Oconee Nuclear Station Corrective Actions for Generic Letter 2004-02

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Acronym List

BS	building spray
CCI	Control Components, Incorporated
CFD	computational fluid dynamics
CFT	core flood tank
ECCS	emergency core cooling system
GL	generic letter
GR	Guidance Report
GSI	Generic Safety Issue
HPI	high-pressure injection
LPI	low-pressure injection
LOCA	loss-of-coolant accident
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
ONS	Oconee Nuclear Station
PWR	pressurized water reactor
QA	quality-assured
RBS	Reactor Building spray
RBES	Reactor Building Emergency Sump
RCS	reactor coolant system
RMI	reflective metallic insulation
RTV	room-temperature vulcanizing
SE	safety evaluation
ZOI	zone of influence

1.0 BACKGROUND

1.1 Introduction

The U.S. Nuclear Regulatory Commission (NRC) is auditing, on a sample basis (related to reactor type, containment type, strainer vendor, NRC regional office, and sump replacement analytical contractor), licensee corrective actions for Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," dated September 13, 2004 [1], for approximately ten commercial pressurized water reactors (PWRs). The purpose of the audits is to verify that the implementation of Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on PWR Sump Performance [2]" sump strainer and related modifications bring those reactor plants into full compliance with 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-water Nuclear Power Reactors," and related requirements, and to draw conclusions as to the probable overall effectiveness of GL 2004-02 corrective actions for the 69 U.S. operating PWRs.

In response to NRC GL 2004-02 [1], PWR licensees are designing and implementing new strainers in their plants in order to resolve the GSI 191 [2] sump performance issue by December 31, 2007. Oconee Nuclear Station (ONS), which is operated by Duke Energy, proceeded to contract for design and installation of new strainers in all three units. The Control Components, Incorporated (CCI) replacement "pocket strainers" have approximately 5200 ft^2 surface area for Units 2 and 3 and 4800 ft^2 for Unit 1. Unit 3 was selected for focus for the audit because that Unit was the lead for the baseline analyses.

The audit is intended to yield benefits to both the NRC and industry. For the NRC these include:

> The audit will help NRC staff determine the adequacy of the new strainer design and the contractor resources needed for future reviews, audits, and/or inspections.
>
> The NRC staff can identify generic GSI-191 issues that need to be further addressed and clarified through future interactions with strainer vendors, other licensees, and the PWR Owners Group.

Benefits envisioned for the licensee and industry include:

> Feedback from the audit will assist Duke in resolving the GSI-191 PWR sump issue.
>
> Lessons learned from the audit will help the industry identify, focus on and prioritize the issues impacting resolution of GSI-191.

The audit commenced on March 13, 2007, when Duke presented an overview of the GSI-191 Project to the staff audit team. Following review of the presentation materials [3] and other documents provided during the overview session, the onsite portion of the audit commenced on March 26, 2007, with the staff audit team exiting the site on March 29, 2007. Table 1 lists key NRC staff, licensee staff and contractors, and NRC consultants and identifies attendance during audit meetings.

Table 1 ONS Audit Meetings

Name	Organization	Title/ Area	Project Over-view 3/13/2007	Audit Onsite Entrance 3/26/2007	Audit Onsite Exit 3/29/2007
John Lehning	NRC/SSIB	Debris Transport/ Characteristics; Upstream	x	x	x
Paul Klein	NRC/DCI	Chemical Effects	x		
Ralph Architzel	NRC/SSIB	Team Leader	x	x	x
Steven Unikewicz	NRC/DCI	Downstream Components	x	x	x
Tom Hafera	NRC/DSS		x		
Steve Smith	NRC/SSIB	Strainer Headloss	x	x	x
Shanlai Lu	NRC/SSIB	Strainer Headloss	x	x	x
Clint Shaffer	NRC - ARES Corp	Baseline	x	x	x
Ted Ginsberg	NRC-BNL	NPSH	x	x	x
Matt Yoder	NRC/DCI	Coatings	x	x	x
Alexander Tsirigotis	NRC/DE	Structural	x		
Walt Jensen	NRC/DSS	Fuel/Core	x		
Lenny Olshan	NRC/DDRL	Project Manager	x		
Michael Scott	NRC/SSIB	Branch Chief	x		
Andy Hutto	NRC/Region II	Resident Inspector			x
Charles Peabody	NRC/Region II	DRS/Eng Br 1			x
Frank Eppler	Duke/ONS	Mech Civil Eng. Primary Sys	x	x	x
Jason Patterson	Duke/ONS	Mech Civil Eng. Pri Supr	x	x	x

Table 1 ONS Audit Meetings

Name	Organization	Title/ Area	Project Over- view 3/13/2007	Audit Onsite Entrance 3/26/2007	Audit Onsite Exit 3/29/2007
Russ Oakley	Duke/ONS	Reg Compliance	x		
Lenny Azzarello	Duke/ONS	Acting Eng Div Mgr		x	x
Steven Copps	Duke/ONS	Mech&Civil Eng Mgr		x	x
Ken Isley	Duke/NGO/MPS	Coating Program		x	x
Brant Elrad	Duke/ONS	Eng/Civil Eng		x	
Bob Gamberg	Duke/New Plant Dev	Primary Systems		x	x
Bob Meixell	Duke/ONS	Reg Compliance		x	x
Judy Smith	Duke/ONS	Reg Compliance		x	x
David Wilson	Duke/ONS	Mech Civil Eng. Primary Sys		x	x
Patsy Earnhardt	Duke/ONS	Modification Eng		x	x
Greg Saxon	Duke/ONS	Eng - Primary Systems		x	x
Bob Heineck	Duke/ONS	Mod Eng Eng Suprv		x	
Abe Lofti	Duke/ONS	Civil Eng		x	
Martin Hemphill	Duke/ONS	Modifications - Civil		x	
James Weast	Duke/ONS	Reg Compliance		x	
Urs Blumer	CCI	Sr Consultant Nuc S			x
Bruce Hamilton	Duke/ONS	Site Vice President			x
Jerry Wermiel	NRC/DSS	Deputy Director			x
Cory Gray	Duke/ONS	Reg Compliance			x
Beth Durham	Duke/ONS	Mod Eng PM/PC Sup			x

The audit provided an opportunity for the NRC to: (1) review the basis, including the detailed mechanistic analysis and design documents, for the proposed new strainer design (2) identify areas that may need clarification or generic resolution. The following technical categories related to sump performance were reviewed and discussed:

Debris generation	Debris transport
Coatings	Debris characterization
System head loss	Chemical head loss
Modifications	Upstream and downstream effects
	Net positive suction head (NPSH) for emergency core cooling system (ECCS) pumps

The staff reviewed the design documents provided by the licensee and interacted with the licensee and its vendors to develop a thorough understanding of major aspects of the design and analysis.

During the course of the audit, staff examined detailed aspects of the ONS new strainer design noting general conformance to the approved staff guidance [8], but also identified issues related to the licensee's implementation and plans that need to be assessed as part of the licensee's completion of corrective actions for GL 2004-02 [1]. These are discussed and identified as Open Items throughout this audit report, and were communicated to the licensee during the audit meetings and telephone conferences. The licensee is expected to address and document resolution of these Open Items in conjunction with its efforts to respond to GL 2004-02 [1].

1.2 Bulletin 2003-01 Response

To reduce post-LOCA (loss-of-coolant accident) sump clogging risk during continued operation until resolution of GSI-191 at operating PWRs, on June 9, 2003, the NRC issued Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors" [4] to all PWR licensees. Overall, the ONS Bulletin 2003-01 response [6], dated August 7, 2003, was initially evaluated by the staff as clear and comprehensive. It specifically addressed the six interim compensatory measure categories of Bulletin 2003-01. The NRC staff reviewed and initially closed the response in a letter dated March 30, 2004 [10]. After the NRC staff closed Bulletin 2003-01 for ONS, concerns regarding degraded containment coatings, adequate remediation efforts and adequacy of Bulletin 2003-01 interim compensatory measures were raised by NRC staff. Based on this information, the NRC staff re-opened its review of the ONS Bulletin 2003-01 response and requested additional information by letter dated March 30, 2005 . The licensee for ONS submitted a response to this request on April 29, 2005 [17]. On August 15, 2005, the NRC staff submitted another request for additional information via electronic mail. The licensee responded to the NRC request by letters dated August 16, 2005, and October 13, 2005.

The NRC staff reviewed the information provided by ONS and found that the interim compensatory measures put in place reduce the interim risk associated with potentially degraded or nonconforming ECCS recirculation functions until an evaluation to determine compliance is complete. As part of this review, documented in a letter dated January 20, 2006 [12], the staff identified key areas (particularly regarding coatings) that required further evaluation in order to successfully resolve GSI-191 at ONS. During the GL 2004-02 audit the NRC staff reviewed the actions taken by the ONS licensee to address the

unresolved issues identified in the Bulletin 2003-01 closure letter [12]. A description of the various coatings test activities conducted by ONS is provided in the coatings evaluation section of this report (pg 56). Although there are open items in the area of coatings as a result of ongoing industry efforts, the NRC staff considers ONS actions to be responsive to and meet the intent of Bulletin 2003-01.

1.3 Generic Letter 2004-02 September 2005 Responses

In response to the NRC staff's information request in Generic Letter 2004-02, the ONS licensee submitted several correspondences including the following:

- a 90-day response dated March 1, 2005, which discussed the planned methodology for analyzing the performance of the Reactor Building Emergency Sump (RBES) and the schedule for conducting a plant walkdown [47],
- a second response dated September 1, 2005, which discussed the licensee's analyses and planned modifications to ensure adequate RBES performance [48],
- a letter dated April 7, 2006, which stated that the licensee planned to submit responses to the staff's requests for additional information by December 31, 2006 [49], and
- a letter dated June 28, 2006, which updated commitments from the licensee regarding GL 2004-02 [49].

Prior to the audit, the staff reviewed these correspondences from the licensee and further reviewed the GL 2004-02 requests for additional information the staff sent to the licensee in a letter dated February 9, 2006 [51], to determine whether technical issues identified in these requests for additional information could be resolved through the audit review.

Through the submittals listed above, the licensee provided responses to the information request in GL 2004-02 [7]. These submittals described the activities performed by the licensee to ensure that the ECCS and containment spray system recirculation functions will be in compliance regarding the post-accident debris issues associated with GSI-191, including the following:

- Reactor Building (containment) walkdowns to quantify potential debris sources
- debris generation and transport analyses
- calculation of net positive suction head margin
- revised containment overpressure calculation
- strainer structural analysis
- chemical effects analysis
- downstream effects analyses
- upstream effects evaluation
- modifications to administrative controls such as the plant coatings and labeling programs

The licensee stated that the methodology used for analyzing the performance of the RBES was Nuclear Energy Institute (NEI) 04-07 [7], as amended by the associated staff safety evaluation (SE) [8]. A summary of the licensee's analysis was presented in the September 2005 GL 2004-02 response [48]. The September 2005, GL 2004-02 response further stated that the methodology in NEI 02-01 [29] was used for performing the Reactor Building walkdown.

The licensee’s September 2005 GL 2004-02 response stated that passive replacement sump strainers designed by CCI, will be installed at ONS [48]. These strainers were described as having a surface area of approximately 5000 ft^2 and 1/12-inch diameter perforations [48].

The licensee's September 2005 GL 2004-02 response [48] contained the following nine commitments:

1. A baseline evaluation has been performed for ONS by ENERCON Services, Inc. This evaluation was performed using the guidance of NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology, Revision 0," as amended by the NRC SE for that methodology. The evaluation will be complete by June 30, 2006.

2. A downstream effects evaluation will be completed for ONS by Stone & Webster. This evaluation will be performed using the methodology provided by WCAP-16406-P, “Evaluation of Downstream Sump Debris Effects in Support of GSI191." Some exceptions to the WCAP-16406-P methodology may be taken. The evaluation will be complete by June 30, 2006. Any additional plant modifications or procedure changes associated with this evaluation will be completed by December 31, 2007.

3. Chemical effects will be evaluated to confirm that sufficient margin exists in the final sump design to account for this head loss. The evaluation will be complete by June 30, 2006. Any additional plant modifications or procedure changes associated with this evaluation will be completed by December 31, 2007.

4. A modified RBES strainer and supporting structure will be installed in the fall of 2005 for ONS Unit 2, the spring of 2006 for Unit 3, and the fall of 2006 for Unit 1.

5. A walkdown of Reactor Building using the guidance of NEI 02-01, "Condition Assessment Guidelines: Debris Sources Inside PWR Containments" was completed for ONS Unit 3 in the fall of 2004, and a confirmatory walkdown of Unit 1 was completed in the spring of 2005. A similar confirmatory walkdown will be performed for Unit 2 in the fall of 2005.

6. The plant labeling process will be enhanced to ensure that any additional labels or signs placed inside the Reactor Building are evaluated to ensure that potential debris sources are evaluated for potential adverse impacts to the RBES and ECCS functions. This corrective action will be completed by December 31, 2007.

7. Enhancements will be made to the ONS coatings program. These enhancements include increased programmatic controls to clearly document and trend evaluations performed on degraded coatings and to ensure that potential coating debris is evaluated for impact on the ECCS and RBES functions. This corrective action will be completed by December 31, 2007.

8. Fibrous insulation installed on the cooling water piping associated with each unit's 'B' Auxiliary Reactor Building Cooling Unit will be removed. This insulation removal is complete for Unit 1. The insulation will be removed in the fall of 2005 for Unit 2 and the spring of 2006 for Unit 3.

9. Duke will evaluate the modification process to determine if additional controls are needed in order to maintain the validity of inputs to analyses performed in resolving GSI-191 concerns. This evaluation will be completed by June 30, 2006.

The discussion in the licensee's GL 2004-02 response is generally based upon underlying analyses and calculations that the staff reviewed in detail during the audit review. As a result, discussion on the technical issues addressed in the GL 2004-02 response is deferred to the appropriate audit report sections that address the licensee's underlying analyses.

1.4 ONS Reactor Building Walkdown

On November 8–9, 2005, a staff team traveled to ONS to review the licensee's ongoing analyses and activities in response to Generic Letter 2004-02 [65]. The primary focus of the trip was informational presentations given by the licensee and subsequent technical discussions between the staff and the licensee. However, the staff's trip had been scheduled to coincide with the Unit 2 refueling outage during which the replacement sump strainer was being

Figure 1 ONS Unit 2 Reactor Building Emergency Sump During the Replacement Strainer Installation

installed. The Unit 2 strainer replacement project was the first strainer modification project at ONS. The staff took the opportunity to tour the Reactor Building, focusing upon the ongoing installation of the replacement strainer, the area around the RBES, and other areas in the Reactor Building containment of significance to the performance of the sump strainer, such as the flowpaths by which water would drain to the Reactor Building basement. At the time of the staff's visit, the installation of the replacement strainer designed by CCI, was nearly complete.

Figure 1 is a photograph taken by the licensee on November 7, 2005, which depicts the RBES with the installation of the replacement strainer in progress on the day before the staff arrived at ONS.

2.0 DESCRIPTION OF INSTALLED/PLANNED CHANGES

In response to NRC GL 2004-02, ONS removed the existing trash racks and screens and installed a new strainer designed by CCI. The diameter of the strainer holes is intended to ensure that any debris that can pass through the strainer will not cause blockage or excessive wear to components in the ECCS flow path or the Reactor Building spray (RBS) system. This includes pumps, valves, nozzles, and the nuclear fuel. The new strainer is a passive component, and the only identified failure mode is structural failure. The strainer assembly was designed specifically for ONS.

The following text in Section 2.1 is excerpted and paraphrased from the Duke Power Company Engineering Change OD300050, Install Filter System on Unit 3 Emergency Sump [24]. Figures 2, 3 and 4 are schematics/drawings that represent the major features of the new sump design. These changes represented a significant part of the audit review. This review did not encompass ongoing design changes to the high-pressure injection pumps, low-pressure injection pumps, reactor building spray pumps, high-pressure injection system or the low-pressure injection system.

2.1 Reactor Building Emergency Sump Strainer Modification

The ONS design change removed the existing trash rack and screen assembly from the RBES and added a strainer assembly with a much larger filtering surface area as an enhancement under the current licensing basis.

The previous screen consisted of a grating “trash rack” and a filtering screen. The new strainer assembly consists of perforated metal to provide the filtering capability. To accommodate the new strainer, sump level instruments and the low-pressure injection (LPI) suction piping containment isolation valve test flange storage locations were relocated. As an aid to outage foreign material exclusion controls and to enhance personnel safety during sump access, a permanently installed strainer was added to the 4-inch sump drain.

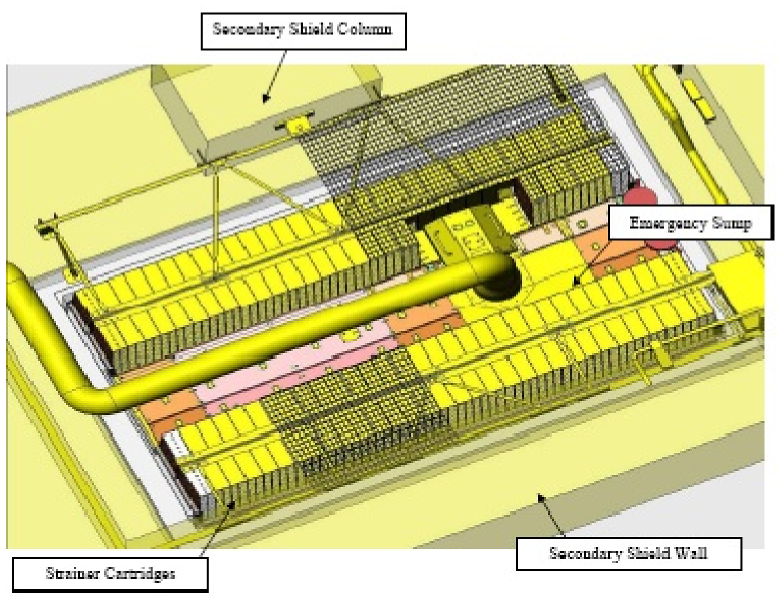

Figure 2 Schematic of New CCI Strainer Arrangement

The design change was not treated as a modification to meet a new methodology for evaluations of PWR

containment sumps. Separate activities will be performed to address the license basis and methodology changes after all analyses and modifications to support the new methodology are complete, thus establishing a connection between this design change and the final response to GL 2004-02.

2.1.1 Screen Replacement

The previous Unit 3 RBES screen/strainer assembly was described in the ONS LPI design basis document (OSS-0254.00-00-1028) as follows:

Each Unit at Oconee Nuclear Station is designed with an emergency sump which is provided with vertical screens around a rectangular frame. The screen area is approximately 100 ft^2 with 0.12" square mesh openings, and the trash rack area is approximately 40 ft^2.

.... The original B&W design criteria for the sump was that it permit a maximum average fluid velocity of approximately one foot per second when considering a 50% sump screen area blockage.... Without a postulated single failure, the head loss would be .... approximately 0.03 feet.

The trash racks were below the Reactor Building basement floor and covered during normal operation by a combination of checker-plate and grating.

The screens and trash racks were removed and replaced with a fabricated strainer assembly made of perforated stainless steel (see Figure 3). The perforations are 2.1 millimeters in diameter and fabricated with a +0.0/-0.1 mm tolerance, equivalent to an opening of 0.0787 to 0.0827 inches. The assembly is an array of strainer “pockets” extending from the bottom of the sump above the Reactor Building floor level by approximately three feet. The array of pockets are arranged in two sets of stacks separated by a center walkway above the low-pressure injection pump suction flanges which are located in the bottom of the sump. The floor sections of the walkway are removable to allow access to the suction flanges for containment leak rate testing. Grating was provided above the stacks to protect the strainers during inspections or other access, except for an approximately four foot long section of the stack in the northeast corner. Seal plates around the perimeter of the assembly and around the level instruments protect against bypass leakage of particles not passing through the strainer perforations. To ensure that particles and fibers larger than the strainer perforations do not bypass the strainer, the gap between connecting or interfacing strainer pieces was controlled to be 1.0 mm or less.

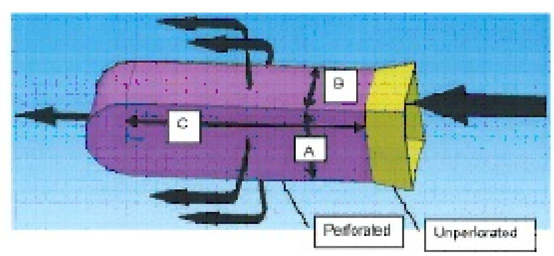

Figure 3 CCI Strainer Cartridge

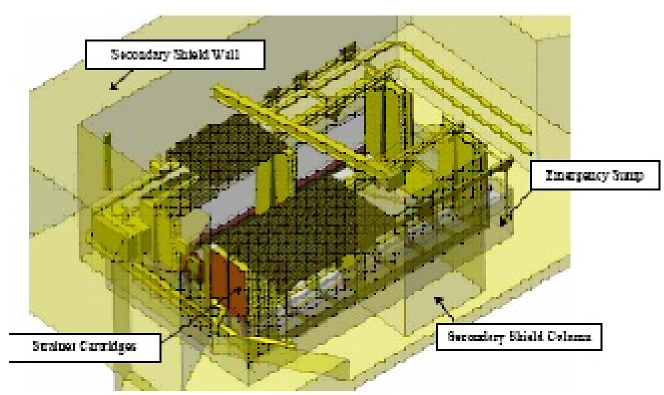

Figure 4 RBES Strainer Arrangement

The pocket openings are each approximately 4 by 3 inches and are arranged to form a grid array, which prevent large objects from entering the individual pockets where the water passes through the screen material. Each “pocket” contains approximately 204 in^2 of screen material, and the total effective screen area of the strainer assembly is approximately 5200 ft^2. The ends of each stack are covered with perforated strainer material. The screen area of the replacement strainer is more than 50 times larger than that of the previous screens. The head loss of the replacement strainers has been evaluated and is less than the specified limit of 0.1 foot, assuming 50 percent blockage (0.05 foot at design flow).

The previous configuration had a trash rack made of floor grating to separate large debris before the flow reached a screen surface. The new strainer does not have a separate trash rack, but the grid formed by the faces of the screen pockets performs an equivalent function, thus protecting the new screens from large debris.

The area of the previous trash rack was approximately 40 ft^2. The area of the grid openings on the new strainer is approximated as follows:

Individual Pocket Opening: 4.29" High by 2.76" Wide = 4.29 x 2.76 / 144 = 0.082 ft^2
Number of Pockets per Inner Stack: 13 High x 64 Wide x 2 stacks = 1664
Number of Pockets per Outer Stack: 15 High x 64 Wide x 2 stacks = 1920
Approximate Grid Area: (1664 +1920) x 0.082 ~ 294 ft^2

The head loss through the new strainer at 50% blocked is 0.05 ft of water. The head loss through the previous screen at 50% blocked was 0.03 ft of water.

Due to the large releases of water into the Reactor Building resulting from a postulated large-break LOCA, the sump can fill very quickly. To ensure that the new screen sections don’t lift during such rapid sump filling, air vents (small sections of perforated strainer material) are included in the sub-floor section design.

The flow conditions used for design of the strainer assembly included backflow from the borated water storage tank into the sump through one or both of the LPI suction lines. These flow conditions could create significant flow impingement loads on the floor of the strainer assembly in the areas above the LPI suction flanges. To protect the floor from such loads, flow impingement plates were installed above and mounted on the LPI suction flanges.

The new strainer assembly includes a support framework attached to the Reactor Building floor, sump floor and walls and nearby walls and columns.

The replacement strainer assembly is seismically qualified, including the flow impingement plates, as is the existing screen assembly. The replacement strainers are passive with no moving parts or power requirements.

The replacement strainer assembly, including supports and floor sections, is constructed of 304 stainless steel; welded parts are constructed of 304L stainless steel. Grating located above the strainer stacks is stainless steel.

2.1.2 Flow Impingement Plates / Test Flanges

The flow impingement plates are a new design feature. The flow impingement plates protect the strainer assembly floor from flow into the RBES through the LPI suction lines from the borated water storage tank. The strainer assembly, and therefore the flow impingement plates, is required to function during a LOCA with a break of any size. The plates were changed from carbon steel to stainless steel and are slightly thinner. Also, the test taps were relocated from the top of the plates to the edge. The plates have no safety-related function as part of the piping system pressure boundary. During normal operation, the plates are mounted above the LPI suction flanges. The head loss with the new flow impingement plates is included in the overall strainer head loss determination.

2.1.3 Level Instrument Relocation

Due to the configuration of the new strainer assembly, level instruments 3LPILT-0003P and 3LPILT-0112 were relocated from their current location on the sides of the sump to the end of the sump. The level instruments are designed to be Regulatory Guide 1.97 (Revision 2), Type B, Category 2 instrumentation, and seismically qualified.

2.1.4 Addition of 4-inch Drain Strainer

The capability exists to provide flow from the letdown storage tank to the RBES through the 4-inch sump drain line to allow post-LOCA flow from the tank to the RBES. As an aid to foreign material exclusion controls, and to enhance safety of personnel while working in the RBES, a strainer was installed over the 4-inch drain.

2.2 License Bases Changes

As a result of this change, wording modifications to the ONS Updated Final Safety Analysis Report, Safety Technical Specifications [25], Selected Licensing Commitments, and system Design Basis Documents were made. The wording changes primarily replaced the term "trash racks" with the term "strainer". The Selected Licensing Commitments and Design Basis Documents also included further descriptions of the new strainer assemblies and mirror the descriptions provided in the design change documents [22, 23, and 24].

3.0 BASELINE EVALUATION AND ANALYTICAL REFINEMENTS

3.1 Break Selection

The ONS calculational report prepared by Alion, entitled "GSI 191 Baseline Analysis for Oconee Units 1, 2, and 3," [57], documents the assumptions and methodology the licensee applied as part of the overall break selection process to determine the limiting break for the ONS units. The ONS baseline analysis evaluated a number of break locations that would require long-term recirculation to identify the location that presents the greatest challenge to the post-accident sump performance. All reactor coolant system (RCS) piping and unisolable connected piping was considered. Based on this evaluation, the licensee determined that postulated secondary side breaks would not require recirculation. Following a secondary side break, the primary

purpose of the safety injection systems is to ensure that the core remains covered during the RCS volume reduction caused by the initial cooldown and to provide additional reactivity to maintain the reactor in the subcritical mode. In addition, the ONS safety analyses show that safety injection is not required for boration or inventory control for longer than 10 minutes. The unaffected steam generator is used to remove decay heat until entry into residual heat removal mode. The licensee concluded that ECCS recirculation is not necessary to maintain long-term decay heat removal for either a main steam or feedwater line break.

The licensee break selection analysis identified two breaks in the 36-in. diameter hot leg piping as having the largest potential for generating debris. The licensee distinguished between a hot leg break nearer the reactor vessel and about mid-height with respect to the steam generator and a break at the top of the steam generator. These two hot leg locations became the two break selections analyzed by the licensee.

ONS Break 1: A break in the hot leg at the bottom of the Loop B steam generator was found to generate the largest quantity of debris. The pressurizer attaches to Loop B. The zone of influence (ZOI) associated with the reflective metallic insulation (RMI) on the RCS piping encompasses the majority of the steam generator compartment from this location.

ONS Break 2: A break in the hot leg near the top of the Loop A steam generator had the potential to generate fibrous debris from the fibrous insulation installed on the 3B Aux Cooler. However, subsequent to the completion of the ONS baseline break selection analysis, all fibrous insulation potentially within a ZOI, including the 3B Aux Cooler insulation, was replaced. Therefore, the licensee concluded that the Break 2 break selection is not as significant a break as Break 1.

The licensee concluded that the selection of Break 1 results in a conservative estimate for the RMI insulation debris for which the radius of a spherically shaped ZOI is considerably larger than the characteristic dimension of the steam generator compartment, other than the compartment height. No other break selection would result in the prediction of a larger quantity of RMI debris. The large-break LOCA high-energy piping is all located within the secondary shield wall. With the removal of all fibrous insulation from within a potential RCS ZOI, no fibrous insulation debris would be generated. The only potential small-break LOCA outside the secondary shield wall is the 2½-in. RCS to letdown heat exchanger piping, which is located well below any fibrous insulation. With respect to ZOI coatings debris, the thickest qualified coatings are associated with the concrete floors and walls for which more surface area exists near the bottoms of the steam generators than near the tops. Therefore, the licensee concluded that the selection of Break 1 for coatings debris also results in a conservative estimate for qualified coatings debris.

Staff Evaluation

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 Guidance Report (GR) [7] and associated NRC Staff Evaluation (SE) [8] provide the criteria to be considered in the overall break selection process in order to identify the limiting break. In general, the principal 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: debris generation, debris transport,

debris accumulation, and sump screen head loss. Two attributes of break selection that are emphasized in the approved evaluation methodology that 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, if applicable.

The NRC staff reviewed the licensee's overall break selection process and the methodology applied to identify the limiting break. Specifically, the NRC staff reviewed ONS Calculation No. ALION-REP-DUKE-2736-02 [57] against the approved methodology documented in Sections 3.3 and 4.2.1 of the SE and GR. The NRC staff found the licensee's break selection evaluation acceptable because it was generally performed in a manner consistent with the SE-approved methodology. Deviations from the staff-approved methodology were considered to be reasonable based on the technical basis provided by the licensee. A detailed discussion is provided below.

3.2 Debris Generation/Zone of Influence

ONS baseline calculational report, ALION-REP-DUKE-2736-02 [57] analyzed the debris generation from different break locations. The licensee's selected ZOI radius for insulation destruction was sized to the insulation with the lowest destruction pressure, which was Diamond Power Specialty Company Mirror insulation. The licensee applied the SE-approved radius of 28.6 pipe diameters (SE Table 3-2) corresponding to a destruction pressure of 2.4 psi. Destruction pressures have not been determined specifically for the National Thermal Insulation RMI. The licensee assumed that the destruction of National Thermal Insulation would not produce a greater quantity of transportable debris than that associated with the Mirror insulation. When the ZOI for LOCA Break 1 is overlaid onto composite piping plans, its comparison clearly demonstrates that the ZOI would encompass nearly the entire steam generator compartment. The licensee adapted the staff-accepted GR baseline guidance (SE Section 3.4.3.3) for the conservative two group size distribution of 75% small pieces and 25 percent large pieces.

In order to reduce fibrous debris, the licensee has replaced the insulation within the steam generation compartments with RMI. The licensee removed all fibrous insulation within reach of a postulated ZOI. The RMI is primarily Mirror insulation with standard banding manufactured by Diamond Power Specialty Company, with the exception of the RMI on the replacement steam generators that are insulated with National Thermal Insulation. All of the RMI is stainless steel. The licensee concluded that the combination of the conservatively large ZOI and the conservatively small RMI size distribution results in a conservative estimate of transportable RMI debris.

The identified potential LOCA-generated debris, other than the RMI debris, included fiberglass cable wrap and some caulking. Some fiberglass cable wrap installed on nuclear instrument cables was identified, for which complete destruction would result in approximately 0.4 ft^3 of fibrous debris for each of two cables (0.8 ft^3 total). The licensee assumed that adhesive and room-temperature vulcanizing (RTV) caulking located within a ZOI would fail as particulate debris resulting in 30.5 lbs of particulate debris. Both the fiberglass cable wrap and the caulking particulate were assumed to be fines that would transport completely to the strainers.

The LOCA-generated debris was specifically evaluated for ONS Unit 3 and assumed to apply to Units 1 and 2 as well. Because the licensee has removed all fibrous insulation from the ZOIs of all three units, the primary source of fibrous debris is latent debris, for which separate walkdown reports were prepared for each unit. The LOCA-generated debris quantities (less coating debris) are summarized in Table 2.

Table 2 Bounding LOCA-Generated Debris Quantities (Less Coatings)		
Debris Type	Debris Form	Quantity
RMI	Metallic	113,415 ft^2
NI Fiberglass Cable Wrap	Fibrous	0.8 ft^3
Paint Caulking	Particulate	30.5 lbs

Other sources of potential debris at ONS include latent debris and coatings debris, which are discussed in Sections 3.4 (pg 18) and 3.8 (pg 56), respectively, and chemical effects precipitants, which are addressed in Section 5.4 (pg 72).

Staff Evaluation

The objective of the debris generation/ZOI process is to determine, for each postulated break location: (1) the zone within which the break jet forces would be sufficient to damage materials and create debris; (2) the amount of debris generated by the break jet forces; and (3) the size characteristics of the postulated debris. Sections 3.4 and 4.2.2 of the GR [7] and the NRC SE [8] provide the methodology to be considered in the ZOI and debris generation analytical process.

The staff reviewed the licensee's ZOI and debris generation evaluations and the methodology applied. Specifically, the staff reviewed the ONS baseline calculational report [57] against the approved methodology documented in sections 3.4 and 4.2.2 of the staff's SE. Because the licensee has removed the fibrous insulation material within ZOIs from all three units, the licensee’s ZOI analysis focused on RMI debris generation, while three walkdown reports provided the debris source of latent fiber and particulate. The licensee applied the material-specific damage pressure for Diamond Power Specialty Company Mirror insulation and the corresponding ZOI radius/break diameter ratio as shown in Table 3-2 of the staff SE. The licensee also assumed that the RMI insulation material made by National Thermal Insulation has the same destruction characteristics as the Diamond Power Specialty Company Mirror RMI. Because RMI debris did not contribute to the head loss during the head loss testing, the staff considers that this assumption is not important, and that the licensee’s approach is reasonable. Therefore, the NRC staff found the licensee's evaluation to be consistent with the approved methodology and that the licensee provided an adequate level of technical justification with respect to ZOI analyses.

3.3 Debris Characteristics

The staff reviewed the ONS licensee's assumptions regarding the characteristics of post-accident debris to provide assurance that the assumed characteristics are conservative with respect to debris transport, debris bed head loss, and other areas of the sump performance analysis. The licensee's discussion of debris characteristics was primarily provided in the baseline analysis report [35]. Additional information concerning the post-accident debris at ONS was provided in the licensee's slide presentation for the audit kick-off meeting [3].

The analyzed debris loading for ONS includes RMI, qualified coatings, unqualified coatings, latent particulate debris, latent fibrous debris, fibrous debris from nuclear instrumentation cable wraps, and foreign materials such as tape, tags, and stickers [3, 35]. The following subsections of this audit report describe the licensee's assumptions regarding the characteristics of these types of debris, with the exception of the characteristics of coatings debris, which are discussed separately in Section 3.8 (pg 56). A summary of the assumed characteristics for non-coatings debris is provided below in Table 3.

Table 3 Summary of Assumed Characteristics for Non-Coatings Debris [35]

Debris Type	Size Distribution	Macroscopic Density (lb_m/ft^3)	Microscopic Density (lb_m/ft^3)	Characteristic Size (µm)
Reflective Metallic Insulation	75% Small Pieces 25% Large Pieces	Not Reported	Not Reported	Not Reported
Fibrous Cable Wrap	100% Small Fines	2.4	94	7.1
Latent Fiber	100% Small Fines	2.4	94	7.1
Latent Particulate	100% Fine Particulate	Not Reported	168	17.3
Caulk Particulate	100% Fine Particulate	Not Reported	65	17.3

3.3.1 Reflective Metallic Insulation

The licensee's baseline analysis stated that the RMI installed at ONS is primarily Diamond Power Mirror Insulation constructed of stainless steel foils with a thickness of 2 mils [35]. However, the licensee noted that the replacement steam generators have been insulated with RMI manufactured by National Thermal Insulation, which was designed and constructed to meet the original Diamond Power insulation drawings [35]. The licensee stated that the size distribution assumed for the RMI debris at ONS is 75% small fines and 25% large pieces, which is consistent with the guidance in NEI 04-07 [7] and Table 3-3 of the staff's safety evaluation (SE) on NEI 04-07 [8]. The staff considers the licensee's assumed size distribution for RMI debris to be acceptable because the assumed distribution is consistent with conservative guidance provided in NEI 04-07 and the staff's SE.

3.3.2 Fibrous Insulation

The licensee's baseline analysis stated that all fibrous insulation in the ONS Reactor Building is located outside of the secondary shield wall [35]. The baseline analysis stated that RCS piping extends outside the secondary shield wall to the letdown coolers, but that all fibrous insulation in the vicinity of the high-energy RCS piping has been removed [35]. The baseline analysis also stated that there is fibrous insulation outside the secondary shield wall on Reactor Building Cooling Unit 3B that could be affected by a pipe rupture at the top of Steam Generator 3A [35]. However, the licensee stated during the onsite audit that this fibrous insulation was replaced subsequent to the completion of the baseline analysis. Based upon the information summarized above, the licensee stated during the audit that there is no fibrous insulation within the ZOI of analyzed pipe ruptures in Reactor Building that require sump recirculation.

In response to a question from the NRC staff, the licensee stated that there is roughly 650 ft^3 of fibrous insulation in the Reactor Building outside of ZOIs for analyzed pipe ruptures. To verify that this fibrous insulation would not be subject to erosion from Reactor Building sprays, the staff further questioned whether the fibrous insulation outside of analyzed ZOIs was protected by jacketing or covers. In response, the licensee stated that this fibrous insulation likely has jacketing or a covering that would prevent its erosion into fines under exposure to Reactor Building spray drainage; however, the licensee could not confirm this statement during the audit.

Following the onsite audit, the licensee provided additional information concerning the fibrous insulation in Reactor Building that is outside of ZOIs. The licensee stated that the majority of this fibrous insulation is jacketed with stainless steel and supported this statement with photographs that had been taken during a Reactor Building walkdown. The licensee also stated that some elbows and short piping runs are not jacketed with stainless steel. Although the information presented by the licensee partially addressed the staff's question, it did not fully address whether the minority of the fibrous insulation outside of ZOIs that lacks stainless steel jacketing could be eroded by sprays after an accident and generate fine fibrous debris that could adversely affect the performance of the replacement strainers. From an additional follow-on discussion with the licensee, the staff understood that the non-stainless-steel-jacketed fibrous insulation was actually coated with a hard substance. The licensee could not confirm whether or not the hard coating would protect the fibrous insulation from Reactor Building spray flows. Therefore, the staff designated it **Open Item 3.3-1** for the licensee to provide adequate justification that unjacketed fibrous insulation outside of the analyzed ZOIs would not adversely impact the performance of the RBES as the result of erosion induced by Reactor Building spray drainage. The staff noted during the discussion with the licensee that this issue could be resolved in a number of ways, including the following: (1) demonstrating that the quantity of unjacketed fibrous insulation is below the threshold of concern for the replacement strainers, (2) demonstrating that the hard coating on the unjacketed insulation is sufficiently robust to resist erosion from Reactor Building spray, and (3) demonstrating that the Reactor Building sprays would not be run long enough to generate adverse quantities of eroded fibers.

3.3.3 Fibrous Cable Wrap

Since the licensee stated that there is no fibrous insulation within the ZOI of analyzed pipe ruptures, the only fibrous material generated within an analyzed ZOI at ONS is a small quantity

of fiber (0.8 ft^3) from nuclear instrumentation cable wraps [35]. The staff considered the licensee's assumed size distribution of 100 percent small fines for the 0.8 ft^3 of fibrous cable wrap to be conservative, since this assumption tends to maximize transport potential and accumulation on the sump strainers.

3.3.4 Latent Fibrous Debris

The licensee assumed that latent fiber comprises 15 percent of the total latent debris loading measured in the Reactor Building. The licensee assumed that latent fibrous debris is composed of 100% small fines. The assumed physical properties for latent fibrous debris are listed above in Table 3.

The properties the licensee assumed for latent fibrous debris are consistent with NUREG/CR-6877 [39] and the NRC SE on NEI 04-07 [8]. Therefore, the staff considers the characteristics assumed for latent fibrous debris to be acceptable.

3.3.5 Latent Particulate Debris

The licensee assumed that latent particulate comprises 85% of the total latent debris loading measured in the Reactor Building. The licensee assumed that latent particulate debris is composed of 100% fine particulate. The assumed physical properties for latent particulate debris are listed above in Table 3.

The properties the licensee assumed for latent particulate debris are consistent with NUREG/CR-6877 [39] and the NRC SE on NEI 04-07 [8]. Therefore, the staff considers the characteristics assumed for latent particulate debris to be acceptable.

3.3.6 Caulk

The baseline analysis considered adhesive caulk and RTV caulk as potential sources of post-accident debris in the Reactor Building. The baseline analysis stated that all caulk within the ZOI of an analyzed piping rupture will be assumed to fail as fine particulate. The licensee stated that adhesive caulk outside the ZOI will be presumed to fail intact and act as surface blockage at the sump strainers. The licensee stated that RTV caulk outside the ZOI will be assumed not to fail based on plant data that indicates that this material is resilient and will remain intact under design-basis post-accident conditions.

The staff considers the licensee's assumption that caulk within the ZOI of a pipe rupture forms 100 percent fine particulate to be conservative. The density of approximately 65 lb_m/ft^3 used for both types of caulk is based upon data from the manufacturer. The staff considered it conservative that adhesive caulk outside the ZOI was considered to fail intact and result in blockage at the sump screen. Based upon these conservatisms, the staff considers the licensee's assumed characteristics for caulk debris to be acceptable. The staff did not evaluate the licensee's data supporting its assumption that caulk outside the ZOI will not fail. However, the staff believes the assumption is reasonable based on general properties of this material, and the other assumptions regarding caulk are sufficiently conservative that the validity of this assumption would not be expected to significantly affect sump performance.

3.3.7 Miscellaneous Debris

The licensee's baseline analysis states that miscellaneous debris sources that could result in sump strainer blockage include metal equipment tags, duct tape, vinyl stickers, plastic tags, plastic covers, and plastic tie wraps [35]. According to Reactor Building walkdown reports, the total quantity of miscellaneous debris was approximately 129 ft^2. The licensee assumed that this material fully transported to the sump strainers and accumulated in such a manner as to block an area on the sump strainers equivalent to 75 percent of the singled-sided surface area of the miscellaneous debris (i.e., 97 ft^2) [35].

The licensee's methodology for characterizing nonporous latent debris by assuming it blocks an area on the sump strainer equal to 75% of the total single-sided area of miscellaneous debris is consistent with Section 3.5.2.2.2 of the staff's SE on NEI 04-07 [8]. Therefore, the staff considers this methodology to be acceptable.

3.3.8 Debris Characteristics Conclusion

The licensee generally made conservative assumptions concerning the characteristics of debris at ONS. However, the staff concluded that adequate justification had not been presented by the licensee to demonstrate that unjacketed fibrous insulation outside of analyzed ZOIs would not be eroded by Reactor Building sprays, potentially resulting in an adverse impact on the performance of the RBES strainer. As a result, the staff designated this issue as Open Item 3.3-1. With the exception of this open item, as described above, the staff considered the debris characteristics assumed by the licensee to be acceptable.

3.4 Latent Debris

The objective of the latent debris evaluation is to quantify the amounts and types of latent debris within the Reactor Building for the ultimate purpose of assessing the impact of this debris on the performance of the RBES. The licensee performed an evaluation of the potential sources of latent debris within the Reactor Building, as recommended by NEI 04-07 [7] and the associated NRC safety evaluation (SE) [8]. The approved baseline methodology for quantifying the mass and characteristics of latent debris inside containment includes the following basic steps: (1) estimating the total containment surface area, including both horizontal and vertical area contributions, (2) surveying the containment to determine the mass of latent debris present, (3) defining the composition and physical properties of the latent debris, (4) determining the fraction of total containment surface area that is susceptible to latent debris buildup, and (5) calculating the total quantity and composition of the latent debris.

As described below, the staff reviewed the methodology, measurements, and calculations that the licensee used to quantify the amounts and characteristics of latent debris in the Reactor Building [40, 58, 59, 57].

3.4.1 Latent Debris Sampling Methodology

The latent debris assessment for all three ONS units relies on data taken largely during the Unit 3 Reactor Building walkdown [40]. This walkdown provided the data used to assess (1) the mass of dirt, dust, and lint within the Reactor Building and (2) the surface area of tape, equipment labels, and other miscellaneous Reactor Building debris (e.g., tie wraps and

caulking). Thirteen tape and label categories were defined, and the walkdown provided data to quantify the amount of debris in each category that was found in each area of the Reactor Building during the walkdown.

The sampling methodology for dirt, dust, and lint is described in the Reactor Building walkdown report for Unit 3 [40]. The surfaces inside containment were divided into vertical and horizontal surface categories, and the surface area of each category was estimated. Ten surface area categories were defined, and the areas associated with each category were calculated [40]. The Reactor Building was surveyed to determine the mass of latent debris present. Four samples were taken using a vacuum cleaning device. Masses of debris were obtained from pre-weight and post-weight measurements. An area of 20 ft^2 was sampled at each location. The average mass of debris per unit area for each one of the four samples was computed. One of the four sampling locations was associated with each of the ten surface categories. The mass for each surface category was computed using the total area for the surface category and the average mass of debris per unit area for the associated sampling location. Ten percent of the vertical surface area was used, as per the guidance in NEI 04-07 [7]. The total mass of latent debris was then obtained by summing the masses computed for each surface category.

The licensee's methodology generally follows the guidance of NEI 04-07 [7] and the staff's SE [8] and is therefore generally acceptable. However, the staff concluded that the use of only four latent debris samples is inadequate to represent the distribution of latent debris in Reactor Building and does not satisfy the intent of Section 3.5.2.2 of NEI 04-07 and the staff's SE, which states that while statistical sample mass collection is an acceptable method for quantifying latent debris mass, several classes of horizontal and vertical surfaces should be sampled, and at least three samples should be taken from each category [9]. The licensee should improve the existing estimate of latent dirt, fiber and dust mass in the Reactor Building by using an adequate number of samples to characterize the spectrum of debris loadings on various categories of containment surfaces. The guidance provided in [9] should be used for selection of the classes of surfaces to be sampled, for the number of samples to be taken for each class, and for defining the characteristics of the sampling device. This issue is designated as **Open Item 3.4-1**.

3.4.2 Latent Debris Quantities

For estimating the quantities of latent fibers and particulate in the Reactor Building, no specific sampling was performed for Units 1 and 2; rather, data from the four samples collected for Unit 3 was applied to these two units. The total mass of latent debris calculated based upon the samples taken from Unit 3 was 95.8 lb_m, for which the licensee followed the SE [8] recommendation that 15 percent of the latent debris be considered as fiber and 85 percent as particulate. This assumption results in 14.4 lb_m of latent fiber and 81.4 lb_m of latent particulate. Because these estimates of latent debris mass are based on sampling from only four collection areas, a significant degree of uncertainty is associated with the licensee's latent debris assessment. One area in which this uncertainty could have significance is the determination of whether or not sufficient fiber exists to form a debris bed capable of effectively filtering chemical precipitates and other debris. The licensee's resolution of Open Item 3.4-1 (see above), in conjunction with the licensee's existing practice of performing Reactor Building washdowns at the beginning and end of each refueling outage, will provide assurance that uncertainties associated with the licensee's existing latent debris estimate do not have an adverse impact on recirculation sump performance.

The licensee's assessment of other miscellaneous debris also depended primarily on the Unit 3 walkdown. Conservative estimates were provided for the amount of miscellaneous Reactor Building debris, which are summarized below in Table 4.

Table 4 ONS Miscellaneous Debris Estimated from Walkdowns

DebrisType	Quantity	Type
Nuclear Instrumentation Cable Wraps	0.75 ft^3	Fibrous
White RTV Caulking	11.02 lb_m	Particulate for breaks within the ZOI [57]
Black Polysilicone Caulking	3.75 ft^2	Particulate for breaks within the ZOI [57]
Tape and Labels	209 ft^2	Treated as sheet-like material
Tie Wraps	20.8 ft^2	Treated as sheet-like material

The staff noted two areas where Unit 2 walkdown data for miscellaneous debris differed from Unit 3:

- The quantity of white RTV caulking assessed for Unit 2 was 57.06 lb_m compared to 11.02 lb_m for Unit 3. The head loss testing specifications were based on the lower value from Unit 3.

- The Unit 2 walkdown report noted 0.41 ft^3 of fibrous material in Reactor Building penetrations [59], but the licensee stated during the onsite audit that all of this fibrous material was since removed.

The staff expects that the licensee's future head loss testing with chemical precipitates will account for any discrepancies between the walkdown data for different units to ensure that bounding debris quantities are analyzed for all three ONS units.

3.4.3 Latent Debris Summary

The licensee's latent debris assessment is generally acceptable because it is generally consistent with the guidance of NEI 04-07 [7] and the staff's SE [8]. However, the staff considered the number of latent debris samples taken by the licensee to be inadequate for obtaining a sufficiently accurate estimate of Reactor Building latent fibrous and particulate debris masses, given the spectrum of debris buildup conditions on various Reactor Building surfaces. The staff designated this issue as Open Item 3.4.1.

3.5 Debris Transport

Debris transport analysis estimates the fraction of post-accident Reactor Building debris that would transport to the RBES strainers. Debris transport would occur through four major modes:

- blowdown transport, which is the vertical and horizontal transport of debris throughout the Reactor Building by the break jet,
- washdown spray transport, which is the downward transport of debris by the Reactor Building sprays and break flow,
- pool-fill-up transport, which is the horizontal transport of debris along the Reactor Building floor by rapidly flowing sheets of water to areas that may be active or inactive during recirculation flow, and
- Reactor Building pool recirculation transport, which is the horizontal transport of debris from the active portions of the Reactor Building pool to the suction strainers through pool flows induced by the operation of the ECCS and RBS system in recirculation mode.

Through the blowdown mode, some debris would be transported throughout the lower and upper Reactor Building. Through the washdown mode, a fraction of the debris in the upper Reactor Building would be washed down to the Reactor Building floor. Through the pool-fill-up mode, debris on the Reactor Building floor would be scattered around, and some debris could be washed into inactive pool 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 to the suction strainers. Through the recirculation mode, a fraction of the debris in the active portions of the Reactor Building pool would be transported to the suction strainers.

The licensee's existing debris transport calculation is contained in the baseline analysis report [35]. The debris transport calculation in the baseline analysis generally follows conservative guidance from the baseline methodology provided in NEI 04-07 [7] and the staff's SE on NEI 04-07 [8]. Exceptions are discussed below.

Subsequent to the completion of the baseline analysis, the licensee performed additional analyses to assess the transportability of coating debris, including a non-quality-assured (non-QA) computational fluid dynamics (CFD) analysis of the flow in the Reactor Building pool during the recirculation phase of a LOCA and transport testing of coating debris in a flume. During the audit, the staff conducted a limited review of the licensee's non-QA CFD analysis [37] and draft report on coating chip transport testing [36]. The licensee stated that an additional quality-assured (QA) CFD analysis is being undertaken to provide formal CFD results that could be credited in the strainer performance analysis; however, this calculation is currently in development and was not available for staff review during the audit. Through these additional analyses, the licensee expects to establish reduced coating debris quantities for future head loss testing or to establish margin for a screen designed to accommodate the existing debris loading.

The staff's debris transport discussion below reviews (1) the licensee's baseline analysis transport calculation, (2) the licensee's non-QA CFD analysis, and (3) the licensee's coating debris transport test program.

3.5.1 Baseline Analysis Debris Transport Evaluation

The baseline analysis states that the licensee analyzed debris transport for two LOCA scenarios [35]. The first scenario was a 36-inch break on the RCS hot leg in the "B" Loop, and the second was a 36-inch break on the RCS hot leg in the "A" Loop [35]. The break in the "B" Loop had been selected because the "B" Loop contains the pressurizer, which would increase the quantity of debris generated by this break in comparison to a similar break in the "A" Loop. The break in the "A" Loop was chosen based upon the presence of fibrous debris on an auxiliary cooler; however, as noted in Section 3.3.2, the licensee stated that this fibrous debris was removed after the baseline analysis was completed. The licensee's discussion of break selection does not explicitly address whether consideration was given to breaks near the recirculation sump which could lead to increased transport fractions.

Based upon the conservative transport assumptions incorporated into the baseline analysis, the staff does not consider the lack of explicit consideration of a pipe rupture near the sump to be significant. As shown in Tables 5 and 6 below, 100 percent of almost all types of debris was assumed to transport to the sump strainers. Therefore, the baseline analysis break selection is appropriate with respect to debris transport. However, in light of the fact that the licensee is in the process of conducting additional analyses that may lead to reductions in the transport fraction for coating debris, additional consideration may be appropriate at the completion of this process to ensure that the breaks analyzed in the existing analysis remain bounding.

The baseline analysis's approach in analyzing debris transport incorporated guidance from NEI 04-07 [7], using assumptions primarily taken from the baseline methodology. The licensee's baseline analysis made conservative debris transport assumptions regarding the fraction of debris in the Reactor Building pool that transports to the sump strainers during recirculation in lieu of using computational fluid dynamics to generate a detailed model of Reactor Building pool flows during recirculation. In particular, the licensee assumed 100 percent transport for all debris types except RMI.

The following subsections discuss the baseline analysis transport methodology in detail, and provide the staff's assessment of the licensee's methodology. Note that the following detailed discussions concerning blowdown and washdown transport (Section 3.5.1.1), pool-fill transport (Section 3.5.1.2), and recirculation transport (Section 3.5.1.3) are only applicable to RMI debris because 100% transport was assumed for all other types of debris in lieu of analysis.

3.5.1.1 Blowdown and Washdown Transport for RMI Debris

The baseline analysis assumed that all RMI debris would be blown down directly to the Reactor Building pool [35]. The licensee stated that this assumption is based on guidance in Section 3.6.3.2 of NEI 04-07 [7], which indicates that small pieces of RMI insulation should be assumed to be blown down to the Reactor Building floor for mostly uncompartmentalized containments. The licensee stated that the ONS containments are mostly uncompartmentalized. As RMI debris is already assumed to be in the containment pool following blowdown, washdown is assumed to have no effect on debris transport [35].

The staff considers the licensee's position that all RMI debris is blown down directly into the Reactor Building pool to be conservative for the purpose of sump strainer sizing because this assumption tends to maximize the quantity of debris reaching the sump strainers. However, this

assumption is not considered realistic, and would not be generally considered acceptable for other purposes, such as for assessing the potential for debris to block choke points in containment such as refueling cavity drains.

3.5.1.2 Pool-Fill-Up Transport for RMI Debris

Based upon Section 3.6.3.2 of NEI 04-07 [7], the baseline analysis states that small pieces of debris will be transported to all areas of the Reactor Building pool, and that some of the small debris pieces will be transported to inactive sumps and other volumes (e.g., the cavity beneath the reactor vessel) that are not significantly perturbed by flow during sump recirculation [35]. The licensee assumed that 3% of small pieces of RMI debris would be sequestered in inactive Reactor Building volumes during the pool-fill-up phase based upon a calculation that determined that inactive volumes constitute slightly less than 3% of the total pool volume at ONS [35]. The staff observed that the licensee's methodology for calculating RMI transport during the Reactor Building pool-fill-up phase is generally consistent with NEI 04-07 [7]. However, the staff noted that, in the same manner as a fraction of the RMI debris may transport to inactive containment volumes during the pool-fill-up phase, some RMI debris may transport directly to the RBES as the Reactor Building pool is filling. The licensee did not explicitly address this potential effect in the baseline analysis [35]. However, the staff expects that the quantity of RMI debris that would transport directly to the RBES is sufficiently small as to be bounded by the substantial conservatisms incorporated into the licensee's baseline analysis transport analysis for RMI during recirculation mode (see Section 3.5.1.3 below). As a result, the staff considered the licensee’s evaluation of debris transport during the pool-fill-up phase to be acceptable.

3.5.1.3 Recirculation Transport for RMI Debris

The baseline analysis states that the baseline guidance in NEI 04-07 [7] recommends that 100 percent of small pieces of debris be assumed to transport to the RBES during recirculation [35]. Although NEI 04-07 further recommends that the transport of large debris be neglected, the NRC staff's SE rejected this position, stating that the transport of large pieces of debris should be explicitly evaluated [8].

In light of this guidance, the licensee assumed 100 percent transport of the available small pieces of RMI during recirculation and performed a simplified analysis in an attempt to show that large pieces of RMI would not reach the sump strainers. The licensee's analysis considered a 13-foot-wide opening, which was stated to be the minimum cross-sectional flow area along the path from the break to the sump, and calculated an average bulk velocity at this location of 0.22 ft/s at the minimum Reactor Building pool height. Based on a comparison of this value to transport test data for RMI debris, the licensee stated that large pieces of RMI debris would not be expected to transport to the strainers and assumed that none of this debris would reach the strainers.

Although, as discussed below, the staff considered the licensee's analysis of RMI debris during the recirculation transport phase to be conservative overall, the staff questioned whether the licensee's methodology for neglecting the transport of large pieces of RMI debris was technically justified. The methodology employed is effectively a simplification of the nodal analysis approach for computing fluid flow, which, as noted in the staff's SE on NEI 04-07 [8], does not consider velocity distributions associated with real fluid flows. To assess whether the licensee's simplified calculation of average flow velocity had non-conservative results, the staff compared

the licensee's simplified flow analysis to contour plots of Reactor Building pool velocity that had been generated by the non-QA CFD analysis performed by the licensee [37]. The CFD analysis showed that, for several CFD scenarios, continuous flowpaths between the pipe break and the RBES experienced flow velocities exceeding 0.22 ft/s, including a maximum continuous velocity for one scenario of over 0.3 ft/s.

Although the licensee’s simplified CFD calculation underestimated the maximum flow velocity approaching the strainer for certain scenarios, the staff's review of flow velocity contour plots from the licensee's non-QA CFD calculation [37] indicated that the calculated RMI transport fraction is still conservative overall. While the NEI 04-07 [7] baseline methodology led to the licensee computing a 73 percent transport percentage for RMI debris, the velocity contour plots showed that substantial portions of the Reactor Building pool would have velocities significantly lower than the incipient tumbling velocity of 0.28 ft/s for RMI debris reported in NUREG/CR-6772 [14], even in the most limiting cases [37]. As a result, a significant fraction of the small pieces of RMI debris conservatively assumed to transport to the sump strainers would realistically not be capable of reaching the strainers. Additionally, as discussed further in Section 3.6 of this audit report (pg 30) and in the licensee's large-scale head loss test report [52], head loss tests for ONS that included RMI debris had lower measured head losses than those conducted without RMI debris. Based upon these observations, the staff considers the licensee's transport results for RMI debris during recirculation to be acceptable.

3.5.1.4 Transported Debris Quantities in the Baseline Analysis

Using the methodology described above, the baseline analysis calculated the quantities of debris transporting to the sump strainers that are provided in Tables 5 and 6 below [35].

Table 5 Debris Quantities from Hot Leg, Loop “B” [35, 3]

Debris Type	Quantity Destroyed	Transport Fraction	Transported Quantity
RMI	113,415 ft^2	0.73	82,739 ft^2
Qualified Coatings	1,466.3 lb_m	1	1,466.3 lb_m
Unqualified Coatings	437.4 lb_m	1	437.4 lb_m
Latent Particulate Dirt / Dust	81.4 lb_m	1	81.4 lb_m
RTV Caulk Particulate	30.5 lb_m	1	30.5 lb_m
Latent Fiber	6 ft^3	1	6 ft^3
Cable Insulation Fiber	0.8 ft^3	1	0.8 ft^3
Tape, Labels, Cable Ties, etc.	97 ft^2	1	97 ft^2

Table 6 Debris Quantities from Hot Leg, Loop "A"			
Debris Type	Quantity Destroyed	Transport Fraction	Transported Quantity
RMI	90,665 ft^2	0.73	66,185 ft^2
Qualified Coatings	873.8 lb_m	1	873.8 lb_m
Unqualified Coatings	437.4 lb_m	1	437.4 lb_m
Latent Particulate Dirt / Dust	81.4 lb_m	1	81.4 lb_m
Latent Fiber	6 ft^3	1	6 ft^3
Cable Insulation Fiber	0.8 ft^3	1	0.8 ft^3
Tape, Labels, Cable Ties, etc.	97 ft^2	1	97 ft^2

Tables 5 and 6 show that for all types of debris other than RMI, 100 percent transport was conservatively assumed. For RMI debris, 73 percent transport was calculated, which the staff also considered conservative based upon the discussion in Section 3.5.1.3 above.

Table 7 compares the quantities of debris currently analyzed as reaching the sump strainers (as reported the licensee's slide presentation for the pre-audit kick-off meeting [3]) to the debris quantities that were assumed for the licensee's previous large-scale head loss testing [52]. This head loss testing was performed in 2005 by CCI, and did not include chemical precipitates.

Table 7 Comparison of Current Strainer Debris Loading to Tested Quantities		
Debris Type	Quantity Analyzed to be at Sump	Quantity Assumed for Head Loss Testing
RMI	82,793 ft^2	111,148 ft^2
PARTICULATE DEBRIS		
Qualified Coatings	2,006.9 lb_m	1,051.5 lb_m
Unqualified Coatings	437.4 lb_m	600.3 lb_m
Latent Particulate Dirt / Dust	107.1 lb_m	100 lb_m
RTV Caulk	41.6 lb_m	11.1 lb_m
TOTAL PARTICULATE	2593 lb_m	1762.9 lb_m

Table 7 Comparison of Current Strainer Debris Loading to Tested Quantities		
Debris Type	Quantity Analyzed to be at Sump	Quantity Assumed for Head Loss Testing
FIBROUS DEBRIS		
Latent Fiber	7.875 ft^3	6.25 ft^3
Cable Insulation Fiber	0.8 ft^3	1 ft^3
Fibrous Insulation	0 ft^3	16.7 ft^3
TOTAL FIBER	8.675 ft^3	23.95 ft^3
Tape, Tags, Labels, Cable Ties, etc.	325 ft^2	325 ft^2 [a]
[a] Test strainer area scaling subtracted the area of the surface debris from the replacement strainer area prior to determining the test module specifications.		

Table 7 shows that, for debris types other than particulate, the quantities of debris used for head loss testing bounded the quantities calculated to reach the sump strainers using the conservative methodology described above. In particular, the quantity of fibrous debris used for head loss testing was approximately 2.7 times greater than the quantity currently analyzed as reaching the strainers. On the other hand, the amount of particulate debris used for head loss testing is only about 2/3 of the currently analyzed quantity, primarily due to an increase in the quantity of qualified coatings assumed to fail during a LOCA. Recognizing these discrepancies, the licensee stated that the baseline analysis was intended to provide a "snapshot in time" of the RBES performance analysis. The licensee verbally stated during the audit that a "living document" would be generated subsequently to capture the existing configuration of the plant into the future.

As discussed further in Section 3.6 of this report (pg 30), the licensee is planning to perform additional head loss testing with chemical precipitates to complete the strainer qualification process. The licensee stated during the onsite audit that the planned test will conservatively account for the current quantity of failed coatings and other post-LOCA debris. Specifically, the staff understood that the licensee's intention is to add over 200 percent of the currently analyzed quantity of coating debris for the head loss test in the form of chips ranging in size from less than 75 μm to 4 mm. Note that, for the initial testing, all of the coating debris had been represented with fine particulate surrogate debris.

The staff considers the use of 200 percent of the expected coating debris loading for the planned head loss test to be acceptable, since it conservatively bounds the expected quantity of coating debris for ONS. The acceptability of using coating chips having the size distributions the licensee is considering is discussed separately in Section 3.8 of this audit report (pg 56). Thus, in light of the licensee's existing plans to conduct additional head loss testing with a bounding

quantity of coating debris, the staff did not designate the quantity of coating debris used in the licensee's previous head loss tests as an open item.

3.5.2 Non-Quality-Assured Computational Fluid Dynamics Calculation

The staff reviewed a non-QA CFD calculation performed by the licensee to assess the flow pattern in the Reactor Building pool during the recirculation phase of a design-basis LOCA [37]. Since the CFD calculation was non-QA, the licensee did not credit the calculation as part of the strainer design basis. However, based upon discussions with the licensee during the onsite audit, the staff understood that the licensee is currently developing a separate QA CFD calculation. In conjunction with the coating debris transport testing described below, the staff understood that the QA CFD calculation may be used as part of the strainer design basis or to establish strainer design margin. As a result, the staff reviewed the non-QA CFD calculation in order to provide feedback that the licensee could consider in developing the QA CFD calculation.

The non-QA CFD simulation had been performed with the CFX code by ANSYS, the code vendor. Based on a limited review, the staff concluded the calculation was reasonable overall and considered it an effective scoping study to assess Reactor Building pool flows during the recirculation phase of a LOCA. Specific staff comments on the non-QA CFD calculation were provided to the licensee during the audit for consideration with respect to the QA CFD calculation that is being developed. These comments are summarized in the list below. The staff recognized that, depending upon how the results of the QA CFD calculation are used and the overall conservatism in the licensee's methodology, some of the comments may be either inapplicable or below the threshold of significance with respect to the QA CFD analysis. Since the non-QA CFD simulation is not being relied upon to support the replacement sump strainer design basis, none of the comments made by the staff are designated as open items in this audit report.

- A detailed description of the methodology underlying the CFD results was not included in the non-QA calculation [37]. While a detailed description of the methodology is not necessary for a scoping study, the staff expects that QA CFD calculations used to demonstrate the adequacy of the sump strainer design will include a description of the methodology used, such as boundary conditions, underlying assumptions, and accompanying technical justifications.

- The water level used in the non-QA CFD calculation was 4 ft [37], which is conservatively lower than the value of 4.4 ft stated in the baseline analysis [35].

- The non-QA CFD analysis considered twelve different cases, and, in half of these cases, blockage of floor drains was explicitly modeled [37]. Consideration of multiple break locations and drain blockage conditions increases confidence that the limiting Reactor Building pool flow conditions will be addressed by the analysis.

- Although the total number of cells in the non-QA CFD model appeared reasonable, adequate justification was not presented to demonstrate that the mesh had been adequately refined.

- The non-QA CFD model did not explicitly model the influx of kinetic energy from fluid falling into the Reactor Building pool (i.e., drainage from the pipe break and Reactor

Building sprays) [37]. Kinetic energy from incoming fluid may affect the turbulence and velocity profiles of the Reactor Building pool flow in a localized area surrounding the fluid's entry point.

- The non-QA CFD calculation included a fairly detailed model to account for the locations where RBS drainage would enter the Reactor Building pool [37]. At most spray drainage entry locations, the drainage appeared to have been treated as a uniform, dispersed flow. Based upon observations from previous audits [53, 54], the staff suggested that the potential for concentrated streams of water to enter the Reactor Building pool should be considered if localized effects on velocity and turbulence could have a non-negligible impact on the overall debris transport results.

- The non-QA CFD results did not include turbulent kinetic energy contours [37]. In addition to velocity contours, turbulent kinetic energy contours may be important in determining the transport behavior of debris in the Reactor Building pool, particularly with respect to settling and resuspension.

- The non-QA CFD calculation assumed that 0.4 ft/s is the transport velocity for coating chips [37]. The staff noted that both NRC testing and the ONS plant-specific testing could provide additional transport velocity data for coating debris, and the licensee indicated that this data would be considered, as appropriate, in the revised CFD calculation.

- The non-QA CFD model considered debris transport during sump recirculation. As appropriate, the licensee should consider whether an assessment of blowdown, washdown, and/or pool-fill transport is necessary to support the calculation of coating debris transport during the recirculation phase of a LOCA.

As described above, the staff's comments on the non-QA CFD calculation were provided to the licensee for consideration in performing the QA CFD calculation that is currently being developed. None of the staff's comments on the non-QA CFD calculation are designated as open items because this calculation was not used to demonstrate the adequacy of the replacement strainers. The licensee's baseline analysis transport calculation, which is currently credited to demonstrate the adequacy of the replacement strainers, is discussed above in Section 3.5.1 of this report.

3.5.3 Coating Debris Transport Testing

As described further in a report concerning the staff's September 2006 trip to observe testing conducted by CCI, the ONS licensee sponsored plant-specific coating debris transport testing [38]. The test matrix included tests of coating debris transport in suspension with water, tests of coating debris tumbling along a flume floor, and quiescent settling rate tests. The size ranges of the tested coating debris included (1) chips between 1 mm and 4 mm, (2) chips between 280 µm and 1 mm, (3) chips between 75 µm and 280 µm, (4) chips less than 75 µm, and (5) particulate approximately 10 µm in diameter. The four size ranges of coating chips tested had been generated from an epoxy that was identified as Carboline 890. The licensee selected stone flour as a surrogate material to represent coating particulate debris.

CCI tested coating debris transport in their multi-functional test loop, with the length of the flume extended to over 20 ft [38]. A flume-averaged velocity of roughly 0.3–0.4 ft/s was established during the transport testing to represent the maximum continuous velocity between the break location and the sump that was predicted via CFD simulation (presumably the non-QA calculation discussed above in Section 3.5.2 of this report).

As noted in the September 2006 trip report [38], the staff observed one of the tests of coating chip transport in suspension with water. Based upon the staff's limited observations of this test and additional discussions with licensee personnel present during the test, the staff provided feedback for the licensee to consider in applying the test results to the replacement sump strainer analysis [38]:

- Velocity profiles in the test flume should be representative of the velocity profiles predicted by CFD for the Reactor Building pool.
- Turbulent kinetic energy in the relatively quiescent test flume should be representative of the turbulent kinetic energy predicted by CFD for the Reactor Building pool.
- The possibility of coating chips transporting while floating on the surface of the Reactor Building pool should be considered.
- Uncertainties in the analysis of washdown transport that could affect predictions of where coating debris will enter the Reactor Building pool should be considered.

Near the end of the onsite audit, the licensee provided the staff a copy of the draft version of the coating debris transport test report [36]. The staff did not perform a detailed review of this report because (1) the report was considered draft material, (2) the existing ONS analysis had not incorporated the results of the coating debris tests into the larger transport analysis (due in part to the fact that the QA CFD calculation had not been completed), and (3) time constraints. Two staff comments on this report are provided below, which supplement the four bulleted points listed above.

Following the presentation of the tumbling transport velocity test results, Section 6.1 of the CCI test report provides a simplified calculation of the average water velocity in front of the strainer. The average velocity calculated in the test report (roughly 0.28 ft/s) is lower than the measured velocities necessary to cause significant movement of multiple coating debris particles during the tumbling transport tests. Although the staff agrees that a simple calculation of this sort can provide a rough estimate of the flow velocity approaching the sump area, velocity distributions of real flows generally do not have uniform profiles. The non-QA CFD calculation described above, for instance, predicts in certain recirculation scenarios that the value of 0.28 ft/s would be exceeded around a portion of the RBES due to non-uniformities in the Reactor Building pool flow distribution. Therefore, while the simplified calculation in Section 6.1 of the CCI report is informative, the staff does not consider it to provide sufficient basis for concluding that no coating debris would transport to the sump strainers via tumbling.

It was not clear to the staff whether the licensee planned to apply coating debris transport test data obtained for stone flour particulate to particulate debris from epoxy coatings or only for particulate from zinc primer coatings. Based upon information provided in the licensee's head loss test report [36], the density of the stone flour used by CCI is approximately 167 lb_m/ft^3. The

staff compared this density value to the density values of for epoxy paint (94 lb_m/ft^3) and zinc primer (457 lb_m/ft^3) provided in the baseline analysis [35]. Considering these densities, the stone flour particulate would be expected to transport more readily than the more dense zinc primer, which would represent a conservative application of the transport test data. However, application of the stone flour data to the less dense epoxy primer in a future transport analysis could represent a nonconservatism that would require additional justification.

As documented at the beginning of Section 3.5 (pg 21), the licensee's baseline sump analysis does not credit settling of coatings. Therefore, comments in this section regarding the coatings transport testing sponsored by the licensee do not impact the staff's conclusions regarding the licensee's baseline methodology. Rather, these remarks should be considered should the licensee plan in the future to credit settling of coatings or otherwise use the test results in its solution of record.

3.5.4 Debris Transport Summary

The NRC staff reviewed the licensee's existing baseline analysis debris transport calculation [35] to determine its consistency with the sump performance methodology approved in the staff's SE [8]. The staff's review found that the analysis was generally consistent with the SE and concluded that the debris transport results calculated in the licensee's baseline analysis incorporate substantial conservatism.

The staff's audit of the licensee's debris transport evaluation also included limited reviews of a non-QA CFD calculation [37] and a coating debris transport test report [36]. Although the non-QA CFD calculation will not be used to demonstrate the adequacy of the replacement strainer design, the staff provided feedback on this calculation that the licensee could consider with respect to the QA CFD calculation that is currently being developed. Although it is presently not integrated into the licensee's debris transport analysis, the staff understood that the licensee plans to use the coating debris transport test report as a basis for determining the quantity of coating debris to use for future head loss testing.

In light of the discussion above, the staff considered the licensee's existing baseline analysis methodology for analyzing debris transport to be conservative and identified no open items in this area.

3.6 Head Loss And Vortex Evaluation

3.6.1 Audit Scope

The audit was conducted on the three ONS Units, with a focus on Unit 3 since that was considered the baseline unit by the licensee. The units are similar in design. The ONS approach is to provide a single design that is applicable to all three units. The licensee's strainer design takes the most conservative inputs and combines them into a single model. For example, the Unit 1 ECCS suction lines are a different configuration than that for Units 2 and 3. The difference results in the Unit 1 strainer having a smaller surface area than the other units. The licensee's strainer head loss evaluation considers the Unit 1 strainer and assumes that this area is also applicable to Units 2 and 3. Thus, where differences between units exist, this audit report references the most limiting or conservative information. Because of the slightly lower strainer area on Unit 1, it is used for most of the discussions in Section 3.6 of this report. Use of

the smaller strainer area results in conservative analyses in areas of flow velocity and debris loading.

The new sump design proposed by the licensee uses CCI pocket strainers installed in the RBES recirculation suction path for the ONS ECC and RBS systems. The design consists of two banks of strainer assemblies installed in the sump pit, the base of which is about 3½ ft below the Reactor Building floor. The strainers extend about 3 ft above the Reactor Building floor. The strainers have grating installed above a majority of the horizontal surface to provide a working platform and physical protection for the strainers during mechanical work in the area. The area between the two banks of strainers is an annulus that contains the suction lines for the LPI pumps that also supply the HPI and RBS trains. After fluid flows through the strainer surface, it flows downward and is discharged into the annulus. The annulus is covered by a stainless steel diamond plate subfloor that is sealed to prevent debris bypass into the fluid that has already passed through the strainer.

Each bank of strainer modules consists of a large array of strainer pockets. Each pocket has openings that are 4.29 inches high by 2.76 inches wide by about 12 inches deep. The rear surface of each pocket is curved. There are a total of 3350 pockets in the ONS strainers plus upper and front side cartridge covers that provide a total surface area of about 4868 ft^2. Of this area, 325 ft^2 are subtracted in the analyses to account for miscellaneous debris (tags, tape, stickers, etc.,) in containment resulting in a final active surface area of about 4543 ft^2 (CCI Strainer Head Loss Calculation, 3 SA-096.029 [70]).

Based on the debris transport calculation, 111,148 ft^2 of RMI, 23.95 ft^3 of fibrous material, 325 ft^2 of tags, tape, etc., 111.1 lb_m of latent particulate matter, and 1651.8 lb_m of coating material are assumed to be transported to the sump region. In addition, it is anticipated that chemical precipitates will be present at the sump region upon initiation of recirculation. Testing completed at the time of the audit included all of the physical debris from the Reactor Building. However, testing of the strainer with postulated chemical effects debris had not yet been performed. Testing performed to date shows that the estimated pressure loss across the strainer assembly is less than the available NPSH margin, and less than the available water level above the strainer. The licensee and strainer vendor are planning to perform integrated chemical effects strainer head loss testing within the next few months. Plans for the upcoming testing were discussed with the licensee and strainer vendor, but are not included in this audit report. As discussed in Section 3.2.2 (pg 16), ONS has reduced the amount of fibrous material that is calculated to transport to the sump. The amount of fiber currently calculated to reach the sump is low. The ONS approach to addressing the strainer issue has been to start with conservative initial evaluation numbers, then to refine the analysis and remove debris to gain margin. The numbers referenced in this section are the ones currently in the baseline transport calculation. For example, the testing was conducted with the full amount of fiber from the transport analysis. The transportable fiber has since been significantly reduced by removing debris from within the Reactor Building.

The licensee installed a strainer that was as large as possible within the available Reactor Building area to maximize the NPSH margin. Subsequently, CCI tested for prototypical head loss using their test flume with a testing module consisting of 120 strainer pockets. This testing assessed the head loss due to the debris on the surface of the strainer. An empirical correlation was used to calculate the clean strainer head loss due to the strainer surface and the stainer internal structure. As part of the prototypical head loss testing program, the licensee evaluated

the susceptibility of the strainers to vortex formation. An analytical evaluation of vortex formation was also completed. The testing and analysis results are documented in the following reports.

"GSI-191 Generic Letter 04-02 Pre-Audit Presentation" Slides presented by Oconee Nuclear Station, March 13, 2007. [3]
Duke Power Co. Engineering Change OD300050, Install Filter System On Unit 3 Emergency Sump, April 17, 2006. [24]
Duke Power Co. U3 RBES Design Input Calculation, OSC-8739, Revision 6, September 21, 2006 [56].
Oconee Nuclear Station Updated Final Safety Analysis Report, Section 6.1, December 31, 2005 [21]
Duke Energy Purchase Specification for Sump Strainer, OSS-0240.00-00-0007, August 30, 2005 [67]
Duke Power Co. Calculation OSC-1969, Revision 7, Post Accident Reactor Building Water Level Following a Large Break LOCA, May 20, 2003 [34]
Oconee Nuclear Station Unit 3 U3 LPI Hydraulic Calculation, OSC-8442, Revision 3 [68]
Oconee Nuclear Station EOP for ECCS Suction Swap to RBES, EP/1/A/1800/001 [63]
CCI Document, Containment Sump Strainer Replacement: Large Size Filter Performance Test Specification, Q.003.35.527, July 4, 2005 [69]
CCI Document, Containment Sump Strainer Replacement: Large Test Loop Filter Performance Report, F-2856, November 11, 2005 [71]
CCI Document, Head Loss Calculation for Oconee Unit 1 Sump Strainers, 3 SA-096.029, September 13, 2006 [70]
CCI Document, Proof of Absence of Vortex and Air Intake, 3 SA-096.009, July 9, 2006 [72]
CCI Document, Containment Sump Strainer Replacement: Small Filter Performance Test Report, F-2854, June 17, 2005 [73]
ALION Document, Oconee Units 1/2/3 Characterization of Events That May Lead to ECCS Sump Recirculation, ALION-REP-DUKE-2736-001, Revision 0, July 13, 2005 [55]
ALION Document, GSI 191 Baseline Analysis for Oconee Units 1, 2, and 3, ALION-REP-DUKE-2736-02, Revision 0, July 13, 2005 [57]
Enercon Services Document, Containment Walkdown Procedure for Potential Sump Screen Debris Sources at Oconee Power Station Units 1, 2, and 3 (and walkdown results), DUK002-PROC-01, Revision 0, May 19, 2005 [40]

The NRC staff reviewed these reports during the onsite audit and focused its audit effort in the following technical areas:

- System characterization and the design input to the head loss evaluation;
- Prototypical head loss test module design, scaling, surrogate material selection and preparation, testing procedures, results and data extrapolation;
- CCI clean strainer head loss calculation methodology and results; and
- Vortex testing procedures and the vortex formation evaluation results.

The staff evaluation regarding these four areas is provided below.

3.6.2 System Characterization-Design Input to Head Loss Evaluation

The licensee evaluated LOCA scenarios and identified events that may lead to ECCS recirculation through the RBES. The three units at ONS were analyzed together for simplicity. Where differences between the units occurred, the most conservative inputs were chosen. For example, due to a different sump configuration, Unit 1 has a slightly smaller strainer installed that Units 2 and 3. This strainer size is used in the calculations. This approach allows ONS to maintain a more simplified approach to the analyses associated with their ECCS systems.

The ONS Units use a group of systems to mitigate the effects of design basis accidents. These systems require the use of the RBES to provide a long-term source of water for cooling following a LOCA. The systems requiring RBES operation can be divided into to two subgroups: the ECCS, which provides borated water for injection to the reactor coolant system in the event of primary system break, and the containment heat removal systems, which cool the atmosphere inside the Reactor Building to reduce the Reactor Building internal pressure. The ECCS at ONS consists of three major components: core flood tanks (CFTs), LPI pumps, and HPI pumps. The borated water storage tank is also integral to the proper operation of the system. The borated water storage tank provides a borated water source for the HPI and LPI pumps early in a LOCA, before the Reactor Building pool contains enough water to supply the pump suctions. The borated water storage tank similarly provides a source of water to the Building Spray (BS) pumps early in a LOCA.

In order to swap from borated water storage tank injection to the RBES recirculation mode, operators are required to realign valves in the LPI system [63]. In addition to the valve realignments, operators are required to throttle HPI flow to maintain LPI flow below the point where NPSH margin for the pumps is challenged. The HPI pumps take suction from the LPI discharge header. Therefore, the HPI flow affects the total LPI flow. The HPI pumps do not have NPSH issues considering their lower flow design and the head provided by the LPI.

The BS pumps assist in cooling and reducing pressure within the Reactor Building by spraying water from the borated water storage tank as a mist into the Reactor Building. The droplets help to cool and condense the steam that exists within the Reactor Building following a LOCA. The operators' actions to switch the LPI suction from the borated water storage tank to the RBES also allow the BS pumps to take suction from the RBES via the LPI pump discharge. The flow to the BS pumps increases the flow through the strainer and sump piping.

3.6.2.1 Flow Rate

The Reactor Building pool provides a reservoir for an adequate source of water for the LPI and BS pumps following the manual switch over to the sump. The licensee indicated in Reference [56, Appendix B] that for the design LOCA scenario, the maximum flow rate through the RBES strainer is 7839 gpm. The limiting break for flow through the strainer is associated with a core flood line break. This case has two trains of LPI, two trains of RBS, and two HPI pumps running. The HPI pumps are throttled, which limits total flow.

The limiting case for the LPI pump available NPSH (NPSHa) is also a core flood line break, but with a single LPI pump running. The flow in the suction pipe of a single running pump is greater due to lower discharge head. The discharge head is reduced because the other LPI pump is not running, discharging to the common piping. The HPI pumps are throttled in this case as well,

which limits total LPI flow. ONS recently installed flow restrictors in the LPI injection lines upstream of the tees to the core flood lines. The effect of the flow restrictors is to limit flow to a broken core flood line so that adequate flow is delivered to the core via the other core flood line, and flow is reduced, limiting the reduction of NPSH margin. Case 1 of Appendix B of OSC-8739 [56] notes that the flow rate for this case is 4695 gpm. The LPI hydraulic calculation OSC-8442 [68] notes that at 3517 gpm LPI flow there is no NPSH margin for the LPI pump (Unit 3 example). In order to ensure that NPSH margin is not exceeded, ONS requires operators to throttle HPI flow to limit one LPI pump flow to 3100 gpm. The 3100 gpm limit provides some margin and also accounts for instrument uncertainty.

The BS pump limiting case for NPSHa is the same as the limiting case for LPI. This is because the BS and LPI pumps take suction from the same header. Maximizing flow in the LPI suction line also maximizes flow in the common portion of the RBS suction line. The increased flow increases the line losses in the suction pipe. ONS requires containment overpressure to achieve adequate NPSHa for the BS pumps in this scenario. The ONS Updated Final Safety Analysis Report [21] states that 2.2 psi of containment overpressure is credited in the NPSH calculation for LPI and RBS. Based on the current calculations this overpressure is not required for LPI. ONS is currently reperforming the LPI and RBS calculations. The amount of overpressure required may be changed based on the evaluation. To reduce flow in the RBS system, and thereby reduce suction line losses, ONS has plugged some spray nozzles to add resistance. With the added system resistance, maximum RBS flow per train is calculated to be 1200 gpm [61].

Staff Evaluation

The staff reviewed the calculations that describe the LOCA event characterizations and found that the inputs used in the calculations were reasonable and conservative. In general, the conclusions of the calculations for a large-break LOCA can be supported by licensing basis documents and other technical information collected on site. A complete reanalysis of system performance during a LOCA was being completed by ONS at the time of the audit. The staff examined the BS, HPI and LPI pump suction line piping, pump curve data, and the control signal requirements. The staff noted that much of the realignment of the systems from injection to recirculation mode is completed manually by the operators. The original strainer design flow rate was 7500 gpm. However, later analysis determined that the actual maximum flow through the strainer could be 7839 gpm. Strainer testing was accomplished at higher flow rates of 120 percent (9000 gpm) with acceptable results. Since the strainer design flow rate is below the testing flow rate, the testing flow rate of 9000 gpm is found to be acceptable.

3.6.2.2 Sump Water Temperature

Because ONS is in the process of revising its LOCA calculations, a consistent sump temperature during recirculation was not used for all aspects of the analysis. OSC-8442, Unit 3 LPI Hydraulic Calculations [68] indicates that the maximum sump temperature of 220 °F occurs during the injection phase following a LOCA, but that 240 °F will be used for conservatism during recirculation. The Unit 3 RBES strainer input calculation [56] states that the maximum temperature expected in the sump is 240 °F, but that 300 °F should be used for the structural design temperature for the strainer. (This extra margin for structural design is not used in the evaluation of head loss.) The calculation [56] also states that the temperature range for recirculation operation is from ambient to 208 °F. Per the CCI head loss calculation [70], the

design temperature for strainer head loss is 208 °F. More recent analyses show that the sump temperature is actually 239 °F at the start of recirculation.

Staff Evaluation

The staff reviewed the information regarding the bounding sump water temperature for the strainer head loss calculation and the NPSH calculation. The staff agrees that the use of 240 °F as the limiting temperature for the NPSH calculation will yield a conservative value, as discussed in Section 3.7.3 of this audit report.

The most conservative assumption regarding the pool temperature for the strainer head loss calculation alone would be to assume the minimum pool temperature expected during recirculation. This would maximize the strainer head loss due to the higher water viscosity at cooler temperatures. However, the licensee used 208 °F for the strainer head loss calculation. This is the expected sump temperature at the time that reactor building overpressure is no longer allowed to be used by the license.

The head loss calculation is only one part of the determination of NPSHa. NPSHa is lowest at the start of recirculation when sump temperatures are the highest. As the event progresses, the pool cools, reducing the vapor pressure at the pump impeller. This effect tends to dominate the pump NPSHa. Therefore, as temperature decreases, NPSHa increases even though strainer and piping head losses are increasing. The licensee's use of a temperature lower than maximum sump temperature for the strainer head loss term, in conjunction with use of the maximum sump temperature in the rest of the NPSHa calculation, makes the overall NPSH calculation even more conservative and is therefore acceptable.

3.6.2.3 Reactor Building Pool Water Level

The licensee has calculated the volume of water transferred to the Reactor Building from the borated water storage tank combined with the amount of water available to the sump from the CFTs prior to transfer to recirculation mode [34]. The minimum water level for a large-break LOCA is determined to be at 781.28 ft, or 4.40 ft above the floor level of 776.88 ft.

The minimum water level for a small-break LOCA may not include the 1959 ft^3 of water potentially introduced into the Reactor Building from the CFTs. According to the GSI-191 Baseline Analysis for ONS Units 1, 2, and 3 [55] this reduction of volume would reduce the water level by 0.2 ft.

The licensee evaluation conservatively assumes that no water from the RCS is added to the pool volume. In addition, minimum volumes, based on Technical Specification minimum levels, are added from the CFTs and borated water storage tank. This is also conservative.

The RBES floor (pit) elevation is 774 ft. The minimum elevation of the water for a large-break LOCA is 781.28 ft. This indicates that there is 7.28 ft of water above the sump floor. The strainers are installed on the floor of the sump pit and extend 6.3 ft above the floor [72]. Based on the above, the minimum strainer submergence is 0.98 ft for a large-break LOCA. This submergence is greater than the maximum corrected head loss across the screen [70]. Therefore, no water vapor flashing is expected to occur inside the strainer during the long-term recirculation phase of a large-break LOCA because the maximum strainer head loss is less than

the minimum submergence during a large-break LOCA. In addition, the CCI calculation [72] documents that the formation of a vortex is extremely unlikely based on an empirical correlation developed for CCI strainers using a form of the Froude Number. The design coverage for the ONS strainer is significantly greater than the depth required by the CCI prediction on vortex formation. Because the maximum head loss is so small at ONS, the staff did not evaluate the void fraction downstream of the debris bed.

CCI did not document the water level above the strainer during testing. However, no evidence of air ingestion into the strainer occurred during testing. The strainer is supplied with a perforated top plate because the submergence of the ONS strainer is much greater than that predicted for vortex formation. On designs that approach the potential for vortex formation CCI does not provide top plates that are perforated, but uses solid plate.

Staff Evaluation

The staff reviewed the analysis determining the minimum Reactor Building water level [34]. The calculation contains some conservatism as discussed above, but specific factors that affect level were not considered. These include the lack of consideration of holdup due to the spray droplets falling from the spray headers and the holdup due to condensation films on Reactor Building horizontal and vertical surfaces. In addition to the omissions, there is a potential for complete or partial blockage of the drains in the reactor annulus and the refueling cavity. Some holdup was considered for these areas, but blockage of the drains would result in additional holdup. The potential for the CFTs to remain full for a period of time following a small-break LOCA is further addressed in Section 3.7.1. These items are considered part of Open Item 3.7.1 in the NPSH margin section of this audit report (pg 54).

The issues described above may be minor influences on the minimum pool level. However, ONS has little excess NPSH margin. Therefore, these issues should be considered in the NPSH margin calculations. It is unlikely that strainer submergence and head loss will be significantly affected by the sump level based on current head loss testing, but when chemical precipitates are added the effects could be significant.

Conclusion

As discussed above, the staff reviewed the analysis determining the estimated sump water temperature, minimum Reactor Building pool water level and the maximum flow rate through the sump for the strainer head loss calculation. Because these design inputs were developed either based on the previous licensing basis calculations or bounding values selected for the head loss evaluation, the staff considers them acceptable with the exception of the items noted in the area of sump pool level. The issues with sump pool level are addressed in detail in Section 3.7 of this report (pg 54).

3.6.3 Prototypical Head Loss Testing

In order to demonstrate that the new strainer head loss for the most limiting LOCA case is less than 0.1 feet, which ONS used as a design input, the licensee contracted with CCI to perform prototypical head loss testing. A module of the strainer containing 120 pockets was placed in a large test flume approximately 8.5 ft wide, 9.8 ft high, and about 14.4 ft long [CCI Large Filter Performance Test Specification 74]. The CCI strainer module was installed near the center of

the flume. The assembly was connected to a pump suction line mounted horizontally which passed through a divider plate, then the end wall of the flume. The pump discharge header was mounted in front of the strainer module. The header had holes drilled in it to distribute the flow more evenly in the test compartment.

Pressure transmitters, a flow meter and thermocouples were installed to measure the head loss, total flow rate and the water temperature. Several head loss tests were run to measure the response of the strainer to varying debris loads and flow rates. The tests included a design basis case and cases with additional fibrous debris, particulate debris, and higher flow rates. The staff reviewed the test plan, the test report and the interpretation of the test results.

The tests were run at varying flow rates. ONS originally anticipated that the maximum strainer flow would be 7500 gpm. Later ECCS hydraulic analyses showed that the actual maximum post-accident flow rate would be 7839 gpm. Tests were scaled based on the original 100% flow rate (7500 gpm) and on the 120 percent flow rate (9000 gpm). The results of the head loss tests scaled to 9000 gpm show that the design maximum of 0.1 ft of head loss is not exceeded for the worst case expected LOCA debris loading at the design accident temperature. The 9000 gpm flow rate tested is well in excess of the actual expected maximum flow rate of 7839 gpm.

3.6.3.1 Debris Types, Quantities, and Characteristics

The specification of the debris quantities and characteristics is important to the choice of debris surrogates and debris preparation for the head loss testing. The predicted quantities of debris used to determine the amount of debris for head loss testing are shown in Table 8. The potential debris accumulation on the replacement strainers was conservatively determined by quantifying debris within the ZOI of interest and adding latent and coating debris. Except for RMI, all debris was assumed to reach the strainer. This table illustrates that the debris types and quantities used in the tests are conservative with respect to the plant debris assessments. The table in this section is based on the quantities used for testing that came from the baseline analysis. These are the quantities that were actually used for baseline testing. The Transport Section of this report (3.5 (pg 31)) discusses updated information that the licensee may use to support future testing or increase calculated margin. As previously discussed, ONS removed all of the fibrous insulation within the ZOI after the baseline analysis was performed. The amounts of fibrous debris in this section are based on the larger baseline amounts prior to removal. The debris loads actually used in the tests were scaled down from the plant debris loads based on the ratio of the strainer areas (i.e., 181/4871 = 0.037).

Table 8 Comparison of ONS LOCA-Generated Debris and Test Debris			
Type of Debris	**Baseline**	**Test Specification**	**Test Surrogate** (Before Area Scaling)
Metallic Debris			
RMI Insulation (Stainless Steel Foils)	82,793 ft^2	111,148 ft^2	111,148 ft^2
Fibrous Debris			
Fibrous Insulation [a]	16.9 ft^3	16.7 ft^3	Owens Corning Fiberglass Insulation 23.95 ft^3
Nuclear Instrument Fiberglass Cable Wrap	0.8 ft^3	1.0 ft^3	
Latent Fibers [b]	6.0 ft^3	6.25 ft^3	
Particulate Debris			
Latent Particulate	81.4 lb_m	100 lb_m	Stone Flour 1762.9 lb_m
Epoxy and Epoxy Phenolic Coatings	1613.9 lb	Qualified and Unqualified Coatings 1651.8 lb_m	
Inorganic Zinc Primer Coatings and Cold Galvanizing	233.0 lb_m		
High Temperature Aluminum	44.1 lb_m		
Alkyds Enamels	12.7 lb_m		
RTV Caulk	30.5 lb_m	11.1 lb_m	
Surface Area Debris			
Foreign Materials [c]	72 ft^2	325 ft^2	None

a – Fibrous insulation was removed from Reactor Building subsequent to test specifications
b – Fiber volume based on a bulk density of 2.4 lb_m/ft^3
c – Test strainer area scaling subtracted the area of the surface debris from the replacement strainer area prior to determining the test module specifications.

The baseline evaluation document reviewed by the staff was an earlier scoping analysis that was out of step with the ONS debris assessment at the time of the audit. The staff noted that the ONS GSI-191 resolution was an ongoing process. The licensee stated an intention to develop a living document that would replace the current baseline analysis document. In particular, ONS has removed all of the fibrous insulation within postulated ZOIs, which substantially reduced the bounding volume of fibrous debris from that tested.

The total quantity of particulate in the test specification was 252.7 lb_m less than the baseline total estimate (baseline estimate of 2015.6 lb_m compared to the test specification of 1762.9 lb_m). The reason for this difference is not clear, but it appears that some of the RTV caulking baseline estimate was subsequently treated as surface area debris rather than as particulate.

The Owens Corning fiberglass insulation used in the testing is similar to the fibrous insulation that was removed from the Reactor Building. As such, it was an excellent representation for the plant fibrous debris. In regard to the latent fibers, the Owens Corning insulation has a greater density than that typically applied to the latent fibers. This consideration is conservative, because the original estimate of latent fibers was a mass measurement that was converted to volume using the generally accepted density of 2.4-lb_m/ft^3, resulting in an estimate of 6 ft^3 of fibrous debris. Since the Owens Corning fiberglass is denser, the mass of fibrous debris added to the tests was more than the calculated mass of latent fibers.

The surrogate RMI debris was manufactured from 2-mil-thick stainless steel foil by a mechanical process that generated metallic debris that appears to be reasonably prototypical of LOCA-generated small piece debris.

The only non-prototypical aspect of the head loss testing surrogate debris is the material density of the stone flour relative to densities of the coatings particulate. The average density for the mixture of test particulate is about 110-lb_m/ft^3 compared to the stone flour density of 167.4-lb_m/ft^3 [74]. The stone flour makes a good surrogate for the latent particulate, which typically has about the same density, but the stone flour is considerably heavier than the epoxy, aluminum, or alkyd coatings and the flour is much lighter than the inorganic zinc coatings. The effect of the density non-prototypicality involves its effect on particle settling within the test tank. The stone flour particles would settle more rapidly than would the epoxy, aluminum, or alkyd particles, resulting in less particulate in the debris bed; and conversely settle less rapidly than the inorganic zinc. The licensee response to this issue was that the stone flour was so fine that the flour tended to not settle in the test tank, therefore this non-prototypicality is not an issue. The validity of this response is questionable because it was based on subjective observations of debris behavior within opaque water, and not on post-test measurements to support the observation. For example, the debris on the bottom of the test tank following the test could have been analyzed.

The test specification report [74] states that the stone flour specific surface area corresponds to a sphere diameter of 7.7 microns, which is significantly less than the GR guidance of 10 microns for coatings particulate. This determination was based on an analysis of a size distribution for the stone flour. In general, in the presence of a fibrous bed, smaller particulate sizes will result in increased head loss, so the smaller particulate should result in conservative measurements.

The head loss characteristics of the surrogate test debris appear to reasonably represent the postulated plant debris and are conservative with respect to the SE-accepted guidance. The one exception is the non-prototypical density of the stone flour. However, because the CCI test procedure introduces the flour near the strainer surfaces, thereby reducing the impact of near-field settling, the staff believes that this non-prototypicality is not likely to be significant. In addition, the amount of coatings estimated to reach the strainer at ONS is conservative since the testing assumed transport of all coatings debris to the strainer.

3.6.3.1.1 RMI Debris Head Loss Assessment

ONS calculations predicted large quantities of RMI debris accumulating at the replacement strainers (i.e., 82,793 ft^2) based on 73 percent transport. However, testing used 100 percent transport numbers, as shown in Table 8. Head loss testing was conducted both with and without RMI. The testing demonstrated that head loss is greater if RMI is not introduced into the test loop. The head loss testing demonstrated that the RMI debris acted as a pre-filter, reducing the amount of fibrous and particulate debris that reached the strainer. Due to the design of the sump and strainer at ONS, it is possible that some RMI could fall into the sump below the Reactor Building floor, but it is unlikely that RMI would enter the upper pockets of the strainer. During small-scale testing it was observed that RMI could increase head loss, but this was only when the column above the small strainer section was filled with RMI. This is not a credible scenario in the plant application. ONS is using the non-RMI test results for prediction of head loss as a conservative measure.

Staff Evaluation

The staff evaluated the large-scale and small-scale test plans and the test results. Based on the results of the tests with and without RMI, and other similar testing observed by the staff, the testing without RMI is accepted as a conservative basis for head loss for the ONS strainers. The head loss caused by RMI observed during the small-scale testing is not representative of actual plant conditions. The staff agrees that it is conservative to use non-RMI test results.

3.6.3.1.2 Tapes and Labels Head Loss Assessment

The licensee predicted relatively small quantities of foreign material debris in the form of tapes, labels, tags, etc., that could obstruct portions of the replacement strainers should this debris actually accumulate. The actual predicted area of this type of debris is 72 ft^2, based on the ONS baseline analysis (ALION-REP-DUKE-2736-02) page 48. If pieces of tape or label debris were to adhere to a screen surface, then the available surface area for the accumulation of other debris would be reduced. Therefore, it would take less fiber to create a fibrous layer on the replacement strainers. In its head loss testing, ONS subtracted 325 ft^2 from the area of the plant strainer to allow for tapes and labels.

Staff Evaluation

The subtraction of an area 4.5 times the predicted debris area is clearly conservative from a head loss perspective. Although the 72 ft^2 predicted as miscellaneous debris is less than has previously been observed by the staff at other sites, the 325 ft^2 should bound any of this type of debris in the Reactor Building. The staff considers the treatment of miscellaneous foreign debris at ONS to be conservative and acceptable from a test scaling perspective.

3.6.3.1.3 Fiber/Particulate Head Loss Assessment

The licensee stated that the potential for fibrous debris in the ONS Reactor Buildings is minimal. Testing was conducted using latent debris of 6.25 ft^3, fibrous insulation debris of 16.7 ft^3, and fiberglass cable insulation wrap of 1.0 ft^3. After the test specification was written, ONS removed the 16.7 ft^3 of fibrous insulation from the ZOI. Therefore, testing was conducted with a

conservative amount of fibrous material (more than 3 times what is now expected). Because of the relatively large screens installed and the very low amount of fiber in the ZOIs, a thin bed did not form during the tests. The amount of particulate debris added was not conservative relative to that which would be generated during a LOCA because of recently updated coating destruction quantities. However, 100 percent of the smaller value was assumed to reach the sump. Because no fibrous bed formed on the strainer, the particulate material added during the test did not result in an unacceptable head loss. In addition, it is unlikely that a larger amount of particulate would have resulted in an unacceptable head loss because of the small amount of fiber present during testing. As discussed elsewhere in this report, ONS plans to perform integrated chemical testing using the most recent postulated debris quantities including coatings as particulate and chips. The integrated testing will account for the change in predicted failed coating quantities.

Staff Evaluation

Based on the fact that the licensee's testing added about 300 percent of the actual fibrous debris expected to arrive at the strainer and a fibrous bed did not form, it is very unlikely that any type of evenly distributed fiber bed will form on the ONS strainer. Because the formation of a fiber bed is unlikely, the use of particulates as coating debris could be non-conservative. Based on discussions held during the audit, the staff understood that the licensee planned to perform testing using chips as coating debris during upcoming chemical effects testing. Although the coatings debris was added as particulate, the likelihood of paint chips creating a large head loss across the strainer is small. Because the ONS strainers are large, the approach velocities, both to the screen area and the circumscribed area, are very low. For the ONS strainer design, the average maximum screen approach velocity, corresponding to the maximum predicted post-accident flow rate (7859 gpm) and 4800 ft^2 of strainer surface, is 0.0036 ft/s. From the dimensions provided in the test specification document [74], the area ratio between the screen area and the open face of an individual pocket is about 14.8. This implies that the average maximum approach velocity directly in front of each pocket would be about 0.054 ft/s. This velocity is substantially slower than velocities required for lifting such debris as RMI, typical paint chips, or even fibrous shreds from the Reactor Building pool floor onto the strainer. In addition, if a piece of debris such as a paint chip entered a pocket, the screen approach velocity is much too slow to keep such debris fixed on the upper or vertical pocket surfaces. Therefore, it is likely that only the fine debris, such as individual fibers and particulate, can effectively accumulate uniformly enough to cause a significant head loss.

One other issue associated with head loss testing was evaluated as part of this audit. During upcoming chemical effects testing, ONS plans to add coating debris as chips as described above. If the fibrous debris combined with chemical precipitates are able to form a thin bed, it is possible that adding the coatings as chips will not be conservative because the particulate debris would tend to filter out on the bed thus increasing head loss. Approved GR/SE guidance regarding assumed forms of coatings debris was formulated prior to some knowledge that has recently been acquired in the area of chemical precipitates and their effect on strainer head loss. The staff discussed this issue and considered that for ONS, the amount of fiber was so small that the formation of a thin bed, even with chemical precipitates, is unlikely. The staff has recently provided guidance to the industry with respect to the addition of coating surrogates during strainer head loss testing. Licensees should test with coatings debris in a form that is conservative with respect to head loss, and should justify the methods chosen.

Based on the above, the testing that has been completed is acceptable to demonstrate that the strainer will perform its intended function if required. Chemical precipitate testing is still required to ensure that the strainer will function with the chemical loading predicted by industry guidance. The staff finds the fibrous and particulate debris testing conducted to date to be acceptable.

3.6.3.2 Scaling Methodology, Testing Procedures and Test Results Interpretation

3.6.3.2.1 Scaling Methodology

There are a total of 3584 pockets in the ONS Unit 3 strainer, plus upper and front side cartridge covers that provide a total surface area of about 5196 ft^2 . Of this area, 325 ft^2 are subtracted to account for latent tags, tape, stickers, etc. in the Reactor Building resulting in a final active surface area of about 4871 ft^2 (CCI Strainer Head Loss Calculation [70]). The test strainer consisted of an array of 120 pockets with a total area (including upper and side covers) of 181 ft^2. The test array pockets were slightly smaller in depth than the plant strainer pockets, but this was accounted for in the total area of the strainer. The height and width of the test pockets was the same as the plant strainer pockets. The overall scaling factor for the testing was 27.0 and was based on the areas described above. The scaling factor was calculated based on the larger strainer area of Units 2 and 3. If the Unit 1 strainer area were used, the scaling factor would have been about 25.2. The test results were later corrected to the more conservative Unit 1 strainer area using a strainer area correlation [70].

During the test, all the debris was introduced into the flume directly upstream of the strainer. Therefore, no credit was taken for near-field debris settlement. Assuming uniform debris distribution, CCI scaled the total debris loading based on the ratio between the total testing module surface area and the actual screen surface area. The screen approach velocity was varied from slightly below the highest predicted flow to well above the flow. The strainer pockets and the arrangement of the pockets in the test setup were similar to the actual stainers installed at ONS.

Staff Evaluation

The testing module was scaled assuming no near-field debris settlement in the test flume. A uniform debris distribution assumption was used to scale the debris loading. The test screen approach velocity was varied to encompass flow rates from 100% to 140% of those at the plant. Most tests were run at 120 percent (velocities scaled to 9000 gpm), which is well above the maximum expected flow rate. The staff considers the licensee's scaling methodology acceptable for the following reasons:

- The scaling factor methodology was appropriate, because the test debris load was scaled based on active strainer area and the full plant debris load
- No near-field settlement was credited
- The highest screen approach velocity tested was bounding

3.6.3.2.2 Testing Procedures

Prototypical head loss testing was performed by the strainer vendor following their specific testing procedures, including specific debris addition procedures, and testing implementation procedures. The testing procedures for the large-scale testing included the following:

Test Loop Description
Scaling Factor Development
Debris Preparation
Nominal Flow Rate Calculation
Test Performance

CCI conducted both small- and large-scale head loss tests to assess the potential head loss across the ONS replacement strainers. The staff reviewed the test specification documents [75, 74] and the test reports [73 , 71] for both the small- and large-scale test programs. The test procedures were presented in the test specifications. The test reports presented the head loss results. The small- and large-scale tests were conducted with test modules consisting of six and 120 strainer pockets having screen areas of 8.03 ft^2 and 160.7 ft^2, respectively. The large-scale testing total area was 180.6 ft^2 due to additional area added for top and side strainer covers. The test data included the flow rates, pressure differentials, and water temperature.The staff reviewed the CCI test procedures for introduction of debris, the test termination criteria, and the test matrix. In the small-scale testing, conducted in a small vertical test loop, virtually all introduced debris would accumulate on or within the six-pocket test strainer module.

For large-scale testing, the CCI approach was to introduce the debris in close proximity to the 120-pocket test strainer module to reduce debris settling within the tank. While this approach effectively reduces near field settling, it does not eliminate such settling completely. Post-test photos clearly show both fibrous and particulate debris on the tank floor. The downside to the CCI approach is that introducing the debris in close proximity to the strainer could introduce non-prototypical debris distribution for the fine suspended debris. Post-test photos of accumulated fibrous debris appear to indicate a skewed accumulation toward an end row of pockets. However, conversations with the licensee on this topic indicate that the debris distribution in the photos appears to be more unbalanced than it actually was. Since the CCI strainer design does not include some type of orifice flow control design to ensure uniform approach flow, the strainer debris accumulation could be more concentrated near the pump suction. The unbalanced concentration observed in the photos may be prototypical of the replacement strainers. Additionally, the rate of introduction of fibrous debris in the proximity of the strainer module can influence the compaction of the accumulated fiber bed, and the water flow carrying the debris into the tank can affect the debris accumulation if that flow were directly toward the strainer screens.

For the test results reviewed during the onsite audit, it is likely that test head losses were not greatly affected by these concerns regarding non-uniformity, because a range of fiber quantities including quantities well above the predicted 100 percent loading were tested, and all test head losses remained within the licensee test acceptance criterion of 0.1 ft of water (after adjusting to the design water temperature). However, for future head loss testing, when only the bounding quantity of fibers associated with latent and nuclear instrument cable wrap debris will be introduced (i.e., reflecting the removal of other fibrous material from the Reactor Building), the uniformity associated with the transport of individual fibers could become more important.

The CCI method of introducing debris in the proximity of the test module is conservative with respect to the introduction of large debris. The ONS tests during which RMI debris was introduced show RMI debris accumulation even in the upper pockets, but only because the debris was artificially introduced close enough for the RMI to be pulled into a pocket before natural settling would drop the debris to the tank floor. In the ONS Reactor Building, the sump flow velocities would not be sufficient for RMI debris to effectively accumulate within the upper pockets. The same considerations would apply to paint chips, as well.

The termination criteria for the ONS testing were apparently not specified in the testing documents. It would have been useful to view examples of the actual head losses versus test time to ascertain whether or not CCI was obtaining reasonably steady-state conditions prior to declaring either test terminations or step increases in an ongoing test, but this information was not made available to the staff for review.

The significance of termination criteria is highlighted by comparing ONS large-scale Test 10 with Test 9. After Test 9 with its nominal 100 percent fiber load and a nominal 100 percent particulate load was completed, an additional 25 percent of particulate debris was added to the tank and the test was allowed to continue running overnight. The resultant test head loss increased from 2.8 to 6.3 mbar, and the tank water cleared substantially, indicating increased filtration efficiency. The increased head loss could be attributed to both the extra particulate and the increased filtration efficiency filtering out the finer particles. Theoretically, assuming a uniform debris bed, a 25 percent increase in particulate in the debris bed would lead to an approximate 25 percent increase in head rather than more than the doubling of head loss that occurred. It seems much more likely that the doubling in head loss was the result of the longer overnight operation, apparently resulting in more complete filtration of the particulate. A possible contributor to the increased filtration is that the fibrous debris was able to accumulate in the weaker spots of the debris bed over a longer term, and it is possible that fibers worked free from the debris bed and subsequently through the strainer, recirculated, and accumulated once again but more uniformly, thereby increasing overall bed filtration efficiency.

While it is not possible to actually resolve the processes making the largest contributions to the doubling of the head loss, these tests do highlight the importance of not prematurely concluding a test before a relative steady state has been achieved. The licensee used the results for Tests 6 and 9 head loss result for strainer qualification [71] because those tests were based on nominal debris loads, but in light of the overnight increased filtration, the more conservative approach would have been to use the Test 10 results. (Note that the head loss calculation [70] references Tests 6 and 7, but the actual results were from Test 9 after the flow rate had been held at 140 percent in Test 8. This was verified with the licensee and CCI during the onsite audit.)

CCI conducted the ONS large-scale tests with fiber loads that were up to eight times the design bases fiber load for their containments. Further, the fiber loading for the tests was based on the baseline analysis that included fibrous insulation debris that has subsequently been removed from the Reactor Building. The stated purpose of testing with up to eight times the calculated fiber load was to demonstrate margin available for potential issues in the future that may result in additional fiber in the ONS Reactor Buildings. This approach, however, will not be valid unless the planned testing with chemical effects precipitants also tests with elevated quantities of fibrous debris.

Based on the head loss testing results available at the time of the onsite audit, the test head losses appear to demonstrate that the head loss associated with the replacement strainers

would be less than the test specification maximum head loss of 0.1 ft of water. However, chemical precipitants were not used in these tests; therefore, the qualification of the replacement strainers depends on the planned head loss testing with the chemical precipitants.

Staff Evaluation

The staff reviewed the key testing procedures affecting the head loss measurement and concluded that they were properly applied to ONS strainer head loss tests. Although improvements to the testing protocol could have been made by the licensee and its contractor, there was adequate conservatism in the testing (e.g., use of extra fibrous debris and higher than nominal flow rates) to account for the issues identified in this Section, including the lack of specified termination criteria. The staff considers the test procedures adequate to determine a conservative debris head loss associated with the strainer.

3.6.3.2.3 Test Results Interpretation

The ONS prototypical strainer test program consisted of two test runs. Both runs were conducted using the design basis debris loading, then placing additional debris into the test stream. The first test was run without RMI, and the second run added RMI prior to placing the fibrous and particulate debris into the test tank. The clean strainer head loss was measured during the first test at scaled flow rates up to 140 percent prior to the introduction of debris into the flume. The flume was then drained and a small percentage of particulate was placed on the floor prior to refilling the flume to be more prototypical of what would be existing during a real event. The second test, with RMI, was performed similarly except that the clean strainer head loss was not remeasured. For both tests, particulate loading was increased to 125 percent of the design basis amout. Fiber loading was increased to 800% of the design value during the non-RMI test and to 400 percent of the design value during testing with RMI.

For the tests the 100 percent flow rate was initially based on 7500 gpm. Flow rates of 120 percent (or 9000 gpm) were also used at the nominal debris loading base cases. After the test specification was completed, ONS discovered that their actual 100 percent flow would be greater than 7500 gpm, perhaps as high as 9000 gpm. During the first set of test cases the strainer head loss approached the design limit. Therefore, after Test Case 8, the 100 percent flow case was changed to be 9000 gpm. For the remainder of the testing, except for steps 20 - 22 during RMI testing, 100 percent flow correlated to 9000 gpm. This is documented in the test record [71].

As discussed above, the actual 100 percent flow rate for ONS has now been determined to be 7839 gpm. Based on the scaling factor of 27, the 7500-gpm scaled flow rate is 278 gpm. The 9000-gpm scaled flow rate is 333 gpm. The measured head loss results are summarized in Table 9 [71]

The clean strainer head loss measured in the test flume was not fully representative of what would be experienced in the plant. The test strainer module was similar to the plant strainer, but had some differences in geometry. The additional losses associated with flow impingement plates added at the entrances to the ECCS piping were not included in the test. These losses were calculated and added to the debris head loss. The addition of some portion of these calculated losses added conservatism because some of the losses associated with the strainer were included as debris losses. Basically the measured clean strainer head loss was accounted

for twice. It was counted once by measurement as part of the test and once by calculation. Since the flow impingement plates were not part of the test facility, they were not accounted for twice. The impingement plate loss was calculated and added to the test results.

Table 9 Head Loss Test Results				
Test Step	Test Module Flow Rate (gpm)	Clean Strainer Head Loss (ft)	Nominal Debris Loaded Loss (ft)	Average Fluid Temperature (°F)
4	349	0.006	N/A	54
9	324	N/A	0.0918	59
23	323	N/A	0.0037	56.3

The test was conducted based on the larger strainer areas of Units 2 and 3. The results of the testing were subsequently corrected to the smaller strainer size of Unit 1 in the Head loss Calculation [70]. The calculation considered both the change in strainer area and the change in debris thickness that result from the smaller strainer area. The result of the calculation indicates that the head loss at the Unit 1 strainer size would be 0.1093 ft of water.

Based on the measured head loss test data, the licensee used an extrapolation methodology to calculate the debris bed head loss at the specified fluid temperature. The licensee assumed that the head loss is directly proportional to the absolute fluid viscosity. Therefore, the predicted debris bed head loss is much lower than the 0.1093 ft of water noted above when it is corrected to 208 °F. A curve was plotted and the head loss was calculated for several temperatures on the curve. The closest temperature to 208 °F for which a head loss was determined was at 212 °F, where the head loss was calculated to be 0.0257 ft.

Staff Evaluation

Because of changing design inputs (the testing was being developed at the same time the strainers were being designed) the testing was performed using a scaling factor for the Unit 2 and 3 strainers. This was not conservative since the Unit 1 strainer is smaller than the other units. However, the result from the testing was corrected to a value for the Unit 1 strainer. The staff found the correction for the smaller strainer area to be appropriate. The extrapolation methodology for debris head loss evaluation is acceptable because the licensee used standard methodology based on the assumption that the debris bed head loss is directly proportional to the absolute viscosity. The licensee determined during testing that the use of results without RMI is more conservative since the RMI tends to act as a prefilter, preventing some of the fiber and particulate debris from reaching the strainer. The licensee used the more conservative results without RMI to determine the final debris head loss result. The staff therefore finds the licensee evaluation of the head loss testing results to be acceptable.

3.6.4 Clean Strainer Head Loss Calculation

The ONS LPI pumps have a relatively low NPSH margin; therefore, the licensee chose to install the strainer with the largest possible strainer surface flow area. The CCI strainer design uses pockets to increase the available surface area to distribute any debris. The very large surface

area results in extremely low head loss across the surface. However, there are internal losses associated with the strainer. There is one loss associated with the flow of the fluid vertically through the internal plenum. In addition, there is an exit loss associated with the flow leaving the plenum and entering the open sump volume. In addition to the losses associated directly with the strainer, there are losses associated with impingement plates that were added to the ECCS suction pipes. These impingement plates are considered part of the strainer because they were not in place prior to strainer installation. The impingement plates are basically blind flanges that stand off from the opening of the suction pipe by about 8 inches. The licensee and its strainer vendor calculated the total clean strainer head loss for these components using a standard single-phase hydraulic analysis [70]. The staff review of these two aspects of the clean strainer head loss calculation is discussed in the following subsections.

3.6.4.1 ECCS Suction Pipe Impingement Plate Head Loss

The new CCI strainer assembly is attached to an open sump. For Units 2 and 3, the exit from the sump to the ECCS is through vertical 20-inch outside diameter strainer discharge piping. At the exit from the sump, there is a conical transition from a relatively large opening to the 20 inch pipe. The impingement plate is installed about 8 inches above this opening. Each of the units has two suction pipes attached to the sump volume. For Unit 1, one pipe is attached as described for Units 2 and 3, and one is attached horizontally. The horizontal pipe also had an impingement plate added as part of the strainer installation. The horizontal pipe does not have a conical inlet and the impingement plate is about 15 inches away from the opening.

CCI performed the hydraulic analysis using industry-standard methodology [Idelchik, 76]. The fluid velocity was calculated based on a single-phase flow assumption and the continuity equation. The calculations found that the vertical configuration losses were greater than the single horizontal configuration entrance loss on Unit 1. Therefore, all entrance losses were conservatively assumed to be the loss for the vertical suction pipe configuration. Calculations were performed for various flow conditions. The loss ranged from 0.0158 ft for 3750 gpm to 0.0358 ft for 5640 gpm [70 pg. 12].

Staff Evaluation

The licensee performed the head loss calculation for the attached piping and fittings using standard hydraulic analysis methods [Idelchik 76]. Without vapor flashing inside the strainer, a single-phase fluid is assumed. Because the head loss across the strainer is less than the strainer submergence, the assumption of a single phase fluid is valid. The analysis methods used were for liquid phase and therefore appropriate. Therefore, the overall approach is considered reasonable.

3.6.4.2 Clean Strainer Array Head Loss

The CCI pocket strainer uses hundreds of pockets to filter the water prior to its entrance to the sump. Because of the extremely low velocity across the large pocket surface area, the head loss across this surface is negligible. After the water passes through the pockets it must flow vertically down to the plenum that discharges the water to the sump. The pockets at the top of the strainer array are closer together than those at the bottom. A cross section of the module would reveal a triangular cross section for this vertical channel. CCI performed a CFD analysis to determine the friction coefficient for the vertical channel. The CFD analysis found the friction factor to be 0.05 which corresponds to a high surface roughness.

CCI calculated the flow area and exit velocity for various strainer flows (5000, 7500, and 9000 gpm) using geometrical relationships.

The exit loss factor from the channel into the plenum was assumed to be 1 because the volume of the plenum is so much larger than the channel. This implies that all dynamic head will be lost upon exit into the plenum.

The overall losses for these portions of the strainer were calculated for various flow rates [70 pg. 10]. This calculation is a standard textbook hydraulic calculation.

Staff Evaluation

The staff reviewed the CCI head loss report for the ONS strainer. The CCI calculations were based on accepted hydraulic correlations and a CFD analysis which provided a reasonable friction factor. The staff therefore finds the clean strainer head loss calculations, including the calculation for the losses attributable to the ECCS piping entrance, to be acceptable.

3.6.5 Vortex Evaluation

The licensee and its strainer vendor investigated the possibility of vortex formation as part of the strainer array testing program. Testing has shown vortexing to be very unlikely as long as the strainer is submerged. However, CCI has observed vortex formation following the formation of a debris bed on the strainer if pumps are stopped and restarted. Stopping the pump releases air that has accumulated inside the strainer. The air release can clear small localized areas of the strainer. When the pump is restarted, high velocity flow areas are created in the cleared areas. These high-velocity zones are more prone to causing vortex formation. CCI evaluated the margin to vortex formation for the ONS strainer using empirical data and a relationship between the Froude number and relative height of water above the strainer. The strainer is predicted to have at least 1 ft of submergence during a large-break LOCA. Based on the evaluation, there is a large margin between operating conditions and the formation of a vortex. Open Item 3.7.1 (pg 54) was identified with respect to the pool level calculation. However, it is not likely that the pool level will decrease sufficiently to adversely affect vortexing for the strainer because of the large margins available. CCI provided an analysis that showed that the strainer application at ONS would not vortex even with a substantially reduced submergence [72].

During large-scale testing no vortexing was observed even with test flows scaled to 140% of nominal. NRC staff observed portions of the large-scale testing [77].

For applications where vortexing is a potential concern, CCI includes solid top plates on the strainer instead of the perforated plate. Use of the solid plate reduces strainer area, but based on test experience eliminates the potential for vortex formation. For the ONS strainers the analysis showed so much margin to vortex formation that a perforated plate was used. Most CCI designs use the perforated plate because vortexing is unlikely with the pocket strainer design

Staff Evaluation

The staff reviewed the licensee's evaluation regarding vortexing, and also observed a portion of the plant-specific large-scale testing conducted on a strainer module similar to those installed in the plant. For verifying whether vortexing occurs, the licensee concluded that the test module was identical to the plant strainer modules. The staff agreed with the licensee and the strainer vendor that vortex formation at the currently predicted sump level, and even at decreased levels, would not occur because there is a large margin between the operating sump level and the predicted sump level at which vortexing will occur. The prediction is based on testing performed by the strainer vendor. The testing resulted in a correlation using the Froude number to predict vortex formation for various strainer submergences. The ONS strainer has a large margin to vortex formation based on the CCI testing and correlation.

3.6.6 Head Loss and Vortex Evaluation Conclusions

Head Loss Evaluation

The results of the head loss tests scaled to 9000 gpm show that the design maximum of 0.1 ft of head loss is not exceeded for the worst case expected LOCA debris loading at the design accident temperature. The 9000 gpm flow rate tested is well in excess of the actual expected maximum flow rate of 7839 gpm. This provides some conservatism to the evaluation. Because there are two ECCS suction lines from the common sump the licensee conducted evaluations for worst case asymmetric pipe flows in these lines [70].

The licensee performed plant-specific prototypical strainer head-loss testing. The system input evaluation, the testing matrix, the testing procedures and the results were reviewed during the audit. Because the estimated head loss based on the maximum measured head loss combined with the calculated clean strainer head loss is very low, the staff considers that the new CCI strainer will likely not cause significant head loss to challenge the ECCS NPSH margin (without consideration for any potential head loss change due to chemical effects).

Vortex Evaluation

The licensee and its strainer vendor performed plant-specific prototypical strainer head loss testing and observed no vortex formation during the tests. In addition, CCI performed an evaluation to determine the margin of the ONS strainer to vortexing. The vortex evaluation was reviewed during the audit, and portions of the strainer testing were witnessed by the NRC staff. Because the estimated margin to vortex formation is very large, the staff has confidence that the new CCI strainer will not cause air entrainment that would challenge the ECCS pump operation.

3.7 Net Positive Suction Head - Emergency Sump Recirculation

The licensee performed NPSH margin calculations for pumps credited with taking suction from the RBES to provide long-term cooling to the reactor core and Reactor Building following postulated accidents. At ONS, these pumps include the LPI pumps and the BS pumps.

The staff reviewed the significant models and assumptions of the licensee's NPSH calculations and discussed these calculations with licensee personnel during the audit. The staff's review used guidance provided by Regulatory Guide 1.82 [9], Generic Letter 97-04 [19], the NRC Audit

Plan [20], NEI 04-07 [7], and the SE on NEI 04-07 [8].

3.7.1 Summary of NPSH Margin Calculation Results

Table 10 below summarizes the assumed operating conditions and NPSH margin results for the LPI and BS pumps. The LPI pump results reported here were computed for Unit 1 [60] based upon the licensee's assessment that the Unit 1 NPSH margins are bounding for Units 2 and 3 [55]. The BS pump results were developed as a composite model of all three units. The licensee considered this composite model to be applicable to all three units [61].

Table 10 Assumed Operating Conditions and NPSH Margins for LPI and BS Pumps

Pump	Sump Water Temperature (°F)	Pump Flow Rate (gpm)	NPSHa (ft)	NPSHr (ft)	NPSH Margin (ft)
LPI [60]	240	3389	14.55	14.2	0.35
BS [61]	240	1195	16.72	17.2	-0.5

The staff found the licensee's descriptions within the NPSH calculations to be limited and to lack adequate discussion of some important technical assumptions and models [60, 61]. The licensee compensated for the calculations' lack of description by providing additional information and clarifications during the onsite audit. The licensee performed NPSH calculations for the LPI and BS pumps using a model and input parameters that the licensee considered to be conservative and applicable to both a large- and a small-break LOCA. Thus, the licensee considered the results presented in Table 10 as applicable for both large- and small-break LOCA conditions.

Table 10 shows that the NPSH margin computed for the LPI pumps is small, and the margin computed for the BS pumps is negative. To address the calculated negative NPSH margin for the BS pumps, the licensee currently credits a limited amount of containment overpressure to demonstrate that these pumps will function as required to mitigate the limiting design-basis LOCA. The licensee's credit for containment overpressure is discussed further in Section 3.7.3 of this audit report.

The models, assumptions, and results for the licensee's NPSH calculations are discussed in further detail below.

3.7.2 Summary of NPSH Margin Calculation Methodology

The definition of NPSH margin from Regulatory Guide 1.82 [9] is the difference between the NPSH available (NPSHa) and NPSH required (NPSHr). Regulatory Guide 1.82 defines NPSHa as the total suction head of liquid, determined at the first stage impeller, less the absolute vapor pressure of the liquid. Regulatory Guide 1.82 defines NPSHr as the amount of suction head, over vapor pressure, required to prevent more than 3% loss in total head of the first stage of the pump (due to factors such as cavitation and the release of dissolved gas) at a specific capacity. For convenience, NPSH values are generally reported as pressure heads, in units of feet of water.

In general, the NPSHa is computed as the difference between the containment atmosphere pressure and the vapor pressure of the sump water at its assumed temperature, plus the height of water from the surface of the containment pool to the pump inlet centerline, minus the hydraulic losses for the flow path from the flow inlet at the containment floor to the pump inlet nozzle (not including the head loss contribution from the sump strainer and debris bed, which are accounted for separately).

The results in Table 10 are based upon the assumption that the Reactor Building atmosphere pressure is equal to the sump water temperature (i.e., there is no contribution from containment overpressure). As discussed subsequently, however, containment overpressure is credited for the BS pumps in a separate calculation. The staff previously approved the limited amount of overpressure the licensee currently credits for the BS pumps [66].

The height of water from the surface of the Reactor Building pool to the pump inlet centerline was obtained from a combination of plant isometric drawings and calculations of minimum water level on the Reactor Building floor. This calculation approach is standard practice, and is therefore considered acceptable.

The licensee computed hydraulic piping losses using the KYPipe software package [62], a standard single-phase fluid hydraulics methodology. This piping network software is a standard engineering tool and is similar to the Crane methodology [18]; therefore, its use for the hydraulic piping network flow calculations is considered reasonable.

Based on the audit review, the staff concluded that standard definitions and methodologies associated with NPSH margin analysis were generally used in the licensee's documents (with the exception of taking credit for containment overpressure, as discussed above). A more detailed discussion of the main parameters influencing the calculated NPSH margin for the LPI and BS pumps is presented in Section 3.7.3.

3.7.3 Parameters Influencing NPSH Margin

The licensee's approach for calculating a conservative NPSH margin for the LPI and BS pumps was to identify a set of bounding scenarios and parameters applicable to both large- and small-break LOCA conditions.

<u>Emergency Core Cooling System Configuration at Switchover to Recirculation</u>

The ECCS and RBS system consist of two independent trains [21]. Each ECCS train contains one HPI pump, one LPI pump, and one core flood tank. The ECCS also contains spare LPI and HPI pumps.

Each train of the ECCS and RBS system draws water from the borated water storage tank during the injection phase of a LOCA, and from the RBES during the recirculation phase of the LOCA. At the time of switchover to sump recirculation, plant operators transfer the suctions of the LPI and BS pumps from the borated water storage tank to the RBES, which serves as their long-term water source. The discharge side of each of the two LPI trains has a connection to the suction of the HPI pumps for use during the recirculation phase of a LOCA when primary system pressure requires high injection pressure.

Pump Capacities

The NPSH margin calculation entails specification of the limiting pump flow rate which enters the calculation. The assumed pump flow rate influences the calculated NPSH through its effect on the hydraulic losses on the suction side of the pump and its effect on the NPSHr. The licensee's NPSH calculations [60,61] provide a limited explanation of the basis for the somewhat complex process used to select bounding pump flows. The following is the licensee's basis for the analysis that was performed in the NPSH calculations for the LPI and BS pumps, as related by the licensee verbally during the onsite audit and in a follow-up telephone conversation. The staff based its conclusions on this verbal information because the logic is clearly valid based on engineering principles. However, the licensee should document this information.

LPI pumps

Based upon calculations using the LPI system hydraulic model, the licensee stated that a core flood line break the scenario which maximizes the flow rate for an LPI pump. This break is postulated to occur between the reactor vessel and the check valve on the core flood tank line that is nearest to the vessel. The system hydraulic calculation [60] conservatively calculates LPI discharge flow under the assumption that the RCS is at zero pressure. Additional conservative assumptions used in calculating the LPI discharge flow are that one LPI train is assumed not to be functioning, and both trains of HPI and BS pumps are assumed not to be functioning. The functioning LPI train is selected as the one which provides the least hydraulic resistance [60]. While the core flood line break is a small-break LOCA scenario, the licensee stated that the flow rate computed based upon the above assumptions also applies as an upper limit for a large-break LOCA, since the LPI flow is computed assuming that it discharges to an RCS at zero pressure, rather than through a flow path that includes the reactor vessel internals and flow to the break. The licensee stated that the flow rate of 3389 gpm shown in Table 10 was computed [60] using the above assumptions.

The staff considers these arguments concerning the maximum achievable LPI flow rate to be physically reasonable. With one functioning train rather than two, the LPI flow through the operating train could be greater than with both pumps functional because of the potential for reduced back pressure on the fluid in the system. Further, valving out the HPI pump from the train would force more of the flow through the break, a lower resistance path than flow into the vessel. In the event that the HPI pump was not valved out with one LPI pump operational, the operator would throttle the HPI flow so that the LPI flow is less than 3100 gpm [63]. The staff also considers it physically reasonable that the computed flow rate of 3389 gpm bounds the case of a large-break LOCA for the same reasons as cited above.

The hydraulic head loss calculations for the LPI NPSH margin were calculated assuming 3389 gpm for the LPI flow and 1200 gpm for the RBS flow [60]. The RBS flow rate is acceptable since it represents a conservative estimate, as described below.

Building Spray Pumps

The maximum flow rate generated by the BS pumps was computed using the hydraulic model of the ECCS and RBS system. For this calculation Train 'B' was used, based

upon previous calculations that showed that Train 'B' was limiting in terms of NPSH margin. The ECCS and RBS system hydraulic model used conservative pump characteristics for the Train 'B' BS pump and the predicted spray flow rate was 1195 gpm.

For the RBS NPSH margin calculation, it was assumed that the LPI flow rate is 3495 gpm, a conservative estimate of the LPI pump flow rate, based upon the discussion above. Thus, the hydraulic head loss calculation for the BS pump was computed using a flow rate of 1195 gpm for the BS pump and 3495 gpm for the LPI pump.

Minimum Water Level

The minimum water level was computed assuming that both the borated water storage tank and the core flood tanks discharge their water inventories during a LOCA [34]. For computation of the minimum Reactor Building water level, it is conservatively assumed that the RCS remains full, corresponding to a break at the top of a hot leg above the steam generator. The borated water storage tank is assumed to be drawn down to a level of 6 feet, based upon Section 6.1.3 of the ONS Updated Final Safety Analysis Report and the plant emergency operating procedures [63].

The CFT volumes are based upon the minimum Technical Specification requirements and incorporate an instrument uncertainty correction. The CFTs begin to flood the core during a LOCA once the RCS system pressure decreases to 600 psig [21, Section 6.3]. The LPI pumps are automatically actuated at a primary system pressure of 550 psig [21, Section 6.3]. However, since the LPI pumps have a dead head pressure of approximately 185 psig [21, Section 6.3], they would not provide flow to the core until the RCS pressure decreases below this value.

For a large-break LOCA the RCS pressure would rapidly decrease, and the CFTs would discharge completely prior to the operation of the LPI pumps in recirculation mode. Based upon the adjusted Reactor Building floor area and the CFT liquid volumes assumed in the licensee's minimum water level calculation, each CFT would provide a contribution of roughly 0.11 ft (1.4 inches) to the water level inside the Reactor Building.

For a small-break LOCA the RCS pressure would decline more slowly. Although full credit for the CFT inventory is taken in the minimum water level calculation, the licensee's calculations do not adequately justify the assumption that both CFTs would have fully discharged by the time the RCS pressure is reduced to the discharge pressure of the LPI pumps. Based upon simplified approximations using ideal gas laws, the staff expects that most of the water volume of the CFT with the intact discharge line would be drained by the time the LPI pumps begin providing flow to the RCS. Although the staff's simplified estimates suggest that the licensee's minimum water level may allow a slight nonconservative credit for water volume from the CFTs over a limited range of RCS pressures, based upon the LPI pump curve provided by the licensee [61], the staff judged that the reduced NPSHr for the LPI pump flows achievable in this pressure range would more than offset the slightly reduced water volume contribution from the CFTs with respect to the resulting NPSH margin values. Furthermore, the staff noted that, for the core flood line break case analyzed by the licensee as the most bounding case for NPSH margin, the CFT associated with the broken line would completely dump into the Reactor Building. In light of the above discussion, the staff considered it likely that the licensee's crediting of the full CFT inventory for small-break LOCAs would not have a significant nonconservative effect on the

NPSH calculation. However, the staff considered it worthwhile for the licensee to provide confirmation and clarification within the NPSH calculation that this issue is not significant.

The water level above the mean Reactor Building floor level was computed from the net water volume reaching the Reactor Building floor and an adjusted floor surface area occupied by the water. The net water volume was calculated using the water volume discharged from the storage tanks, and adjusted by subtracting water deposited in various sumps, cavities, canals and spray headers. Volume adjustments were made due to irregular Reactor Building geometry such as a sloping floor and curvature at the floor-to-wall joints. Replacement of hot primary system water with cool water from the Borated water storage tank was considered. Areas and volumes of structures in the Reactor Building were also included.

Two sources of water holdup that were not considered that could reduce the licensee's calculated water level to a limited extent are (1) holdup of spray water droplets in the Reactor Building atmosphere, and (2) holdup of water as condensate film on surfaces of the Reactor Building. While it is expected that these factors will involve minor corrections, the NPSH margins are relatively small for ONS. Thus, even small corrections could have a non-negligible effect on the magnitude of the calculated margin, and these two sources of water holdup, along with the basis for conclusions regarding the impact of CFTs on sump level during a small-break LOCA, are designated as **Open Item 3.7.1**. This audit report discusses an additional water holdup mechanism associated with upstream debris accumulation in the refueling canal and reactor cavity drains. Depending upon the resolution of Open Item 5.2-1 (pg 64), the licensee should consider whether additional hold up in the refueling canal and/or reactor cavity should be accounted for in the minimum water level calculation.

The minimum water level above the mean Reactor Building floor level is computed in [34] as 4.4 ft of water. The water level used in the NPSH margin calculation[60, 61] is 4.27 ft. The calculated NPSH margins in Table 10, therefore, contain a small conservatism as a result of this difference, which may be of sufficient magnitude to address the issues identified in Open Item 3.7-1 (pg 54).

Based upon the discussion above, the staff considered the calculated minimum water level as reasonable, provided that the licensee's resolves Open Item 3.7.1, involving spray droplets held up in the Reactor Building atmosphere and water held up in condensate films on Reactor Building structures.

Sump Water Temperature

The NPSH calculations for both the LPI and BS pumps were computed assuming a sump water temperature of 240 °F [60, 61]. This temperature is based upon RELAP5 calculations for a large-break LOCA [64]. The calculations show that, at the time of switchover to recirculation cooling, the sump water temperature is 239 °F. This temperature subsequently decreases to approximately 140 °F later in the transient.

The choice of the upper limit value of 240 °F from the analyzed range of sump water temperatures discussed above minimizes the water density, thereby maximizing the static head of liquid. This choice also minimizes the kinematic viscosity of the sump water and, hence, minimizes head losses through the ECCS piping network. Both of these factors result in modest increases to NPSHa as the sump water temperature is increased.

Although by itself, this observation suggests that the minimum sump water temperature in the expected range would be most appropriate for the NPSHa calculation, the staff considered the licensee's choice of 240 °F for the sump pool temperature to be appropriate because this value bounds the expected sump pool temperature at the time of the switchover to sump recirculation, when the NPSH margin reaches its minimum. Subsequently, the NPSH margin for the LPI and BS pumps would quickly increase because of the licensee's credit for containment overpressure (i.e., the difference between the Reactor Building pressure and the vapor pressure of the sump fluid). The available overpressure would quickly increase following the switchover to sump recirculation due to the removal of heat from the water in the containment sump pool via the LPI coolers and, initially, the spraying of relatively warm sump water into the containment atmosphere. The increase in available overpressure following switchover would substantially outweigh the less significant temperature dependence associated with the static head and frictional loss terms discussed in the previous paragraph. Therefore, the staff considered the choice of 240 °F for the containment sump pool temperature to be appropriate for the licensee's NPSH calculation.

Containment Pressure

In a safety evaluation dated July 19, 1999, the staff approved the use of a limited quantity of containment overpressure in the licensee's NPSH calculations for the BS and LPI pumps [66]. Specifically, the staff concluded that crediting 2.2 psig above the saturated vapor pressure for the time period of approximately 3000 to 30000 seconds (approximately 7.5 hours) after a hot-leg LOCA is acceptable for ONS.

Based on the positive NPSH margin presented above in Table 10 for the LPI pumps, the licensee stated that crediting containment overpressure for the these pumps is not currently considered necessary. However, during the audit, the licensee stated verbally that the LPI pump NPSH calculations had not been finalized and that some credit may still be necessary.

As shown in Table 10, the licensee's calculations demonstrate that the BS pumps' NPSH margins are negative without credit for containment overpressure. The magnitude of their negative margin (0.5 ft) is smaller than the 2.2 psig (approximately 5 ft) of overpressure credit allowed in the current licensing basis for ONS. However, as noted in the staff's quicklook report from a November 2005 site visit [65], licensee calculations show that the switchover to recirculation could occur earlier than had previously been analyzed. As a result, for the most limiting accident condition, overpressure credit would be required at 2600 seconds rather than the 3000 seconds that was approved in the staff's safety evaluation dated July 19, 1999 [66].

To address this issue, the licensee is performing a revised analysis to support crediting containment overpressure for the BS pumps for the more limiting case of a switchover to RBES recirculation at 2600 seconds. Although part of the licensee's calculations supporting the crediting of containment overpressure were provided to the staff during the audit, a formal review of these calculations was not within the audit scope.

NPSHr and the Hot Fluid Correction Factor

NPSHr values for both the LPI and BS pumps were determined based upon conservative pump characteristics, which considered variations in the frequency of the incoming emergency

electrical power supplied by the Keowee hydroelectric station [60, 61], source of emergency power for ONS. The hot fluid correction factor for NPSHr is not used in the analysis, as per NRC guidance [9]. The licensee's specification of NPSHr is acceptable because it follows conservative NRC guidance.

3.7.4 Net Positive Suction Head Summary

The licensee performed the NPSH margin calculations using standard definitions and a standard single-phase hydraulics methodology. The assumptions and the selection of physical parameters that provide the numerical basis for the calculations generally follow conservative guidance provided by Regulatory Guide 1.82 [9]. The staff considered the values of the parameters used in the NPSH calculations to be largely reasonable. The licensee currently credits a limited amount of containment overpressure for the BS pumps, which was previously approved by the NRC staff. The licensee is revising the existing overpressure calculation, but the revised calculation was not within the scope of the audit review. In conclusion, based upon the discussion above, the staff considered the NPSH margin results computed by the licensee to be sufficiently conservative provided that the licensee acceptably resolves Open Item 3.7.1 (pg 54).

3.8 Coatings Evaluation

3.8.1 Coatings Zone of Influence

The quantities of LOCA-generated qualified coatings debris assumed by the licensee were based on applying the spherical ZOI model. The NRC SE recommends a ZOI for qualified coatings with an equivalent radius of 10 length/diameter for the largest pipe. The controlling quantity of qualified coatings debris is based on a 10 length/diameter ZOI radius about a 36-inch hot leg break. The staff finds the licensee's treatment of the ZOI for coatings acceptable because it is consistent with the NRC SE.

3.8.2 Coatings Debris Characteristics

The NRC staff's SE addresses two distinct scenarios for formation of a fiber bed on the sump screen surface. For a thin bed case, the SE states that all coatings debris should be treated as particulate and assumes 100 percent transport to the sump screen. For the case in which no thin bed is formed, the staff's SE states that the coating debris should be sized based on plant-specific analyses for debris generated from within the ZOI and from outside the ZOI, or that a default chip size equivalent to the area of the sump screen openings should be used.

The existing analysis and headloss testing for ONS was performed with fine particulate representing coatings debris (10 micron spheres in the analysis and stone flour as a fine particulate surrogate in the headloss testing). The licensee justified the use of particulate based on the analysis and the headloss testing assumption that there would be a sufficient amount of fibrous debris to create a continuous fiber bed across the strainer surface.

ONS representatives stated that coating chips will be used (rather than particulate) for the final analysis and headloss testing. The planned testing will be based on a fiber source term that is more representative of the actual amount of fiber present in the Reactor Buildings. The reduced fiber load is not sufficient to create a continuous fiber bed across the strainer surface, therefore

the use of coating chips is justified. See the discussion in Section 3.6.3.1.3 (pg 41), regarding the staff's recent guidance on the form of coatings debris to be assumed for head loss testing.

The licensee plans to use a range of sizes for coating chips. The amount of each size chip that arrives at the strainer surface will be determined based on transport testing that ONS has performed. Based on discussions with ONS representatives, the staff finds that the plans for headloss testing with coating chips and the transport testing used to determine the amount of coatings chips will be conservative with respect to the treatment of debris from protective coatings. Further discussion of debris transport and headloss testing is provided in Sections 3.5 (pg 21) and 3.6 (pg 30).

During interaction with PWR licensees for resolution of GSI-191, the NRC staff has questioned the current industry method of assessing qualified coatings. The staff has asked licensees to either justify that their assessment techniques can accurately identify the amount of degraded qualified coatings in containment, or assume all of the coatings fail. The licensee stated that they will rely on the results of an ongoing test program conducted by the Electric Power Research Institute and the Nuclear Utilities Coatings Council to validate their assessment techniques at ONS. The referenced testing will subject both visually sound and visually degraded coatings to physical testing (i.e., adhesion tests) in an attempt to show that visual assessments are capable of identifying coatings that would not remain adhered during a design basis accident. The results of this testing have not been completed, and therefore have not been reviewed by the NRC staff. ONS was scheduled to participate in the Nuclear Utilities Coatings Council testing during May 2007. Assessment of qualified coatings is identified as **Open Item 3.8-1** pending industry validation testing and NRC staff review of the results.

Although assessment of qualified coatings remains an open item for ONS, the staff noted that the licensee has performed significant remediation of degraded coatings and has additional remediation planned for future refueling outages. ONS has also performed some physical testing of containment coatings using knife-cut adhesion tests in accordance with ASTM Standard D6677. The knife-cut adhesion tests provide a quick, qualitative assessment of the material condition of the coatings without quantifying the level of degradation. Elcometer adhesion testing will be performed in accordance with ASTM Standard D4541 at ONS as part of the Nuclear Utilities Coatings Council testing in May 2007. Finally, ONS has removed several pieces of coated spray support steel from the Reactor Building and subjected those samples to a simulated design basis accident test in accordance with ASTM Standard D3911. The simulated design basis accident tests address staff concerns that some of the inorganic zinc primer that is left in service following failure of the epoxy topcoat may itself fail and become a debris source. The test shows that only a thin layer on the surface of the exposed inorganic zinc primer fails during the simulated design basis accident test. The licensee indicated that it will include a portion of all exposed inorganic zinc in the Reactor Building in the analysis and headloss testing. The amount of inorganic zinc included as debris is based on the reduction in dry-film thickness determined by the design basis accident test. The NRC staff is encouraged by the efforts to remediate degraded coatings and characterize the condition of existing coatings at ONS. The activities described above, including the pending Nuclear Utilities Coatings Council tests, will help to resolve Open Item 3.8-1 on coating assessment and provide a basis for conservatively accounting for all debris associated with unqualified and degraded qualified protective coatings.

4.0 DESIGN AND ADMINISTRATIVE CONTROLS

4.1 Debris Source Term

Section 5.1 of NEI 04-07 [7] and the NRC staff's accompanying safety evaluation (SE) [8] discuss five categories of design and operational refinements associated with the debris source term considered in the sump performance analysis.

- housekeeping and foreign material exclusion programs
- change-out of insulation
- modification of existing insulation
- modification of other equipment or systems
- modification or improvement of coatings program

The SE states that these additional refinements should be evaluated for their potential to improve plant safety and reduce risks associated with sump screen blockage. The staff's discussion below describes the licensee's procedures and planned or completed actions in each of these areas.

4.1.1 Housekeeping and Foreign Material Exclusion Programs

The staff reviewed ONS procedures for Reactor Building inspection and closeout [41, 42]; an excerpt from the plant engineering directives manual regarding changes to the plant configuration [43]; procedures for material condition, housekeeping, cleanliness, and foreign material exclusion [44]; a site directive for Reactor Building material control [45]; and a problem investigation process report regarding labels and signs in the Reactor Building [46]. These plant documents provide administrative controls to ensure that the debris source term affecting the recirculation sump following a LOCA is bounded by the existing analysis. Based upon the administrative controls for plant modifications and Reactor Building materials listed above, the licensee indicated that materials introduced into the Reactor Building as part of the plant modification process (e.g., insulation materials, equipment signs and tags) would be reviewed to ensure that adverse interactions with the RBES would not occur.

In discussions during the audit, the licensee stated that Reactor Building washdowns are conducted at the beginning and end of each refueling outage as part of the plant cleanliness program. The licensee's September 2005 GL 2004-02 response further indicates that the washdown uses high-pressure water spray to clean the Reactor Building from the fourth floor to the basement [48]. The licensee stated that, although measurement and quantification of the washed-down debris is not performed, conducting a washdown at the beginning of a refueling outage provides a rough indication of the quantity of latent debris and foreign material in the Reactor Building following an operating cycle. In light of the fact that the licensee does not conduct periodic, quantified measurements of latent debris in the Reactor Building, the staff considered the licensee's practice of conducting Reactor Building washdowns at the beginning and end of each refueling outage to be adequate to provide assurance that assumptions in strainer analyses regarding Reactor Building cleanliness remain valid.

The licensee stated that washdown procedures contain precautions for avoiding inadvertent debris washdown into the RBES. However, based upon discussions with the licensee during the

audit, it was not clear that the licensee's procedures contain adequate precautions to ensure that washdowns do not result in washed-down debris entering and clogging fluid drainage paths (e.g., floor drains) in the Reactor Building upstream of the strainers. The staff designated it **Open Item 4.1-1** for the licensee to ensure that any drainage paths necessary for ensuring adequate water drainage to the RBES do not become blocked as a result of the Reactor Building washdown procedure.

With the exception of Open Item 4.1-1, the staff found that the licensee's housekeeping and foreign material exclusion programs appear to adequately control their respective processes for maintenance of the debris source term as needed to maintain adequate ECCS strainer functionality.

4.1.2 Change-Out of Insulation

During the audit, the licensee stated that the primary type of thermal insulation in the ONS Reactor Building has historically been reflective metallic insulation. The licensee stated that, subsequent to the original plant design and construction, fibrous insulation was installed in the Reactor Building as a replacement for some of the original RMI insulation. As part of the licensee's activities in response to Generic Letter 2004-02 [1], the licensee has removed fibrous insulation that has been identified as being within the zones of influence of analyzed pipe ruptures. In particular, the baseline analysis report [35] indicates that fibrous insulation potentially vulnerable to pipe ruptures in the vicinity of the letdown heat exchangers has been removed. In addition, the baseline analysis report refers to fibrous insulation on Reactor Building Cooling Unit 3B as being potentially vulnerable to a pipe rupture at the top of Steam Generator 3A. However, based upon discussions with the licensee during the audit, since the completion of the baseline analysis report, the fibrous insulation on Reactor Building Cooling Unit 3B has been removed from the Reactor Building. As a result of the licensee's previous change-out of limited quantities of fibrous insulation, the licensee stated that there is currently no fibrous insulation within the ZOIs of analyzed pipe ruptures.

4.1.3 Modification of Existing Insulation

The licensee has not committed to the modification of existing insulation to reduce the debris source term as a corrective action in response to GL 2004-02.

4.1.4 Modification of Other Equipment or Systems

As described above in Section 4.1.1 above, the licensee plans to implement changes to the plant labeling process to ensure that any additional labels or signs installed inside the Reactor Building are evaluated as potential debris sources. As stated in Section 1.3 of this report (pg 5), the licensee's planned improvements to the plant labeling process were considered a licensee commitment in the September 2005 response to GL 2004-02 [48].

4.1.5 Modification or Improvement of Coatings Program

In the slide presentation provided to the staff during the audit kick-off meeting [3], the licensee described activities associated with improving the condition of coatings used in the ONS Reactor Buildings. The licensee's slides stated that a long-term strategy has been developed to reduce

the quantity of failed coatings and identified the surface areas of remediated coatings and of coatings planned for remediation by 2007, which are listed in the table below:

Table 11 Reactor Building Coating Remediation at ONS [3]		
Unit	Remediated Surface Area (ft^2)	Additional Surface Area Planned for Remediation Through 2007 (ft^2)
1	8700	N/A
2	9400	~ 7,000
3	1570	~ 5,000 – 10,000

Table 11 demonstrates that the licensee has made significant efforts to improve the condition of the coatings used inside the Reactor Building. The licensee noted that some of the coatings that are scheduled for remediation are in locations that are relatively difficult to access, such as the Reactor Building dome. The licensee also noted during the audit that the RBES had been recoated. As noted previously in Section 1.3 of this report (pg 5), the licensee's September 2005, GL 2004-02 response further indicates that the licensee is planning to enhance the ONS coatings program by documenting and trending evaluations performed on degraded coatings [48].

4.2 Screen Modifications

Section 5.3 of the approved GR 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 of strainers, and active strainers. The 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.

Staff Evaluation:

Based on the review described in Section 3.0 of this audit report, the staff believes that the new sump design will be able to accommodate the maximum volume of debris. The specific design features appear to preclude the formation of a thin bed with insulation and latent debris. However, Open Item 6.4-1 (pg 73), has been identified that relates to the possibility of a thin bed when chemical precipitants are considered.

5.0 ADDITIONAL DESIGN CONSIDERATIONS

5.1 Sump Structural Analysis

The RBES strainers for the ONS Units 1, 2 and 3 have been enlarged as part of the resolution of GL 2004-02. The RBES strainers for each unit are in the RBES enclosure located in the basement floor of the Reactor Building (elevation 774 ft.). The strainers are surrounded on three sides by a 4 ft wall and partially protected on the other side by a 4'x6' column. The new, larger RBES strainers extend above the floor level by approximately 3.5 feet. See Figure 5 for a representative composite drawing.

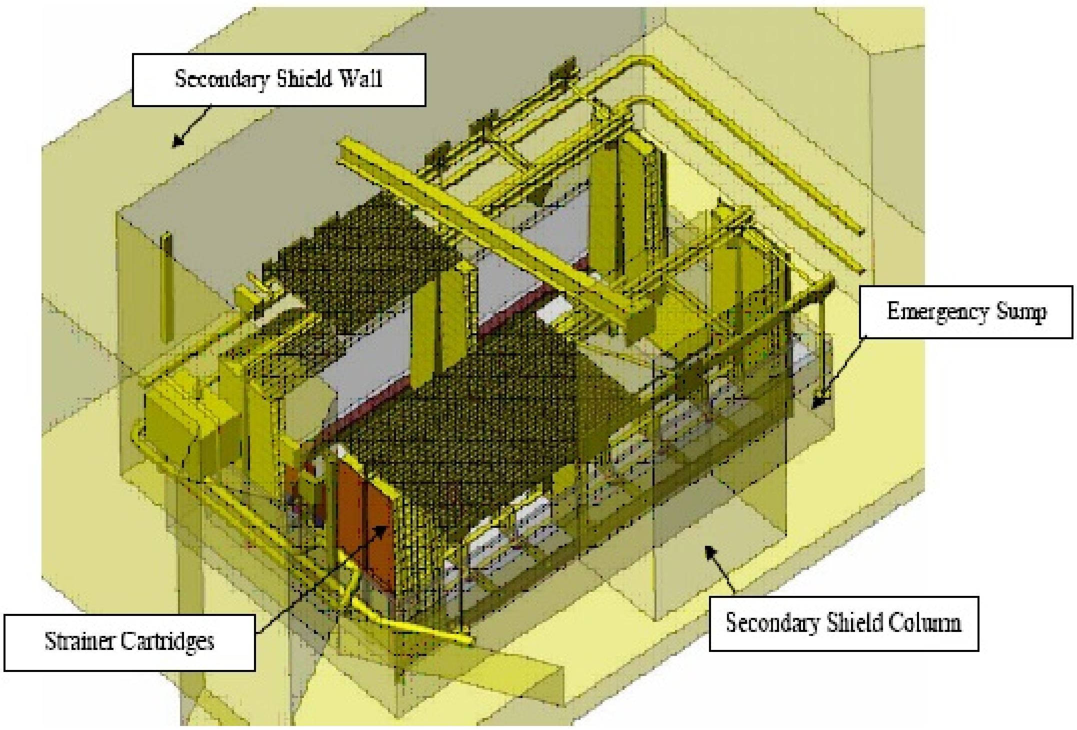

Figure 5 Composite Drawing of Sump Structures

The licensee issued Engineering Design Change Packages for the RBES strainer replacements at ONS. The licensee evaluated the safety impacts of the packages by screening pursuant to 10 CFR 50.59, considering the Updated Final Safety Analysis Report, and concluded that the design change does not adversely impact sump design functions. The licensee concluded that the structural calculations performed as part of the Engineering Design Change Package have shown satisfactory results.

The staff reviewed the following calculation reports (contained in OSC-8021) which are referenced in the structural portion of the Engineering Design Change Package:

1. CCI calculation numbers 3SA-096.006 (Units 2 and 3) [91] and 3SA-096.028, (Unit 1) [92]. See below.

2. Calculations S-007 (Unit 1) [93], S-001 (Unit 2) [94], and S-004 (Unit 3) [95], which were performed by Stone & Webster. These calculations evaluate the interface loads of the anchor plates (obtained from [91] and [92]) and the effects they have on the surrounding wall, building column and to the sump floor and sump walls to which they attach. Based on these evaluations, the licensee concluded that all structures to which the new emergency strainer support system is attached are acceptable per the design criteria of ONS.

3. Miscellaneous other calculations that were also performed to account for the structural integrity of the supports of the relocated level instruments, grating for the catwalks above the strainers and for the stainless steel wire woven cloth used to fill the gaps between the end units and the sump walls. Based on its calculations, the licensee has demonstrated that these items have adequate structural integrity.

The licensee's contractors performed finite element method analysis using the computer structural analysis software ANSYS to provide structural evaluations of the RBES strainers and their support structures. These structural evaluations are detailed in CCI calculation reports [91 (Units 2 & 3)] and [92 (Unit 1)]. Loads considered in the analysis included:

> Design differential pressure (ΔP), design temperature (for evaluating material properties), structural weight (W =strainer cartridges + support structure + grating), debris weight (WD), live load (LL) and Earthquake (Response Spectra) which conservatively uses a hypothetical earthquake (HE) defined by a floor response spectra at 0.10 g as opposed to the design basis earthquake defined by a floor response spectrum at 0.05 g.
>
> The load combinations considered were: 1) W + LL, 2) W + LL +HE (pool dry), 3) W + WD + ΔP and 4) W + WD + ΔP + HE (pool filled).
>
> For the pool filled condition the hydrodynamic water masses of the strainer components were considered in addition to the steel mass.

The structural evaluations showed that all stresses were below code allowable limits.

Double-ended ruptures in high-energy piping can result in rapid pipe movement (pipe whip) in any unrestrained direction of the broken piping segments caused by escaping high-energy fluid (jets). Pipe whipping impact and jet impingement can potentially damage the new larger RBES strainers, which extend above the basement floor level of the Reactor Building by approximately 3.5 feet. The licensee walked down the plant to identify all the piping that if broken could potentially damage the new strainers. Stone & Webster evaluated the walkdown and other information to determine if a rupture of any of the piping systems in the vicinity of the RBES strainers could impact the safety-related design function of the strainers to operate during LOCA events for pipe breaks of any size [Stone & Webster calculation reports 96, 97, and 98]. Based on these evaluations, the licensee concluded that the design function of the RBES strainers would not be compromised by jet impingement or pipe whip from any of the lines in the vicinity of the RBES strainers.

The safety-related function of the new larger RBES strainers could also be impacted by missiles generated by high-energy piping and equipment. Missiles are generally items such as valve

stems, valve bonnets, instrument thimbles, and nuts and bolts. Walkdown information, photos and plant documentation were used by Stone & Webster to identify and evaluate potential sources of missiles in the RBES area and determine if there was a design impact on the new strainers [Stone & Webster calculation reports 99, 100, and 101]. Based on these evaluations the licensee determined that the operability of the RBES strainers will not be compromised during a LOCA by missiles.

Based on review of the licensee's evaluation as discussed above, the staff finds that the licensee has met the intent of Generic Letter 2004-02 for structural integrity by providing reasonable assurance that the RBES strainer replacements at ONS Units 1, 2, and 3 are structurally adequate to perform their intended safety-related design function to operate without compromise during a LOCA event.

5.2 Upstream Effects

The staff reviewed the Reactor Building walkdown report for ONS Unit 3 [40], the baseline analysis [35], and several plant drawings of the Reactor Building in order to evaluate the licensee's treatment of debris blockage at Reactor Building drainage flow chokepoints and other upstream effects.

Attachment D to the Reactor Building walkdown report [40] describes the walkdown conducted to assess Reactor Building transport pathways for water and debris. Among the objectives of this report were identifying potential chokepoints in the Reactor Building and verifying that drainage paths are currently unblocked.

Attachment D to the walkdown report provides a general description of the ONS Reactor Building [40]. As summarized below, the licensee described therein how water discharged from the Reactor Building spray headers located at plant elevation 944'-0" would pass through various plant elevations en route to ultimately draining into the Reactor Building pool formed on elevation 777'-6".

At elevation 861'-6", the licensee stated that the flooring is primarily grating above the steam generator cavities and around the outer edge of the Reactor Building [40]. There is an ungrated area between the steam generator cavities that extends down to the operating floor and the fuel transfer canal. Portions of the floor at this elevation are concrete, and these solid portions drain to grated areas with no obstructions. Therefore, the licensee stated that spray flow will either pass through gratings into the steam generator compartments or will fall onto the operating floor at elevation 844'-6". The floor on either side of the refueling canal outside the steam generator compartments at elevation 844'-6" is described as consisting of both grating and concrete.

The licensee stated that the upper refueling canal floor can drain into the deep end of the refueling canal or to the reactor vessel cavity through the vessel seal plate annulus down through the vessel annulus and into the reactor cavity through two 3-inch annulus drain lines [40]. Water that reaches the lower refueling canal floor drains to the normal sump through two 4-inch lines. Water that reaches the reactor cavity drains to the normal sump through one 4-inch line. The licensee's walkdown report [40] identified that the details concerning these flowpaths and associated hold up assumptions are documented in the minimum post-LOCA Reactor Building water level calculation [34]. The licensee assumed that hold up in the refueling canal and reactor cavity would occur to the extent necessary to develop flow through the reactor

cavity and refueling canal drains [40]. During the audit, the licensee provided plant drawings depicting these drainage flowpaths.

The licensee stated that the flooring at elevation 825'-0" is completely composed of grating [40]. There are no large openings to the elevation below with the exceptions of the open hatchway above the equipment hatch and an open area on the southeast edge of the Reactor Building.

The licensee stated that elevation 797'-6" is the first major floor level above the basement elevation. The flooring at this elevation is composed of a combination of concrete and gratings [40]. The licensee stated that solid flooring at these elevations can drain to the grated area with no obstructions. The only large opening at this elevation was stated to be a hatch that is kept open during normal operation. Other smaller openings exist, and these openings are generally surrounded by concrete or steel curbing 3 to 6 inches in height.

Attachment D to the walkdown report also includes a specific discussion of the water drainage pathways within the steam generator compartments [40]. The licensee stated that the steam generator compartments have a vertical span between elevations 797'-6" and 861'-6". These enclosures are described as being completely covered with grating at the 861'-6" elevation. The licensee stated that the steam generator compartments are congested, noting that there are multiple levels of grated partial platforms within the compartments, but no solid floors. The licensee did not identify any water hold up concerns in the steam generator compartments.

The licensee's discussion of water flows at the basement level elevation of 776'-6" stated that there are no completely enclosed areas [40]. Although several physical features were noted that may be able to hold up some pieces of large debris (e.g., grating and curbs), the licensee stated that the flow of water to the sump would not be impeded.

Based upon the licensee's descriptions of the Reactor Building elevations in Attachment D to the walkdown report that are summarized above, with one exception described below, the staff concluded that water drainage in the ONS Reactor Buildings would not generally be susceptible to being trapped in unanalyzed hold up locations.

The staff's review of upstream effects focused upon the drainage flowpaths through the refueling canal and reactor cavity because of the potential for the drains to these large volumes to act as chokepoints for retaining substantial quantities of water if the drains were to become blocked by debris. The licensee did not provide sufficient information during the onsite audit for the staff to conclude that water hold up in the refueling canal and reactor cavity volumes due to debris blockage of drainage flowpaths would not be a concern for ONS. As a result, the staff designated **Open Item 5.2-1** for the licensee to address the potential for debris accumulation to result in blockage or partial obstruction of the refueling canal and reactor cavity drainage flowpaths. In particular, the staff noted that the licensee should consider the potential for these flowpaths to be blocked both by (1) one or more large pieces of debris and (2) by an accumulation of smaller pieces of debris. The staff also noted that a break on the reactor vessel could apparently generate debris in the reactor cavity with the potential for blocking the reactor cavity drains.

The staff noted during the audit that this Open Item 5.2-1 could be resolved in several ways, including the following: (1) demonstrating that debris capable of blocking these flowpaths could

not be transported to the flowpaths, (2) demonstrating that the existing design of the flowpaths is adequate to prevent hold up greater than assumed in existing calculations, (3) demonstrating that additional water hold up in the refueling cavity and reactor cavity would not result in adverse affects, and (4) installing a strainer over floor drains or other flow restrictions that is designed to prevent debris blockage of credited drainage flowpaths.

5.3 Downstream Effects

5.3.1 Downstream Effects - Core

The acceptance criteria for the performance of a nuclear reactor core following a LOCA are found in Section 10CFR50.46 of the Commission's regulations. The acceptance criterion dealing with the long-term cooling phase of the accident recovery is as follows:

> Long-term cooling: After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

At the request of the industry, the NRC staff provided additional interpretation for 1) the requirements and acceptance criteria for long-term core cooling once the core has quenched and reflooded and 2) for the mission time that should be used in evaluating debris ingestion effects on the reactor fuel [26].

Following a large break in the reactor system after the core has been recovered with water, long-term cooling at ONS will be accomplished by the low-pressure and high-pressure injection pumps. These pumps initially take suction from a storage tank containing borated water. When that source of water becomes depleted, the suction to the low-pressure pumps will be switched to the RBES to recirculate water to the reactor system. At that time the Reactor Building will contain all the water spilled from the reactor system and that added by the RBS. The core cooling mode by which water from the RBES is continually added to the reactor system and is recirculated as it spills from the break may be required for an extended period of time. During this long-term cooling period any debris which is washed into the RBES and which is passed through the sump screens will have a high probability of being pumped into the reactor system.

Generic Letter 2004-02 requests that holders of operating licenses for pressurized-water reactors perform evaluations of the ECCS and the containment spray recirculation functions. These evaluations are to include the potential for debris blockage at flow restrictions within the ECCS recirculation flow path downstream of the sump screen. Examples of flow restrictions which should be evaluated are the fuel assembly inlet debris screens and the spacer grids within the fuel assemblies. Debris blockage at such flow restrictions could impede or prevent the recirculation of coolant to the reactor core leading to inadequate long-term core cooling.

NRC staff concerns for debris blockage of the reactor core are primarily related to the recovery following the largest postulated reactor system piping breaks. For smaller break sizes the goal of plant operators would be to fill the reactor system and establish closed loop cooling using the decay heat removal system. Recirculation of sump water might not be required for small break sizes and if recirculation were needed, the requirements would be less than for large breaks. The amount of sump debris following a small break is expected to be less than that which would

be generated following a large break. The audit evaluation therefore emphasized long-term cooling following large piping breaks.

Following a large-break LOCA at ONS, the high-pressure ECCS pumps are aligned to inject into the reactor cold legs. The low-pressure ECCS pumps inject directly into the reactor downcomer. If the break were in a reactor system hot leg, the ECCS water would be forced through the reactor core toward the break. Core flow, including a small amount of core bypass flow, during the long-term cooling period would be equal to the total ECCS flow. If all ECCS pumps were assumed to operate, ECCS flow into the reactor system through the reactor vessel and into the core would be maximized. The maximum flow condition is evaluated since it provides the greatest potential for debris transport to the reactor core and subsequent lodging within flow restrictions.

Following a large cold leg break with injection into the reactor downcomer and cold legs, water will flow into the core but the rate of core flow will be limited by the pressure needed to overcome the flow resistance of steam generated by the core in reaching the break and by the static head of the water in the core. Eventually the rate of ECCS water reaching the core will be limited to that needed to replenish what is boiled away. The excess will be spilled out of the break, including that water injected into the intact cold legs which will flow around the upper elevations of the downcomer and reach the break without passing through the core. The long-term cooling period following a large cold leg break represents a minimum core flow condition. Core blockage by debris under these conditions would add to the resistance which must be overcome for the ECCS water to reach the core and lead to additional spillage from the break.

For the evaluation of potential core blockage following a hot leg or a cold leg break, the licensee stated that it will use the methodologies of WCAP-16406-P [28] and WCAP-16793-NP. The WCAPs describe how particulate debris with a density that is heavier than water will settle in the reactor vessel lower plenum and not be passed into the core for a sufficiently low flow velocity. The WCAPs also describe how fibrous debris with a density approximately the same as water would be carried along with the recirculated sump water but would be filtered by the sump screens and by screens located at the inlet to the fuel bundles. WCAP-16406-P and WCAP-16793-NP were recently submitted as a topical reports for NRC staff review. The staff plans to complete the review of these topical reports in 2007.

The licensee contracted with Areva to provide additional considerations by which the methodologies of WCAP-16406-P and WCAP-16793-NP can be applied to ONS [See reference 79]. Areva evaluated the amount of additional flow resistance from debris which would still allow adequate water to enter the core for core cooling. Before ECCS water can enter the core at ONS, the water first must flow from the downcomer, turn in the lower plenum and flow upward to the core. The lower plenum provides a location of low flow velocity where heavier debris might settle without reaching the core. Areva expanded on the methodology of WCAP-16406-P to provide additional equations to be used to evaluate the settling process. Areva developed a methodology by which local heating might be calculated in the event that debris lodged between the fuel bundle grid straps and the fuel rods. If debris were to be deposited behind the fuel bundle grid straps, local hot spots might be generated. The staff did not review the Areva methodology in detail, but from an abbreviated review of this material, the staff believes that the Areva methodology will be beneficial in evaluating the post-LOCA consequences from debris injection into the reactor vessel at ONS. The staff will review the licensee's application of

WCAP-16406-P during the staff's review of the GL 2004-02 supplemental responses, when the licensee's evaluations of long-term core cooling have been completed.

Following a large cold leg break, continued boiling in the core will act to concentrate the debris and chemicals in the water between the core coolant channels. Chemical reaction of the debris with the pool buffering agents and boric acid from the ECCS water in the presence of the core radiation field might change the chemical and physical nature of the mixture. Heat transfer might be affected by direct plate out of debris on the fuel rods and by accumulation of material within the fuel element spacer grids. Neither WCAP-16406-P nor the Areva methodology deals with the effect of chemicals in the recirculated water on core heat transfer or the possible precipitation of chemicals and debris by the boiling process. During a meeting with the PWR owners group April 12, 2006, to discuss issues associated with downstream effects on reactor fuel, the owners presented plans to develop another topical report with a more detailed fuel evaluation methodology. The licensee has stated that they will rely on the ongoing program by the PWR owners group for evaluating the effects of chemicals and debris on reactor core heat transfer during the long-term cooling period.

At a meeting with the PWR owners group February 7, 2007, the staff provided the owners with a list of considerations which should be addressed to resolve GSI-191 for the reactor core:

1. Methodology should account for differences in PWR RCS and ECCS designs.

   Examples:
   - Combustion Engineering plants with smaller recirculation flows may produce extended core boiling long after hot leg recirculation begins. The extended boiling period may impact concentration of debris in core, plate-out, etc..
   - Use of pressurizer spray nozzles for hot-leg recirculation should be evaluated for the potential of clogging with debris.
   - Upper Plenum Injection plants with no cold leg recirculation flow may have no means of flushing the core following a large hot leg LOCA and may need special consideration.

2. Hot spots may be produced from debris trapped by swelled and ruptured cladding.

   - Debris may collect in the restricted channels caused by clad swelling, and at the rough edges at rupture locations.
   - FLECHT tests have shown that swelled and ruptured cladding may not detrimentally affect the cladding temperature profile. The FLECHT tests did not include post-LOCA debris.

3. Long-term core boiling effects on debris and chemical concentrations in the core should be accounted for.

   - The evaluations should be similar to post-LOCA boric acid precipitation evaluations.
   - They should account for the change in water volume available to mix with constituents concentrated by the core from debris accumulation.
   - Partial blockage of the core creates alternate circulation patterns within the reactor vessel and will affect the concentration analysis.

- Will the solubility limits be exceeded for any of the material dissolved in the coolant that is being concentrated by boiling in the core?

4. Debris and chemicals which might be trapped behind spacer grids could potentially affect heat transfer from the fuel rods and should be evaluated.

   - Analyses show that a partially filled spacer grid produces only a moderate cladding temperature increase even if only axial conduction down the cladding is considered.
   - Similar analyses show that a completely filled spacer grid with only axial conduction will result in unacceptable temperatures.
   - A physical basis for determining to what extent the spacer grids can trap debris, and the ability for the debris to block heat transfer needs to be provided.
   - e evaluation needs to include the chemical and physical processes which may occur in the core during the long-term cooling period.

5. Consideration should be included for plating out of debris and/or chemicals on the fuel rods during long-term boiling.

   - Long-term boiling in the core following a large-break LOCA may last for several weeks for some designs depending on the ECCS flow and core inlet temperature.
   - The concentration of materials in the core, and the potential for plate out on the fuel rods (boiler scale) from this material should be determined.
   - When the composition and thickness of the boiler scale has been determined, the effect on fuel rod heat transfer should be evaluated.

6. The licensees need to address whether high concentrations of debris and chemicals in the core from long-term boiling can affect the natural circulation elevation head which causes coolant to enter the core.

   - For a large cold leg break, the density difference between the core and the downcomer determines the hydrostatic driving head, and consequently the flowrate into the core.
   - As boiling continues, a high concentration of debris and chemicals in the core may increase the core density and reduce the flow into the core.

7. If hot spots are found to occur, the licensee should address cladding embrittlement. Applicable experimental data for the calculated condition and type of the cladding should be presented to demonstrate that a coolable geometry is maintained.

Staff Evaluation:

The licensee continues to evaluate the post-LOCA consequences of debris ingestion into the reactor system and its effect on long-term core cooling. The licensee has stated that they will use the results from generic evaluations currently being performed by the PWR Owners Group. Although downstream evaluations were in progress during the audit, the licensee has not made any final conclusions as to whether the cores at ONS could be blocked by debris following a LOCA, and this area is incomplete and is identified as **Open Item 5.3-1**. The staff will review the

application of this methodology for ONS and the licensee's conclusions when they are submitted. The PWR Owners Group recently submitted WCAP-16793-NP, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid [102]." The staff expects to complete review of this document in 2007. ONS may choose to use this WCAP to support its analysis of in-vessel debris issues.

5.3.2 Component (Ex-Vessel) Evaluation

SE Section 7.3 DOWNSTREAM EFFECTS (Audit Guidelines)

The GR provides licensees the following guidance on evaluating the flowpaths downstream of the RBES for blockage from entrained debris. The GR and associated SE identify the following aspects to be included in the downstream evaluation:

- Flow clearance through the sump screen should be identified to determine the maximum size of particulate debris to be used in downstream component evaluations.

- An evaluation of wear and abrasion of surfaces in the emergency core cooling and containment spray systems based on flow rates to which the surfaces will be subjected and the grittiness or abrasiveness of the plant-specific ingested debris.

- A review of the effects of debris on pumps and rotating equipment, piping, valves, and heat exchangers downstream of the sump. In particular, any throttle valves installed in the ECCS for flow balancing should be evaluated for potential blockage.

- Long-term and short-term system operating lineups, conditions of operation, and mission times should be defined. For pumps and rotating equipment, an assessment of the condition and operability of the component during and following its required mission times should be performed.

- Component rotor dynamics changes and long-term effects on vibrations caused by potential wear should be evaluated, including the potential impact on pump internal loads to address such concerns as rotor and shaft cracking (NUREG/CP-0152 Vol. 5, TIA 2003-04 [80]).

- System piping, containment spray nozzles, and instrumentation tubing, should be evaluated for the settling of dusts and fines in low-flow/low fluid velocity areas. Include such components as tubing connections for differential pressure from flow orifices, elbow taps, and venturis and reactor vessel/RCS leg connections for reactor vessel level. Consideration should be given to any potential impact that matting may have on instrumentation necessary for continued long-term operation.

- Valve and heat exchanger wetted materials must be evaluated for susceptibility to wear, surface abrasion, and plugging that may alter the system flow distribution.

- Heat exchanger degradation resulting from plugging, blocking, plating of slurry materials must be evaluated with respect to overall system required hydraulic and heat removal capability.

- An overall system evaluation, integrating limiting conditions and including the potential for reduced pump/system capacity resulting from internal bypass leakage or through external leakage should be performed.

- Leakage past seals and rings caused by wear from debris fines to areas outside containment should be evaluated with respect to fluid inventory and overall accident scenario design and license bases environmental and dose consequences.

NRC Staff Audit:

ONS used PWR Owners Group WCAP-16406-P Evaluation of Downstream Sump Debris Effects in Support of GSI-191, Revision 0 [78] in their assessment of their ECCS and components. Revision 1 [28] to the PWR Owners Group document is currently under review as a topical report by the staff. The licensee evaluations of the ECCS and effects on downstream component are preliminary; based in part on the generic methodology of WCAP-16406-P, currently under review by the NRC staff. The need to complete the evaluation and to consider how the conclusions and findings associated with the staff's topical review need to be applied to the evaluation of post-LOCA downstream effects for ONS is identified as **Open Item 5.3-2**.

The staff reviewed a list and marked-up flow diagrams of all components and flowpaths considered to determine the scope of the licensee's downstream evaluation (pumps, valves, instruments, and heat exchangers, etc.). ONS representatives provided a complete and thorough listing and evaluation of instrument tubing connections. The licensee evaluation was complete and well organized. All system components and flowpaths were considered and evaluated. Piping and instrumentation drawings, operations procedures and supporting calculations were reviewed by the staff with no design discrepancies discovered.

In accordance with SE Section 7.3, the staff reviewed design and license mission times and system lineups to support mission critical systems. The mission times for HPI, LPI, and RBS, were defined as 48 hours, 30 days and 7 days, respectively. The licensee did not provide a clearly defined basis or supporting documentation for these mission times, which is identified as **Open Item 5.3-3**. Line-ups, flows and pressures used to bound downstream evaluations were in all cases conservative with respect to review and evaluation of downstream components.

The staff also reviewed small-, medium-, and large-break LOCA scenarios to assess system operation. ECCS operation during small- , medium-, and large-break LOCAs appeared to be adequate. Flows and pressures achieved meet the requirements of the ONS accident analysis.

The staff reviewed the licensee's analysis of the extent of air entrainment (apart from vortexing) , and concurs that there is no significant air entrainment issue with the ECCS that would either impact ECCS pump operation or cause air pockets in ECCS piping. The licensee adequately addressed the potential for waterhammer and slug flow.

The ONS characterization and assumed properties of bypass debris in ECCS post-LOCA fluid (abrasiveness, solids content, and debris characterization) is ongoing and not yet complete. For the initial assessment [61], 100 percent pass-through was assumed. The result of the initial assumption is the potential clogging of cyclone separators, plugging of seal flush lines and excessive wear of LPI pump wear rings and hubs. The licensee is continuing to evaluate the

properties of the post-LOCA ECCS fluid and is considered part of Open Item 5.3-2 above relating to incomplete status of downstream evaluation [References 81, 82 and 83].

The staff reviewed design documents to verify opening sizes and running clearances. No discrepancies were noted, and information was correctly incorporated into the licensee evaluation [61].

The SE identifies the vulnerability of the high-pressure safety injection throttle valves to clog during ECCS operation. ONS has CCI Drag® Valves in the HPI system and CCI Drag® Valve Flow Restrictors in the LPI system [Drawings 84, 85 and 86]. The susceptibility of these valves to clog or plug, thus reducing system flow, was under review by the licensee at the time of the audit [87]. The staff examined the following operating procedures, which provide direction such that if an operator chooses to open a throttle valve to maintain ECCS flow, there is adequate indication and alarm;

Emergency Procedure EP/3/A/1800/001 K, Forced Cooldown, Revision 35
Emergency Procedure EP/3/A/1800/001 Enclosure 5.1, ES Actuation
Emergency Procedure EP/3/A/1800/001 Enclosure 5.5, Pzr and LDST Level Control
Emergency Procedure EP/3/A/1800/001 Rule 2, Loss of SCM
Emergency Procedure EP/3/A/1800/001 Rule 4, Initiation of HPI Forced Cooling

The licensee provided a listing of the materials of all wetted downstream surfaces (wear rings, pump internals, bearings, throttle valve plug, and seat materials). The staff examined this list and checked materials of construction by reviewing design drawings and licensee technical manuals. The staff noted that HPI pump internals are in the process of being hardfaced [88]. The modifications were initiated in 2001 as part of an overall HPI pump improvement process. This information was correctly incorporated into the HPI pump evaluations. The HPI pump modifications [88] include removing the 9th stage impeller, changing the pumps from 24-stage to 23-stage pumps. The staff found that not all design drawings were updated to indicate this aspect of the change. Operating and maintenance manuals correctly depict the changes [90]. The licensee generated PIP O-07-01675 [89] to investigate and track this discrepancy.

SE Section 7.3 notes the potential to clog or degrade, equipment strainers, cyclone separators, or other components. ONS has cyclone separators on its HPI, LPI and BS pumps. The licensee is currently reviewing the pumps' design and operation and has not yet completed its assessment. Therefore, this area was not reviewed as part of this audit and is considered part of Open Item 5.3-2 above relating to incomplete status of downstream evaluation.

The SE states that a review and assessment of changes in system or equipment operation caused by wear (e.g., pump vibration and rotor dynamics) should be performed. Also an assessment of whether the internal bypass flow increases, thereby decreasing performance or accelerating internal wear should be completed. These subjects are not addressed in WCAP-16406-P, Revision 0. Conclusions and findings related to staff topical review of WCAP-16406-P, Revision 1 need to be applied to the evaluation of post-LOCA downstream effects for ONS as noted in Open Item 5.3-2 above.

The licensee's evaluation [61] notes that there may be primary and backup seal leakage from the HPI, LPI and BS pumps into the Auxiliary Building. The staff identified **Open Item 5.3-4** as

this leakage was not quantified nor was an evaluation of the effects on equipment qualification, sumps and drains operation, nor room habitability performed.

The licensee conservatively defined the range of fluid velocities within piping systems. ONS staff will re-assess ECCS flow balances and flow-induced vibration after they complete their assessment of the HPI throttle valves and the LPI flow restrictors. No information on this subject was available for NRC staff review and is considered part of Open Item 5.3-2 above relating to incomplete status of downstream evaluation.

ONS personnel adequately reviewed system low points and low flow areas and found no settlement areas. Non-pump component wear evaluations appropriately used maximum system flows.

The licensee is continuing to evaluate the properties of the post-LOCA ECCS fluid [81, 82 and 83]. The ONS current evaluation of ECCS heat exchangers [61] will be re-assessed following those evaluations.

5.4 Chemical Effects

The staff reviewed the licensee's chemical effects evaluation, comparing it with the guidance provided in Section 7.4 of the GSI-191 SE [8].

The ONS Reactor Building insulation materials include mostly reflective metallic insulation with low amounts of fiber. ONS uses trisodium phosphate to buffer the pH of a post-LOCA Reactor Building pool. There are substantial amounts of aluminum inside the Reactor Building, primarily from duct-work, that contribute to plant-specific chemical effects.

Evaluation of ONS's plant-specific chemical effects is a work in progress. The ONS approach to determining the plant-specific amount of chemical precipitate is based on the methodology in WCAP 16530-NP, "Evaluation of Post-Accident Chemical Effects In Containment Sump Fluids to Support GSI-191 [31]." Because the primary source of chemical precipitates at ONS results from the large amount of aluminum in the Reactor Building, Duke Energy has performed corrosion testing of commercially pure aluminum and Alloy 3003 in both sodium borate and trisodium phosphate environments. Duke Energy plans to use the results of these tests to refine the quantity of precipitates predicted by the WCAP 16530-NP chemical model. During the audit, ONS personnel arranged for Duke Energy corporate staff to describe the aluminum corrosion test data to the NRC staff. Preliminary test results indicate that ONS-specific inputs significantly reduce the amount of precipitate relative to those predicted in the WCAP base model since aluminum passivated in a relatively short time in the trisodium phosphate environment and the corrosion rates for Alloy 3003 were somewhat lower than those for commercially pure aluminum.

Head loss tests that include chemical precipitates will be conducted for ONS at CCI's test facilities. ONS indicated that they intend to credit the Duke data on aluminum alloy corrosion rates and passivation of aluminum by phosphate when determining the total amount of chemical precipitates to add to the head loss test. The chemical precipitates were planned to be formed within the CCI test tank through chemical addition, rather than being prepared prior to testing and then added during the test according to the WCAP-16530 protocol. During the audit the NRC staff provided some comments related to the ongoing chemical effects evaluations at ONS:

- Although the chemical effects evaluation is not yet complete, the licensee is being proactive in performing additional testing to evaluate certain plant-specific corrosion issues related to chemical effects.

- Based on Duke Energy testing, ONS intends to take credit for reduced chemical precipitate formation relative to predicted amounts from the WCAP-16530-NP base model. Separate from this audit, NRC expects to receive a copy and comment on a PWR Owners Group test report supporting refinements to the WCAP-16530 base chemical model. In general, NRC staff expectations for licensees intending to perform integrated head loss testing with reduced amounts of precipitate relative to the WCAP base model predictions include: (1) a technical basis (e.g., test results) will be provided to justify the amount of chemical precipitate included in head loss testing, and (2) the rationale for why the overall chemical effects tests remain conservative with the reduced amount of chemical precipitate will be documented.

- For tests that inject chemicals into the test flume to produce precipitates during head loss testing, the staff notes the following expectations:

    1. Precipitate that is formed via an injection technique should be verified to have representative settling and filterability properties relative to those expected to form in a post-LOCA environment.

    2. The amount of precipitate that forms (or the amount of injected chemical, such as aluminum, that remains dissolved) during the head loss test should be measured. In other words, it should be verified that the amount of precipitate that is calculated to form during the test is actually present during the test.

    3. Precipitation reaction kinetics or other time dependent effects should be understood and considered in the test termination criteria. For example, running the test for additional time would not result in a significantly higher head loss due to additional precipitate formation, changes in precipitate hydration, etc.

In summary, the ONS chemical effects evaluation is still in progress. Therefore, resolution of chemical effects is **Open Item 5.4-1**. Within the resolution of chemical effects, the NRC staff indicated there is a general question related to the potential for coatings to contribute to chemical effects through interaction with the pool environment (i.e., the potential for some of the coatings to change into material that would affect head loss in a non-conservative manner). Although the PWR Owners Group has indicated that it will address this question on a generic basis, no resolution had been reached at the time of this audit. Therefore, it remains part of the chemical effects open item for ONS. If this issue is resolved in a generic manner, it will no longer be considered as part of an open item for ONS.

6.0 <u>Conclusions</u>

The Oconee Nuclear Station has responded to NRC's Bulletin and Generic Letter GL 2004-02 according to the required schedule. New CCI basket strainers, with an effective surface area of 4868 ft^2, have been installed in Unit 1, with slightly larger strainer areas for Units 2 and 3.

An overall conclusion as to the adequacy of the licensee's corrective actions in response to Generic Letter 2004-02 will be contained in a future letter to Duke from the NRC Office of Nuclear Reactor Regulation. This letter will consider licensee responses to GL 2004-02 requests for additional information, as well as future licensee GL 2004-02 supplemental responses reporting closure of the open items in this report and completion of GL 2004-02 corrective actions at ONS.

Appendix I Open Items

Open Item 3.3-1 : Unjacketed fibrous insulation erosion (pg 16)

The licensee identified that there are some piping runs and elbows insulated with unjacketed fibrous materials in the Reactor Building. The licensee had not evaluated the effect of erosion of such fibrous insulation induced by Reactor Building spray drainage.

Open Item 3.4-1: Latent Debris Sample Size (pg 19)

The staff considered the number of latent debris samples taken by the licensee, at four locations within the Unit 3 Reactor Building, to be inadequate for obtaining a sufficiently accurate estimate of containment latent fibrous and particulate debris masses, given the spectrum of debris buildup conditions on various Reactor Building surfaces.

Open Item 3.7-1: Water hold up and CFT Volume effects on Minimum Water Level (pg 54)

The licensee did not consider two sources of water holdup that could reduce the calculated water level. These sources include (1) holdup of spray water droplets in the Reactor Building atmosphere, and (2) holdup of water as condensate film on surfaces of the Reactor Building. The licensee has also not documented a basis for its conclusion that the full CFT volume would be available early during a small-break LOCA.

Open Item 3.8-1: Justify Coatings Visual Assessment (pg 57)

The licensee relies on visual inspection as part of their qualified coatings assessment program. The industry has not submitted documentation for staff review that shows that visual assessment is an acceptable method for identifying degraded qualified coatings.

Open Item 4.1-1: Potential for drainage path blockage as a result of Reactor Building washdown (pg 59)

The licensee's procedures do not contain adequate precautions to ensure that Reactor Building washdowns do not result in debris entering and clogging fluid drainage paths in the Reactor Building upstream of the strainers.

Open Item 5.2-1: Potential for Blockage or Obstruction of Drainage Flow Paths (pg 64)

The licensee did not provide sufficient information for the staff to conclude that water hold up in the refueling canal and reactor cavity volumes due to debris blockage of drainage flowpaths would not be a concern for Oconee Nuclear Station. The staff noted that the licensee should consider the potential for these flowpaths to be blocked both by (1) one or more large pieces of debris and (2) an accumulation of smaller pieces of debris.

Open Item 5.3-1: Downstream Effects-Core Blockage (pg 68)

Although downstream effects evaluations were in progress during the audit, the licensee has not made any final conclusions as to whether the cores at ONS could be blocked by debris following a LOCA, and this area is incomplete.

Open Item 5.3-2: Downstream Effects Evaluations Preliminary (pg 70)

The licensee evaluations of the downstream effects of debris on systems and components are preliminary; based in part on the generic methodology of WCAP-16406-P which is under review by the NRC staff. The licensee needs to complete its evaluations in and conclusions and findings of the staff review need to be applied to the Oconee Nuclear Station evaluation of post-LOCA downstream effects.

Open Item 5.3-3: Mission times for Fluid Systems (pg 70)

The mission times for high-pressure injection, low-pressure injection, and Reactor Building spray were defined as 48 hours, 30 days and 7 days, respectively. The licensee did not provide a clearly defined basis or supporting documentation for these mission times.

Open Item 5.3-4: Quantification and assessment of downstream effects that cause seal leakage (pg 71)

The licensee did not quantify seal leakage associated with downstream effects into the auxiliary building, nor evaluate the effects on equipment qualification, sumps and drains operation or room habitability.

Open Item 5.4-1: Evaluate Chemical Effects (pg 73)

The licensee's chemical effects analysis was incomplete at the time of the audit. Also, the licensee has not evaluated the contribution of coatings to chemical effects by: (1) leaching constituents that could form precipitates or affect other debris; and (2) changing form due to the pool environment.

Appendix II References

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