport of debris from the active portions of the containment pool to the suction strainers through pool flows induced by the operation of the emergency core coolant system and containment spray system.

Through the blowdown mode, some debris would be transported throughout the lower and upper containment. Through the washdown mode, a fraction of the debris in the upper containment would be washed down to the containment floor. Through the pool fill-up mode, debris on the containment floor would be scattered around, and some debris would be washed into inactive volumes, such as the reactor cavity, which do not participate in recirculation. Thus, any debris that enters an inactive pool would tend to stay there, rather than being transported onto the suction strainers. Through the recirculation mode, a fraction of the debris in the active portions of the containment pool would be transported to the suction strainers.

The licensee analyzed debris transport in "Fort Calhoun Station Debris Transport Post-LOCA," ALION-REP-OPPD2522-003, Revision 0 [65]. The licensee stated that the debris transport methodology used for FCS is based on the methodology reported in NUREG/CR-6808 [18], NUREG/CR-6762 [11,13,14] , NUREG/CR-6224 [15], and the NRC Safety Evaluation on the NEI Guidance Report (GR) [2]. The licensee used logic trees to calculate the transport of debris from the ZOI to the sump strainers by the blowdown, washdown, pool fill, and recirculation modes [65]. The licensee's logic trees were based upon the generic model recommended by the GR, dividing debris into two size distributions (fines and large pieces) [65]. One modification to the licensee's logic tree structure was an additional branch for large pieces of frangible debris to allow modeling of erosion [65]. Despite this modification to the formal logic tree structure, the licensee did not actually model debris erosion explicitly, since a branching fraction of zero was apparently assumed for erosion in all cases [65]. The licensee proceeded to quantify the logic trees to calculate transport of the following types of debris: fibrous insulation, RMI, calcium silicate insulation, and qualified coatings [65].

The licensee considered the FCS containment to be highly compartmentalized, in accordance with the guidance set out in the GR [71]. The licensee stated that each steam generator is separated by block walls in addition to the reactor coolant pumps [65]. As a result, a break in one steam generator bay does not communicate with the other bay [65]. Further, there are no openings in the bioshield wall other than pipe and instrument penetrations [65].

The licensee identified two limiting, postulated breaks for the sump strainer debris source term in ALION-REP-OPPD2522-003 [65]. The LBLOCA was analyzed as being the break that produces the largest and most varied quantity of debris, while the SBLOCA was analyzed as being the break with the most direct path to the sump [65]. An LBLOCA on the reactor coolant system hot leg in Steam Generator B Bay was chosen as the limiting break because it was found to produce the largest amount of fibrous and calcium silicate debris [65]. A break in Steam Generator B Bay was also found to be the LBLOCA having the shortest transport path to the recirculation sump [65]. The licensee stated that a detailed transport analysis was not performed for the SBLOCA case, since, for this case, it was assumed that 100% of the debris generated would transport to the sump [65].

The licensee's overall approach followed guidance from the GR, using assumptions from both the baseline methodology and analytical refinements. In particular, the licensee applied an analytical refinement to analyze debris recirculation transport, using FLOW-3D, a computational fluid dynamics (CFD) computer code, to calculate the fraction of debris that transports from the containment pool to the sump strainers during recirculation. As discussed subsequently, the staff also noted that the licensee took several exceptions to the approved guidance. Specific issues associated with the license's debris transport analysis identified by the NRC staff during the course of the audit review are described below in the following sections of this report.

3.5.1 Debris Settling During Large-Scale Head Loss Testing

The results of the licensee's debris transport analysis estimate the fraction of debris that would reach the suction strainers. During the large-scale suction strainer head loss testing, this quantity of debris analyzed as reaching the suction strainers was added to the test tank; however, due to settling in the tank, a significant fraction of the added debris did not actually reach the test strainer. Visual observations by the NRC staff during a visit to the test facility estimated that, during one test, approximately two-thirds of the debris added to the tank had settled onto the floor, and that approximately one-third had reached the test strainer.

NRC Staff Evaluation:

The NRC staff considers the crediting of debris settling during large-scale head loss testing to be an exception to the baseline methodology, since the debris transport fractions in the NRC staff's SE were accepted according to the assumption that the transported debris would actually reach the sump strainers, rather than settling in the vicinity of the sump [2]. Further, the NRC staff considers this exception to be quite significant, in that the settling of such a large fraction of the transported debris would have a major impact in reducing the head loss due to debris calculated from the test. As such, the NRC staff expects that a robust technical basis be presented to demonstrate that the transport properties of the debris and the flows simulated in the test tank are conservative with respect to the actual properties expected for the FCS containment pool.

Near the end of the NRC staff's pilot audit review, the licensee provided additional information to support the crediting of debris settling by explaining that the large-scale head loss test conditions are prototypic of actual plant conditions [69]. The licensee stated that the FCS module tests were performed with actual plant water levels and scaled flow rates [69]. Accordingly, the licensee stated that no scaling is necessary to interpret the debris settling during the head loss testing [69]. The licensee further stated that the approach velocity to the FCS strainers is very low based on the strainer design, and that debris that could not be carried to the test strainer based on the test velocities would settle onto the floor near the strainer [69]. The licensee stated that at FCS, debris would have a similar opportunity to settle between containment doorways and the strainers [69]. The licensee also stated a conservatism regarding the head loss testing, which was that containment sprays in the plant would enter the pool from above, thereby tending to encourage debris settling [69].

After reviewing the information supplied by the licensee, the NRC staff was not convinced that sufficient justification had been provided to support crediting of debris settling during large-scale strainer head loss testing. First, the justification presented by the licensee, summarized above, is entirely qualitative. In light of the significant effect of debris settling, the NRC staff believes a more detailed, quantitative analysis may be warranted. Without quantification, the NRC staff cannot judge the relative importance of various test conditions that may effect debris transport in both conservative and nonconservative ways.

Second, the NRC staff considers large-scale head loss testing to be in equal measure a test of debris transport. However, while the test flow conditions may be representative of the actual plant flow conditions at the strainer surface, the NRC staff has not seen sufficient evidence that the flow conditions in large-scale test facilities farther away from the strainer are similarly being designed and monitored to assure a conservative modeling of actual plant flow conditions. Effects related to both the scaling of the strainer module and differences in the geometry of the test tank and the FCS containment building could significantly influence debris settling. The licensee has not presented an adequate technical basis, such as a comparison of computational fluid dynamics calculations and/or test measurements, to demonstrate that the debris transport conditions realized during large-scale testing are generally representative of those in the FCS containment pool. Thus, while large-scale head loss testing that accurately models the suction strainer surface may provide a technical basis for demonstrating that debris cannot be lifted up onto the vertical surface of a strainer, without properly modeling and controlling the upstream flow conditions simultaneously, such tests should not be assumed to provide sufficient basis to credit debris settling on a tank or flume floor upstream of the strainer.

Third, in the course of the audit review, the NRC staff was not provided sufficient information to perform a full verification of the adequacy of the licensee's test procedures, including the procedures for debris procurement, preparation, and addition to the test tank. For instance, the NRC staff lacked sufficient information to verify the adequacy of the debris transport parameters (e.g., densities) of certain surrogate test materials procured by the licensee, including calcium silicate, Kaowool, and silicon carbide. Concerning the addition of debris to the test tank, the NRC staff noted that, although the licensee's procedures specified that debris should be added uniformly around the strainer, nonuniformities were apparent. Further, although debris quantities appeared to be scaled based upon head loss concerns, the NRC staff was not clear as to whether scaling had been applied to the debris addition to the tank to achieve representative debris concentrations. Given these uncertainties regarding nonuniformity and concentration, the NRC staff could not confirm, for instance, that coating particulate was being added to the test tank in a manner that would preclude artificial bulk agglomeration and settling. The NRC staff notes that variations in the test procedures may have a significant influence upon the test results.

In summary, the NRC staff's review of this issue concluded that, while some degree of debris settling during large-scale testing may be realistic, the licensee and its vendors had not provided sufficient information in the course of the pilot audit review to justify crediting the substantial effect of this phenomenon. Further details regarding the NRC staff's concerns are provided in Appendix II to this audit report.

3.5.2 Debris Transport Properties

The licensee provided information and assumptions concerning the transport properties for sources of debris at FCS in calculation ALION-REP-OPPD2522-002 [66].

NRC Staff Evaluation:

As described below, the NRC staff identified several issues with respect to the debris transport properties assumed by the licensee.

3.5.2.1 Nukon® Tumbling Velocity

In calculation ALION-REP-OPPD2522-002 [66], the licensee stated in Section 2.1 that one of the debris types applicable to FCS is Nukon® low-density fiberglass insulation. In Table 3.9.1 of this calculation, the licensee listed the incipient tumbling velocity used for low-density fiberglass. In response to a question from the NRC staff, the licensee stated that this value was based on the incipient transport velocity for another type of low-density fiberglass. The licensee further stated that the two types of low-density fiberglass insulation are generally accepted to have identical characteristics [68].

The NRC staff identified the application of the given value for the incipient tumbling velocity of Nukon® low-density fiberglass as being a nonconservative assumption, since Table 3.1 of NUREG/CR-6772 specifically reports that the incipient tumbling velocity for Nukon® is 0.12 ft/s [50]. As such, the NRC staff considers the licensee's technical basis for applying data from another type of low-density fiberglass to Nukon® insulation to be insubstantial and continues to expect that licensees use the most pertinent data available in conducting sump performance analyses.

3.5.2.2 Temp-Mat Tumbling Velocity

In Table 3.9.1 of the calculation ALION-REP-OPPD2522-002 [66], the licensee listed the incipient tumbling velocities that were used for Temp-Mat high-density fiberglass insulation. In response to a question from the NRC staff, the licensee noted that, although a specific brand was not identified, the test material used to derive the incipient tumbling velocities in Table 3.9.1 was high-density fiberglass insulation [68].

The staff's concern with the incipient test velocities used by the licensee is that the pieces of debris tested to support these velocity results had not been properly prepared. The high-density fiberglass had been prepared by first slicing the insulation into squares having an area of 1 in^2 and then further slicing these squares into 1/8-inch-thick layers. These flat, layered pieces tended to lie flat along the floor during the transport tests, which reduced the projected area perpendicular to the flow direction, thereby generating lower drag forces. As a result, transporting these pieces necessitated a higher velocity than would be expected for shredded high-density fiberglass debris that may be more prototypical of debris generated during a LOCA. Therefore, the NRC staff considers the transport velocities used by the licensee for Temp-Mat to be nonconservative. A similar issue was identified and discussed in the staff's pilot audit report for Crystal River, Unit 3 [70].

3.5.2.3 Temp-Mat and Cerafiber® Flotation

The NRC staff noted that the capability of a piece of debris to float for a significant period of time following an accident could substantially affect its transportability within the containment sump pool. Depending on plant-specific conditions, floating pieces of debris may transport to the suction strainers much more readily than pieces that have sunk to the containment floor. For this reason, the NRC staff questioned whether the licensee has evidence that Temp-Mat and Cerafiber® debris of any size will become saturated with water such that it will sink and subsequently transport by sliding along the floor, rather than by floating upon the containment pool surface.

The staff noted that the licensee assumed 60% of the Temp-Mat within the ZOI was fragmented into fines, and 40% was considered to be broken into large pieces. As large pieces of Temp-Mat have not been tested extensively to determine their transport properties, it is not clear whether they will remain floating for a significant length of time. The licensee further noted that at FCS, locked screen doors are located at each steam generator bay entrance [67]. Until the pool level reaches approximately six inches, the licensee stated that some pieces of Temp-Mat could float past the doors [67]. During the remainder of pool filling and recirculation, however, the licensee stated that the screen mesh would prevent large pieces of floating debris from transporting from these bays to the sump strainers, thereby limiting the quantity of floating debris available for transport [67]. The licensee further stated that its analysis addresses some fraction of transport for large pieces of fibrous debris [67].

The NRC staff considers the licensee's response to contain insufficient information to support a firm conclusion on this issue. First, although it would appear to be on the order of several minutes, the duration during which large pieces of Temp-Mat or Cerafiber® may float out of the steam generator bay is not clear to the NRC staff. Although this period seems relatively short, the NRC staff notes that the pool flow within the steam generator bay during this time apparently would be oriented preferentially outward. The flow rates, velocities, and other pertinent characteristics of the flow during this initial period have not been provided to the NRC staff. The staff is concerned that if Temp-Mat were to remain floating for even half an hour, it would seemingly be capable of transporting to the sump strainers by floating. The NRC staff recognizes that the licensee's analysis does account for a limited amount of transport of large pieces of fibrous debris. Further, the NRC staff recognizes that, once a thin-bed has formed, the transport of additional fibrous debris to the suction strainers may often have a significantly smaller incremental effect on head loss than the transport of additional particulate material and, depending upon the assumptions made, could even provide benefit. Additional information regarding the information gaps noted above would help resolve uncertainties regarding the transportability of Temp-Mat and Cerafiber® via floatation.

3.5.2.4 Paint Chip Tumbling Velocity

In Table 3.9.1 of the calculation ALION-REP-OPPD2522-002 [66], the licensee listed the incipient tumbling velocity for epoxy paint chips of the size distribution expected for FCS. The NRC staff noted that a reference to support the assumption velocity was not provided, and, in a question to the licensee, further stated that this velocity apparently corresponds to paint chips of a greater density and thickness than the chips assumed in the analysis. The NRC staff noted that paint chips that are thinner or less dense than those tested, such as the epoxy chips analyzed by the licensee, would likely tumble at a lower velocity than that assumed in Table 3.9.1.

The licensee responded that the incipient tumbling velocity used was based upon existing data [68]. Since the completion of the transport analysis for FCS, the vendor that performed the transport analysis has formulated a theoretical derivation to extrapolate the existing data to paint chips of other densities and thicknesses than those that were tested [68]. The licensee further stated that any follow-up calculations performed for FCS will account for the paint chips' actual density and thickness [68].

While the NRC staff believes that specifically accounting for density and thickness would improve the accuracy of the licensee's paint chip transport analysis, the licensee did not provide the revised methodology and its basis for NRC staff review. Therefore, the NRC staff could not determine whether the revised methodology would be conservative. Further, the NRC staff notes that existing FCS calculations do not employ the revised methodology, and would thus appear to remain nonconservative in their treatment of paint chip transport.

3.5.3 Erosion of Fibrous Debris

In Section 4.1.3 of the calculation ALION-REP-OPPD2522-003 [65], the licensee examined the degree to which large pieces of fibrous debris would be subject to erosion. The erosion rate assumed by the licensee was significantly lower than the rate provided in Appendix III of the NRC staff's SE [2 ,67]. The licensee justified the reduced erosion rate assumed in its analysis by pointing out that the pool depth at FCS would be greater than the depth of the test tank used to generate the data supporting the SE's erosion rate [65]. The NRC staff questioned this justification, since the erosion rate would seem to be correlated more with pool turbulence than depth, and the licensee had not provided a comparison of the turbulence in the two pools.

The licensee responded to the NRC staff's question by stating that it is not considering performing an analysis to compare the turbulence of the test conditions to that of the FCS containment pool [67]. The licensee stated that a 10% allowance for erosion of fibrous debris is already accounted for by the GR's baseline methodology, applicable to Nukon®, Temp-Mat, and Cerafiber® [67]. The licensee stated that the NRC staff's SE concluded that this assumption was acceptable [67]. The licensee further stated that the containment sprays will only be operated for a minimum of 5 hours prior to termination [67].

Although the licensee is correct in noting that the baseline methodology accepted by the NRC staff did not specify that the erosion of fibrous debris should be treated explicitly, the NRC staff's acceptance of the baseline methodology's assumptions was based upon these assumptions being "taken as a whole." The NRC staff further noted that "an analytical refinement that decreases the degree of conservatism on a particular assumption has the potential to alter the package balance such that the degree of conservatism is reduced or even reversed to nonconservatism." As a result of the licensee's use of an approved analytical refinement to the baseline methodology (i.e., using CFD analysis for debris transport), as well as exceptions to the approved methodology, including the crediting of debris settling during large-scale head loss testing and the assumption that unqualified coatings outside the ZOI fail as chips rather than 10-μm particles, the NRC staff was not convinced that the licensee's evaluation had maintained the baseline methodology's inherent conservatism. Further, it is not clear to the NRC staff that the 10% allowance for erosion noted by the licensee is fully substantiated by the GR and Section 3.4.3.2 of the SE, which discusses debris characterization. In light of the exceptions taken to the baseline methodology and the incompleteness of the licensee's response to the NRC staff's question above, the NRC staff concluded that sufficient justification had not been provided in the course of the pilot audit review to confirm the acceptability of the licensee's treatment of the erosion of fibrous debris.

3.5.4 Computational Fluid Dynamics Analysis

For the purpose of the NRC staff's audit review, the licensee provided the FLOW-3D input deck that was used to perform the CFD analysis of debris transport in the FCS containment pool. The licensee's CFD analysis was performed to predict the steady-state containment pool velocity field that would exist during the recirculation phase of a postulated LBLOCA. The main objective of the NRC staff's review was to evaluate the adequacy of the physical assumptions and numerical approaches used in the CFD analysis to ensure that it conservatively predicted debris transport within the containment sump pool. The NRC staff's review consisted of three parts: (1) an examination of the assumptions and explanations provided in the licensee's written report concerning the results of the CFD analysis, (2) an onsite audit review of the vendor that performed the CFD analysis, and (3) the execution of the FLOW-3D code using the input deck provided by the licensee and several variations thereof.

As described below, during the course of the audit review, the NRC staff identified several issues with respect to the CFD analysis performed by the licensee.

3.5.4.1 Containment Spray Modeling

The licensee judged that containment spray droplets would, for the most part, drain uniformly throughout the containment [66]. The licensee stated that this conclusion is partially based on indications that the floors are primarily grating and open to flow [66]. The licensee further noted that some spray drainage occurs through the refueling cavity via a 4-inch line [66]. However, the licensee stated that the drain line is located far from the suction strainers [66]. The licensee input containment spray source parameters into the CFD code in an attempt to equate the influx of mass and energy in the CFD model to the expected actual values [66].

NRC Staff Evaluation:

First, the NRC staff questioned the conservatism of the licensee's assumptions concerning the degree of uniformity with which the containment spray flow was modeled as entering the containment pool. In particular, the staff was concerned that a large amount of spray flow may collect in the refueling cavity, from which it would be returned to the containment pool in a concentrated stream via a 4-inch drain line rather than as a diffuse flow. The staff noted that this concentrated flow from the refueling cavity drain line would result in an increased flow velocity along particular debris transport pathways to the suction strainers, as compared to what the licensee had modeled in its CFD input deck.

In response, the licensee agreed that the CFD modeling of the flow from the refueling cavity drain line was not conservative. A more conservative model would effectively speed up the flow through throughout the debris transport pathways in the containment pool [68]. The licensee further noted that the fraction of containment spray drainage that would enter the containment pool through the refueling cavity drain would be significant [68]. The licensee indicated that any follow-up calculations performed for FCS would conservatively model the refueling cavity drain line [68].

The NRC staff agrees with the technical analysis in the licensee's response, but notes that the existing FCS CFD model remains nonconservative in its modeling of the refueling cavity drain.

The NRC staff further notes that FCS calculation FC07010, “Calculation of Design Basis Minimum Containment Post-RAS Water Level, [53]” seems to indicate that other nonuniformities in the containment spray flow could be significant as well. This calculation states that “more than half of the water droplets will not fall unimpeded to the 994' floor but first land on the 1045' or 1060' floor. The water then becomes a ‘waterfall’ that cascades to the lower levels. As such they reach a much higher terminal velocity” [53]. This description from calculation FC07010 [53] appears inconsistent with assumptions underlying the containment spray CFD model, and the NRC staff believes that uncertainty remains as to whether the licensee’s treatment of the containment sprays would be conservative, even if the refueling cavity were to be explicitly modeled.

Second, the NRC staff identified a concern that the licensee’s CFD model did not introduce the quantity of kinetic energy that would be expected from the influx of containment spray droplets to the FCS containment pool. The NRC staff’s concern was prompted by the identification of an apparent error in the licensee’s derivation of kinetic energy flux conservation. The result of the apparent error was that a significantly reduced kinetic energy flux was calculated by the vendor. However, the staff noted that other aspects of the CFD modeling for the containment sprays added conservatism to compensate for this error. As a result, the NRC staff concluded that, despite lacking a sound theoretical basis, the licensee’s modeling of the containment spray flow would be conservative with respect to debris transport in areas where the actual spray flow would be diffuse, because it would overestimate the actual turbulent kinetic energy induced near the pool floor.

The NRC staff performed two sample analyses using the FLOW-3D CFD code to confirm the above conclusion. These analyses confirmed the NRC staff’s judgment that diffuse containment spray droplets would not significantly influence the velocity field near the pool floor. As such, they further confirmed that the licensee’s modeling of diffuse containment spray flow as conservative with respect to the prediction of debris transport.

In summary, regarding the licensee’s modeling of the containment sprays in its CFD input deck, the NRC staff found that (1) the licensee’s method of transferring kinetic energy into the pool, although lacking a sound theoretical basis, appears to be conservative for areas of the containment pool where the containment spray flow enters in a diffuse manner and (2) adequate technical basis was not presented in the course of the NRC staff’s audit to justify the assumptions concerning the degree of uniformity with which the FCS containment spray enters the pool.

3.5.4.2 Break Flow Exit Velocity

In Section 3.6 of the calculation ALION-REP-OPPD2522-002 [66], the licensee derived the free fall velocity used in the CFD analysis to model the flow entering the containment pool from the broken reactor coolant system pipe. In its review of this calculation, the NRC staff identified that the CFD model did not appear to account for the initial velocity of the fluid in the pipe as it exited the break. The licensee subsequently stated that any follow-up calculations for FCS will account for the fluid exit velocity [68].

NRC Staff Evaluation:

Although the fluid exit velocity is generally small compared to the free fall vertical velocity component (induced by gravitational acceleration) with which the break flow impacts the pool surface, the NRC staff considers it conservative to model the break exit velocity. Modeling the break exit velocity ensures that important physical quantities, such as energy, are conservatively predicted in the CFD analysis. For FCS, the NRC staff found that explicitly modeling the break exit velocity did not have a significant effect upon the total kinetic energy transferred from the break flow to the containment pool; however, this finding may not hold for other plants.

The NRC staff also noted that, if the flow exiting the break were to impact structures as it fell to the containment pool, momentum would be dissipated, in which case, the licensee's model would be conservative. In the case that the break spilled directly into the pool, however, this assumption would merely be realistic, rather than conservative. Also, through additional modeling assumptions, the licensee has added conservatism to its modeling of break flow.

In light of these conservatisms, the NRC staff concluded that the licensee's modeling of break flow appears reasonable, despite the small nonconservatism of neglecting the break flow exit velocity.

3.5.4.3 Spatial Resolution of the Mesh Block Nodalization

In the CFD input deck the licensee provided in conjunction with the pilot audit review, the NRC staff examined the vertical mesh spacing used to model the fluid layer directly above the containment floor. The staff's review noted that in numerical simulations of open-channel flow, it is customary to employ finer mesh spacing near the floor, where viscous shear dominates energy dissipation in the fluid boundary layer. Although the NRC staff did not consider it necessary to resolve the boundary layer for this type of debris transport calculation, a defensibly conservative estimate of the velocity field in the layer of fluid just above the containment pool floor is essential for analyzing debris transport, since this velocity field constitutes the primary influence upon the motion of tumbling debris. In this fluid layer, having a height above the floor that is on the order of the size of debris pieces that may tumble along the floor, the NRC staff considers it essential to provide a defensibly conservative estimate of the fluid velocity.

The audit material provided by the licensee did not offer a technically defensible basis that the mesh spacing in the CFD input deck leads to results that conservatively represent the velocity field that would influence the motion of debris tumbling along the containment pool floor. However, in its review, the NRC staff performed a limited investigation of the effect of introducing into the input deck a finer mesh spacing near the containment floor to more accurately predict the local velocity field. The NRC staff's CFD runs using the finer mesh spacing showed lateral velocities along the containment floor that were generally slightly lower than the velocities computed for the mesh spacing employed by the licensee. Thus, the results of the NRC staff's limited investigation indicate that the mesh spacing used by the licensee above the containment floor appears reasonable for FCS. However, the NRC staff notes that a relatively coarse nodalization near the containment floor may not be conservative in general.

3.5.4.4 Computational Fluid Dynamics Summary

In the course of its review of the licensee's CFD analysis, discussed above, the NRC staff identified several areas where analytical assumptions made by the licensee had not been accompanied by sufficient supporting technical justification or appeared nonconservative. The NRC staff's discussion also identified several conservatisms in the licensee's analysis. Weighing both the identified conservatisms and nonconservatisms, overall, the NRC staff found that the licensee's CFD model appeared reasonable for the purpose of estimating debris transport in the FCS containment pool.

3.5.5 Conservatisms in the Debris Transport Analysis

In addition to various conservatisms previously noted in the above discussion, the NRC staff identified several other substantive sources of conservatism in the licensee's debris transport analysis.

The licensee assumed that large and small pieces of insulation would be uniformly distributed between the locations where they would be destroyed and the suction strainers. In actuality, the NRC staff expects that the multi-directional flows occurring during blowdown and pool fill-up transport would tend to disburse this debris throughout containment, including areas with reduced transport potential. Therefore, by distributing the debris only over areas between the location where it is destroyed and the suction strainers, the conservatism of the debris transport analysis will be enhanced.

For many types of debris, the licensee used transport fractions for blowdown and washdown transport that appear conservative. For instance, the licensee followed the baseline guidance in generally assuming that 75% of debris generated from insulation would be transported to the lower containment, and only 25% would be transported to the upper containment. This assumption appears conservative because blowdown transport would not be expected to have such a bias in moving debris toward the lower containment, where the debris would tend to have a more direct path to the suction strainers than if it were blown into the upper containment.

Another conservative baseline assumption adopted by the licensee was the assumption that 100% of the small fines of fibrous and particulate material in active containment pools would transport to the suction strainers. Although small fines of fibrous and particulate material are expected to have a very high transport fraction, clearly the assumption of complete transport for these types of debris is conservative. Similarly, the licensee stated that 100% of the debris generated during an SBLOCA was assumed to transport to the suction strainers [65]. The most limiting SBLOCA analyzed by the licensee is located in the vicinity of the suction strainers. Although large debris transport fractions are expected to result from the shortness of the transport path and the turbulence that the break flow may cause in the pool near the suction strainers, once again, assuming complete transport is clearly conservative.

3.5.6 Debris Transport Summary

The NRC staff reviewed the licensee's debris transport analysis to determine its consistency with the sump performance methodology approved in the NRC staff's SE. The NRC staff's review found that the analysis was generally consistent with the SE and identified both conservative and potentially nonconservative assumptions in the licensee's methodology.

Among the more significant assumptions that are potentially nonconservative, the NRC staff was particularly concerned with the crediting of debris settling during large-scale head loss testing. The NRC staff considered the licensee's justification for taking credit for this phenomenon to be inadequate. The NRC staff also noted other potential nonconservatisms, including assumptions regarding (1) the tumbling velocities used for Nukon®, Temp-Mat, and paint chip debris, (2) the possibility of Temp-Mat and Cerafiber® debris floatation, (3) the erosion of fibrous debris, and (4) the representation of the containment spray flow in the CFD model using a uniform distribution.

Among the more significant conservative assumptions made by the licensee, the NRC staff noted the following: (1) the introduction of the containment spray flow through the floor, which introduced energy to resuspend or tumble settled debris, (2) the assumption that small and large pieces of insulation debris would be uniformly distributed between the locations where they would be destroyed and the suction strainers, (3) the general use of conservative transport fractions for blowdown and washdown transport, (4) the assumption of complete transport for all small fines of fibrous and particulate material introduced into active pools, and (5) the assumption of complete debris transport for an SBLOCA.

Overall, the licensee's methodology for analyzing debris transport appears reasonable if a technical justification exists to validate the crediting of debris settling during large-scale head loss testing and the other potential nonconservatisms identified by the NRC staff.

3.6 Head Loss

3.6.1 Head Loss Audit Scope

The NRC staff review of the FCS head loss evaluation focused on the head loss testing being performed by GE Energy (GE) at the Continuum Dynamics, Inc. (CDI) test facility in Ewing, NJ. Because the licensee had not completed the design of their replacement strainer modules and did not provide supporting head loss evaluation documentation, the NRC staff was not able to review either the adequacy of the replacement design or the head loss evaluation. Specific aspects of the head loss evaluation, such as sump water level, sump water temperature, strainer module approach velocities, head loss calculations, and scaling of head loss test data were therefore not reviewed.

3.6.1.1 Test Facility Audit

On August 29 through September 1, 2005 the team observed head loss testing being performed by GE at the CDI test facility in Ewing, NJ for FCS. This audit and the review of several GE and CDI head loss test documents provide the basis for head loss portion of this audit report. A detailed report for the test facility audit can be found in Appendix II. During this test facility visit, GE conducted two tests of the basic design planned for implementation at FCS. The audit team had the opportunity to watch the installation of several disks of the test module, the filling of the test tank, the introduction of the debris, and the post-test tank conditions and partial disassembly the strainer with the final accumulation of debris. The post-test examination of the observed test demonstrated that a large portion of the debris had settled to the test tank floor rather than accumulating on the test strainer module. As such, the test involved both debris transport and head loss issues in the same test. The debris accumulation on the strainer had a rather uniform appearance and the outer layer debris surface appeared to be primarily an accumulation of particulate and appeared to be a thin-bed debris accumulation. GE provided the NRC a proprietary sample of the debris bed and proprietary photos of the first day's testing observed by the NRC through OPPD following the visit [72].

Application of GE Head Loss Testing to Licensee Sump Blockage Resolution

NRC staff discussions with GE and CDI personnel made it apparent that the head loss testing being conducted by GE will provide the primary basis for the adequacy of the final strainer design. The observed debris settling within the test tank was referred to as a 'near field effect' that is being assumed to reflect a realistic debris transport that would actually occur within the plant sump following a postulated LOCA. The debris loads introduced into the test tank in the GE head loss tests were based on FCS debris transport analyses performed by Alion Science and Technology.

NRC Staff Evaluation:

The NRC staff noted several issues regarding the validity of the near field effect that had not been adequately addressed. These include: (1) the applicability of the CDI test tank hydraulic flow conditions (velocity and turbulence) to the corresponding postulated FCS sump pool flow conditions; (2) the applicability of the debris introduction procedures employed by GE in the CDI tank to represent the transport of debris into the recirculation sump area at FCS; and (3) the scaling of the measured head loss data to plant accident conditions.

This near field effect has not been considered in any of the previously reviewed or accepted pressurized water reactor sump blockage guidance, therefore its acceptance represents an enhancement to the guidance. Acceptance of the near field effect places the sump blockage evaluation outside the acceptance of the GR Baseline Guidance. Because acceptance of the near field effect has a substantial impact on the results of head loss testing, and if accepted, will likely to be adapted by other licensees, the NRC staff needs to thoroughly review the near field effect before accepting or rejecting the effect. Including the near field effect in the evaluation results in the overall debris transport fractions to the screens effectively multiplies the Alion calculated transport fractions by the near field effect transport fractions. For example, for Temp-Mat, Alion predicted 57% of the ZOI Temp-Mat transported to the sump screens. The near field effect transport fraction was speculated at 30%, thus resulting in an estimated overall transport fraction of 17%. A likely question for fibrous debris is whether a transport fraction of 17% for Temp-Mat adequately represents the fraction of the fibrous debris destroyed into suspended fiber due to LOCA debris generation and the subsequent erosion of Temp-Mat in the sump pool that would reach the strainer surface.

The basis offered by FCS, GE and CDI regarding the applicability of the near field effect to the FCS sump pool was: (1) the strainer module approach velocity, which was designed to be essentially identical to the future plant strainers; and (2) the test tank pool turbulence generated by the returning recirculation flow (which needs to be compared to plant sump pool turbulence which would include spray drainage). In order to properly validate the appropriateness of the near field effect for FCS, more supporting information should be provided to the NRC staff to ensure the near field effect is representative of the plant and not an artifact of testing. Such supporting information could include, but not be limited to: (1) performing alternate testing procedures; (2) observing fibrous debris transport (without particulate); (3) separate effects testing of particulate transport at comparable flow conditions; and/or (4) comparative CFD analyses of flows velocities and turbulence levels. One example of an alternate testing procedure could be the testing of fibrous debris without obscuring the water with particulate. Such testing could verify that the fibrous shreds cannot generally lift from the tank floor onto the strainer. That is, settled pieces of fibrous debris (this also would apply to RMI and paint chip debris) could exist where the required velocity to lift debris from the pool floor onto the screens is not present in the test tank.

The NRC staff is concerned that focusing testing on one basic general testing procedure, for instance, introducing all debris before the pumps are operated, could bypass potential important aspects of near field effect. Although, some of the debris in a LOCA scenario would accumulate in the sump prior to operating the recirculation pumps, substantial quantities of debris would enter the sump pool after recirculation started; e.g., (1) small and fine debris blown into the upper containment levels could take a while to be washed back down to the sump; (2) erosion of fibrous debris and calcium silicate in the pool would occur over a long term; and (3) failure of unqualified coatings is potentially a long term process. Such alternate time-dependent accumulations could be explored by altering test procedures; e.g., (1) introducing debris with the pump running, as well as with the pump off; (2) introducing the fibrous debris separate from the particulate debris; i.e., allowing the fibrous debris to fully accumulate prior to introducing the particulate (possible bed stratification); and (3) introducing the silicon carbide, zinc filler, and sand particulates in a wet slurry instead of in the bulk dry form observed during the pilot audit to ensure the particulate does not agglomerate upon introduction to form a sludge, which could unrealistically reduce particulate transport within the tank.

During the audit, it was noted that the time required for high density fiberglass insulation debris (Temp-Mat) to sink in water at the postulated FCS sump temperature had not been determined. Therefore, the assumption that Temp-Mat debris would approach the strainer by tumbling along the pool floor rather than floatation transport has not been substantiated for either large or small piece debris. This assumption is the basis for the CDI debris preparation procedure of pre-soaking the fibrous debris prior to introducing it into the test tank. If the Temp-Mat small piece debris were to remain buoyant long enough to float to the strainer, then this behavior could negate the near field effect.

The NRC staff notes that a thin-bed head loss, akin to the test observed, seems likely to be the limiting head loss accumulation. Therefore, it will be beneficial for GE/CDI to provide more information or justification to ensure that sufficient in-depth testing has been conducted for thin-beds so that the worst case thin-bed test is achieved and bore hole phenomena are well understood.

The NRC staff noted that the observed head losses in the head loss charts provided for review appeared to still be increasing when the tests were terminated. The NRC staff is concerned that the test termination criterion might not be sufficiently conservative to ensure the maximum head losses are measured or can be inferred.

During the audit, the NRC staff noted that one of the SBLOCA cases has a break location right above the sump region and FCS relies on the LBLOCA case to bound the head loss data for the SBLOCA case. Due to the anticipated higher turbulence near the sump during the SBLOCA, NRC staff was concerned with the validity of using the LBLOCA tests results, which assumed no additional turbulence after the recirculation starts. During a follow-up meeting with GE, CDI and FCS at the NRC Offices, the NRC staff was told that FCS had decided to remove all the calcium silicate thermal insulation material from the piping system above the sump and perform sector head loss tests to justify the SBLOCA head loss. The NRC staff considers this an appropriate approach. However, without more detailed documents, the NRC staff cannot pass final judgement on this issue.

3.6.1.2 Head Loss Scaling

There are two types of scaling involved: (1) scaling estimated plant debris loads down to head loss testing debris loads, and (2) scaling test head losses to alternate plant debris loadings, flow rates, and water temperatures.

NRC Staff Evaluation:

The test personnel stated the debris loads used in the tests were scaled down from the predicted plant debris loads by the area of the strainer tested to the area of the strainer proposed for implementation in the plant to preserve the debris loading per unit area. This method of scaling predicted plant conditions to comparable test conditions is a valid approach that has previously been found acceptable.

The staff raised a question regarding the GE scaling equation as to whether the near field effect is the same for the test as that for the plant in terms of geometry, flow path, fluid velocity field, temperature, debris loading, and debris type. GE and CDI should provide further justification to support the use of this scaling equation to extrapolate the module head loss test results. The NRC staff acceptance of this scaling equation should be limited to specific conditions that ensure conservatism, which include: (1) a demonstration of equal of higher near sump debris transport in the test tank compared with in the plant; (2) the debris bed composition needs to be similar with respect to types of debris, bed thickness, and bed effective specific surface area; (3) the debris bed compression of the test needs to be greater than the compression in the plant; (4) the strainer approach flow velocity in the test needs to be equal to or greater than the plant strainer approach velocity; (5) the debris bed in the test needs to be acceptably continuous.

3.6.1.3 Head Loss Conclusions

Overall, the NRC staff considers that the GE/CDI plant-specific head loss tests is the proper way to proceed with the new strainer design, if sufficient justification is provided. As the FCS new strainer testing and design have not reached the final stage, the NRC staff cannot comment on the acceptance of the overall head loss evaluation since neither the final strainer module design nor the head loss evaluation was available for review. The NRC staff audit of the GE head loss testing for FCS resulted in several areas that require more supporting information before the NRC staff can accept the test procedures. In particular, the near field effect inherent in the GE testing has not been validated sufficiently for the plant LOCA scenarios. Due to the importance of this issue, the NRC staff finds that substantially more experimental and/or analytical validation is needed for acceptance. Because a thin-bed debris accumulation appears to be the limiting head loss for FCS, the NRC staff would like to review more in-depth thin-bed testing to ensure the worst case thin-bed has been achieved and to ensure reproducibility. The NRC staff is concerned that the GE scaling equation will not produce conservative scaling results unless limiting application criterion are clearly defined to compensate for assumptions inherent in the derivation of the equation.

3.6.2 Net Positive Suction Head (NPSH) for Containment Sump Recirculation

3.6.2.1 Audit Scope

The NRC staff audited the following the following documents that were provided by the licensee concerning the calculation of NPSH margin:

1. “Post-RAS NPSH Adjustments for CS and HPSI Pumps,” Calculation FC06676 [59].
2. “Calculation of Design Basis Minimum Containment Post-RAS Water Level, Revision 0,” Calculation FC07010 [53].
3. “ABB-CE Evaluation of Containment Spray Pump Net Positive Suction Head Accounting for Sump Subcooling,” Calculation FC05977 [58].
4. “Table of Pressures used in Calculating NPSH.”

In the performance of its review, the NRC staff also reviewed the licensee’s response to (GL) 97-04, “Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps,” as well as selected subsequent correspondence between the NRC staff and the licensee regarding that GL . As the bulk of the licensee’s NPSH calculations had not been modified since the NRC staff documented its review of the response to GL 97-04, the present audit review did not generally reevaluate their adequacy, except where necessary to provide assurance that these calculations were adequate for addressing the technical issues associated withGSI-191. In addition, the NRC staff reviewed Section 6.2, “Engineered Safeguards, Safety Injection System,” of the FCS Updated Safety Analysis Report, Revision 14. Finally, the NRC staff also reviewed a slide presentation provided by the licensee at a public meeting on July 14, 2005.

3.6.2.2 Summary of Recirculation Configuration and Net Positive Suction Head Parameters

The new containment sump suction strainer design proposed by the licensee has not been finalized. However, the planned design consists of an array of large passive stacked disk strainers with a total area of approximately 2800 ft^2 [51].

According to the licensee's response to GL 97-04, the pumps taking suction from the containment recirculation sump following a design-basis LOCA are the HPSI pumps and the containment spray (CS) pumps [61]. There are two separate ECCS pump suction headers. One header supplies two HPSI pumps and one CS pump; the other supplies one HPSI pump and two CS pumps. Each suction header also supplies one low-pressure safety injection pump; however, the low-pressure safety injection pumps shut down upon receipt of a recirculation actuation signal. The licensee's GL 97-04 response further stated that the capability exists to operate the HPSI pumps in "piggyback" configuration by connecting the CS pump discharge to the suction of the HPSI pumps. Although this mode of operation would be less demanding with respect to NPSH for both sets of pumps, it is not credited in accident analyses. Therefore, the NPSH margin for the "piggyback" configuration was not evaluated by the NRC staff in conjunction with either the GL 97-04 review or the present pilot audit review.

As described further below, the licensee provided a tabular summary of parameters that are pertinent to the NPSH margin for the CS pumps, which are the most limiting pumps with respect to NPSH margin. The table provided by the licensee was editorially adapted by the NRC staff into the table below [52]. The NRC staff further notes that the degree to which the parameters for the proposed design had been finalized by the licensee was not clear at the time of the audit.

Parameter	Existing Screen Licensing/Design Basis	Proposed Strainer Licensing/Design Basis (Large-Break LOCA)	Proposed Strainer Licensing/Design Basis (Small-Break LOCA)
Containment Overpressure Head ($P_{cont} - P_{vap}$) (ft)	8.99	8.99	8.99
Height of Water Static Head (Z) (ft)	23.55	24.71	24.16
Piping/System Head Loss ($h_{f\ max}$) (ft)	3.87	3.87 *	3.87 *
Clean Strainer Head Loss (h_{clean}) (ft)	0.27	0.20	0.20
Strainer Head Loss Due to Debris Bed (h_{debris}) (ft)	0	1.012	1.012 **
NPSH available (ft)	28.4	28.618 *	28.068 *
NPSH required (ft)	27.3	27.3 *	27.3 *
NPSH margin (ft)	1.1	1.318	0.768
Flow (gpm)	Strainer A : 4000 Strainer B : 6650	Strainer A : 4140 Strainer B : 6700	Strainer A : 4140 Strainer B : 6700

* *Approximate because of slight change in flow rate.*

** *small-Break LOCA head loss conservatively uses large-break LOCA value but is expected to be lower and may be bounded by large-break LOCA.*

In the above table, the first column represents the condition of FCS with (1) the existing sump screen design that relies upon the 50%-screen-blockage assumption and (2) the existing drainage gap in the refueling cavity past the reactor vessel seal ring. The second and third columns are representative of a LBLOCA and an SBLOCA, respectively, once the proposed large passive strainers have been installed. The second and third columns do not incorporate the 50%-blockage assumption and further assume that a proposed modification has been performed to install spacers under the reactor vessel seal ring to reduce the hold up of water in the refueling cavity.

Although the licensee did not provide an analogous table for the HPSI pumps, the NPSH parameters pertinent to these pumps were available in Calculations FC06676 [59] and FC07010 [53], as well as Revision 14 of the FCS Updated Safety Analyses Report, Section 6.2. Compiling this information to facilitate comparison, the table below contains what the NRC staff believes to represent approximate NPSH parameters for the HPSI pumps [3, 9, 13]. Once again, the degree to which the parameters for the proposed design have been finalized was not clear to the NRC staff.

Parameter	Existing Screen Licensing/Design Basis	Proposed Strainer Licensing/Design Basis (Large-Break LOCA)	Proposed Strainer Licensing/Design Basis (Small-Break LOCA)
Containment Overpressure Head ($P_{cont} – P_{vap}$) (ft)	8.99	8.99	8.99
Height of Water Static Head (Z) (ft)	24.13	25.29	24.74
Piping/System Head Loss ($h_{f\ max}$) (ft)	5.84	5.84 *	5.84 *
Clean Strainer Head Loss (h_{clean}) (ft)	0.27	0.20	0.20
Strainer Head Loss Due to Debris Bed (h_{debris}) (ft)	0	1.012	1.012 **
NPSH available (ft)	27.01	27.23 *	26.68 *
NPSH required (ft)	13.9	13.9 *	13.9 *
NPSH margin (ft)	13.1	13.3	12.8
Flow (gpm)	Strainer A : 4000 Strainer B : 6650	Strainer A : 4140 Strainer B : 6700	Strainer A : 4140 Strainer B : 6700

* *Approximate because of slight change in flow rate.*

** *SBLOCA head loss conservatively uses LBLOCA value but is expected to be lower and may be bounded by LBLOCA.*

The above tables employ the licensee's definition of available NPSH (NPSHa), which is apparently as follows:

$$NPSHa = P_{cont} - P_{vap} + Z - h_{f\ max} - h_{clean} - h_{debris},$$

where the variables in this equation are identified in the tables above. Then the NPSH margin (NPSHm) can be defined in the usual way, as the difference between the NPSHa and the required NPSH (NPSHr):

$$NPSHm = NPSHa - NPSHr.$$

From comparing the two tables above, it is apparent that the HPSI pumps have a significantly larger NPSH margin than the CS pumps, primarily due to their substantially lower NPSHr.

In the table of NPSH parameters provided by the licensee for the CS pumps, the licensee indicated that the calculated NPSHm values are considered as margin available to address any head loss that may be experienced due to chemical effects [52]. The licensee further noted that actual expected chemical effects head loss is being developed.

3.6.2.3 Calculation of Net Positive Suction Head Margin

3.6.2.3.1 Net Positive Suction Head Available

(A) Containment Overpressure Head (P_{cont} - P_{vap})

> In the resolution of GL 97-04, the licensee submitted a letter dated April 15, 1999, which provided information concerning the calculation of the containment overpressure head [57]. In an attachment to this letter, the licensee stated that ABB-Combustion Engineering had used the CONTRANS code to compute the amount of overpressure that would exist as a function of time for the first 100,000 seconds (approximately 27.8 hours) following an LBLOCA. The licensee further stated that out of a matrix of 27 cases, a best estimate maximum safeguards scenario was found to provide the most limiting overpressure head, which was found to be approximately 36 ft. Of this limiting overpressure head, the licensee decided to credit only 25%, or 8.99 ft, for conservatism.

(B) Height of Water Static Head (Z)

> The licensee calculated the minimum post-accident containment water level in Calculation FC07010, “Calculation of Design Basis Minimum Containment Post-RAS Water Level, Revision 0,” dated May 27, 2005 [53]. As this calculation was not previously examined under the NRC staff’s GL 97-04 review, the NRC staff performed a detailed audit review.
>
> The water source terms considered in the calculation were (1) the safety injection and refueling water tank, and, depending upon the accident scenario, (2) the reactor coolant system and (3) the safety injection tanks (SITs).
>
> The water hold up or loss terms considered in the calculation were (1) water vapor in the containment atmosphere, (2) condensation on heat sinks, (3) containment spray droplets in the containment atmosphere, (4) hold up on containment floors and gratings, (5) the water volume required to fill the CS system headers to establish spray flow, (6) leakage from the ECCS and CS system, (7) hold up in the refueling cavity, and (8) hold up in the reactor cavity.
>
> Calculation FC07010 [53] computed a minimum water level for ten cases, including LBLOCAs and SBLOCAs, full and minimum ECCS and CS flows, break locations on the hot leg and at the top of the pressurizer, and the current configuration of a drainage

path along the reactor vessel seal ring as well as the configuration that would exist following a proposed modification to reduce hold up in the refueling cavity.

Based upon the inputs and assumptions described above, the licensee concluded that the minimum containment water level would be as stated in the table below [53]. The reported minimum pool depths arise from assuming full engineered safeguards (i.e., all HPSI and CS pumps operating), since under these conditions, water hold up in the containment is maximized.

Case	Existing Reactor Vessel Seal Ring Gap	Proposed Reactor Vessel Seal Ring Gap
LBLOCA Minimum Pool Depth (ft)	3.41	3.96
SBLOCA Minimum Pool Depth (ft)	2.86	3.41

(C) Piping/System Head Loss ($h_{f\ max}$)

The licensee calculated the suction header losses for the CS and HPSI pumps in Calculations FC05977 [58], "ABB-CE Evaluation of Containment Spray Pump Net Positive Suction Head Accounting for Sump Subcooling," and FC06676 [59], "Post-RAS NPSH Adjustments for CS and HPSI Pumps."

As these calculations had been evaluated as part of the NRC staff's efforts on GL 97-04, and, further, the licensee had not yet updated them to account for flow adjustments or other perturbations that may occur due to the proposed modifications to the suction strainers, the NRC staff did not perform a detailed audit review of this section of the calculation.

(D) Clean Strainer Head Loss (h_{clean})

The clean strainer head loss term for the existing sump screen design was determined in Calculation FC06676 [59], "Post-RAS NPSH Adjustments for CS and HPSI Pumps." This calculation conservatively models the existing screen as a mesh within the 30-inch sump suction opening.

The licensee apparently did not provide a finalized clean strainer head loss for the proposed strainer design. However, the head loss from the clean strainer and the strainer system piping is estimated as being approximately 0.20 ft [2, 4]. Information provided by the licensee indicated that the clean strainer head loss will be derived from a combination of clean strainer head loss test data obtained during large-scale debris head loss testing and analytical predictions using classical methods [54].

(E) Strainer Head Loss Due to Debris Bed (h_{debris})

The licensee's calculations of the head loss due to debris upon the suction strainer are discussed in this audit report in Section 3.6.1.

3.6.2.3.2 Calculation of Net Positive Suction Head Required

The licensee calculated the NPSHr for the CS and HPSI pumps in Calculations FC05977 [58], "ABB-CE Evaluation of Containment Spray Pump Net Positive Suction Head Accounting for Sump Subcooling," and FC06676 [59], "Post-RAS NPSH Adjustments for CS and HPSI Pumps."

As these calculations had been evaluated as part of the NRC staff's efforts on GL 97-04, and, further, the licensee had not yet updated them to account for flow adjustments or other perturbations that may occur due to the proposed modifications to the suction strainers, the NRC staff did not perform a detailed audit review of this section of the calculation.

NRC Staff Evaluation:

For completeness, in addition to providing an assessment related to the present audit review, the NRC staff will also briefly summarize, where appropriate, conclusions from an earlier NRC staff assessment of the licensee's NPSH calculation that were generated during the GL 97-04 review.

(I) Net Positive Suction Head Available

As stated previously, the licensee used the following equation to compute the NPSHa:

$$\text{NPSHa} = P_{cont} - P_{vap} + Z - h_{f\ max} - h_{clean} - h_{debris}$$

To verify the adequacy of the licensee's application this equation, the NRC staff questioned whether the proposed suction strainers would be fully submerged at all times during the recirculation phase of postulated accident scenarios requiring recirculation. The NRC staff further requested that the licensee provide the minimum postulated submergence. In response, the licensee stated that the current design goal is a minimum submergence of 4 inches during an SBLOCA [64]. For an LBLOCA, the licensee stated that the minimum submergence would be approximately 10.5 inches. The licensee's response confirmed that the above equation is valid for FCS, and that additional failure mechanisms that apply to partially submerged strainers are not applicable to FCS.

(A) Containment Overpressure Head (P_{cont} - P_{vap})

During its review of FCS's response to GL 97-04, the NRC staff first identified that the licensee had begun crediting containment overpressure. FCS was originally licensed in accordance with Safety Guide 1, without any credit for containment overpressure [55]. In 1992, however, the licensee discovered that an as-built hydraulic analysis of the CS system was not available [55]. When a reanalysis was performed, the licensee concluded that the CS pumps' flows would be significantly greater than had been originally analyzed [55]. To compensate for the increased NPSHr necessary to support the increased flows, the licensee inappropriately applied the 10 CFR 50.59 process to take credit for containment overpressure without NRC staff review [55]. (The NRC staff position concerning overpressure or subcooling credit has been and continues to be that staff review is required for any increase in overpressure credit, whether it be an increase in the amount of overpressure required, or an extension of the time over which the overpressure will be credited.)

The staff's GL 97-04 review further focused upon the analysis used to support the licensee's overpressure credit. In computing the available containment overpressure, the licensee employed the CONTRANS code to determine the two essential inputs: containment pressure and sump pool temperature. However, when the CONTRANS code was originally reviewed by the NRC staff, in a topical report evaluation dated April 6, 1976, the NRC staff noted that it would not be used to calculate containment overpressure for ECCS evaluations [60]. During the GL 97-04 review, the NRC staff expressed further concerns that, although the CONTRANS simulations had calculated containment overpressure as a function of time for approximately one day, the entire NPSH calculation had not been performed in a time-dependent manner [55]. Additionally, the licensee's rationale for applying containment overpressure credit for the HPSI pumps was not clear to the NRC staff, since overpressure was not necessary to demonstrate adequate NPSH margin for these pumps (as is apparent from an earlier table in Section 3.6.2.2 of this report) [55].

Following the submission of additional information and further interactions with the licensee, the NRC staff concluded that the information request of GL 97-04 had been satisfied [56]. Prior to closing GL 97-04, however, the NRC staff reviewed the FCS Individual Plant Examination, in part to evaluate the effect of the loss of the CS pumps due to insufficient subcooling [56]. Although the NRC staff did not independently verify that an overpressure head of 8.99 ft would be available for the CS pumps, the NRC staff's calculations showed that the HPSI pumps would remain functional even if the CS pumps were lost [56]. As such, the NRC staff concluded that assurance is provided "that the requirements of 10 CFR 50.46, i.e., long term core cooling, will be met even if the containment spray pumps are lost due to insufficient subcooling following a large-break LOCA" [56]. On the basis of this conclusion and insights from the FCS Individual Plant Examination, the FCS Updated Safety Analyses Report, and information provided by the licensee, the NRC staff closed the GL 97-04 review effort for FCS [56].

For the present pilot audit, the NRC staff did not reexamine the licensee's existing containment overpressure analysis. However, as part of the pilot audit review, the NRC staff noted that the existing CONTRANS calculations did not analyze the available containment overpressure beyond approximately the first 28 hours following a postulated accident. Although the CONTRANS calculations generally appear to have quite steady results over all but the early stages, it is not clear that the results would be valid at times significantly longer than one day. Furthermore, information provided by the licensee suggested that overpressure credit would be used, in part, to provide margin against the possible head loss due to chemical effects [52]. As recent experiments have indicated, the head loss due to chemical effects may build up gradually over time periods on the order of days or weeks. Consequently, the NRC staff requested that the licensee provide further information to justify the implicit assumption that containment overpressure will continue to be available in the long-term to provide margin for chemical effects.

The licensee responded that, following a recirculation actuation signal , both the ECCS and CS flow rates and the containment sump temperature will tend to decrease with time, thereby increasing NPSH margin [64]. The licensee stated that the CS pumps would normally be secured after approximately five hours of operation, and that the HPSI stop and throttle criteria would normally result in operators reducing HPSI flow to less than the value assumed to occur immediately following the recirculation actuation signal recirculation actuation signal . The licensee further provided examples to show that the cooling of the containment sump pool water would increase the amount of overpressure available. At a containment pressure of 14.2 psia, with the sump pool at 195EF, the licensee stated that an overpressure head of approximately 9.12 ft would be available. If the pressure remained steady and the sump pool cooled to 160EF, the available overpressure head would increase to approximately 22.5 ft. The licensee’s response further expressed a distinction between its definition of overpressure (i.e., containment pressure exceeding the existing pressure present prior to an accident) and the atmospheric subcooling that may be present when the containment pressure is greater than or equal to atmospheric pressure. For clarity, however, throughout the present audit report, the NRC staff has maintained consistency with GL 97-04, using the term containment overpressure solely to denote “containment pressure that is above the vapor pressure of the sump fluid” (i.e., $P_{cont} – P_{vap}$) [62].

Although the licensee’s response provided a simplified analysis explaining why NPSH margin is expected to increase with time, as the sump pool temperature and recirculation sump flow rates are reduced, the NRC staff believes that further supporting analysis and documentation may be warranted if the current overpressure credit is to be extended beyond the time period that has been analyzed and documented in the licensee’s existing NPSH calculations.

(B) Height of Water Static Head (Z)

Regarding the licensee’s calculation of minimum containment water level (Calculation FC07010 [53]), the NRC staff found that the licensee had generally done an adequate job of considering potential water loss and hold up. The NRC staff’s audit review identified two questions, however.

The first question concerned the licensee’s assumption regarding leakage from the ECCS and CS systems. Within Calculation FC07010 [53], the licensee states that, “[f]or the purposes of calculating containment minimum water level, it is assumed that the plant is placed on shutdown cooling within 24 hours of the start of the LOCA." Consistent with this statement, the licensee computed sump leakage by doubling the limiting hourly leakage rate from the FCS Technical Specifications (3800 cm^3/hour) and multiplying by 24 hours. Inasmuch as the mission time of the recirculation sump is generally considered to be 30 days, the NRC staff questioned whether the transition to shutdown cooling can be made within 24 hours for all postulated LOCAs. If such an assurance were not possible, the NRC staff requested that the licensee justify the conservatism of the assumed sump leakage term.

The licensee responded that situations may occur for which the transition to shutdown cooling may not occur within 24 hours, and that some leakage would presumably occur whenever the ECCS and CS system operate [64]. However, the licensee indicated that the effect of any expected leakage rate on containment water level would be more than offset by the termination of the CS pumps, which is normally performed after approximately 5 hours of operation following a LOCA. The licensee stated that, following the shutdown of the CS pumps, water flow into the refueling cavity and reactor cavity will cease, and hold up in these volumes will no longer be a concern. In the minimum safeguards case in Calculation FC07010 [53] , the licensee indicated that roughly 6,083 ft^3 (45,500 gallons) of water is assumed to be held up in the refueling cavity and reactor cavity. Adding to this quantity other volumes of water that CS termination would restore, including the water held up in falling spray droplets (49 ft^3) and run-off from containment surfaces (462 ft^3), the licensee indicated that approximately 11 inches of water would be added to the sump level in the minimum safeguards case. In comparison, an apparently conservative leakage rate of approximately 10 ft^3 per day over 30 days would result in a water loss of approximately 300 ft^3, which represents roughly half an inch in sump level.

Although the degree of precision associated with the simplified analysis in the licensee's response is unclear, the NRC staff considers that the large margin discussed in the response provides confidence that the leakage rate assumed in the Calculation FC07010 [53] does not result in this calculation being nonconservative.

The second question concerned the licensee's assumptions regarding the thermal expansion and contraction of the water inventory added to the containment sump pool. In Calculation FC07010 [53], the licensee assumed 160EF as the temperature for calculating the thermal expansion of the safety injection refueling water storage tank inventory added to the containment pool. Although it did not seem to be explicit in Calculation FC07010 [53], it appeared to the NRC staff that the licensee had further assumed 160EF as the temperature for computing the thermal contraction of the reactor coolant system volume added to the containment pool. The NRC staff requested that the licensee justify the assumption that the use of 160EF in computing thermal expansion and contraction provides conservative results. As noted in the previous paragraph, the licensee stated that the containment minimum water level is based on a 24-hour period. Yet the NRC staff noted that the best-estimate maximum safeguards CONTRANS calculation used to support the licensee's overpressure credit indicates that the containment temperature after 24 hours would be approximately 100EF. The reason for this discrepancy was not clear to the NRC staff. Furthermore, as noted in the previous paragraph, the mission time for the ECCS is generally considered to be 30 days. Therefore, the NRC staff requested that the licensee justify that its computations of thermal expansion and contraction provided conservative results.

The licensee responded to the NRC staff's request in a manner similar to the previous question, by showing that the static head gained by securing the CS pumps would substantially exceed the static head lost from the thermal contraction of the sump pool water [64]. The licensee indicated that if the sump pool were to cool from 160EF to 100EF, a reduction in volume of approximately 600 ft^3 would occur, roughly corresponding to a 1-inch reduction in sump pool level. In contrast, as discussed above, securing the CS pumps would result in a sump pool level increase of approximately 11-inches. The licensee further stated that the conservatism identified in its containment minimum water level calculation may be considered later in demonstrating additional NPSH margin for chemical effects.

As before, although the degree of precision associated with the simplified analysis in the licensee's response is not clear, the NRC staff considers that the margin discussed in the response provides confidence that the licensee's assumption concerning pool temperature does not result in Calculation FC07010 [53] being nonconservative (even when simultaneously considering the apparently conservative leakage assumption discussed in the preceding paragraphs, which would reduce containment water level by approximately another half inch). The NRC staff further agrees that a formal time-dependent or long-term calculation of minimum containment water level may identify conservatisms that could be useful in addressing long-term head loss that may arise from chemical effects.

(C) Piping/System Head Loss ($h_{f\ max}$)

As the NRC staff had reviewed this area of the licensee's calculation under GL 97-04, a detailed review was not performed for the pilot audit review. However the NRC staff requested that the licensee specify whether the roughness factors used to calculate the piping friction loss term account for the possible effect of material aging. The licensee responded that the existing design basis analysis (Calculation FC05777) does not include the variation of the piping friction factors due to aging [64]. The licensee stated that the HPSI, low-pressure safety injection , and CS system piping is constructed of stainless steel and is not used during normal plant operation. As a result of the construction materials and the service conditions, the licensee stated that no appreciable corrosion or increase in roughness would be expected to occur due to aging. Although the licensee's response appears reasonable, the NRC staff did not verify its appropriateness.

(D) Clean Strainer Head Loss (h_{clean})

Insufficient information was available for the NRC staff to perform an audit of the licensee's clean strainer head loss methodology.

(E) Strainer Head Loss Due to Debris Bed (h_{debris})

The NRC staff's audit of the licensee's debris bed head loss calculations is provided in Section 3.6.1 of this report.

(ii) Net Positive Suction Head Required

The NRC staff did not perform a detailed review in this area, since the GL 97-04 review appeared to have adequately reviewed the licensee's existing calculations. However, the NRC staff noted that in Calculation FC06676 [59], slightly different flow rates were used for the CS pumps in the calculation of NPSHr than for the calculation of friction losses. The NRC staff verified that these slight variations led to negligible differences in the final results.

Returning to the general subject of NPSH margin, the licensee observed that existing calculations may not precisely reflect the condition of the plant once the proposed strainers are installed, due to perturbations in flow rates and other parameters [52]. In response to a question from the NRC staff regarding this observation, the licensee responded that plans exist for updating affected NPSH calculations once the proposed suction strainers have been installed [64]. The licensee stated that, as part of the FCS design change process, all calculations, drawings, and associated configuration control documents are required to conform to the as-built condition of the plant. The licensee further stated that the final design calculation will include strainer head loss performance data based upon plant-specific test data, final strainer design, updated data for chemical effects, updated data for qualified coatings in the ZOI, and the final insulation configuration. The licensee also stated that any additional design or analysis change that affects NPSH will be included.

In summary, with the exception of the licensee's crediting of containment overpressure, the NRC staff's audit review found that the methods used by the licensee in computing NPSH margin appeared to be generally conservative and consistent with applicable regulatory guidance, including Regulatory Guide 1.82, Revision 3. Further, the NRC staff had previously reviewed the licensee's NPSH calculations, including the licensee's crediting of containment overpressure, in response to GL 97-04 and found them to be sufficient to support GL closure [56]. The NRC staff's current audit review primarily considered whether the NPSH calculations remained sufficient in light of the technical issues associated with GSI-191. The audit review found that the licensee's NPSH calculations have not been finalized, but that the current approach and general methodology with respect to NPSH appear reasonable. (Note however, that this conclusion does not apply to the licensee's methodology for calculating the head loss across debris beds. The licensee's methodology and calculations of debris bed head loss are discussed in Section 3.6.1 of this audit report.) Finally, the NRC staff noted a concern that supplementary supporting analysis and documentation may be warranted if the licensee plans to extend the current containment overpressure credit beyond the time period for which it has been analyzed and documented in existing NPSH calculations.

4.0 DESIGN AND ADMINISTRATIVE CONTROLS

4.1 Debris Source Term

Section 5.1 of the GR and SE discuss additional refinements for licensees to consider as part of their overall sump evaluations. These additional refinements could improve plant safety and reduce the risks associated with sump screen blockage. Specifically, this section addresses the following five categories for design and operational refinements; however, there may be other refinements that would also meet the intent of this section of the evaluation methodology:

- Housekeeping and foreign material exclusion programs
- Change-out of insulation
- Modify existing insulation
- Modify other equipment or systems
- Modify or improve coatings program

NRC Staff Evaluation:

The licensee provided very limited information to address each of these candidate refinements as part of this pilot audit. Each refinement is discussed here:

- Housekeeping and Foreign Material Exclusion Programs

  The licensee did not provide any specific information to review. The licensee did implement these types of programs through their Bulletin 2003-01 response.

- Change-out of Insulation

  FCS Calculation No. FC06985 [49], “Fort Calhoun Station Debris Generation Post LOCA,” assessed the impact of installing new insulation types on the replacement steam generators and pressurizer in the 2006 refueling outage. As such, the licensee evaluated two debris load cases:

  - Steam generators and pressurizer insulated with RMI
  - All calcium silicate inside the bioshield replaced with low density fiberglass insulation such as THERMAL-WRAP® or Nukon®.

  The NRC staff agrees that the licensee should consider the change-out of the calcium silicate insulation as a means to reduce the debris source term.

- Modify Existing Insulation

  As noted above, the licensee proposes some modification of existing insulation (Calcium Silicate).

- Modify Other Equipment or Systems

The staff performed a minimal review of this area. One modification of existing equipment was evaluated concerning drainage from the refueling cavity.

- Modify or Improve Coatings Program

  The NRC staff did not review the coatings area as part of this pilot audit.

4.2 Screen Modifications

Section 5.3 of the NEI GR and NRC staff SE provides guidance and considerations regarding potential sump screen designs and features to address sump blockage concerns. Specifically, the attributes of three generic design approaches are addressed. These include passive strainers, backwash strainers, and active strainers. The NRC staff SE does not specifically support any single design, but rather emphasizes two performance objectives that should be addressed by any sump screen design:

- The design should accommodate the maximum volume of debris that is predicted to arrive at the screen, fully considering debris generation, debris transport, and any mitigating factors (e.g., curbing)

- The design should address the possibility of thin-bed formation.

NRC Staff Evaluation:

The licensee informed the NRC staff that they will be installing a GE passive strainer design; however, the licensee did not provide reviewable design information regarding the proposed FCS sump modifications. The NRC staff did attend full scale module testing of the FCS prototype sump screen design at the Continuum Dynamics Incorporated testing facility in Ewing, NJ as part of this pilot audit. This testing included scaled debris loading based on the maximum volume that is predicted to arrive at the sump screen. The licensee also informed the NRC staff that the design will address the possibility of thin-bed formation by analyzing this case in the head loss analyses. However, at the time of the conclusion of this pilot audit, the licensee had not completed their head loss analyses. The head loss calculations and Engineering Change Packages were not provided to the NRC staff for review during the audit period.

Based on the limited information reviewed as part of this pilot audit, the NRC staff is not able to reach a conclusion regarding the adequacy of the FCS sump screen design at this time. The licensee was in the middle stages of design; therefore, definitive conclusions could not be reached by the audit team. Also, because the adequacy of the new screen design and other proposed modifications is highly dependent on the acceptability of the various analyses that establish the screen design (i.e., debris generation, debris transport, debris accumulation, and head loss), further design changes could be necessary as the licensee finalizes the various ongoing aspects of the sump performance evaluation.

5.0 ADDITIONAL DESIGN CONSIDERATIONS

General guidance for considerations to be used when performing a structural analysis of the containment sump screen is contained in Section 7.1 of the NEI GR [71] and the approved NRC staff Safety Evaluation [2]. General items identified for consideration include (1) verifying maximum differential pressure caused by combined clean screen and maximum debris load at rated flow rates, (2) geometry concerns, (3) sump screen material selection for the post accident environment, and (4) the addition of hydrodynamic loads from a seismic event.

Review of sump screen structural analysis was not included in the scope of this audit plan as part of the limitations applied to this Pilot Plant review. Documentation provided by FCS indicated that the structural analysis would be included in the detailed design phase, and was not available for review at this time.

5.1 Upstream Effects

The objective of the break selection process is to evaluate the flowpaths upstream of the containment sump for holdup of inventory which could reduce flow to and possibly starve the sump. Section 7.2 of the GR [71] and the safety evaluation of the methodology [2] provide the guidance to be considered in the upstream effects process to evaluate holdup or choke points which could reduce flow to and possibly cause blockage upstream of the containment sump. The GR identifies two parameters important to the evaluation of upstream effects: (1) containment design and postulated break location, and (2) postulated break size and insulation materials in the ZOI.

FCS Calculation No. FC07010 [53], “Engineering Calculation: Calculation of Design Basis Minimum Containment Post-RAS Water Level,” was used in the review for this section. The NRC staff reviewed this calculation to ascertain that the licensee evaluated the flow paths from the postulated break locations and from containment spray washdown to identify and take measures to alleviate potential choke points in the flow field upstream of the sump. The NRC staff also reviewed the above document to verify that the licensee considered water holdup in the placement of any curbs or debris racks intended to trap debris before reaching the sump.

NRC Staff Evaluation:

The NRC staff reviewed the licensee’s head loss evaluations. Specifically, the NRC staff reviewed FC Calculation No. FC07010 [53], considering the approved NEI methodology documented in Section 7.2 of the SE. The NRC staff noted during the audit that the licensee’s upstream effects evaluation was performed in a manner consistent with the approved methodology.

The licensee determined the possible places that water may be prevented from reaching the containment basement due to holdup on upper levels of the containment building. During a refueling outage a walkdown was performed to examined the potential for pool formation on the concrete portions of the floor elevations above El. 994’ or on major pieces of equipment located inside containment. The conclusion was that it is not credible that a pool of any significance would be formed. Although pooling is not considered to be credible, water may be held up as a thin layer of water on floor surfaces and grating, and prevented from reaching the containment basement.

During a LOCA containment spray will deposit a part of its flow into the refueling cavity. A drain line at the bottom of the cavity drains the collected water to the 994’ floor, and to the containment sump. Some water will be retained in the refueling cavity and reactor cavity. The licensee determined the extent of water retention in these two areas (Calculation No. FC07144). When the retention reaches the 1013’ level, at the reactor flange, the water finds a second means to exit the refueling cavity past the reactor seal ring. A gap is created by thermal expansion allowing water to flow into the reactor cavity. For future design, the licensee is considering install spacers under the seal ring to insure that a larger gap will exist as a second flow path at 1013’, to establish a higher sump pool level.

5.2 Downstream Effects

5.2.1 Downstream Effects - General

Guidance for considerations when evaluating downstream effects are contained in Section 7.3 of the NEI GR [71] and the approved SE [2].

NRC Staff Evaluation:

The NRC staff reviewed the following documents related to downstream effects during the audit:

1. Document 2005-08220, GSI-191 Downstream Effects - Flow Clearances, Rev. 0
2. FCS Updated Safety Analyses Report Section 6.2, Engineered Safeguards, Safety Injection System, Rev. 18 [63]
3. OPPD FCS, GL 2004-02 Pilot Plant Overview Meeting, July 14, 2005
4. Calculation FC06676 [59] Post-RAS NPSH Adjustments for CS and HPSI Pumps, Rev. 0
5. Calculation FC05977 [58] ABB-CE Evaluation of Containment Spray Pump Net Positive Suction Head Accounting for Sump Sub-cooling, Rev 0

The NRC staff reviewed the list of all components and flowpaths considered to determine the scope of the licensee’s downstream evaluation (pumps, valves, instruments, and heat exchangers, etc.). The listing provided appeared to be complete. However, the listing was based on a review of existing documentation and not on walkdown information. It is expected that the licensee will field-verify components and flowpaths at some time in the future. The licensee reviewed appropriate references and had a good starting point for their Phase 2 analysis.

The NRC staff examined design and license mission times and system lineups to support mission critical systems and these seemed reasonable based on NRC Staff review of the Updated Safety Analyses Report and Emergency Operating procedures. A discussion of LOCA scenarios was also provided. The audit team notes that a table to summarize this information would be helpful for future assessments.

The licensee appropriately recognized that their HPSI throttle valves will require further effort to assess their vulnerability to clogging during post-LOCA operation. They also identified areas of potential blockage and materials of construction of other downstream components. The consideration of wear and therefore, potential leakage, will be addressed in their Phase 2 analysis.

The licensee did not address the subject of the potential for air entrainment during ECCS operation. FCS indicated that this will be addressed in their Phase 2 analysis. Therefore, no assessment of the adequacy of the ECCS for this aspect was performed.

The FCS Phase 1 evaluation lists the components and materials of all wetted downstream surfaces (wear rings, pump internals, bearings, throttle valve plug, and seat materials). However, characterization and properties of ECCS post-LOCA fluid (abrasiveness, solids content, and debris characterization) is not yet complete. Therefore FCS will perform their assessment as part of the Phase 2 evaluation once testing is complete.

The licensee performed a fairly comprehensive study of the opening sizes and running clearances of their pumps and valves. The study appears to be complete. Their approach to initial screening and susceptibility is reasonable and consistent with current industry guidance.

FCS performed a paper review of system low points and low-flow areas. The review appears to be complete, but physical walkdowns may be necessary to validate the conclusions.

FCS performed an initial assessment of flow velocities. Their initial assessment seems reasonable. However, further review will be needed once information regarding the constituents of the downstream fluid are better known. FCS indicated that this will be addressed in their Phase 2 evaluation,

FCS did not address equipment strainers, cyclone separators and other components, nor did they assess potential changes in system or equipment operation caused by wear (i.e., pump or system vibration, flow balances, rotor dynamics etc.). FCS indicated that issues will be addressed in their Phase 2 analysis.

FCS performed a paper review of their instruments and instrument tubing and determined them not an issue due to existing configuration. While the configuration has not been verified by walkdown, the paper review appears reasonable and complete.

FCS has appropriately noted a potential susceptibility with downstream heat exchangers. This issue is to be addressed in their Phase 2 evaluation.

In summary, the licensee performed a Phase 1 evaluation only. Their submittal is partially complete with an approach that appears reasonable. They have completed an initial review and generated a listing of components to be reviewed in greater detail in Phase 2. This Phase 2 review will be performed after completion of the strainer testing and will be based on the determination of the constituents and properties of the downstream fluid.

5.3.2 Downstream Effects - Fuel Only

Guidance for considerations when evaluating downstream effects on vessel internals and reactor fuel are also contained in Section 7.3 of the NEI GR [71] and the approved NRC staff Safety Evaluation [2]. General items identified for consideration include flow blockage associated with core grid supports, mixing vanes, and debris filters; and impact on flowpaths between the downcomer and upper plenum.

No other refinements are provided in other sections of the SE.

Documentation provided by FCS indicated that the evaluation of downstream effects on reactor fuel was not complete, and that this evaluation would be performed in accordance with Westinghouse Owners Group WCAP-16406-P “Evaluation of Downstream Sump Debris Effects in Support of GSI-191", issued on June 30, 2005 at a later date. To determine and evaluate the methodology of this review, the NRC staff requested a copy of WCAP-16406-P from the Westinghouse Owners Group. A copy was provided to the NRC staff on August 5, 2005.

NRC Staff Evaluation:

The NRC staff reviewed the WCAP evaluation methodology and has identified a number of areas where clarification or additional information is required. Request for this additional information was provided to the Westinghouse Owners Group via email on 9/30/05, with copies provided to FCS for information. The NRC staff and the Westinghouse Owners Group are currently discussing the best method for resolution and communication of issues with this evaluation methodology so that it might provide a standardized approach for addressing this item.

Therefore, NRC staff is unable to completely assess the ability of the FCS ECCS sump screen to perform as required with the proposed modifications to prevent undesirable downstream effects from impacting reactor fuel or vessel internals performance at this time.

The comprehensive tool for evaluation of downstream effects on reactor fuel (WCAP-16404-P) contains a number areas where technical issues relevant to injection of debris-laden water into the reactor vessel and core region need additional information, clarification, and development. Ongoing testing and methodology development efforts are expected to provide additional inputs needed to complete this evaluation.

6.0 CONCLUSIONS

The pilot audit outcome goal was to provide a report that would inform external stakeholders, FCS, and the NRC review team on lessons learned during implementation of the approved methodology to aid in resolving the pressurized water reactor sump performance issue.

To accomplish this goal, the NRC staff audited key decision points in the SE, considered the level of information available to make a preliminary technical judgement on each decision point, and what the engineering judgements are that can be drawn for each decision point. The level of information was characterized as (1) robust, (2) partially complete with an approach that appears reasonable, or (3) limited to the point where an informed technical judgement on that particular decision point would be subject to large uncertainty.

The NRC staff is exploring both domestic and international sources for technical information that could minimize uncertainty in its decision making. Any such information obtained during the resolution of GSI-191 will be used to support NRC staff reviews.

The following list reflects the NRC staff technical judgments of the aforementioned key decision points. (Areas of limited audit review are not included below.):

3.1 Break Selection

Level of information - Level 2: Partially complete with an approach that appears reasonable.

The licensee's evaluation of break selection appears to be reasonable. The evaluation was generally performed in a manner consistent with the approved SE methodology. Deviations from the NRC staff approved methodology were judged by the NRC staff to be acceptable based on the technical basis provided by the licensee (with head loss remaining to be considered). The licensee has not provided the NRC staff with any head loss calculations for the breaks analyzed, and the licensee has not provided the NRC staff with any of the head loss calculations to demonstrate that the thin-bed effect can be accommodated by the sump screen design.

The criterion used to define the most challenging break is the estimated head loss across the sump screen. Therefore, all phases of the accident scenario must be considered for each postulated break location, including debris generation, debris transport, debris accumulation and sump screen head loss. As such, should the licensee need to revise other phases of their analyses (e.g., debris transport - transport fractions), then they should also reassess the limiting break location to verify that it remains limiting.

3.2 Debris Generation/Zone of Influence (Excluding Coatings)

Level of information - Level 2: Partially complete with an approach that appears reasonable.

The licensee's ZOI evaluation appears to be reasonable. The evaluation was generally performed in a manner consistent with the approved SE methodology. Deviations from the NRC staff approved methodology were judged by the NRC staff to be acceptable based on the technical basis provided by the licensee (with several issues identified). The licensee stated in a request for additional information response that they will revise the debris generation calculation to clearly state the debris generation and ZOI methodologies applied and incorporate consistent treatment of pipe diameter for ZOI calculations. Additionally, the licensee should include a qualitative description of the methodology applied regarding partial walkdown area applications in the calculation.

3.3 Debris Characteristics

Level of information - Level 2: Partially complete with an approach that appears reasonable.

The evaluation was generally performed in a manner consistent with the approved SE methodology. Deviations from the NRC staff approved methodology were judged by the NRC staff to be acceptable based on the technical basis provided by the licensee (with issues identified by the staff). The licensee agreed to revise FCS Calculation No. FC06985 [49] to accurately reflect the Nukon® size distribution applied, the licensee stated that the RMI debris size distribution discussion would be relocated to the FCS debris generation calculation (FC06985 [49]), given more time for this audit, the NRC staff would have pursued questions regarding the licensee's calcium silicate scanning electron microscope analysis, the NRC staff expects to receive information regarding Nukon® vs. THERMAL-WRAP® material and may review this as part of the GL closeout process, and the NRC staff identified an issue regarding float-transport of Temp-Mat insulation material.

3.5 Debris Transport

Level of information - Level 2: Partially complete with an approach that appears reasonable.

The NRC staff reviewed the licensee's debris transport analysis to determine its consistency with the sump performance methodology approved in the NRC staff's SE. The NRC staff's review found that the analysis was generally consistent with the SE and identified both conservative and potentially nonconservative assumptions in the licensee's methodology.

Among the more significant assumptions that are potentially nonconservative, the NRC staff was particularly concerned with the crediting of debris settling during large-scale head loss testing. The NRC staff considered the licensee's justification for taking credit for this phenomenon to be inadequate. The NRC staff also noted other potential nonconservatisms, including assumptions regarding (1) the tumbling velocities used for Nukon®, Temp-Mat, and paint chip debris, (2) the possibility of Temp-Mat and Cerafiber® debris floatation, (3) the erosion of fibrous debris, and (4) the representation of the containment spray flow in the CFD model using a uniform distribution.

Among the more significant conservative assumptions made by the licensee, the NRC staff noted the following: (1) the introduction of the containment spray flow through the floor, which introduced energy to resuspend or tumble settled debris, (2) the assumption that small and large pieces of insulation debris would be uniformly distributed between the locations where they would be destroyed and the suction strainers, (3) the general use of conservative transport fractions for blowdown and washdown transport, (4) the assumption of complete transport for all small fines of fibrous and particulate material introduced into active pools, and (5) the assumption of complete debris transport for an SBLOCA.

Overall, the licensee's methodology for analyzing debris transport appears reasonable if a technical justification exists to validate the crediting of debris settling during large-scale head loss testing and the other potential nonconservatisms identified by the NRC staff.

3.6.1 Head Loss

Level of information - Level 3: Limited information

Overall, the NRC staff considers that the GE/CDI plant-specific head loss tests are proper ways to proceed with the new strainer design, if sufficient justifications are provided. As the FCS new strainer testing and design have not reached the final stage, the NRC staff cannot comment on the acceptance of the overall head loss evaluation since neither the final strainer module design nor the head loss evaluation was available for review. The NRC staff audit of the GE head loss testing for FCS resulted in several areas of issues that require more supporting information before the NRC staff can accept the test procedures. In particular, the near field effect inherent in the GE testing has not been validated sufficiently for the plant LOCA scenarios. Due to the importance of this issue, the NRC staff finds that substantially more experimental and/or analytical validation is needed for acceptance. Because a thin-bed debris accumulation appears to be the limiting head loss for FCS, the NRC staff would like to review more in-depth thin-bed testing to ensure the worst case thin-bed has been achieved and to ensure reproducibility. The NRC staff is concerned that the GE scaling equation will not produce conservative scaling results unless limiting application criterion are clearly defined so compensate for assumptions inherent in the derivation of the equation.

3.6.2 Net Positive Suction Head for Containment Sump Recirculation

Level of information - Level 2: Partially complete with an approach that appears reasonable.

With the exception of the licensee's crediting of containment overpressure (described in detail in Section 3.6.2 of this audit report), the NRC staff found that the methods used by the licensee in computing NPSH margin appeared to be generally conservative and consistent with applicable regulatory guidance, including Regulatory Guide 1.82, Revision 3. Although the licensee's overpressure credit is inconsistent with existing regulatory guidance, the NRC staff had previously focused upon this issue during its review of responses to GL 97-04 and concluded that the licensee had provided sufficient information to support the closure of the GL for FCS [56]. The NPSH calculations provided to the NRC staff generally had not been updated to reflect the as-built condition that will exist once the proposed suction strainers have been installed. However, the information provided by the licensee during the course of the audit review suggests that the licensee's current methodology appears to account for technical issues associated with GSI-191 and further seems to provide a reasonable approach for updating existing NPSH calculations. Finally, the NRC staff noted a concern that supplementary supporting analysis and documentation may be warranted if the licensee plans to extend the current containment overpressure credit beyond the time period for which it has been analyzed and documented in existing NPSH calculations.

4.3 Screen Modifications

Level of information - Level 3: Limited information

The licensee’s overall screen modification approach appears reasonable. However, because the adequacy of the new screen design and other proposed modifications is highly dependent on the acceptability of the various analyses that drive the screen design (i.e., debris generation, debris transport, debris accumulation and head loss), further design changes could be necessary as the licensee finalizes the various ongoing aspects of the sump performance evaluation.

5.2.1 Downstream Effects

Level of information - Level 2: Partially complete with an approach that appears reasonable

The FCS licensee performed a Phase 1 evaluation only. Their submittal is partially complete with an approach that appears reasonable. They have completed an initial review and generated a listing of components to be reviewed in greater detail. This Phase 2 review will be performed after completion of the strainer testing and will be based on the determination of the constituents and properties of the downstream fluid.

Attachment 1 References

1.	GL 04-02	NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," dated September 13, 2004.
2.	191 SE	GSI-191 SE, Revision 0, "Safety Evaluation of NEI Guidance on Pressurized Water Reactor Sump Performances," dated December 6, 2004.
3.	NUREG-0897	NUREG-0897, "Containment Emergency Sump Performance," U.S. Nuclear Regulatory Commission, dated October 1985.
4.	DG-1107	Draft Regulatory Guide DG-1107, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, transmitted December 2002.
5.	NRCB, 1994	Bulletin 93-02, Supplement 1, "Debris Plugging of Emergency Core Cooling Suction Strainers," dated February 18, 1994
6.	NRCB, 1995	Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode," dated October 17, 1995;
7.	NRCB, 1996	Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," dated May 6, 1996
8.	GL 85-22	NRC Generic Letter GL 85-22, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," dated December 3, 1985
9.	GSI 191	GSI-191, "Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance," prioritized September, 1996.
10.	NRCB, 2003	NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors," dated June 9, 2003.
11.	NUREG/CR-6762	NUREG/CR-6762, "GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," Vol. 1, dated August 2002.
12.	NUREG/CR-6770	NUREG/CR-6770, "GSI-191: Thermal-Hydraulic Response of Pressurized Water Reactor Reactor Coolant System and Containments to Selected Accident Sequences," dated August 2002.
13.	NUREG/CR-6762	NUREG/CR-6762, Vol. 3, "GSI-191 Technical Assessment: Development of Debris Generation Quantities in Support of the Parametric Evaluation," dated August 2002.
14.	NUREG/CR-6762	NUREG/CR-6762, "GSI-191 Technical Assessment: Development of Debris Transport Fractions in Support of the Parametric Evaluation," Vol. 4, dated August 2002.

15.	NUREG/CR-6224	NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," dated October 1995.
16.	RG 1.82-3	Regulatory Guide 1.82, Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," dated November 2003.
17.	ANSI/ANS 58.2	ANSI/ANS 58.2-1988, "American National Standard Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture," 1988.
18.	NUREG/CR-6808	NUREG/CR-6808, "Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance," U.S. Nuclear Regulatory Commission, February 2003.
19.	URG SE	Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Bulletin 96-03, Boiling Water Reactor Owners Group topical Report NEDO-32686, "Utility Resolution Guidance for ECCS Suction Strainer Blockage" (Docket No. Proj0691), dated August 20, 1998.
20.	URG	GE Nuclear Energy, "Utility Resolution Guide for ECCS Suction Strainer Blockage," by BWR Owners' Group, NEDO-32686-A, October 1998.
21.	NUREG/CR-6369	NUREG/CR-6369, Vol 1: Drywall Debris Transport Study-Final Report; Vol 2: Drywell Debris Transport Study: Experimental Work-Final Report, February 1998.
22.	NUREG/CR-6773	NUREG/CR-6773 (LA-UR-02-6786), "GSI-191: Integrated Debris-Transport Tests in Water Using Simulated Containment Floor Geometries," Los Alamos National Laboratory, December 2002.
23.	CS, 2004a	C. J. Shaffer, M. T. Leonard, B. C. Letellier, J. D. Madrid, A. K. Maji, K. Howe, A. Ghosh, and J. Garcia, "GSI-191: Experimental Studies of Loss-of-Coolant-Accident-Generated Debris Accumulation and Head Loss with Emphasis on the Effects of Calcium Silicate Insulation," Los Alamos National Laboratory report LA-UR-04-1227 (April 2004).
24.	MD, 2004	Mei Ding, Amr Abdel-Fattah, Stewart Fischer, Bruce Letellier, Kerry Howe, Janet Garcia, and Clint Shaffer, "Characterization of Latent Debris from Pressurized-Water-Reactor Containment Buildings," Los Alamos National Laboratory report LA-UR-04-3970 (June 2004).
25.	CS, 2004b	Clinton J. Shaffer and Bruce C. Letellier, "GSI-191 Technical Study: Detailed Blowdown/Washdown Transport Analysis for Pressurized-Water-Reactor Volunteer Plant," Los Alamos National Laboratory report LA-UR-04-4338 (June 2004).
26.	LA-UR-03-6415	Los Alamos National Laboratory, ADAMS Accession No. ML033230260.
27.	GL 87-11	NRC Generic Letter 87-11, "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements," dated June 19, 1987.
28.	SRP	NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.

29.	Rao, 2001	D. V. Rao, B. Letellier, C. Shaffer, S. Ashbaugh, and L. Bartlein, "GSI-191: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," Draft Technical Letter Report, Los Alamos National Laboratory, July 2001.
30.	NUREG/CR-6371	Shaffer, C., W. Bernahl, J. Brideau and D. V. Rao, BLOCKAGE 2.5 Reference Manual, NUREG/CR-6371, U.S. Nuclear Regulatory Commission, December 1996.
31.	NEI, 2002a	NEI 02-01, "Condition Assessment Guidelines: Debris Sources Inside Pressurized Water Reactor Containments," dated April 19, 2002.
32.	NEI, 2002b"	NEI Draft Evaluation Methodology Ground Rules" dated December 12, 2002.
33.	10 CFR	Code of Federal Regulations, Title 10 "Energy."
34.	RG 1.174	Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," dated July 1998.
35.	MEB 3-1	Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," included in NUREG-0800, "Standard Review Plan," Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping."
36.	NUREG-0588	NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Revision 1, July 1981.
37.	RG 1.1	Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," November 2, 1970.
38.	GL 91-18	Generic Letter 91-18, "Information to Licensees Regarding NRC Inspection Manual on Resolution of Degraded and Non-Conforming Conditions," Revision 1, October 8, 1997.
39.	ANSI/HI 1.1-1.5	ANSI/HI 1.1-1.5 - 1994, "American National Standard for Centrifugal Pumps for Nomenclature, Definitions, Application and Operation."
40.	OPG, 2001	Ontario Power Generation, "Jet Impact Tests—Preliminary Results and Their Applications," N-REP-34320-10000-R00, April 2001.
41.	NIS96	National Institute of Standards and Technology, standard reference database 10, version 2.11, 1996.
42.	WEI83	Weigand, G. G., et al., "Two-Phase Jet Loads," NUREG/CR-2913, Sandia National Laboratory (prepared for the U.S. Nuclear Regulatory Commission), January 1983.
43.	DIN04	Ding, M., et al., "Characterization and Head-Loss Testing of Latent Debris from Pressurized-Water-Reactor Containment Buildings," prepared for the U.S. Nuclear Regulatory Commission by Los Alamos National Laboratory, LA-UR-04-3970, August 2004.
44.	NUREG/CP-0152	Proceedings of the Eighth NRC/ASME Symposium on Valve and Pump Testing, paper titled "Design, Testing and Implementation of Modifications to the Davis-Besse HPI Pumps for Debris Laden Water

		Operation" by Robert Coward and Stephen Kinsey, July 12, 2004, MPR Associates and Robert Schrauder, John O'Neill and Stephen Osting, First Energy Nuclear Operating Company.
45.	TIA 2003-04	Task Interface Agreement (TIA) No. 2003-04, TAC No. MC0584, "Evaluation of Davis-Besse Modifications to the High Pressure Injection Pump and Associated Mock-up Testing," dated February 11, 2004.
46.	IN 96-27	NRC Information Notice 96-27, "Potential Clogging of High Pressure Safety Injection Throttle Valves During Recirculation," dated May 1, 1966.
47.	IN 93-34	NRC Information Notice 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," dated April 26, 1993.
48.	IN 93-34S1	NRC Information Notice 93-34, Supplement 1, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," dated May 6, 1993.
49.	OPPD/FC06985	FCS Calculation No. FC06985, “Fort Calhoun Station Debris Generation Post LOCA,” draft.
50.	NUREG/CR-6772	NUREG/CR-6772, "GSI-191: Separate-Effects Characterization of Debris Transport in Water," August 2002.
51.	OPPD/	Omaha Public Power District, “Generic Letter 2004-02 Pilot Plant Overview Meeting,” July 14, 2005.
52.	OPPD/NPSH	Omaha Public Power District, “Table of Pressures Used in Calculating NPSH.”
53.	OPPD/FC07010	Omaha Public Power District, “Calculation of Design Basis Minimum Containment Post-RAS Water Level, Revision 0,” Calculation FC07010.
54.	GE, 2005a	General Electric, GE Proposal No. 1724-JXDA7-TP1, Revision 8 (Proprietary), 2005.
55.	LRW, 1999	Wharton, L. R., “Assessment of Fort Calhoun Response to RAI Pertaining to NRC Generic Letter 97-04, ‘Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps,” February 23, 1999.
56.	JNH, 2000	Hannon, J. N., “Close out of NRC Generic Letter 97-04, ‘Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps,’ for Fort Calhoun Station,” February 3, 2000.
57.	SKG, 1999	Gambhir, S. K., “Additional Information to Support NRC Review of Response to Generic Letter (GL) 97-04, Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps,” April 15, 1999.
58.	OPPD/FC05977	Omaha Public Power District, “ABB-CE Evaluation of Containment Spray Pump Net Positive Suction Head Accounting for Sump Subcooling,” Calculation FC05977.

59.	OPPD/FC06676	Omaha Public Power District, “Post-RAS NPSH Adjustments for CS and HPSI Pumps,” Calculation FC06676.
60.	NRC, 1976	U.S. NRC, “Description of the CONTRANS Digital Computer Code for Containment Pressure and Temperature Analysis, Amendment Nos. 1, 2 and 3 (Nonproprietary),” April 6, 1976.
61.	SKG, 1997	Gambhir, S. K., “90 Day Response to NRC Generic Letter 97-04,” December 31, 1997.
62.	GL 97-04	U.S. NRC, Generic Letter 97-04, “Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps,” October 7, 1997.
63.	OPPD/UFSAR	Omaha Public Power District, “Fort Calhoun Station Updated Safety Analysis Report,” Revision 14.
64.	OPPD, 2005a	Omaha Public Power District, Responses to Staff Questions on NPSH (via email), August 2005.
65.	OPPD, 2005b	Omaha Public Power District, "Fort Calhoun Station Debris Transport Post-LOCA," ALION-REP-OPPD2522-003, Revision 0, July 22, 2005.
66.	OPPD, 2005c	Omaha Public Power District, "Fort Calhoun Station LOCA Pool CFD Transport Analysis," ALION-REP-OPPD2522-002, Revision 0, 2005.
67.	OPPD, 2005d	Omaha Public Power District, "[Response to] Request for Additional Information on Fort Calhoun Pilot Plant Audit Regarding Calculation ALION-REP-OPPD2522-003," September 2005.
68.	OPPD, 2005e	Omaha Public Power District, "[Response to] Request for Additional Information on Fort Calhoun Pilot Plant Audit Regarding Calculation ALION-REP-OPPD2522-002," September 2005.
69.	GE, 2005b	General Electric Company, "GE Response to NRC/FCS Strainer Pilot Plant Audit Discussion," September 2005 (Proprietary).
70.	CR3 SE	U.S. Nuclear Regulatory Commission, "Crystal River Unit 3 Pilot Plant Audit Report: Analyses Required for the Response to Generic Letter 2004-02 and GSI-191 Resolution," June 29, 2005.
71	NEI 2004b	NEI Guidance Report 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Volume 1 - "Pressurized Water Reactor Sump Performance Evaluation Methodology," December 2004
72	GE, 2005c	GE-OPPD-CSS-010, “Information Requested in Support of Fort Calhoun Pilot Plant Audit,” Proprietary, September 12, 2005.