CRYSTAL RIVER UNIT 3

PILOT PLANT AUDIT REPORT

ANALYSES REQUIRED FOR THE RESPONSE

TO GENERIC LETTER 2004-02

AND GSI-191 RESOLUTION

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Acronym List

ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ARL	Argonne Research Laboratory
BS	building spray
BWST	borated water storage tank
BTP	Branch Technical Position
BWR	boiling water reactor
BWROG	Boiling Water Reactor Owners' Group
B&W	Babcock and Wilcox
B&WOG	Babcock and Wilcox Owners' Group
CFD	computational fluid dynamics
CR3	Crystal River 3
DBA	design basis accident
DDTS	Drywell Debris Transport Study
DP	differential pressure
EC	Engineering Change
ECCS	emergency core cooling system
EEQ	electrical equipment qualification
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
GL	generic letter
GR	NEI 04-07 Volume 1, PWR Sump Performance Evaluation Methodology (Guidance Report)
GSI	Generic Safety Issue
HPI	high-pressure injection
HPSI	high-pressure safety injection
ICET	integrated chemical effects tests
ICM	interim compensatory measure
IOZ	inorganic zinc
LANL	Los Alamos National Laboratory
LBLOCA	large break loss of coolant accident
L/D	length/diameter
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
NRR	Office of Nuclear Reactor Regulation
PWR	pressurized water reactor
RAI	Request for Additional Information
RB	reactor building
RBES	reactor building emergency sump
RCS	reactor coolant system

RG	Regulatory Guide
RMI	reflective metal insulation
SBLOCA	small break loss of coolant accident
SEM	scanning electron microscope
SE	Safety Evaluation
SP	special procedure
SRP	Standard Review Plan
TKE	turbulence kinetic energy
TSP	trisodium phosphate
UNM	University of New Mexico
WOG	Westinghouse Owners Group
ZOI	zone of influence

1.0 BACKGROUND

1.1 Introduction

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

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

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

Benefits envisioned for the industry include:

> Feedback from the lessons learned will enable the industry to make high-quality submittals by September 2005. High-quality submittals will need less NRC staff resources to review and take regulatory actions.
>
> NRC staff clarifications regarding GL 2004-02, if needed, early during the resolution of GSI-191.
>
> Reduced need for audits because the NRC staff will become familiar with the approach taken for the resolution of GSI-191.
>
> Fee waiver for NRC review of license amendment requests related to GSI-191 for pilot plants.
>
> Enable the industry to focus and prioritize the open items impacting the GSI-191 resolution.

Crystal River Unit 3 (CR3) volunteered to be the first pilot plant reviewed. The CR3 licensee supplied documentation relating to its proposed analyses and planned changes. During the period from April 5, 2005 until June 16, 2005, the NRC staff conducted a pilot audit of the CR3 implementation of its plant-specific evaluations responsive to GL 2004-02. On April 5-6, 2005, the NRC commenced the audit with a 2-day meeting between the NRC staff and the licensee with its vendors to go over the supplied documents and references in detail.

The meeting provided an opportunity for the NRC to: (1) review the basis, including the detailed mechanistic analysis and design documents, for a proposed submittal (2) identify areas that may need improvement or clarification and (3) ask questions of the licensee. The following categories of the submittal were reviewed and discussed:

Debris generation	Debris transport
Coatings	Debris characterization
System head loss	Chemical head loss
NPSH for ECCS pumps	Upstream and downstream effects
Modifications	

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

Name	Affiliation
Industry	
Paul Infanger	Progress Energy – CR3
Terry Austin	Progress Energy – CR-3
Wes McGoun	Progress Energy - Raleigh
Jay Basken	Enercon Services
Robert Bryan	Enercon Services
Peter Mast	Alion Science and Technology
NRC	
David Solorio	NRC
Mark Kowal	NRC
Ralph Architzel	NRC
Martin Murphy	NRC
Thomas Hafera	NRC
Hanry Wagage	NRC
Shanlai Lu	NRC
Joseph Golla	NRC
Steven Unikewicz	NRC
Angela Lavretta	NRC
Paul Klein	NRC
Brenda Mozafari	NRC
Ruth Reyes-Maldonado	NRC
Christopher Hunter	NRC
Matt Yoder	NRC (not at initial mtg)
Richard Lobel	NRC

Name	Affiliation
John Hannon	NRC
Louise Lund	NRC
Randy Moore	Region II
NRC Contractors	
Bruce Letellier	Los Alamos National Laboratory
Clint Shaffer	ARES Corporation
Vesselin Palazov	Information Systems Laboratories, Inc.

Following this meeting the NRC staff continued in-depth review of the CR3 documentation. Many telephone conferences were held to clarify the methods and calculations and to communicate NRC concerns and questions. These included February 22, 2005, to plan for the audit, and followup telephone conferences on April 20, 21, and 27, May 4 and May 23, and June 6, 15, and 16, 2005.

Crystal River Unit 3 (CR3) sponsored head loss testing performed by Alion Science and Technology at the Alion test facility in its vertical head loss test loop in Chicago, IL. On May 18, 2005, the NRC staff and its contractors visited the Alion Head Loss Test Facility as part of the pilot audit.

On June 15, 2005, the audit team and its management held a telephone exit conference at the end of the audit. During the course of the audit the team identified issues related to the licensee's implementation and plans that need to be assessed as part of the licensee's closure of GL 2004-02. These are discussed throughout the audit report and were communicated to the licensee during the audit meetings and telephone conferences. The licensee is expected to address and document resolution of these issues in conjunction with its efforts to respond to GL 2004-02. The licensee may address some of the issues in the upcoming September 2005 response letter, as appropriate.

1.2 Bulletin 2003-01 Response

Overall, the Crystal River 3 (CR3) Bulletin 2003-01 response, dated August 8, 2003, was clear, comprehensive and of higher than average quality in comparison with other responses received from industry. It specifically addressed the six interim compensatory measure (ICM) categories of Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-water Reactors:"

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

The licensee stated that licensed operator fundamentals training was then currently ongoing on the indications available for recognition of containment sump screen blockage and appropriate response measures. This initial round of training was to be completed for all licensed operators by August 22, 2003.

A listing of training efforts was provided: (1) classroom and simulator training on containment sump blockage issues to familiarize operators with symptoms of a pending loss of emergency core cooling system/building spray (ECCS/BS) recirculation capability, and the steps necessary for subsequent recovery of net positive suction head (NPSH) margin; (2) specific emergency operating procedure (EOP) step/action training was to have been conducted upon approval of new procedures (and before their implementation on all shifts) by September 5, 2003; (3) training for required technical support center personnel for the revisions being made to Emergency Management Procedures EM-225 and EM-225E was to be completed by September 5, 2003; (4) upgrade of the CR3 plant simulator to support sump screen blockage scenarios (demonstration of the impact on ECCS and BS pump NPSH resulting from progressive sump screen clogging, including the beneficial effects of decreasing flow in accordance with the interim compensatory guidance [see below], with direct visual representation of the indications that are available, the corrective actions that may be taken, and the restoration of pump performance and sump level that are a direct result of those actions); (5) the contents of the operator training lesson plan on sump clogging were listed; (6) the future utilization in CR3 EOPs of a revised Babcock and Wilcox Owners Group (B&WOG) Technical Basis Document, which was to provide the basis for EOP instructions for B&W plants, providing strategies and guidance for addressing symptoms indicative of a clogged containment sump screen following recirculation initiation; (6) a listing of CR3-specific instrumentation available to determine the potential of impending sump screen blockage; (7) a listing of possible indications of sump screen blockage following transition to reactor building sump recirculation; (8) new EM-225, “Duties of the Technical Support Center (TSC) Accident Assessment Team,” Enclosure 4 steps to take if indications of sump blockage are evident (listed); (9) an extensive listing of actions to restore sump recirculation; and (10) additional guidance to Technical support center support personnel to consider aggressive actions if indications of pump distress continue or return (listed), with a note that most of these actions would require declaration of 10CFR50.54(x).

The licensee noted that it has a backwash sump recovery method (discussed as performance of a decay heat “drop line dump to sump” evolution), as well as various other sump clogging response strategies such as cyclic operation of low pressure injection pumps from the reactor building sump.

ICM Category #2: Procedural Modifications, if appropriate, that would delay the switchover to containment sump recirculation (e.g., shutting down redundant pumps that are not necessary to provide required flows to cool the containment and reactor core, and operating the BS intermittently).

The licensee concluded, with justifications, that additional measures to extend the borated water storage tank (BWST) injection time (and therefore delay switchover to sump recirculation) were not appropriate in that they were not in the interest of safety (specifically, operator actions to stop pumps or throttle flow). However, the licensee noted that some measures already directed in the EOPs would be effective in delaying switchover during small and medium sized break LOCAs. Actions are taken to throttle HPI flow in consideration of maintaining minimum subcooling margin, nil-ductility margin, and pressurized thermal shock concerns (EOP-13). The NRC staff notes that efforts by the licensee to conduct aggressive cooldown may avoid switchover to recirculation mode entirely for these smaller sizes of pipe breaks.

The licensee stated that guidance was being added to the EOPs to consider shutting down the second BS train after switchover to recirculation if a symptom of excessive sump blockage is evident and if event recovery conditions permit. The required level of diagnosis was considered by the licensee to be too difficult to implement during the BWST drawdown injection phase.

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

The licensee stated that Technical support center guidance was being revised to direct immediate BWST refill upon completion of switchover to sump recirculation from the injection mode, using the spent fuel pool as a source of borated water. Other strategies included aligning an HPI pump to inject using the residual BWST inventory at twice the decay heat boil-off rate, or at least 200 gpm. Use of the normal charging path using the makeup tank, while maintaining makeup tank level via the reactor coolant bleed tanks or the boric acid storage tanks was included as a less preferred but available option.

The licensee noted that the impact of additional water on reactor building structural integrity would be negligible, and the potential consequent loss of certain post-accident monitoring instrumentation was a lower import than adequate core cooling.

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

The licensee stated that the reactor building inspection procedure SP-324 had been revised to require more detailed reactor building inspections, with six zones inspected by separate teams headed typically by Senior Reactor Operators. The licensee noted that SP-324 requires that all items remaining inside the reactor building should be approved (i.e., they have been qualified and/or evaluated for reactor building sump impact), with full implementation of this procedure by October, 2003. A significant list of additional measures to minimize foreign material in the reactor building was provided.

The licensee stated that a “High Impact Team” had been assembled and devoted to reactor building housekeeping prior to and during refueling outage 13R. The team was responsible for assigning cleaning resources, as well as enforcing cleanliness standards in the reactor building and performing routine inspections. The licensee stated that the team’s intent was to clean all accessible areas, not just task work areas during the time available. The licensee discussed maintenance staff reactor building cleanliness awareness training, with clearly detailed expectations, timed to occur during outage 13R.

ICM Category #5: Ensuring Containment Drainage Paths are Unblocked

The licensee discussed the primarily credited post-LOCA fluid recirculation access hallway/door with mesh gate through the ‘A’ D-ring wall on elevation 95 feet-7 ½ inches, and seven other 12 inch by 12 inch openings in the D-ring walls (“scuppers”), which are inspected during performance of procedure SP-324. The licensee stated that verification of drainage path availability is performed for all floor drains and two fuel transfer canal drains per SP-324, so that there is positive assurance that the deep end of the fuel transfer canal does not hold excessive amounts of water nor prevent its return to the reactor building sump.

The licensee noted that CR3's flood and NPSH analyses do account for the possibility of retained water in the fuel transfer canal, the reactor vessel cavity, and those areas which are served by floor drains.

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

The licensee stated that sump screen integrity is verified on a 24-month refueling outage interval, requiring the ECCS System Engineer to inspect the reactor building sump each outage. The checklist requires acceptability of the complete sump structure, the screen, all fasteners, minimal corrosion of all components, and no evidence of debris in or around the screen or around the screen or ECCS suction piping. The inspection includes both sides of the screen. The licensee noted that a procedure revision had been initiated to add an inspection for gaps and breaches.

The licensee noted that there had been a detailed inspection of the sump assembly in 1997. The strainer was structurally reinforced to support reverse pipe jet flow used in “dump the sump” boron precipitation control measures. During that time the screen and its supporting steel network were verified to be constructed per the design drawings and additional work was performed (e.g., welding, bolting) to be in full compliance with design requirements. At task completion, the licensee’s engineering department inspected the strainer perimeter and the deck plate above to ensure that no gaps exceeded 1/4 inch in width (some gap shimming was performed around piping penetrations through the deck plate to maintain this dimensional requirement).

The licensee stated that during outage 13R (since completed) it would conduct comprehensive containment walkdowns for the purpose of defining the potential sump screen debris source term similar to the activities described in NEI-02-01, Revision 1, “Condition Assessment Guidelines: Debris Sources Inside PWR Containments.”

Conclusion: The licensee’s actions, as described in its August 8, 2003 letter, were responsive to the intent of Bulletin 2003-01.

1.3 Generic Letter 2004-02 90-Day Response

Florida Power Corporation, the licensee (also doing business as Progress Energy-Florida), submitted its 90-day response on March 4, 2004 in accordance with the schedule in Generic Letter (GL) 2004-02. The licensee stated in its 90-day response that they will evaluate sump performance using methods intended to conform to NEI Guidance Report 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Volume 1 - "Pressurized Water Reactor Sump Performance Evaluation Methodology," and Volume 2 - "Safety Evaluation by the Office of NRR Related to NRC Generic Letter 2004-02, Revision 0, December 6, 2004." The licensee will refine the debris transport portion of its analysis, as necessary, by using computational fluid dynamics modeling to decrease predicted debris transport. The following exceptions to the NRC methodology were identified by the licensee and were not all reviewed by the staff during this audit:

- The Zone of Influence (ZOI) for qualified epoxy coatings has been established as a sphere with a radius of four times the pipe break diameter. This radius is less than the radius of ten pipe break diameters recommended in Section 3.4.2.1 of the NRC methodology. However, planned industry testing is expected to support the four diameter ZOI.

- Unqualified epoxy coatings outside the ZOI are assumed to fail as chips, with a characteristic size dependent on coating thickness. Section 3.4.3.6 of the NRC Methodology recommends that unqualified epoxy coatings outside the ZOI fail as particulates with a diameter of 10 microns. However, the 10 micron size is associated with erosion of coatings due to high pressure jet impingement inside the ZOI. Coatings outside the ZOI are not exposed to jet impingement, and therefore the predominant failure mechanism is not erosion.

- CR3 has two quality levels of qualified epoxy coatings. Service Level I coatings are defined as coatings applied to areas within containment where failure in a post-LOCA environment could have a detrimental effect on plant safety. These areas are defined as those where reactor building pool fluid velocity and turbulence that could be experienced during post-LOCA recovery are adequate to transport failed coatings to the sump and contribute to blockage of the sump strainer (reference Florida Power Corporation to NRC letter 3F1 198-02, dated November 6, 1998, Response to Generic letter 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment"). Service Level II coatings are defined as coatings applied to areas outside the Service Level I areas, where transport of failed coatings to the sump during post-LOCA recirculation is unlikely. Service Level II coating products are Design Basis Accident (DBA) qualified and applied with the same work processes and documentation controls as Service Level I coatings, with an exception that verification and inspection activities may be performed by a qualified coatings verifier instead of a Quality Control Inspector. CR3 treats both Service Levels as qualified coatings for the purpose of post-LOCA debris generation, transport, and sump strainer impact considerations.

- Service Level I and II coatings outside the ZOI that are degraded are assumed to fail as chips. The transport of failed coating chips is a function of location within the reactor building pool, pool turbulence, and pool velocity.

The scheduled completion date for the licensee’s evaluation is September 1, 2005. The licensee stated that revisions to analyses after the scheduled completion date may be necessary to address issues that are still unresolved. For example, the method used to account for potential chemical effects may need to be adjusted based on conclusions from the ongoing integrated chemical effects testing.

The licensee reports that they completed the CR3 containment walkdown surveillance during Refueling Outage 13 in the Fall of 2003.

2.0 DESCRIPTION OF PLANNED CHANGES

The following text is excerpted from the CR3 description of Engineering Change (EC) 58982R0 [69]. Figures 1, 2 and 3 are drawings representing the major features of the proposed new sump design. These proposed changes represented a significant part of the audit review:

Reactor Building Sump Strainer, Reactor Building Flow Distribution, and Reactor Building Debris Interception Mods

The intent of the modification is to perform the hardware changes required to bring CR3 into full resolution with NRC GSI-191; however, there are still several unknown inputs that factor into the supporting analyses that will ultimately support this conclusion. Therefore, certain assumptions have been made that require further validation and these are listed in Section B.5 of this EC. A revision to this EC is planned (tracked by A/R 89198-20) when industry developed testing and analyses conclude. It is anticipated this revision will be completed prior to the September 1, 2005, Generic Letter 2004-02 response to the NRC.

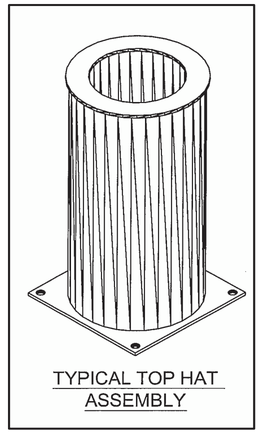

Figure 1 Single Top Hat Strainer Unit

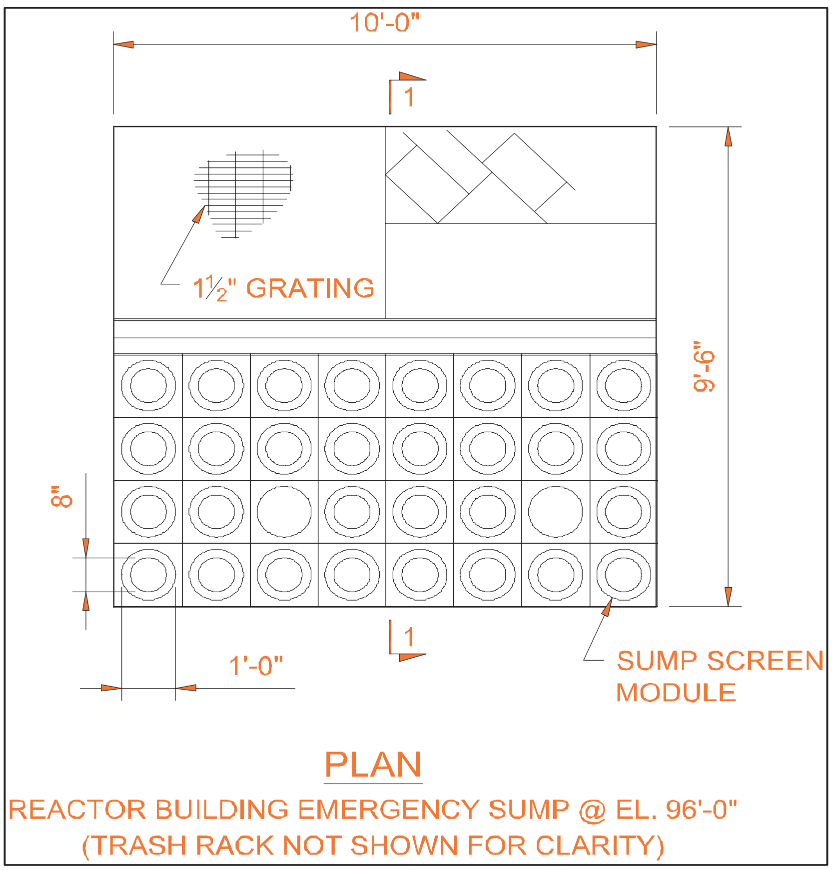

Figure 2 Crystal River Top Hat Strainer Array Top View.

The modifications presented in this EC are to increase the RBES strainer area and to install a

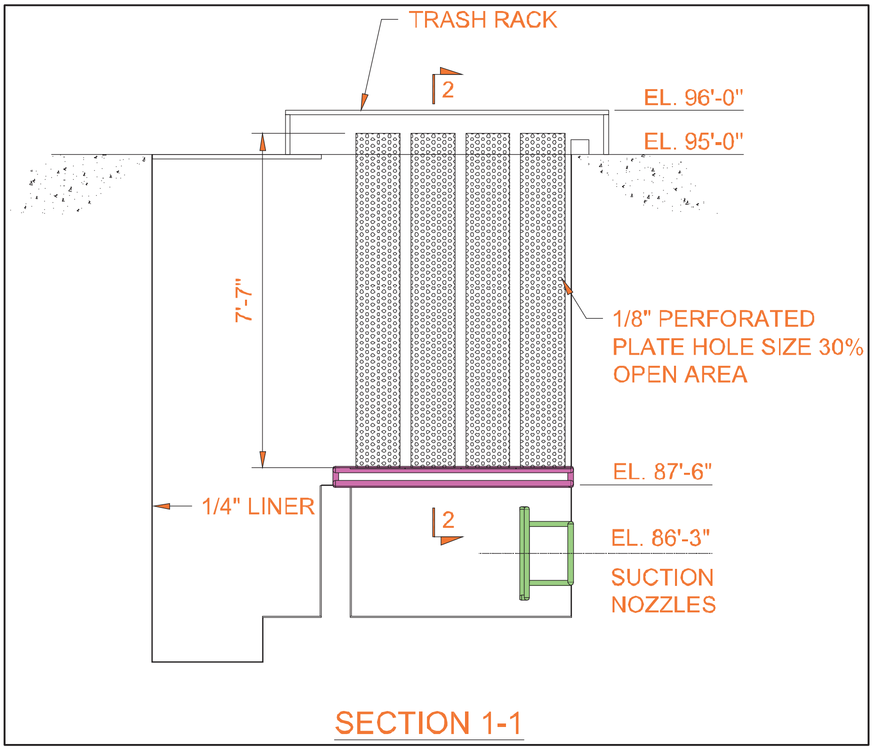

Figure 3 Crystal River 3 Top Hat Strainer Array Side View

debris interceptor and a flow distributor at strategic locations in the reactor building. This EC does not satisfy all of the requirements associated with Generic Letter 2004-02 and the methodology defined in GSI-191 SE (referred to as the SE). This EC will attempt to comply with the Generic Letter and SE to the greatest extent possible at this time. Section B of this EC contains a detailed description of the Generic Letter and SE requirements versus the as-designed replacement strainer assembly. The following provides a list of changes in the strainer design concepts.

Increasing the strainer area from 86 ft^2 to approximately 1139 ft^2.

Reducing the strainer opening size from ¼" x ¼" square to 1/8" diameter holes.

Optimizing flow to the RB sump with a flow distributor, debris interceptor, improved scupper screens (for south side D-ring), and curbs for the purpose of inducing sedimentation and entrapment of debris.

Strainer differential pressure measurement to monitor and react to debris accumulation (flow reduction, Dump-to-Sump strainer cleaning, etc.).

Crediting the torturous path flow pattern from the D-ring pool to minimize transport of debris to the RBES.

Expansion of RBES trash rack area from 55 ft^2 of horizontal surface to 100 ft^2 of horizontal plus 25 ft of vertical surface to minimize the potential to completely block flow to the strainer.

Installation of a trash rack over the 6" fuel transfer canal drain to minimize the hold of reactor building spray inventory within the refueling canal.

Installation of bellmouth flanges to minimize the entrance losses for the flow entering the Decay Heat suction nozzles in the reactor building emergency sump.

Testing of mineral wool insulation to determine head loss characteristics.

Strainer:

The new strainer assembly will be constructed of 32 top-hat style assemblies. Each top-hat will be approximately 7 ft 6 in long and is constructed of perforated plate with a square flange at the bottom for attachment to the supporting structural steel. The top-hats and supporting structural steel will be constructed of corrosion resistant stainless steel alloys. The top-hat strainer assemblies will provide in excess of 1000 ft^2 of strainer area.

Debris Interceptor/Trash Racks:

A trash rack will be located in the refueling canal over the 6" diameter northwest end drain. This will ensure that debris does not plug the drain causing additional holdup of reactor building spray inventory within the refueling canal.

Floor grating will be installed over the top-hat assemblies to prevent damage and provide personnel access to the Letdown Heat Exchanger Rooms. The grating over the top-hat assemblies serves as a trash rack, a vortex suppressor, and a personnel walkway.

A debris interceptor will be located between the D-ring wall and the reactor building wall to minimize the transport of debris from the D-ring area. Additionally, a flow distributor will be installed to help distribute the flow evenly and optimize the velocity across the reactor building floor.

To prevent large pieces of debris from transporting out of the D-ring scuppers on the south side of the reactor building, bypassing the debris interceptor, perforated plate will be installed over these scupper holes.

Bellmouth Flange:

The modification will install a bellmouth flange on each of the 14" decay heat suction nozzles in the RBES. The purpose of the bellmouth flange is to reduce the frictional entrance losses to the decay heat suction nozzles. The reduction in entrance losses improves the net positive suction head available.

Assumptions:

A zone of influence with a radius equal to 4 pipe diameters for qualified coatings is utilized in lieu of 10 pipe diameters as stated in the SE. This is an unconfirmed assumption in the debris generation calculation M04-0004 [3].

Since chemical effects testing is not complete, margin for an increase in approximately 30 - 50% head loss across the new sump strainer is provided. This is an unconfirmed assumption in the sump screen head loss calculation M04-0007 [6].

The reactor building spray and decay heat removal system cyclone separator outlet throttle valves are assumed to be approximately 3/16" open. This is an unconfirmed assumption in the downstream effects calculation M04-0016 [7].

It is assumed that the critical instrument lines in the Building Spray, HPI, and low pressure injection systems are routed in a manner that would not allow debris to settle in these lines. This is an unconfirmed assumption in calculation M04-0016 [7].

In the downstream effects analysis, volumetric concentrations of less than 1% abrasive and soft, fibrous material will be contained in the process fluid downstream of the reactor building sump screens is assumed. This is an unconfirmed assumption in calculation M04-0016 [7].

The SE [2] stated that the potential for long slivers traveling axially through the perforated sump screen holes and then reorienting and clogging at any flow restriction downstream be considered. It is assumed that blockage of the downstream components due to these long hard slivers (i.e. slivers that are not flexible like fibers) is not an issue. This is an unconfirmed assumption in downstream effects calculation M04-0016 [7].

It is assumed that the Decay Heat Pumps (Model Number 8HN-194) and the Building Spray Pumps (Model Number 6HND-134) manufactured by Worthington Pump International will perform their intended post-LOCA safety function without significant degradation with a volumetric fraction of up to 1% debris. This is an unconfirmed assumption in downstream effects calculation M04-0016 [7].

The fiber and particulate matter traveling through the perforated plate holes in the Reactor Building Sump Screens has the potential to cause some wear and abrasion of ECCS and BS system valves as well as the Decay Heat Heat Exchangers. It is assumed that any wear and abrasion will not adversely impact the function of these components for the post-LOCA mission time required of these components. This is an unconfirmed assumption in downstream effects calculation M04-0016 [7].

It is assumed that the Borg-Warner cyclone separator utilized within the Decay Heat and Building Spray systems are effective at removing insulation fibers from the process fluid. This is an unconfirmed assumption in downstream effects calculation M04-0016 [7].

3.0 BASELINE EVALUATION AND ANALYTICAL REFINEMENTS

3.1 Break Selection

The objective of the break selection process is to identify the break size and location that presents the greatest challenge to post-accident sump performance. Sections 3.3 and 4.2.1 of the NEI GR [66] and the approved evaluation methodology [2] provide the criteria to be considered in the overall break selection process in order to identify the limiting break. In general, the criterion used to define the most challenging break is the estimated head loss across the sump screen. Therefore, all phases of the accident scenario must be considered for each postulated break location, including debris generation, debris transport, debris accumulation, and sump screen head loss. Two attributes of break selection that are emphasized in the approved evaluation methodology and can contribute to head loss are (1) the maximum amount of debris transported to the screen, and (2) the worst combinations of debris mixes that are transported to the screen. Additionally, the approved methodology states that breaks should be considered in each high-pressure system that relies on recirculation, including secondary side system piping, if applicable.

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

CR3 Calculation No. M04-0004, "CR3 Reactor Building GSI-191 Debris Generation Calculation," [3], documents the assumptions and methodology the licensee applied as part of the overall break selection process and to determine the limiting break for CR3. Crystal River Unit 3, Calculation Nos. M04-0005, "CR3 Reactor Building GSI-191 Debris Transport Calculation," [4] and Calculation No. M04-0007, "CR3 RB Sump-Head Loss Calculation for Debris Laden Screen," [6] provide assumptions and methods applied for debris transport, accumulation and head loss calculations. Together, these calculations support and inform the limiting break selection process.

NRC Staff Audit:

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 Calculations M04-0004, M04-0005 and M04-0007 against the approved methodology documented in Sections 3.3 and 4.2.1 of the SE. The NRC staff observed that the licensee's break selection evaluation was generally performed in a manner consistent with the approved SE methodology. However, the NRC staff identified one issue that the licensee has agreed to resolve. 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 here.

The NRC staff's review found that the licensee evaluated a number of break locations and piping systems, and considered breaks that rely on recirculation to mitigate the event. As a minimum, the following break locations were considered:

Break No. 1 - Breaks in the RCS with the largest potential for debris.

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

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

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

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

This spectrum of breaks is consistent with that recommended in the staff-approved methodology and is also consistent with regulatory position 1.3.2.3 of Regulatory Guide 1.82, Revision 3 [32].

The licensee considered breaks in the primary reactor coolant system piping, secondary system piping, and other high energy line break piping systems having the potential to rely on ECCS sump recirculation. The licensee reviewed accident analysis scenarios as described in the CR3 Updated Final Safety Analysis Report to determine which accidents and piping systems may require sump recirculation. The licensee concluded that a large break loss of coolant accident and certain small break LOCAs would require sump recirculation. For small breaks, only piping that is 2½" in diameter and larger was considered. The NRC staff found this to be consistent with the approved evaluation methodology, which states that breaks less than 2 inches in diameter need not be considered (discussed in Section 3.3.4.1 of the SE).

During the April 5-6, 2005, audit meeting, the licensee stated that LOCAs are the only break scenarios that may require sump recirculation. Additionally, with respect to secondary side piping system breaks (main steam or feedwater lines), the licensee stated that these breaks do not rely on sump recirculation. The CR3 reactor building cooling fans (units) are safety-related; however, neither the cooling fans nor the reactor building sprays are credited in the licensing basis analyses for secondary side breaks. Therefore, sump recirculation is not needed to mitigate these secondary side breaks. Additionally, during an April 20, 2005, telephone conference, the licensee stated that a loss-of-coolant accident (LOCA) reactor building pressure/temperature profile is used for Environmental Equipment Qualification (EEQ) purposes for CR3. Sump blockage is being addressed for LOCAs and will therefore have no impact on the existing CR3 EEQ profiles. The NRC staff reviewed CR3 Updated Final Safety Analysis Report Sections 5, 6 and 14, and verified that a large break LOCA (LBLOCA) and certain small break LOCAs are the only break scenarios that may require sump recirculation.

The licensee evaluation identified a double ended guillotine break of the RCS hot-leg piping in the North D-Ring compartment as the limiting break. This break generates the largest amount of debris, and also the worst combination of debris. The licensee evaluated all phases of the accident scenario for this break, including debris generation, debris transport, debris accumulation and sump screen head loss.

Section 3.3.5 of the SE describes a systematic approach to the break selection process which includes beginning the evaluation at an initial location along a pipe, generally a terminal end, and stepping along in equal increments (5-ft increments per the SE) considering breaks at each sequential location. However, the CR3 break selection process did not apply such a systematic approach. The licensee stated that due to the size of the ZOI applied in the analyses, and the consequent volume of debris generated, it was not necessary to evaluate 5-ft increments. For the limiting LBLOCA case, the ZOI captures the entire North D-Ring compartment, which was conservatively chosen over the South D-Ring compartment because it contains the pressurizer and associated piping. For this limiting break, the licensee also conservatively assumed the break location that captures the worst location for reflective metallic insulation (RMI). Based on the magnitude of the ZOI applied, which captures the entire North D-Ring compartment, the NRC staff agrees that performing the analysis by considering 5-ft increments is not necessary.

The licensee identified the break with the most direct path to the sump as a break in the 2½ inch diameter RCS letdown piping. For this break, the licensee did not evaluate all phases of the accident, as the head loss calculation was not performed. The NRC staff reviewed the licensee's calculations to understand the technical basis for not calculating the head loss for this SBLOCA case. For this case, the licensee assumed that 100% of debris generated is transported to the sump [4]. The licensee compared the SBLOCA debris generation [3] to the amount of debris transported to the sump for the LBLOCA [6]. The licensee conservatively assumed that the failed coatings and latent debris source terms are the same for the SBLOCA case as for the LBLOCA case [3]. Thus, the NRC staff agrees that the LBLOCA case is bounding for all debris types except for RMI insulation. However, RMI insulation is not a large contributor to head loss. Additionally, the volume of fibrous debris calculated for the SBLOCA case was considerably less than that for the LBLOCA case. On this basis, the NRC staff could deduce that the North D-Ring RCS hot-leg LBLOCA case could bound the small break LOCA case. In an April 20, 2005, telephone conference, the licensee also stated that the head loss for the analyzed thin-bed case bounds both the LBLOCA and the SBLOCA cases. The NRC staff review of this SBLOCA case identified that the technical basis for not completing the SBLOCA head loss calculation is not clearly documented in the calculations and information provided for the NRC staff's audit. As a result of the April 20 conference call, the NRC staff and licensee agreed that it should document further clarification of the technical basis for not performing the SBLOCA head loss calculation.

The licensee also addressed breaks that could generate a "thin-bed." Many possible high energy line breaks at CR3 can be postulated where a small quantity of fibrous debris is generated and transported to the sump, followed by washdown of particulate latent debris potentially resulting in the thin-bed effect. Rather than analyzing specific high-energy line break's, the licensee specifically addressed the thin-bed effect in the CR3 head-loss calculation. The NRC staff agrees that this methodology ensures that the new CR3 sump screen is designed to accommodate formation of a thin-bed of fiber and its associated head-loss effects.

Consistent with the SE, the licensee did not apply the proposed refinement of Section 4.2.1 of the GR which would allow considering only a limited set of break locations delineated in Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," and Standard Review Plan (NUREG-0800) Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping."

In conclusion, based on the above discussions, the NRC staff found that break selection was generally performed in a manner consistent with the approved SE methodology. Deviations from the staff-approved methodology appeared reasonable based on the technical basis provided by the licensee. The NRC staff and licensee agreed that the licensee should document further clarification of the technical basis for not performing the SBLOCA head loss calculation.

3.2 Debris Generation/Zone of Influence (Excluding Coatings)

The objective of the debris generation/zone of influence (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 need to determine the characteristics of the debris. Sections 3.4 and 4.2.2 of the GR [66] and the safety evaluation of the methodology [2] provide the methodology to be considered in the ZOI and debris generation process. In general, the baseline methodology for determining ZOI is based on the ANSI/ANS 58.2 1988 standard [33]. The baseline methodology incorporates spherical ZOIs based on material damage pressures and the corresponding volume-equivalent spherical ZOI radii. Debris generation is then calculated based on the amount of materials within the ZOI. Other sections of the GR and SE provide guidance on particle size distribution and characterization of the debris types.

Section 4.2.2 of the SE discusses proposed refinements that would allow application of debris specific spherical destruction zones, and direct jet impingement modeling. As discussed in Section 4.2.2 of the SE, the NRC staff accepted the application of these proposed refinements for PWR sump analyses.

CR3 Calculation No. M04-0004, “CR3 Reactor Building GSI-191 Debris Generation Calculation,” [3], documents the assumptions and methodology the licensee applied to determine the ZOI and debris generated for each postulated break.

NRC Staff Audit:

The NRC staff reviewed the licensee’s ZOI and debris generation evaluations and the methodology applied. Specifically, the NRC staff reviewed CR3 Calculation M04-0004 against the approved methodology documented in Sections 3.4 and 4.2.2 of the SE. The NRC staff found the licensee’s evaluation to be largely consistent with the approved methodology. However, the NRC staff notes that the licensee has not provided any technical basis for the coatings ZOI assumption (4L/D), as addressed in Section 3.7 of this audit report.

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

The NRC staff’s review concluded that the licensee correctly applied the staff-approved methodology to determine the ZOI to be used for debris generation. The licensee applied the

ZOI refinement discussed in Section 4.2.2.1.1 of the SE, which allows the use of debris-specific spherical ZOIs. This refinement allows the use of a specific ZOI for each debris type identified. Using this approach, the amount of debris generated within each ZOI is calculated and the individual contributions from each debris type are summed to arrive at a total debris source term. The NRC staff concluded in its SE that the definition of multiple, spherical ZOIs at each break location that correspond to the damage pressures for potentially affected materials is an appropriate refinement for debris-generation.

The licensee applied material-specific damage pressures and corresponding ZOI radius/break diameter ratios as shown in Table 3-2 of the SE, with two exceptions. The first exception involves application of a coatings ZOI value of 4.0 ZOI radius/break diameter ratio. The SE recommended a value based on plant-specific testing or a default choice of 10.0 for coatings absent such testing. This coatings ZOI is based on an unverified CR3 assumption. NRC staff review and documentation of this assumption is addressed in Section 3.7 of this audit report.

The second exception involves application of the Nukon® ZOI for mineral wool type insulation. Neither the GR nor the SE provides material-specific destruction pressure or ZOI radius for mineral wool insulation, which is classified as a fibrous type insulation. The licensee's calculation M04-0004 documents that mineral wool is slightly more dense than fiberglass (4.0 lbm/ft^3 vs. 2.4 lbm/ft^3), which is comparable to Nukon® type insulation. The mineral wool at CR3 is also fully encapsulated with stainless steel cassettes similar to RMI. Based on the robust nature of this fibrous insulation system, the licensee assumes that the ZOI for mineral wool is the same as that for Nukon® insulation (17.0 ZOI radius/break diameter ratio). The NRC staff finds the application of the Nukon® ZOI for mineral wool appears reasonable based on mineral wool being denser than Nukon®, and because the mineral wool at CR3 is fully jacketed with stainless steel cassettes. The NRC staff observed that these factors combine to make this a conservative assumption regarding the size of the ZOI for mineral wool. Additionally, for the limiting break (LBLOCA in North D-ring compartment) all the mineral wool in the compartment is destroyed. It is important to note that the debris characterization (e.g., particle size distribution) for mineral wool may not be similar to Nukon®. Debris characterization of mineral wool is addressed in Section 3.3 of this audit report.

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

As discussed in Sections 3.3.1.1 and 3.3.1.2 of CR3 calculation M04-0004, the licensee credits the North D-Ring compartment wall and the primary shield wall as robust barriers in the analysis. The licensee's analysis also credits "shadowing" of potential debris materials in the South D-Ring compartment by the primary shield wall. The licensee's basis for crediting these mechanisms is that there is sufficient open volume in the North D-Ring to allow free expansion and reflection of the break jet, and it is unlikely that the jet will reach the insulated RCS piping and steam generator on the other side of the primary shield with sufficient pressure to destroy the materials. The NRC staff agrees that the North D-Ring wall can be credited as a robust barrier and can be used as a basis for truncating the ZOI. Although the licensee did not perform calculations to quantify credit for “shadowing” of potential debris within the South D-Ring by the primary shield wall, the NRC staff agrees that the primary shield wall is a large component and will tend to dissipate break jet energy from reaching the South D-Ring compartment. The licensee does not credit any “shadowing” in the North D-Ring compartment and assumes all Nukon® and mineral wool insulation within the entire compartment is destroyed. The NRC staff finds this methodology to be consistent with the approved methodology.

In conclusion, based on the above discussions, the NRC staff observed that the licensee’s ZOI evaluation was performed in a manner consistent with the approved methodology. The licensee deviates from the staff-approved methodology only with respect to the protective coatings ZOI, which is addressed in Section 3.7 of this audit report.

3.3 Debris Characteristics (Excluding Coatings)

Quantification of Debris Characteristics

Based on drawings and documentation available for Crystal River 3 (CR3), a containment debris source inventory was drafted by the licensee [76]. The licensee performed a confirmatory walkdown in October 2003 [75], to confirm, correct, or collect the debris source inventory in containment. The walkdown was performed in accordance with plant operating procedures for containment inspection [77]. Information was collected by checklist, photographs, and notes of conditions observed; however, no scaffolding or shielding was included and a limited amount of insulation jacketing was lifted for this effort.

For CR3, the licensee has determined that the potential post-LOCA debris types inside containment (excluding coatings and latent debris, which are evaluated in Sections 3.7 and 3.4, respectively) are as follows:

Nukon®
Mirror®
Mineral wool
RMI
Armaflex

Quantities were also documented for the following materials inside containment:

Marinite board
fire barriers

For mineral wool, the licensee proposes that Nukon® characteristics be applied because mineral wool is denser than Nukon®.

Approximately 30 ft^3 of Armaflex was noted in containment. No destruction pressure was available, therefore 100% was assumed to fail. Armaflex is a foam with very low permeability.

Because it floats, this debris type was not considered in the sump strainer head loss calculations.

Marinite boards are located on the sides of overhead cable trays at only one location, extending 25 ft along cable trays. However, because it was determined to be located outside all ZOIs and is small in quantity, the licensee did not consider Marinite board in the sump strainer head loss calculations.

The fireboards are located in overhead cable trays in containment. In Reference 75, Section 6.1.1.1, the licensee stated that the material composition was not investigated because they are located outside the D-Rings and therefore are considered protected from impact during a LOCA. However, in Reference 3, Section 3.3.1.4, the licensee stated that fireboard material is considered as a potential debris source.

NRC Staff Audit:

The NRC staff reviewed the licensee's containment walkdown [75] to determine the adequacy of the licensee's identification of all potential debris types, besides coatings and latent debris (which are evaluated separately), in containment. The NRC staff looked for debris-specific size and density information. For those debris types that have values based on default values for another known debris type, the NRC staff determined whether debris-specific values were sought, and whether the default values used were adequately justified. The NRC staff also reviewed debris characteristics information to confirm consideration for specific NRC staff concerns provided on page 42 of the SE.

The containment walkdown was performed during the October 2003 outage, by Framatome ANP. The walkdown identified the five debris types listed above as the only applicable debris types. In response to a staff question, the licensee stated that some, but not all, jacketing was lifted to confirm the insulation types during the walkdown. This is discussed further in Section 5.3 of the walkdown report - Framatome document 77-5036736-00 [75]. However, all approved insulation types were physically verified to be in conformance with the plant configuration drawings and specifications by sampling (visual inspection) various insulation assemblies that were removed for in-service inspections, steam generator-pressurizer-reactor vessel maintenance and other activities. Some jacket lifting of accessible insulation did occur. No deviations from the specifications were discovered during sampling, providing confidence based on the sampling performed that the insulation applications and types are as designed. The NRC staff finds this level of effort for containment walkdowns to be reasonable.

The NRC staff's review found that the licensee's size distribution assumptions deviate from that approved in the SE for each of the debris types. The licensee applies size distributions for fibrous debris (both Nukon® and mineral wool) of 20% small fines and 80% small pieces (< 6" on a side). For RMI insulation, the licensee applies a size distribution comprised of 6 size groupings as shown in Table 3.3-5 of the CR3 Calculation M04-0004. Table 3-3 of the SE provides the recommended size distributions for various debris types. For Nukon®, the SE approved size distribution is 60% small fines and 40% large pieces. For "generic mineral wool," the SE recommends 100% small fines. For RMI, the SE recommends 75% small fines and 25% large pieces. Small fines are defined in the SE as any material that could transport through gratings, trash racks, or radiological protection fences. The SE assumes the largest openings of the gratings, trash racks, or radiological protection fences to be less than a nominal

4 inches (less than 20 square inches total open area). The SE classifies the remaining material that cannot pass through gratings, trash racks and radiological protection fences as large pieces. Each of these deviations will be discussed here.

For Nukon®, the NRC staff observed that the licensee's applied size distribution appears reasonable (20% small fines and 80% small pieces). A realistic (not necessarily conservative) size distribution for Nukon® would be represented by 6% individual fibers, 16% small piece debris (shreds or cotton ball sized), and 78% large piece debris (cannot pass through typical grating) based on Nukon® debris generation testing, a ZOI sized to 6 psi destruction pressure (see SE Appendix II), and the ZOI radii determined in SE Appendix I. Therefore the NRC staff observed that the CR3 distribution of 20% individual fibers, 80% shreds, and 0% large piece debris is very conservative for Nukon®.

The licensee assumes that mineral wool behaves similar to Nukon®. The licensee's justification for this is that mineral wool is also a fibrous type insulation and is denser than Nukon®. The SE (see Appendix II, Section II.3.1.4) provides partial support for the use of Nukon® size distribution for other fibrous insulation types with destruction pressures that are higher than the Nukon® destruction pressure. At this time, the NRC staff does not have adequate information to conclude that mineral wool will behave similar to Nukon®, and that the size distribution applied for mineral wool is appropriate. Based on data provided in NUREG/CR-2982, there is a potential for mineral wool to float for a longer time than Nukon®. The effect of this is that the mineral wool debris could potentially transport by floating over the CR3 debris interceptor and reach the sump. The CR3 analysis assumes that the mineral wool sinks and transports on the floor similar to Nukon®. Therefore, much of this debris is assumed to be captured by the debris interceptor. Therefore, an assumption of 100% small fines, consistent with the SE, would be a more conservative assumption for mineral wool. The fact that the CR3 mineral wool is metal encapsulated is not sufficient justification to treat mineral wool as jacketed Nukon® because the CR3 jackets have no visible banding and RMI with standard bands can be ripped off with pressures as low as 4 psi. Actual differences in the behavior and debris size distributions for these materials can also impact downstream components (downstream effects).

In response to an NRC staff question on this issue, the licensee provided information stating that the D-Ring door has a cyclone fence mesh with 2.5" square openings which would contain floating debris larger than this size within the D-Ring. Considering the licensee's assumption, 20% of the mineral wool could potentially pass through the D-Ring door. However, applying the SE assumption, significantly more mineral wool could be available to pass through the door and transport to the sump. A simple test to determine whether the CR3-specific mineral wool floats would be useful to verify the CR3 assumption. This issue is also discussed in the debris transport section of this report.

For RMI, the licensee assumes a size distribution comprised of 6 size groupings as shown in Table 3.3-5 of CR3 Calculation M04-0004. The licensee's technical justification for using this size distribution testing data is documented in NUREG/CR-6808. Additionally, when considering the SE definitions for small fines and large pieces of debris (small fines < 4"), the size distribution being applied for CR3 is 71% small fines and 29% large pieces, which is reasonable compared to the SE recommended values of 75% small fines and 25% large pieces. Based on this, the NRC staff observed that the licensee's RMI size distributions appear reasonable.

The NRC staff agrees with the licensee's conclusions regarding Marinite board. It is not considered because it is outside of the ZOI. Marinite board is a material that is not easily destroyed and thus tends to break into larger debris pieces. Additionally, it tends to sink and requires high pool velocities for transport.

For Armaflex, the NRC staff agrees with the licensee's assessment that it does not need to be considered in the CR3 calculations. Armaflex is a closed-cellular insulation material whose debris will float in water. The licensee stated that the CR3 sump strainer is submerged well below the water line during sump recirculation. Therefore, Armaflex debris should not impact sump strainer head loss.

In summary, the NRC staff observed that the licensee's level of effort for the containment walkdowns for identifying and verifying debris types appeared reasonable. Additionally, the NRC staff's review found that the licensee's debris size distribution assumptions, which deviate from those approved in the SE for each of the debris types, appear reasonable except for mineral wool insulation. At this time, the licensee has not provided adequate information for the NRC staff to conclude that mineral wool behaves similarly to Nukon®, and that the debris size distribution applied for mineral wool is appropriate. This issue is also identified in the debris transport section of this report.

Debris Characterization Refinements

The licensee performed debris-specific generation calculations in accordance with the refinements option of the GR and SE instead of using the baseline guidance to use the worst-case insulation type to represent all debris.

NRC Staff Audit:

The NRC staff reviewed refinements to the debris generation evaluation (including values taken from GR Table 4-1) to confirm consideration of NRC staff concerns provided on pages 95 and 96 of the SE. None were applicable to the CR3 approach. Debris-specific calculations for debris generation values were performed by the licensee for the range of breaks identified in Section 3.1 of this report. For uncertainties encountered, conservatisms were introduced. The NRC staff found the results to be well-based.

3.4 Latent Debris

The objective of the latent debris evaluation process is to provide a reasonable approximation of the amount and types of latent debris existing within the containment, and its potential impact on sump screen head loss. Section 3.5 of the NEI GR [66] and the approved SE [2] provide the methodology to be considered for evaluation of latent debris. In general, the GR outlined the following five generic activities to quantify and characterize latent debris inside containment: (1) Estimate horizontal and vertical surface area; (2) Evaluate resident debris buildup; (3) Define debris characteristics; (4) Determine fractional surface area susceptible to debris buildup; and (5) Calculate total quantity and composition of debris. The Safety Evaluation (SE) provided alternate guidance for sampling techniques and analysis to allow licensees to more accurately determine the impact of latent debris on sump-screen performance.

CR3 Calculations No. M04-0004 “Crystal River 3 Reactor Building GSI-191 Debris Generation Calculation” [3], M04-0005 “Crystal River 3 Reactor Building GSI-191 Debris Transport Calculation” [4], M04-0006 “Crystal River 3 Reactor Building LOCA CFD Transport Analysis” [5], and M04-0007 “Crystal River 3 Reactor Building Sump Head Loss Calculation for a Debris Laden Screen” [6] document the assumptions and methodology the licensee applied to determine the amount, type, and impact on sump screen head loss from latent debris.

NRC Staff Audit:

The NRC staff reviewed the licensee’s latent debris evaluations and the methodology applied. Specifically, the NRC staff reviewed CR3 Calculations M04-0004, M04-0005, M04-0006, and M04-0007 against the approved methodology documented in Section 3.5 of the SE. The evaluation for latent debris was performed in a manner considered to be significantly different than the SE approved methodology. However, as discussed below, the NRC staff concluded that this evaluation by the licensee exhibited large conservative values applied in the final evaluation of sump screen performance to bound uncertainties, and the evaluation showed low sensitivity to the overall head loss across the sump screen to the amount of latent debris.

The latent debris source term is provided in CR3 Calculation M04-0004 Table 6 and Section 3.5. The total source term was determined through the collection of two debris samples, a qualitative assessment of the sample areas for classification purposes, and a manual count of labels and tags in containment. From this, a simplified calculation was performed to provide an estimate of the amount of latent debris in containment. To account for uncertainties and provide a bounding estimate for screen evaluation purposes, a large conservative source term was used. The characterization of latent debris followed the guidance in the SE. The large conservative value used for the amount of latent debris transported to the sump screen will override uncertainties in the sampling and calculations.

Evaluation of latent debris transport is included in CR3 calculations M04-0005 and M04-0006. In both cases, the end result was an assumption that all latent debris was transportable, and resulted in transport to the sump screen. This is conservative. One assumption that did not appear to be completely justified is the 10% transport of labels outside the D-ring. However, the conservative assumption of 139 ft^2 of sump screen blockage bounds the value that would occur if 100% of the labels outside the D-ring were to transport so the transport assumption is not an issue.

Evaluation of the impact of latent debris on sump strainer head loss is included in CR3 calculation M04-0007. Conservative assumptions are used for the amount of latent debris and sump screen coverage area from labels for evaluation of sump screen head loss. Additionally, Figure 3-5 shows the sensitivity for head loss from latent debris. This indicates that the amount of latent debris has relatively little impact on the overall sump screen head loss.

Use of only two samples for estimating the amount of latent debris does not provide a strong position for statistical accuracy. This could be compounded by the simplified calculation used to estimate the latent debris coverage for the entire containment. Data to date indicates that latent debris samples can be variable, and sampling errors and outlier data are possible. The vendor report also mentions that dirt/dust accumulations were observed in overhead and inaccessible areas. The licensee is planning for collection of additional confirmatory samples during the next refueling outage.

In conclusion, the NRC staff found that the evaluation for latent debris was performed in a manner significantly different than the SE approved methodology. However, the NRC staff has concluded that this evaluation by the licensee exhibited large conservative values applied in the final evaluation of sump screen performance to bound uncertainties, and the evaluation showed low sensitivity to the overall head loss across the sump screen to the amount of latent debris. Additional samples for latent debris would be advantageous for strengthening current estimates and evaluation parameters.

3.5 Debris Transport

Debris transport analysis estimates the fraction of debris that is transported from debris sources within the zone of influence (ZOI) and as latent debris outside the ZOI to the sump screens. Debris transport occurs by four major modes as follows:

- Blowdown transport is the vertical and horizontal transport of debris to all areas of containment by the break jet.

- Washdown spray transport is the vertical (downward) transport of debris by the containment sprays and break flow.

- Pool fill transport is the horizontal transport of debris by break flow and containment spray flow to areas that may be active or inactive during recirculation flow.

- Recirculation transport is the horizontal transport of debris from the active portions of the containment floor (recirculation) pool to the sump screen by the flow through the emergency core coolant system (ECCS).

During the blowdown mode, some debris would be transported to the upper containment. During the washdown mode, a fraction of the debris in the upper containment would be washed back down to the containment floor. During the pool fill-up mode, debris on the containment floor would be scattered around, and some debris would be preferentially washed into the reactor cavity or directly onto the sump screens as the cavities below the floor elevation fill up. The reactor cavity is considered an inactive pool because the water in this volume would not recirculate. Thus, any debris that moves in would stay there without being transported onto the sump screen. During the recirculation mode, debris would be transported from various areas of the pool to the sump screens.

The licensee analyzed debris transport in CR3 Reactor Building GSI-191 Debris Generation Calculation (CR3 Calculation No. M04-0005) and CR3 RB LOCA Pool CFD Transport Analysis (CR3 Calculation No. M04-0006) [4, 5]. The licensee's debris transport methodology is based on the methodology reported in NUREG/CR-6762, Vol. 4 and in Section 3.6.3 of the NEI Sump Evaluation Methodology (the guidance report or GR) [30, 66]. The licensee used a logic tree to calculate the debris transport from the ZOI to the sump strainer by blowdown, washdown, pool-fill, and recirculation modes. The licensee's logic tree used a four-size debris distribution (fines and small, large, and intact pieces) while the generic logic tree recommended in the GR used a two-size debris distribution (small fines and large pieces). The licensee quantified the logic tree to calculate transport of seven types of debris: mineral wool, low density fiberglass, stainless steel reflective metallic insulation (RMI), epoxy paint inside the ZOI, epoxy paint outside the ZOI, dirt/dust, and latent fiber.

The licensee used an analytical refinement to analyze debris recirculation transport. This refinement involved using a computational fluid dynamics (CFD) computer code to calculate the fraction of debris that moves from the reactor building pool to the sump strainer during recirculation.

The licensee designed a flow diverter and debris interceptor to reduce debris being transported onto the sump strainer. Using CFD analysis, the licensee showed that the flow diverter and debris interceptor were effective in reducing the amount of debris that moves onto the sump strainer.

The licensee identified two limiting, postulated breaks for the sump strainer debris source term in its CR3 Reactor Building GSI-191 Debris Generation Calculation [4]. The break during the limiting small break loss of coolant accident (SBLOCA) had a direct path from the break location to the sump. The break during the large break loss of coolant accident (LBLOCA) produced the largest and most varied quantity of debris. Due to the proximity of the SBLOCA break location to the sump strainer, the licensee conservatively assumed that 100% of the debris generated within the ZOI was transferred to the sump strainer. Thus, no debris transport analysis was needed for this case. The debris transport analysis presented was for the limiting LBLOCA.

The NRC staff reviewed CR3 Calculation Nos. M04-0005 and M04-0006 [4, 5] in accordance with the NRC-approved PWR sump performance evaluation methodology presented in References 2 and 66. The NRC staff identified the following areas of debris transport that the licensee needs to address to substantiate its analysis.

3.5.1 Fibrous Debris Size Distribution

The blowdown flow would destroy a portion of the ZOI fibrous insulation into small-piece and individual-fiber debris. The insulation near the break location would be completely destroyed but the insulation near the ZOI boundary could be nearly intact, including pillows blown free from the piping. For most transport analyses, it is conservative to assume a finer debris size distribution than would be realistically expected. An exception is if large debris pieces that tend to remain buoyant (for a period longer than the time it takes to transport to the sump strainer) subsequently sink near the sump strainer. Large debris pieces are likely to remain buoyant for a longer time than small debris pieces because relatively more air can be trapped in large debris pieces.

The licensee assumed 20% each of Nukon® and mineral wool insulation in the ZOI were destroyed into individual fibers and the remaining fibrous insulation in the ZOI became small debris pieces sized less than 6 inches.

NRC Staff Audit:

The size distribution given in SE Appendix II was for low density fiberglass insulations, which includes Nukon® [2]. Using the CR3 assumed destruction pressure of 6 psi, the NRC staff calculated that low density fiberglass debris distribution in the ZOI would consist of 22% small fines and 78% large pieces. Note that in this case, small fine debris included both individual fibers and small piece debris. The following table compares the CR3 debris distribution to that given in the SE:

Debris Size Category	CR3 Distribution [4]	SE Appendix II Nukon® [2]
Individual Fibers	20%	22%
Small Pieces	80%	
Large Pieces	0%	78%

The CR3 size distribution for Nukon® is conservative because of the large fraction of small pieces, which are amenable for transport. However, the licensee used Nukon® debris generation data for mineral wool without justification. The NRC staff is unable to determine the validity of the CR3 size distribution for mineral wool because no debris generation data is available. During the April 5-6, 2005, audit meeting, the NRC staff noted and the licensee agreed to justify the debris size distribution used for mineral wool.

3.5.2 Plant-specific Debris Blowdown/Washdown Transport Fractions

SE Appendix VI shows a calculation of plant-specific debris blowdown and washdown transport for a volunteer plant.

The licensee used data from an event report of a dissimilar plant without justification or results from non-plant-specific analyses as follows:

- Used Barseback event data to assume that 25% of the debris blown into the upper containment would be retained on structures sheltered from the spray flows.

- Used NUREG/CR-6762, Vol. 4, to support washdown debris transport fractions [30].

- Assumed that 25% of the individual fibers blown into the upper containment would be retained on structures in the upper containment.

NRC Staff Audit:

The licensee did not justify debris transport fractions used as follows:

- The licensee did not justify the applicability of Barseback debris transport data to CR3. The NRC staff considers that a justification should involve determining whether the containment sprays impacted the debris retained in the Barseback drywell and calculating the corresponding impact on debris retained in the upper containment of CR3.

- The parameters given in NUREG/CR-6762 were not intended for plant-specific analyses. As stated in its Abstract [30]: "Substantial uncertainty associated with the debris transport estimates is inherent due to the complexity of the analysis and the availability of appropriate data. Due to limitations of information, these estimates are not considered best-estimate plant-specific values." The licensee did not justify using parameters from NUREG/CR-6762.

- The CR3 analysis assumed that 25% of the individual fibers blown into the upper containment were retained on structures in the upper containment. This retention fraction is significantly larger than the one calculated for the volunteer plant (7%) (SE Appendix VI, Table VI-18) [2]. The licensee did not justify its plant-specific retention assumption.

During the April 5-6, 2005, audit meeting, the NRC staff noted and the licensee agreed it had not justified the debris transport fractions used as given above.

3.5.3 Debris Introduction into Reactor Building Pool

Key considerations in determining a sump pool debris transport fraction are where and when the debris enters the sump pool. If the debris enters the pool near the sump, it would be transported more readily to the sump strainer during the sump pool recirculation than if it entered away from the strainers. GR Section 3.6.3 recommends assuming that the debris is uniformly distributed in the containment water volume for the baseline analysis [66]. The NRC staff accepted this assumption for the baseline analysis but not for analyses involving analytical refinements for pool debris transport: "The GR-recommended debris transport model in Section 4.2.4, which assumes a uniform distribution of debris across the sump floor, is not acceptable because the debris entrance into the pool is not uniform." (SE Section 4.2.4) [2]

The licensee used the following assumptions to estimate debris transport within the reactor building pool:

- Transportable insulation debris would be uniformly distributed throughout the entire volume of water in the containment.

- Small pieces (<6") of Nukon® and mineral wool insulation debris are spread out in the pool over an area at least as large as the footprint of the destruction zone associated with the insulation. The footprint is defined as the projected area of the ZOI onto the sump floor.

- The area over which the debris is spread extends back from the sump along the primary paths of flow to the sump.

NRC Staff Audit:

The uniform debris introduction assumption, although not generally acceptable, appears to be conservative for the CR3 containment. At CR3, more debris would likely enter the pool at locations well away from the recirculation sump (i.e., inside the D-ring, from the refueling pool drainage that also drains inside the D-ring, the stairwell, and the floor drains), and the sheeting flow during the pool fill would tend to move substantial quantities of debris to the opposite side of the inner D-ring floor rather than to the recirculation sump. Therefore, using the uniform debris distribution assumption causes debris to move artificially closer to the recirculation sump than would likely occur in reality. The uniform debris introduction assumption is conservative for the CR3 containment.

3.5.4 Buoyancy of Fibrous Debris

Although fibrous debris soaked with water would sink in a sump pool because the fibers are denser than water, dry fibrous debris would float due to air trapped within the fibers. The time period for fibrous debris to become soaked with water and sink depends on the size of the debris pieces and the water temperature. The lower viscosity of hotter water makes it easier for hot water to penetrate the fibers than for cold water. In 1982, the Alden Research Laboratories tested the buoyancy effects of fibrous debris on transport and found the following (NUREG/CR-2982) [67]:

- Fiberglass insulation readily absorbed hot water and sank rapidly (within 20 to 30 seconds in 120°F water).

- Mineral wool shreds (from the Maine Yankee Plant) sank within 10 to 20 minutes in 120°F water (Table 4.1). However, mineral wool shreds from an unnamed source did not sink in 10 days (Table 4.1). The test with mineral wool from the unnamed source was conducted at 120°F but the temperature was not maintained throughout the test.

- Mineral wool in whole and half pillows took days to sink at 140°F (Table 4.2).

The licensee did not address the buoyancy effects of fibrous debris.

NRC Staff Audit:

Although Nukon® will readily sink in a hot sump pool, the behavior of mineral wool is not clear because the available data are not conclusive (NUREG/CR-2982) [67]. If mineral wool shreds in the CR3 reactor building pool remain afloat, they could transport over the debris interceptor and continue toward the recirculation sump area. The shreds may subsequently sink onto the recirculation strainers. The licensee assumed all of the fibrous debris would become either individual fibers (100% pool transport) or shreds smaller than 4 inches that are capable of being transported along the floor (i.e., no large debris pieces - Reference 66, Section 3.6.3.1). However, the debris generated closer to the boundary of the ZOI would consist of large pieces because of the destruction of insulation occurring at lower pressure. Therefore, the licensee's assumption concerning debris size distribution may not conservatively account for the possibility of larger mineral wool debris pieces being afloat and transported on the sump strainer. During a teleconference on April 21, 2005, the NRC staff noted and the licensee agreed to address the possibility of larger mineral wool debris pieces being afloat and transported onto the sump strainer.

3.5.5 CFD Turbulence Suspension Evaluations

The NRC-sponsored Drywell Debris Transport Study (DDTS) benchmarked its turbulence kinetic energy (TKE) model against flume transport debris suspension data (NUREG/CR-6369, Vol. 3) [37]. This model was developed to determine the kinetic energies required to keep specific types of debris in suspension.

The licensee used a minimum TKE equal to 3/2 times the debris settling velocity squared as the criterion for determining if debris would settle on the reactor building pool floor.

NRC Staff Audit:

The licensee's TKE model contains uncertainties. Such uncertainties can be addressed by benchmarking it against data similar to those used in DDTS or other data. During the April 5-6, 2005, audit meeting, the licensee agreed to address the TKE model uncertainties. The NRC staff suggested that the licensee benchmark the CR3 TKE model against data in NUREG/CR-6369 [37] or any other source to address the model uncertainties.

3.5.6 Incipient Tumbling Velocity

After settling on the sump pool floor, debris would move along the floor if the local flow velocity were sufficiently high. NRC-sponsored separate effects tests at the University of New Mexico (UNM) measured the incipient tumbling and bulk minimum transport velocities needed to transport debris settled on the floor (NUREG/CR-6772 [82]). The incipient tumbling velocity is the minimum flow velocity needed to move individual debris pieces. The bulk minimum transport velocity is the minimum flow velocity needed to move a bulk of debris pieces. Because the final result of incipient debris motion over the long term is similar to bulk movement, use of the incipient tumbling velocity rather than the bulk minimum transport velocity to calculate debris transport is conservative. In the UNM tests, the tumbling velocity was measured for uniform flow where dampening pads were used to dampen turbulence. When turbulence was not dampened, the incipient tumbling velocity for Nukon® was 0.06 ft/s (NUREG/CR-6772, Table 3.1).

The licensee listed the incipient tumbling velocities that were used in Table 4.9.1 of CR3 Calculation No. M04-0006. The tumbling velocities used for the fibrous debris of Nukon® (<6") and mineral wool are 0.3 and 0.5 ft/s, respectively. The tumbling velocities used for stainless steel RMI pieces and epoxy paint chips were 0.28 and 0.4 ft/s, respectively.

NRC Staff Audit:

The licensee used higher tumbling velocities for Nukon®, mineral wool, and epoxy paint chips than those measured in NRC-sponsored tests, resulting in less calculated debris transport onto the sump strainer than if the NRC test data had been used.

- Nukon® - The licensee used an incipient velocity for Nukon® of 0.3 ft/s based on Alden Research Laboratory (ARL) transport test data summarized in Section 5.2.8 of NUREG/CR-6808 [34]). NUREG/CR-6808 also summarized transport data from the NRC-sponsored UNM separate effects tests, which provide an incipient tumbling velocity for Nukon® of 0.12 ft/s. A major difference between the two tests is in the Nukon® debris preparation method for the tests. The UNM prepared debris using a leaf shredder. The ARL prepared debris by cutting insulation into ½" x ½" x 1/8" pieces. The ARL pieces layed flatter on the floor than the UNM pieces causing the ARL pieces to need a higher bulk velocity because of lower projected area perpendicular to the flow direction resulting in lower drag forces. The UNM incipient tumbling velocity of 0.12 ft/s would be more conservative for this analysis than the ARL data.

- Mineral Wool – The licensee used an incipient velocity for mineral wool of 0.5 ft/s. Table 4.3 of NUREG/CR-2982 data lists this value for high density fiberglass of size 4" x 1" x 1" but a value of 0.3 ft/s for shreds of mineral wool (Type 1 debris in the report [67]). The debris in the tests that measured an incipient transport velocity 0.3 ft/s was prepared by slicing the insulation first into 1"x1" squares and then into 1/8"-thick layers. As such, the layered pieces lay flatter on the floor, causing them to move by a higher bulk velocity than some possible LOCA generated debris because of lower projected area perpendicular to the flow direction resulting in lower drag forces. Mineral wool, being denser than Nukon® (4 lb/ft^3 versus 2.4 lb/ft^3), would move at a higher bulk velocity than Nukon®. Therefore, it would be conservative to use an incipient tumbling velocity of 0.12 ft/s, as discussed above, relative to Nukon®.

- Epoxy paint chips - The licensee used an incipient velocity for epoxy paint chips of 0.4 ft/s based on data from NUREG/CR-6772 [82]. The CR3 paint chips are less dense than the UNM paint chips (94 lb/ft^3 versus 110 lb/ft^3), making the CR3 chips likely to move at a lower transport velocity than the tested UNM paint chips.

During the April 5-6, 2005, audit meeting and the teleconference on April 21, 2005, the licensee agreed to reevaluate the incipient tumbling velocities used for Nukon®, mineral wool, and epoxy paint chips.

3.5.7 Debris Interceptor

The upstream flow field generated by the debris interceptor will result in debris accumulation upstream of the interceptor.

NRC Staff Audit:

If lower incipient tumbling velocities are assumed, as discussed in Section 3.5.6, a substantial increase in the calculated amount of debris arriving at the interceptor is likely. Accumulation of large quantities of debris along the debris transport pathways, in particular, the approach to the interceptor, can alter the pool flow characteristics. A layer of debris along the bottom of the pool would reduce the cross sectional flow area and thereby increase the flow velocity. During the April 5-6, 2005, audit meeting, the licensee agreed to address the effect of debris buildup upstream of the debris interceptor on pool flow field and, thus, debris transport.

3.5.8 Erosion of Fibrous Debris in Reactor Building pool

Small scale experiments showed that fibrous debris erodes in a moving pool of water, slowly giving up individual fibers that readily remain in suspension. During integrated tank tests conducted for 4 to 5 hours, continued debris accumulation on the simulated sump screen was collected every 30 minutes (NUREG/CR-6773 [38]). SE Appendix Section III.3.3.3 stated that this continued long term debris accumulation would be likely due to continued erosion of the fibrous debris [2]. An extrapolation of the rates of erosion from these tests indicated that over a 30-day mission time, substantial quantities of the reactor building pool fibrous debris would erode.

The licensee dismissed fibrous debris erosion by stating that Class 5 as well as large pieces would not be subject to further erosion given the relatively quiet pool.

NRC Staff Audit:

Neglecting the fibrous debris erosion would lower the amount of fibrous debris being transported to the sump strainers, resulting in lower head loss across the sump strainer due to debris. The erosion products tend to form uniform debris beds because they are fine pieces. The available debris erosion data are limited and there are large uncertainties. No erosion data is available for mineral wool. During the April 5-6, 2005, audit meeting, the licensee agreed to address the effect of debris erosion on debris transport.

The licensee could account for the potential erosion of fibrous debris within the reactor building pool by either implementing a conservatively high erosion rate over the 30-day mission time or by performing tests to measure realistic long-term erosion rates for the characteristic pool turbulence expected in the CR3 reactor building pool.

3.5.9 Debris Transport Summary

The NRC staff reviewed the licensee's debris transport analyses given in CR3 Calculation Nos. M04-0005 and M04-0006 [4, 5] in accordance with the NRC-approved PWR sump performance evaluation methodology [2, 66].

The NRC staff identified that the licensee used the following conservative assumptions leading to a higher amount of debris being transported to the sump strainer: a Nukon® debris size distribution with more small debris amenable to transport, a large fraction blowdown transported onto the reactor building pool floor, and a uniform debris distribution over ZOI footprint giving a higher amount of debris being transported to the reactor building pool.

The NRC staff identified that the licensee used the following non-conservative assumptions leading to a lower amount of debris being transported to the sump strainer: an upper containment debris retention fraction (25%) for individual fibers which is high, no floating of mineral wool that could be a source of debris transporting to the sump strainer, higher incipient transport velocities that would cause a lower amount of transport of debris on the floor, and no erosion of debris that would provide an addtional source of debris to the sump strainer. The licensee used (1) an unverified assumption that the mineral wool debris size distribution would be similar to that of Nukon® and (2) a TKE model to calculate settling of debris on the reactor building pool floor without addressing model uncertainties. The licensee did not address the effect of debris buildup upstream of the debris interceptor on pool flow field and thus the debris transport. The NRC staff plans for a more detailed review of the pool CFD model subsequent to this audit.

3.6 Head Loss

3.6.1 Head Loss Audit Scope

The new sump design proposed by the licensee uses vertical top hat strainer modules installed within the existing containment sump. The water enters the inner and outer perforated plate surfaces and flows through the annulus created between these two surfaces. The total surface area of perforated plate is 1,139 ft^2. The total interstitial volume between the strainers is 231.9 ft^3. Based on the debris transport calculation,125.9 ft^3 of fibrous material is assumed to be on the surface of the sump strainer. The target pressure loss across the sump would be less than 0.4 ft.

The licensee employed the HLOSS computer code and the uniform debris bed assumption to calculate the head loss across the strainer. To enable the code to predict the pressure drop across the strainer with the presence of mineral wool fibrous material, ALION performed plant specific head loss tests and mineral wool scanning electron microscope (SEM) examination according to the SE. No additional scaled tests were conducted to examine the debris accumulation on the CR3 sump strainer.

The NRC staff reviewed the reports provided in support of head loss calculations, audited the ALION head loss test facility, audited the following list of testing procedures and documents, and performed three sets of confirmatory calculations to audit the calculations.

1. "CR3 RB Sump-Head Loss Calculation for Debris Laden Screen" M04-0007.
2. "Hydraulic Properties of Mineral Wool Insulation Test Report" ALION-REP-LAB-2352-52.
3. "Building Spray and Decay Heat Pump NPSHa/r", Florida Power Corp., Document M90-0021.
4. "Building Spray and Decay Heat Pump NPSHa/r", Crystal River 3, Document M90-0021 Revision 13, Draft 2.
5. "Crystal River Unit 3 Conceptual Design Options for Resolving GSI-191", Enercon Services, Inc., Document FPC111-PRT-001.
6. "RB Sump-Head Loss Calculation for Debris Laden Screen", Crystal River 3, Document M04-0007.
7. "Hydraulic Testing of NUKON® Debris Benchmark Test Plan: Vertical Loop," Draft, ALION-PLN-LAB-2352-56.
8. "Hydraulic Testing of Debris Program Description: Vertical Loop," ALION-PLN-LAB-2352-02.
9. "Hydraulic Testing of Mineral Wool Debris Test Plan: Vertical Loop," ALION-PLN-LAB2352-51.
10. "Test Lab Safety Procedure," ALION-SPP-LAB-2352-21.
11. "Debris Preparation Procedure," ALION-SPP-LAB-2352-22.
12. "Hydraulic Testing of Debris Test Plan Guideline," ALION-SPP-LAB-2352-23.
13. "Vertical Test Loop Fill Procedure," ALION-SPP-LAB-2352-31.
14. "Vertical Test Loop Draining and Cleaning Procedure," ALION-SPP-LAB-2352-32.
15. "Vertical Test Loop Debris Head Loss Procedure," ALION-SPP-LAB-2352-33.

3.6.2 Net Positive Suction Head (NPSH) of the Building Spray and Decay Heat Pumps

CR3 Calculation 90-0021, "Building Spray and Decay Heat NPSHa/r" [10] contains calculations of the building spray pump and decay heat pump net positive suction head (NPSH) available and the margin between the available NPSH (NPSHA) and the required NPSH (NPSHR) for a spectrum of design basis loss-of-coolant accidents (LOCAs). The results of these calculations are summarized in the table below.

	RB sump 12.7 psia[2] 204.7 EF		RB Sump 14.7 psia[3] 212 EF		RB Sump 26.4 psia 243 EF	
Decay Heat Pumps	1A	1B	1A	1B	1A	1B
NPSHR (ft)	13.63	13.62	13.63	13.62	13.62	13.62
NPSHA (ft)	14.36	13.62	14.35	13.61	14.36	13.62
NPSH Margin (ft)	0.73	0	0.72	0	0.74	0
Building Spray Pumps	0.04	1B	1A	1B	1A	1B
NPSHR (ft)	12.77	12.77	12.77	12.77	12.77	12.77
NPSHA (ft)	14.81	14.23	14.8	14.22	14.81	14.24
NPSH Margin (ft)	2.04	1.46	2.03	1.45	2.04	1.47
	Sump Level Elevation (ft)		Sump Level Elevation (ft)		Sump Level Elevation (ft)	
These Levels Produce the Above NPSHA Values	96.69		96.69		96.64	

NOTES:

1) 5.5 ft is the minimum BWST water level

2) Minimum initial containment pressure

3) Licensee stated that this case is run for information since 14.7 psia is the expected pressure prior to the accident

As shown in the table, the licensee analyzed three cases corresponding to three values of sump water temperature. The first case is the temperature at the minimum allowable containment pressure of -2 psig. The second temperature is the saturation temperature corresponding to atmospheric pressure. The third temperature is the maximum predicted sump water temperature. Values of NPSHR and NPSHA are given for each temperature as well as the NPSH margin (NPSHA - NPSHR). Note that the margin is zero for the reactor building spray pumps.

The available NPSH values were determined for a given flood elevation level. The most limiting pumps (NPSH margin equal to zero) are the reactor building spray pumps. However, this flood elevation level is less than the actual calculated flood level, which is determined in a separate calculation (M90-0023 Revision 9 [11]) provided by the licensee. The minimum level from Calculation M90-0023 is 101.65 ft for the maximum sump water temperature of 243 EF. Thus, there is margin in the results of the licensee's NPSH calculations, even though the table shows zero margin for the most limiting pumps. The calculation stated that this difference between the calculated level and the level used in the NPSH calculations is considered the NPSH margin for the purpose of acceptability of NPSH calculation results; i.e., the reactor building water level calculated in Calculation M90-0021 (the NPSH calculation) must be less than the reactor building water level in Calculation M90-0023 (the reactor building water level calculation).

No credit is taken for the effect of the containment accident pressure in increasing the available NPSH, which is consistent with the guidance of NRC Safety Guide 1. The licensee assumes that the containment pressure corresponds to the saturation pressure at the sump water temperature. This assumption is consistent with the guidance of Standard Review Plan (SRP) Section 6.2.2, "Containment Heat Removal," although the licensee did not reference the SRP. This conservative assumption results in the static head of water above the pump suction being the only positive contribution to the available NPSH.

The required NPSH is a function of the pump design for a given pump speed and capacity and is determined by the pump vendor through testing. It is defined as the amount of suction head over the vapor pressure required to prevent more than a 3 percent loss in total head from the first stage of the pump at a specific capacity. The licensee applies the 3 percent head loss to the decay heat pumps but applies a less conservative 5 percent loss to the reactor building spray pumps. The 5 percent head loss criterion is supported by the pump vendor. Using the 5 percent criterion indicates a higher acceptable degree of cavitation. The acceptability of this lower value of required NPSH is based on (1) the cavitation damage resistant pump materials, and (2) the fact that the drop in head with reduced available NPSH is gradual and not a sudden drop (that is, the pump is operating on the "knee" of the pump curve).

The licensee's use of the 5 percent drop in head as a required NPSH criterion appears to be a reasonable approach based on the items discussed above.

Another table in the licensee's calculation, partially reproduced below, shows the difference between required NPSH values for the 3 percent and 5 percent head loss criteria.

Impeller	3 % Breakdown in Total Developed Head @ 1359 gpm (ft)	5 % Breakdown in Total Developed Head @ 1359 gpm (ft)	Difference (ft)
1A Building Spray Pump	14.1	12.8	1.3
1B Building Spray Pump	12.8	11.3	1.5

The above table demonstrates that, if the 3 percent head loss criterion had been used, the 1B reactor building spray pump would also have essentially zero NPSH margin for the specified flow, level, and temperature.

Licensee calculation 97-0132, "CR-3 Containment Analysis," [79] demonstrates that 243 EF bounds the worst case sump water temperature. This is a hot leg break mitigated by one train of ECCS and one reactor building cooling unit. This is consistent with the position of Safety Guide 1 which recommends use of the highest expected sump temperature.

The available NPSH decreases due to the head loss due to pipe friction and fittings in the suction lines to the pumps. The licensee used the methods of Miller [78] which the licensee stated were "more realistic." The licensee applied these methods to adjust for the flow losses of bends, bend-tee combinations, and for the interactions between them. Miller states:

> Closely coupled bends and Ts give rise to interactions. The effect on the loss coefficient K_{ij} depends on which leg of the T the bend is connected, the orientation of the bend in relation to the T, the flow ratio Q_1/Q_3 [Flow ratio between legs of the T. Q_3 is the total flow.] and whether the flow is combining or dividing.

The licensee applied these methods for the following cases:

1. Miller gives the loss values for a Reynolds number of 106. Guidance is provided for correcting the Reynolds number to the calculated value.

2. Miller provides a correction factor for the loss due to the outlet pipe downstream of the bend.

3. A correction factor is applied for the roughness of the bend. The correction factor is f_{smooth}/f_{rough} and is necessary to adjust the values given in the curves provided by Miller. The licensee states that f_{rough} is calculated with the standard Colebrook equation.

4. The licensee uses the data provided by Miller to calculate an adjustment for the interaction between bends. Miller states that bends within less than 30 diameters of each other have losses that do not correspond simply to the sum of the separate losses. It is also necessary to consider other components (fittings) which may be between the two bends.

Miller also provides curves for determining the interaction between bends and Ts. The licensee states (Page 19 of 112 of Calculation M90-0021) that this adjustment is small.

While these factors adjust for real effects, Miller points out that in some cases data are either not available or are sparse. However, if the Miller guidance on the application of these factors is followed, their use is acceptable.

The licensee assumes that all piping is clean commercial steel pipe and assigns a roughness of ε = 1.5E-4. The NRC staff agrees this number is appropriate for clean commercial pipe. (The Miller reference, Table 8.1 proposes a value of 0.98E-4.) The assumption of clean commercial pipe appears appropriate since it is the NRC staff's understanding that this piping has never had flow through it.

The adjustments made applying the Miller methods account for real effects. However, the licensee should assure that application of these methods does not have the effect of significantly reducing overall conservative margin in NPSH since, as Miller acknowledges, there is still uncertainty in his methods.

The licensee listed several conservative assumptions in the NPSH calculations. These are:

The Hydraulic Institute recognizes a temperature correction factor that reduces the required NPSH as the pumped water temperature increases. The licensee did not use this correction and estimated that for a sump water temperature of 204.7 EF, this increases the actual NPSH margin by 0.7 ft.

The licensee stated that the water temperature at the pump suction would actually be lower than the sump water temperature due to heat transfer in the piping between the sump and the pump suction. Assuming that the sump water is saturated at 204.7 EF, if the water at the sump inlet were subcooled to 200 EF, an increase in margin of approximately 2.8 ft would result.

No credit is taken for the increased partial pressure of air at the higher accident temperatures. The licensee has determined that an increase in the temperature of the containment atmosphere from 130 EF to 204.7 EF would provide an additional 4.5 ft of available NPSH.

The impeller installed on Building Spray Pump BSP-1A and used for NPSH calculations requires 1.5 ft more NPSH than the impellers on the other building spray pump.

The CR3 calculation combines the lowest containment water level (resulting from a small break LOCA) with the maximum sump water temperature, maximum ECCS flow rates, and instrument uncertainties that are a consequence of a large break LOCA. This combination of parameters is unlikely.

NRC Staff Audit

The NRC staff reviewed Calculations M90-0021, "Building Spray and Decay Heat NPSHa/r" and Calculation 97-0132, "CR-3 Containment Analysis." The calculations were done using generally acceptable methods. The assumptions were generally conservative. The licensee presented a discussion in Calculation M90-0021 on the conservatism in the calculation. However, several aspects reduced conservatism. One of these was the use of a required NPSH value for the building spray pumps based on a 5% head loss rather than the standard 3% value. Another was the use of the methods of Miller for frictional and fitting head loss calculations. While these methods are more realistic in the sense that they account for expected effects, they are used evidently to reduce head loss and, therefore, would increase NPSH margin. The extent of their contribution isn't explicitly stated. The licensee’s use of the Miller methods appears generally acceptable, considering the overall conservatism in the calculation. The licensee should ensure in any subsequent revisions to the analysis that the reduction in conservatism due to using the Miller methods will not significantly reduce the conservative margin. In a telephone conference with the NRC staff, the licensee stated that it would consider conducting a sensitivity study to assess the magnitude of this effect.

The NRC staff finds the licensee's NPSH calculation methods and results to be acceptable.

3.6.2.1 NPSH Margin Safety Relevance

The existing NPSH margin determines the allowable ECCS sump screen head loss under design-basis recirculation flow conditions. This available margin must be met by the new sump design to resolve NRC Generic Safety Issue 191 " Assessment of Debris Accumulation on PWR Sump Performance".

The NRC staff reviewed relevant parts of the documents listed in Section 3.6.1 to ensure that the methodology was consistent with Regulatory Guide 1.82, Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" [32].

Reference 10, submitted by the licensee, documents five different scenarios (Cases 1 through 5) for assessing the available NPSH margin. These cases constitute a subset of an extended set of ten cases documented in Reference 21. Cases 3 through 5 deal with the recirculation phase of the accident and are discussed below. Cases 3 through 5 are also discussed in Reference 83.

3.6.2.2 Review of the Applied NPSH Margin Methodology

The CR3 sump provides borated water to both trains of ECCS and reactor building spray (RBS) during a postulated design basis LOCA that requires long term containment pool recirculation cooling. The available NPSH margin for the recirculation mode of the ECCS and RBS pumps depends on the following factors:

(1) the pumps' flow rates,
(2) the height of the water level above the pumps' suctions,
(3) the sump water temperature and physical properties,
(4) containment pressure, and
(5) pump NPSH characteristics and connecting piping resistance.

Following LOCA initiation, the CR3 ECCS injects water from the BWST into the reactor vessel. When the BWST indicated level reaches 15 ft, the operator will take action to place high pressure injection in piggy-back with low pressure injection and open the low pressure injection system suction valves from the reactor building sump, thus establishing long term recirculation cooling. The applied methodology properly considered operational conditions for pump available NPSH during the following three consecutive phases of ECCS operation:

(1) pumping water from the BWST,
(2) transitioning to reactor building sump recirculation and
(3) long term sump recirculation.

Specific analysis related to the key factors identified above for the recirculation mode of ECCS operation is presented in the following sections.

In the methodology applied, the first step in the analysis was running the FATHOM computer program to determine the minimum required sump and BWST levels that produce a zero NPSH margin for the decay heat pumps for each case considered. The FATHOM code was used to analyze the ECCS piping connecting the containment sump with the BS pumps, the decay heat pumps and the BWST. The predictions for the minimum required levels were conservatively based on the maximum pump flows (throttled setpoint plus uncertainty) while accounting for the line losses associated with combined uncertainty-adjusted decay heat, RBS, and makeup flows. Any level above the minimum required sump water levels contribute directly to the available NPSH margins determined for the appropriate cases.

3.6.2.2.1 Consideration of Possible ECCS Modes of Operation

Case 1 addresses the ECCS mode of operation when the BWST supplies water for the ECCS, whereas, Case 2 considers the switchover from the BWST to sump recirculation. The sump recirculation mode of ECCS operation is discussed below.

Cases 3, 4, and 5: Reactor Building Sump Recirculation Mode

The licensee's analyses determined that sump recirculation produces the most limiting required NPSH conditions for the decay heat and RBS pumps by large margins. As the available NPSH margin depends strongly on the key factors identified previously (pump flow rate, flood containment level, containment pressure, and sump water temperature), the following sections analyze the conservatism in the determination of each individual parameter of importance for the remaining Cases 3, 4, and 5.

3.6.2.2.2 Consideration of Main Parameters Influencing the Available NPSH Margin

The licensee stated that the available NPSH margin was conservatively calculated assuming the worst conditions in assessing each individual parameter. Thus, conditions that would result in the lowest expected containment flooding water elevations, such as an SBLOCA, minimum BWST inventory, and large hideout water volumes, are considered simultaneously with maximum ECCS pump flow rates, sump water temperatures, and instrumentation uncertainties associated with an LBLOCA event. The associated probability is limited by the probability of the least probable initiating event, in this case the LBLOCA accident. However, this approach provides a certain degree of confidence that the ECCS pumps can handle any probable worst-case scenario. This section specifically considers the approach in determining the pumps' process flow rates, sump water temperature, and containment backpressure.

(A) ECCS Pump Flow Rates

The flow rate for the RBS system was set to 1200 gpm plus the associated controller error during the reactor building sump recirculation phase, thus ensuring that the minimum analyzed flow rate of 1000 gpm would be exceeded. A flow control error of 159 gpm was included, thus producing an error-corrected actual RBS pump flow of 1359 gpm. In some revised analyses, the error-corrected flow potential was revised to 1362 gpm, a value also cited in Reference 4 and applied in head loss calculations. However, the analyses for the available NSPH margin used the value of 1359 gpm, as the 3-gpm difference has a negligible effect on the results. The decay heat flow was set at 2000 gpm, which is the nominal flow rate during the recirculation mode. A flow of 386 gpm was conservatively added to the decay heat flow to account for the uncertainty associated with the flow controllers. Thus, the resulting error-corrected decay heat flow rate amounted to 2386 gpm (the same value is cited in Reference 4 and used in head loss calculations). A 1,600 gpm high pressure injection flow rate was assumed in the analysis.

The maximum flow through the containment sump per ECCS train amounts conservatively to:

1362 gpm (BS flow) + 2386 gpm (decay heat flow) + 600 gpm (makeup flow) = 4348 gpm

The above discussed flow rates were properly used for all cases that assume the ECCS operates in a sump recirculation mode.

(B) Containment Sump Water Temperature

The maximum sump water temperature of 243 EF during the period of sump recirculation was determined from an LBLOCA containment analysis assuming a hot leg break. This limiting sump water temperature was assumed in Case 5. At the same time, the lowest possible sump water temperature was set equal to 204.7 EF as considered in Case 3. This value corresponds to the saturated water temperature at the lowest allowed containment pressure prior to the LOCA (-2.0 psig). This containment pressure was assumed in Case 3. The temperature of 212 EF that was adopted in Case 4 is the sump water saturation temperature at normal atmospheric pressure.

Uncertainties may have been involved in the determination of the upper limiting value for the sump water temperature. Sump water temperature is an important parameter that has a direct effect on key factors needed to be considered for determining the NPSH margin available and thus the adequacy of any sump design. In particular, the sump temperature influences the following parameters:

(1) the strainer head loss via the effect of the water properties, and
(2) the degree of subcooling via the effect of decreasing the fluid vapor pressure.

Reference 4 estimates that reducing the temperature of the sump water from 204.7 EF to 120 EF increases the strainer head loss from 0.30 feet-water to 0.55 feet-water.

Reference 10 states that subcooling the water by 4.7 EF (from 204.7 EF to 200 EF) provides an additional 2.8 ft of NPSH margin. Additionally, the CR3 Operability Concern Resolution Document notes that another 4 EF of subcooling for the water reaching the pump impellers would provide approximately 2.5 ft of NPSH margin.

As a result, the use of the lowest expected saturation temperature would result in the highest pressure drop in the suction piping to the ECCS pumps, and therefore, the most conservative result for the NPSH margin available. Since 204.7 EF is the minimum possible saturation temperature of the containment sump water, it represents the most conservative temperature for NPSH margin assessment purposes.

(C) Containment Backpressure

The containment backpressures considered in Cases 3 through 5 varied between 12.7 psia for Case 3 and 26.4 psia for Case 5. The first value corresponds to the lowest possible containment pressure prior to the LOCA (-2.0 psig). The highest containment backpressure, as considered in Case 5, corresponds to the saturation pressure at the maximum sump water temperature of 243 EF. Case 4 considered a pressure of 14.7 psia, which is the normal pressure expected in the reactor building before an accident.

For Cases 3, 4 and 5, the containment pressure was set equal to the saturation pressure as determined by the assumed temperature of the sump fluid. This approach does not credit any additional containment overpressure due to the partial pressure of the noncondensable gases (mainly air) present in the containment building.

NRC Staff Audit:

Overall, the NPSH margin and head loss calculations have been established to properly bound the LOCA scenarios, considering different cases of ECCS operations. The key system parameters affecting NPSH and head loss calculations have been properly considered.

3.6.3 Debris Types, Quantities, and Characteristics

The CR3 documentation contains an inconsistent use of head loss parameters. The mineral wool specific surface area in the base case was 129,450/ft compared to 160,640/ft for the thin-bed case, which represents a significant difference. The mineral wool glass density is 179-lbm/ft^3 in the base case and 176-lbm/ft^3 in the thin-bed case, which represents a minor difference. Also, the CR3 base case uses 263 lbm of ZOI coatings debris but the thin-bed case uses 258 lbm, a minor difference. In addition, the CR3 base case uses 94 lbm of unqualified coatings debris but the thin-bed calculation assumes 47 lbm, a significant difference. The team questioned why the 94 lbm, considered conservative for the base case calculation, was not used for the thin-bed calculation.

The GR conservatively recommends assuming 10 µm spheres for unqualified coating debris due to substantial uncertainty regarding its debris size. However, CR3 M04-0007 treats unqualified coatings as chips, where Sv = 2/t and t = 1 mil. The specific surface area for 10 µm spheres is 182,900/ft compared to 24010/ft for chips, which is a significant difference. An additional factor relating to the use of 10 µm spheres for unqualified coating debris is that the NUREG-6224 head loss correlation was based on particulates and could under represent the head loss associated with chip debris in certain conditions.

CR3 M04-0007 combines Nukon® and latent fibers with mineral wool in a combined volume calculation of these two debris types together with mineral wool. The calculation assumes the 8 lbm/ft^3 bulk density for mineral wool applies to Nukon® and latent fibers. However, in actuality each type of fibrous debris would maintain its own volume within the bed. Therefore, the M04-0007 calculation incorrectly determines the bed thickness (note the base case thickness is 1.60-inch compared to 1.38-inch from M04-0007). The bed thickness error is not that substantial for CR3 because the fibrous debris is dominated by mineral wool; however, for other plants the distribution could be more evenly split. A more appropriate procedure, according to NRC research and included in NRC software, is to maintain the correct volumes but then determine volume averaged properties for the mixture.

NRC Staff Audit:

Some inconsistent parameters were used by the licensee as the input to the head loss calculation. These inconsistencies should be evaluated and corrected or justified by the licensee. This point was discussed with the licensee.

3.6.4 Screen Head Loss Aspects

De-aeration And Possible Air Trapping

As supplemental approved guidance for the sump design, the NRC staff established a criterion to limit the debris-bed exit air void fraction below 3%. The licensee’s modification documents did not specifically contain information about the bed exit void fraction. Therefore, the NRC staff performed independent calculations and confirmed that the maximum bed exit void fraction is less than the 3% limit.

During the de-aeration evaluation, the NRC staff had concerns on possible air trapping inside the top-hat sump array. The NRC staff initially anticipated that air could accumulate inside the sump, and that a trapped air pocket would self-limit by upward flow penetrating the debris bed. The NRC staff used computational fluid dynamics (CFD) to simulate the air accumulation process in the top hat strainer array (Appendix IV). The preliminary analysis indicates that, if air generation occurs within the debris layers, the possibility of air entrainment downstream of the strainers cannot be ruled out. The calculations from this initial model show that the air has the potential for accumulating and forming coalesced bubbles in the sump region, with little chance of being released by buoyancy to the containment. It is not clear at this point whether the accumulating air will be entrained into the pump as a stream of small bubbles or as large slugs. If large air slugs can be transported into the pump intake, pump performance degradation is possible. The licensee should address this issue and justify its position on whether large air slugs would either not form or not be damaging if transported to the pump intake.

Debris Bed Formation

The CR3 new sump design implements the top-hat strainer array in a deep sump pit. Water and the postulated debris would flow from the top of the pit and downwards to the strainer array. It is unlikely that a uniform debris bed would form. It is not clear whether this issue can be fully resolved without scaled tests. The licensee should be able to justify its uniform bed assumption. The NRC staff performed confirmatory analysis (Appendices I and II) to analyze potential head loss of a non-uniform bed head. If the lower part of the strainer array is assumed to be completely blocked and the remaining portion of the strainer array has a thin bed formed, higher head loss would be expected. The resulting head loss calculated by the NRC staff, however, was not significantly higher than the design limit of 0.4 ft of water. With more conservatism introduced into the calculation by raising the fiber volume by 15%, the strainer head loss calculated is about 0.61 ft of water. The increase of the fiber volume by 15% is based on the approximate 15% uncertainty associated with fiber bed density compared with the as-manufactured density.

The NRC staff has not seen sufficient information to justify the conservatism of the licensee’s uniform bed assumption. Additional technical justification is needed.

3.6.5 NPSH and Strainer Head Loss Conclusion

Based on the audit of Alion head loss test facility, the review of fifteen reports submitted in the head loss calculation area and three sets of confirmatory calculations, NRC staff concluded that the licensee's sump head loss methodology is in general compliance with the guidance of the GR and SE. The reports prepared were generally of high quality and the evaluation appears to be thorough and technically sound. However, several technical issues remain to be resolved by the licensee. The following issues prevented the NRC staff from determining that sufficient conservatism was included in the calculated maximum head loss:

- Uniform Debris Assumption

  The licensee did not provide any scaled test data to justify the conservatism of the uniform debris bed distribution assumption. Based on the flow field within the top-hat strainer array, one could assume a non-uniform bed formation. The licensee should justify the assumption technically.

- Mineral Wool SEM Test

  A scanning electron microscope (SEM) test is needed to identify the micro-structure of mineral wool fibrous material. The licensee used a mineral wool sample from the plant secondary system instead of a containment system. Therefore, the effects of high temperature aging and radiation are unknown. The NRC staff recommends additional SEM verification testing to confirm the current SEM test results using the mineral wool sample from a high temperature containment area.

- Mineral Wool Head Loss Test Procedure

  Part of the fiber debris introduced into the Alion test facility was found floating on the water surface near the debris intake region (See Appendix III). The approved testing procedures did not force all the debris to the bed or deduct the mass from the debris bed at the post-data processing stage. The licensee should quantify the uncertainty associated with possible floating debris at the debris intake for the mineral wool head loss test.

- HLOSS Code

  This computer code is the key element of the head loss calculation methodology. NRC staff confirmatory analyses verified a few points of the calculation and found the results of the sample review acceptable.

- Deaeration and Possible Air Trapping

  A small amount of air bubble formation has been observed during many strainer tests. The NRC staff's confirmatory CFD analysis shows possible air trapping in the top hat strainer array. The licensee should be able to demonstrate that the strainer performance will not be adversely affected, or that no large air slugs would be transported to the pump intake. Section 3.6.4 of this report discusses the need for the licensee to address the issue of possible pump performance degradation from the potential transport of large air slugs into the pump intake.

3.7 Coatings Evaluation

3.7.1 Coatings Zone of Influence

Although the SE states that a 10 length/diameter (L/D) spherical zone of influence (ZOI) should be used as a default value for qualified coatings, the licensee applied a plant-specific coatings spherical ZOI sized with an equivalent radius of 4 pipe diameters (4 L/D). CR3 calculation M04-

0004 stated that this is an unverified assumption and will require further confirmation. The licensee judges the 4 L/D ZOI for coatings to be acceptable based on existing test data used for removing coatings using water jet technology as compared to the blowdown and discharge characteristics of a LOCA. The licensee has left the confirmation of the 4 L/D ZOI as an unverified assumption to be confirmed upon completion of further testing.

NRC Staff Audit:

For determining coatings destruction for CR3, the basic approach used by the licensee of a spherical approximation for the ZOI is consistent with that provided in the NRC staff-approved methodology for coatings. However, the licensee took two deviations from the approved methodology:

- Instead of totaling the amount of coatings on individual lines and components contained within the sphere, the amount of coatings debris assumed in this evaluation is equal to the overall surface area of the sphere, and

- the radius of the ZOI proposed in this evaluation (i.e., 4 L/D), is smaller than the NRC staff-approved default value of 10 L/D.

The NRC staff-approved approach for determining the amount of coatings debris generated during a LOCA is based on the surface area of coatings that exist within the spherical ZOI (i.e. totaling the amount of coatings on individual lines and components contained within the sphere). The licensee determined that the surface area of the sphere is greater than the total surface area of coated lines and components within the sphere, and therefore provides a conservative approximation of that area. Thus, on the basis of its conservatism, the NRC staff finds this approach reasonable. However, if the area of coatings within containment changes such that the surface area of the spherical ZOI is no longer the conservative approach, then the evaluation should be revised accordingly.

The NRC staff-approved approach for determining the radius of the ZOI for coatings is by plant-specific experimentation or by use of a default value (10 L/D). The licensee believes a coatings ZOI radius of 4D to be conservative in light of upcoming generic coatings testing and recent French coatings tests. However, data for neither was available to the NRC staff for review during the audit. Technical justification of the 4 L/D value will need to be reviewed by the NRC staff. Generic coatings testing may be used to justify plant-specific coatings destruction as long as the test(s) are reasonably correlatable to the coatings type, condition, and environment expected during a LOCA at that site. The NRC staff will review this issue further based on the licensee's September 2005 GL response.

3.7.2 Coatings Debris Characteristics

The licensee calculated the amount of coatings debris considering a coatings ZOI of 4D. The licensee's calculations are documented in CR3 Calculation M04-0004 [3]. The licensee identified all qualified, unqualified, and other coatings within the CR3 reactor building. Consistent with Section 3.4.3.3.3 of the NRC staff-approved GR, the licensee assumes that all qualified and unqualified coatings within the coatings ZOI fail as 10 μm fines. The licensee assumes that unqualified coatings outside of the ZOI, except for inorganic zinc, fail as chips, with a failed size equivalent to their applied thickness (Table 3.4-3 of M04-0004).

Additionally, the licensee identified degraded or failed DBA qualified coatings as a result of chemical leaching of the concrete on the interior walls of the D-Rings. The licensee assumed that 30% of the of the coatings on the interior walls of the D-Ring are degraded and will fail in a post-LOCA environment.

NRC Staff Audit:

The licensee assumed that all qualified and unqualified coatings within the coatings ZOI fail as 10 µm fines. The NRC staff finds the licensee's treatment of qualified and unqualified coatings within the coatings ZOI to be consistent with Section 3.4.3.3.3 of the staff-approved GR.

The licensee assumes that unqualified coatings outside of the ZOI, except for inorganic zinc, fail as chips, with a failed size equivalent to their applied thickness (Calculation M04-0004 Assumption 6 and Table 3.4-3 of M04-0004). This is not consistent with the SE Section 3.4.2, which states that all unqualified coatings outside of the ZOI should be treated as failed coating with a particulate size. The NRC staff finds the licensee's treatment of inorganic zinc outside of the coatings ZOI to be consistent with the GR (10 µm fines are assumed). For other coatings outside the ZOI, the licensee refers to a BWROG report, "Images from Failed Coating Debris Characterization Report," which utilized autoclave test data gathered by the BWROG Containment Coating Committee to simulate LOCA exposure and gain insight into post-LOCA failure mechanisms. The results showed that all but the inorganic zinc paint failed in macro-sized pieces. The NRC staff requested additional information on this issue, and the licensee responded by stating that the chip size assumption actually only applies to degraded qualified coatings outside the ZOI. CR3 assumes that all unqualified coatings that are outside the ZOI will transport to the sump (see M04-0006 and Table 1 of M04-0007). These coatings at the sump strainer are modeled as 25 micron particulate for head loss purposes. The NRC staff finds the use of 25 micron particulate size to be reasonable because a particulate size is used, and a significant difference in head loss between 10 and 25 micron particles would not be expected. The licensee stated that this could be clarified in the calculation and in the referenced document.

The licensee identified an additional source of degraded or failed DBA qualified coatings as a result of chemical leaching of the concrete on the interior walls of the D-Rings. This location is outside of the 4D coatings ZOI being applied by the licensee. The licensee assumed that 30% of the coatings on the interior walls of both the North and South D-Rings are degraded and would fail as paint chips in the post-LOCA environment. The licensee's basis for a 30% value is a visual walkdown of the CR3 containment. The licensee stated the this is a conservative estimate because areas within this 30% that have been scraped clean and left uncoated (remediated) have not been removed from the estimate. Based on visual inspection, the licensee assumed that 30% of the D-ring coatings are degraded; however no testing or analysis was performed on the remaining 70% of coating that is assumed to remain qualified to demonstrate that it is not degraded. Visual inspection alone would not sufficiently assess the extent of degradation to the coatings. The NRC staff finds that the licensee should conduct a more thorough assessment of the qualified coatings throughout containment to demonstrate reasonable assurance that the coatings will remain intact and not create a debris source in the event of a DBA. This assessment should address all qualified coatings in containment and should not be limited to the D-Ring coatings. The licensee will need to determine the condition of the qualified coatings in containment by more rigorous methods than visual inspection (i.e., physical tests). For instance, the guidance provided by ASTM D5163, "Establishing Procedures to Monitor the Performance of Service Level I Coatings in an Operating Nuclear Power Plant" suggests additional physical testing should be performed when a visual examination identifies coatings that are suspect, defective, or deficient.

The licensee assumed that the degraded qualified coatings outside of the ZOI fail as chips with a failed size equivalent to their applied thickness and are considered in the transport analysis. The NRC staff agrees that degraded qualified coatings outside the ZOI failing as chips is a reasonable assumption. The licensee refers to a BWROG report, "Images from Failed Coating Debris Characterization Report," which utilized autoclave test data gathered by the BWROG Containment Coating Committee to simulate LOCA exposure and gain insight into post-LOCA failure mechanisms. The results showed that all but the inorganic zinc failed as macro-sized pieces. In response to a NRC staff question, the licensee stated that it needs to clarify the appropriate calculations to reflect how they treat the coatings assumed to be failing as chips and identify the sizes.

4.0 DESIGN AND ADMINISTRATIVE CONTROLS

4.1 Debris Source Term

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

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

NRC Staff Audit:

The licensee addressed each of these candidate refinements in CR3 Calculation M04-0004, Attachment 14 [3]:

- Housekeeping and foreign material exclusion programs

  CR3 Procedure SP-324, "Containment Inspection," currently provides instructions to inspect the reactor building following a maintenance outage, quarterly inspections or other containment entries. This procedure provides instructions to ensure containment integrity, ensure no loose debris is present which can be carried to the containment sump, ensure post LOCA recirculating water flow paths are open and unobstructed, and address other containment cleanliness items. The licensee stated that refined housekeeping methods have not been proposed in detailed sump evaluation calculations, but that inputs from detailed calculations should be referenced in cleanliness programs so that station personnel are aware of the safety-related inputs that were used in the debris head loss calculations. The licensee stated in Calculation M04-0004, Attachment 14, that they plan to identify whether refinements in Administrative Control and Design are required. Additionally, during the April 5-6, 2005, audit meeting, the licensee stated that appropriate procedures will be reviewed and modified to reflect the new sump design and other modifications resulting from the sump performance evaluations.

- Change-out of Insulation

  CR3 Calculation M04-0004 assessed the impact of replacing the mineral wool insulation on the steam generators with reflective metal insulation during the future steam generator replacement. The licensee determined that replacing the mineral wool insulation in the D-Ring compartments with RMI insulation will reduce the insulation debris source term and the effect of the failed coatings source term. The NRC staff agrees that the licensee should consider the change-out of the mineral wool insulation as a means to reduce the debris source term. It should be noted that the licensee did not credit or assume this insulation change-out in its proposed sump design evaluations.

- Modify Existing Insulation

  The licensee did not evaluate or propose modification of existing insulation.

- Modify Other Equipment or Systems

  Modification of existing equipment was not proposed in the detailed licensee evaluations. The licensee stated in Calculation M04-0004, Attachment 14 that forthcoming evaluations will identify if any modifications are necessary (e.g., bell mouth piping configurations for NPSH). The NRC staff agrees that other such modifications should be considered, and notes as discussed in section 2 below the licensee does plan to modify the piping configuration by adding bell mouth flanges to add NPSH margin.

- Modify or Improve Coatings Program

  The licensee did not evaluate or propose improvement of coatings programs.

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 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), and

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

The licensee provided information regarding the proposed CR3 sump modifications. The information provided is documented in Engineering Change (EC) Packages EC-58982 and EC-59476. EC-58952 addresses the reactor building sump screen, flow distribution and debris interceptor modifications, while EC-59476 addresses new reactor building sump level instrumentation modifications.

NRC Staff Audit:

The objective of the proposed CR3 reactor building sump design changes is to improve long term sump recirculation capabilities by addressing post-LOCA debris. The NRC staff reviewed the licensee's Engineering Change packages (EC-58982 and EC-59476) to assess the overall CR3 sump blockage resolution approach. The NRC staff observed that the licensee's overall screen modification approach appears reasonable. However, because the adequacy of the new screen design is highly dependent on the acceptability of the various analyses that establish the screen design (i.e., debris generation, debris transport, debris accumulation and head loss), further design changes could be necessary as the licensee finalizes the various ongoing aspects of the sump performance evaluation. These items are discussed in other sections of this audit report. Examples include the ongoing chemical effects testing, the licensee's unverified assumption on the protective coatings ZOI and the downstream effects analysis (awaiting Westinghouse methodology report). The analyses of these individual aspects of the sump evaluation form the technical basis for confirming adequacy of the new sump screen design and other proposed modifications.

The CR3 reactor building sump design team is comprised of numerous individuals from Progress Energy, with technical support from Enercon and Alion (consultants/vendors). The intent of the modifications and design changes being developed by this team is to ensure that the CR3 reactor building sump complies with applicable regulations, in accordance with GSI-191 resolution methodology. As stated in the previous paragraph, there are still several unknown and/or unverified inputs to the sump screen analyses. As such, the licensee has made certain assumptions that could impact the proposed design modifications. These assumptions, listed in Section B.5 of Engineering Change (EC)-58982R0 [69], capture the three examples listed in the previous paragraph. The licensee will revise these Engineering Change packages when industry developed testing and analyses conclude. The licensee anticipates that these revisions will be completed prior to September 1, 2005, when the response to Generic Letter 2004-02 is due to the NRC.

CR3 EC-58952 provides the appropriate justifications, fabrication details, and installation instructions for a reactor building sump screen that is considerably larger than that currently installed. EC-58952 also includes other modifications intended to ensure that water inventory at the sump is adequate to meet post-LOCA recirculation cooling mode system demand and that the amount of debris that reaches the sump screen post-LOCA is minimized. The following list of proposed changes in design concepts are addressed in EC-58952:

- Increasing the sump strainer area from 86 ft^2 to approximately 1140 ft^2. The proposed new strainer assembly will be constructed of 32 top-hat style assemblies. Each top-hat will be approximately 7'-6" long and constructed of perforated plate formed into cylinders. Thirty of the top-hat assemblies consist of an outer perforated tube with a diameter of 12 inches and an inner perforated tube with a diameter of 8 inches. The recirculation water enters the top-hats through the inner or outer perforated tubes and then flows downward through the annulus region between the tubes. Two of the top-hat assemblies consist of only a 12-inch diameter perforated tube in order to allow for inspection of the sump water box below the sump assembly.

- Reducing the existing strainer opening size from 1/4" square screen mesh to 1/8" diameter holes in the perforated plates.

- Optimizing flow to the sump with a flow distributor, debris interceptor, improved scupper screens (for the South side D-Ring), and curbs for the purpose of inducing sedimentation and entrapment of debris.

- Installing strainer ΔP measurement to monitor and react to debris accumulation (flow reduction, strainer backflow cleaning by Dump-to-Sump, etc.).

- Crediting and engineering the tortuous path flow pattern from the D-Ring pool to minimize transport of debris to the sump.

- Expansion of the sump trash rack area from 55 ft^2 of horizontal surface to 100 ft^2 of horizontal plus 25 linear ft of vertical surface to minimize the potential to completely block flow to the strainer.

- Installation of a new trash rack over the 6" fuel transfer canal drain to minimize the potential for hold of reactor building spray inventory within the refueling canal and refueling pool.

- Installation of bellmouth flanges to minimize the entrance losses for the flow entering the decay heat suction nozzles in the sump.

- Testing of mineral wool insulation to determine its head loss characteristics.

As mentioned previously, the adequacy of the sump screen design is highly dependent on the acceptability of the various analyses that establish the screen design (i.e., debris generation, debris transport, debris accumulation and head loss). The design features listed above have been included as part of the analyses for each individual aspect of the overall sump performance evaluation.

As part of the review for Section 5.3 of the GR and SE, the licensee verified that the proposed design would 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 licensee demonstrated this through analyses performed to evaluate structural design of the sump and in the head loss and NPSH calculations. These analyses are documented in CR3 Calculations S04-0006 for the structural design, M04-0007 for head loss, and M90-0021 for NPSH calculations. These are discussed in more detail in Section 3.6 (Head Loss and NPSH) of this audit report. The staff review of the sump structural analysis (Section 5.1) is ongoing, and the staff will inform the licensee of any issues identified upon completion of the review.

The licensee also demonstrated that the proposed screen design addresses the possibility of a thin-bed of fibrous debris. The licensee analyzed this condition as a separate case in the CR3 Calculation M04-0007, Head Loss analysis. This case assumes that a thin-bed of fibrous debris blankets the entire screen. The results of this case show a head loss that the NRC staff found acceptable with respect to available NPSH margin.

The NRC staff reviewed the licensee's 10 CFR 50.59 screening evaluation for EC-58982. One important item of note is that the licensee is not revising the CR3 licensing basis at this time. The current CR3 licensing basis assumes that 50% of the available surface area of the sump strainer is blocked. The impact of installing a much larger strainer, coupled with the other proposed modifications described in EC-58982, is improved NPSH margin. The licensee has chosen not to revise the current CR3 licensing basis until all remaining unverified assumptions are resolved (e.g., chemical effects, downstream effects, protective coatings ZOI verification) and the design packages are finalized. The licensee is planning to install the new sump strainer during the Fall 2005 outage. Another outage is available in 2007 to complete any additional modifications, should resolution of the remaining items and verification of assumptions determine that additional modifications are necessary. During the April 5-6, 2005, audit meeting, the licensee stated that they will revise the 10 CFR 50.59 evaluation once the design is finalized and will revise the current licensing basis at that time. The current 10 CFR 50.59 screening did not trigger any licensing actions, but depending on the outcome of the remaining testing and verifications, the final engineering change package may require NRC staff review and approval if licensing actions are necessary.

CR3 is also relocating and replacing existing reactor building sump level instruments as described in Engineering Change (EC)-59476. The licensee has determined that redundant reactor building sump level monitoring equipment is not required to meet the Regulatory Guide 1.97 requirements for post-accident operation. Based on this, the EC proposes to remove the Channel A containment sump level transmitters from the CR3 EEQ program. Regulatory Guide 1.97 guidance will continue to be met for post-accident sump level indication by installation of a new level transmitter. The licensee's 10 CFR 50.59 determination concluded that a license amendment request is not required prior to implementing this change.

The audit team determined that crediting the "Dump to Sump" operation for potentially cleaning the ECCS sump screen will most likely not be effective using the current procedure/practice. The licensee indicated that it expects to rewrite the Emergency Operating Procedures and Technical Support Center Procedures this summer to provide a different lineup for "Dump to Sump" operations that will result in a process that will significantly improve the ability to backwash the ECCS sump screen.

The NRC staff has concluded that the licensee's overall screen modification approach appears reasonable. However, because the adequacy of the new screen design and other proposed modifications is highly dependent on the acceptability of the various analyses that establish the screen design (i.e., debris generation, debris transport, debris accumulation, and head loss), further design changes could be necessary as the licensee finalizes the various ongoing aspects of the sump performance evaluation. The licensee stated that it will revise the 10 CFR 50.59 evaluation once the design is finalized and will revise the current licensing basis at that time.

5.0 ADDITIONAL DESIGN CONSIDERATIONS

5.1 Sump Structural Analysis

General guidance for considerations to be used when performing a structural analysis of the containment sump screen is contained in Section 7.1 of the NEI GR [66] and the approved NRC staff Safety Evaluation [2]. General items identified for consideration include (1) verifying maximum differential pressure caused by combined clean screen and maximum debris load at rated flow rates, (2) geometry concerns, (3) sump screen material selection for the post accident environment, and (4) the addition of hydrodynamic loads from a seismic event. Dynamic loads imposed on the related sump screen related structures due to break-jet impingement were not required based on CR3 license requirements. No other refinements were provided in other sections of the SE.

CR3 Calculations No. S04-0006 “Reactor Building Strainer Structural Design” [13], S04-0007 “Reactor Building Trash Rack Structural Design” [14], S04-0008 “Reactor Building Flow Distributor, Debris Interceptor, and Fuel Transfer Canal Drain Trash Rack Structural Design” [15], and S04-0010 “Support for Level Transmitter WD-302-LT” [16] document the calculations and methodology the licensee applied to analyze various parts of the proposed sump screen related modifications for structural capability.

NRC Staff Audit:

The staff review in this area is ongoing, and the staff will inform the licensee of any issues identified upon completion of the review.

5.2 Upstream Effects

The objective of the break selection process is to evaluate the flowpaths upstream of the containment sump for holdup of inventory which could reduce flow to and possibly starve the sump. Section 7.2 of the GR [66] and the safety evaluation of the methodology [2] provide the guidance to be considered in the upstream effects process to evaluate holdup or choke points which could reduce flow to and possibly cause blockage upstream of the containment sump. The GR identifies two parameters important to the evaluation of upstream effects: (1) containment design and postulated break location, and (2) postulated break size and insulation materials in the ZOI.

CR3 Calculation No. M90-0023, “Reactor Building Flooding,” Calculation No. MO4-0006, “CR3 RB LOCA Pool CFD Transport Analysis,” and Engineering Change No. 58982R0, “Reactor Building Sump Strainer, Reactor Building Flow Distribution, and Reactor Building Debris Interception Mode,” were used in the review for this section. The NRC staff reviewed these calculations and documents to ascertain that the licensee evaluated the flow paths from the postulated break locations and from containment spray washdown to identify and take measures to alleviate potential choke points in the flow field upstream of the sump. The NRC staff also reviewed the above documents and interviewed licensee personnel to verify that the licensee considered water holdup in the placement of any curbs or debris racks intended to trap debris before reaching the sump. Within this context, the NRC staff also reviewed the available information to determine that the licensee considered plant-specific insulation and unique geometric features of its containment in evaluating the drainage paths and in the design and placement of any debris racks or curbs.

NRC Staff Audit:

The NRC staff reviewed the licensee's upstream effects evaluations. Specifically, the NRC staff reviewed CR3 Calculations M90-0023 and MO4-0006, considering the approved NEI methodology documented in Section 7.2 of the SE. The NRC staff noted during the audit that the licensee's upstream effects evaluation was performed in a manner consistent with the approved methodology.

In the reactor building flooding calculation (Calculation No. M90-0023) the licensee determined the post-accident steady state flood height within the reactor building for various conditions. Maximum water level cases were studied in order to establish the minimum height for the location of electrical equipment that is not rated for submergence but is needed during post-accident conditions. Minimum water level cases were calculated for input to the ECCS and building spray (BS) pump NPSH calculation. Also, minimum water level for LBLOCA cases was calculated for input to the debris transport and head loss calculations. The purpose of these cases was to provide a flood plane level that could be used in the evaluation of debris transport to the sump.

The NRC staff reviewed the reactor building flooding calculation to determine that reasonable assumptions were used regarding flow path clearance or blockage specifically for the minimum flood elevation cases. The NRC staff found that assumptions were based on conservative judgements about the flow path leading to the sump which result in the minimum volume. Examples of this include:

- The licensee's evaluation of Core Flood Room A at elevation 119' indicates a 4" drain line to the reactor building sump. The licensee's walkdown inspection of this room notes that an opening exists in the floor of this room measuring 12" by 1'-4". Assumption No. 4Z.8 of the flooding calculation stated, "While the drain line could possibly clog, the Core Flood Room will not retain water due to the large opening. Conservatively, 1" of water is assumed to be retained on the floor of the Core Flood Room A."

- The licensee's conservative judgement in the flooding calculation is evident in Assumption No. 4Z.9. This assumes that other floor drains on elevations 119' and 160' are clogged because they are fitted with screens.

- The licensee's critical evaluation of the flow path of water to the sump is reflected in assumption no. 4Z.28 which stated, "All water entering the west side of the fuel transfer canal is assumed to flow to the deep end (east) without accumulating significantly on the 135' elevation... This is because there is a 10" clearance on each side of the Reactor Vessel Seal Plate Assembly...[which] is only a couple of inches higher than the canal floor elevation. Similarly, all water entering the Incore Instrumentation Well is assumed to flow to the deep end of the fuel transfer canal. Reference 5.84 shows that the only restriction to this flow is the Failed Fuel Container socket and there are several inches of clearance on each side of the socket."

Based on these and other judgments by the licensee in the flooding calculation, the NRC staff observed that the licensee incorporated assumptions that appear critically informed because they were based on actual inspection of the containment as well as plant drawings, and the assumptions made by the licensee appear appropriately conservative.

In the computational fluid dynamics (CFD) transport analysis calculation, the licensee presented the inputs, assumptions, methods, and results of its CFD calculation made to predict the velocity and turbulence fields that would develop in the containment pool at CR3 during recirculation through the emergency sump, should an LBLOCA occur. This calculation also provides an estimate of the amounts of insulation and other debris that would transport through the containment pool to the emergency sump. This calculation includes a flow distributor and debris interceptor in its geometry description. The debris interceptor is a planned addition to the CR3 containment.

The NRC staff reviewed the CFD calculation to determine whether the licensee, as in the reactor building flooding calculation, used conservative assumptions in the upstream effects aspects of flow modeling such that choke points are identified and enhancements, if warranted, can be made to provide for adequate drainage from upper portions of containment to the emergency sump.

The NRC staff's review of this calculation indicates that the CFD modeling appears to be thorough and, as with the reactor building flooding calculation, appropriately conservative assumptions were used. Consideration of flow obstructions and their incorporation into the modeling of containment spray drainage and flow from the pipe break was documented in the calculation. In this calculation, conservative assumptions, where appropriate, assumed more flow available for the transport of debris rather than less flow to conservatively calculate flooding level on the containment floor. The calculation shows that the flow path was rigorously determined and modeled. Aspects of this calculation that represent "upstream effects" (rather than debris transport) were correctly and/or conservatively represented. An example of this is represented in the determination of pool transport fraction for large pieces of fibrous insulation debris. Assumptions made in the calculation regarding the potential for this material to be dislocated to areas outside of the D-Ring region included the recognition that the gate at the entrance to the D-Ring area is open at the bottom (this modification was made in response to Bulletin No. 2003-01).

Enhancements were made or will be made to address upstream effects and are included in design changes in the strainer design concept. These are described in Engineering Change No. 58982R0 [69]. The flow distributor and debris interceptor are modeled in the CFD calculation. Flow will be optimized to the RB sump with a flow distributor, debris interceptor, improved scupper screens in south side D-ring and curbs for the purpose of inducing sedimentation and entrapment of debris. Expansion will be made of RBES trash rack area from 55 sq. ft. of horizontal surface to 100 sq. ft. of horizontal plus 25 sq. ft. of vertical surface to minimize the potential to completely block flow to the strainer. Further, installation of a trash rack over the 6" fuel transfer canal drain will be made to minimize the hold up of reactor building spray inventory within the refueling canal.

Based on discussions with the licensee and review of the documents indicated above, the licensee appears to have performed a thorough review of the flow paths leading to the emergency sump screen for choke points, considered the entrapment of debris upstream of the sump screen with regard to the holdup of water, and considered the effect of holdup in planned modifications.

5.3 Downstream Effects

5.3.1 Downstream Effects - General

Guidance for considerations when evaluating downstream effects are contained in Section 7.3 of the NEI GR [66] and the approved SE [2]. The NRC staff reviewed the following documents related to downstream effects during the audit:

Crystal River Unit 3 Conceptual Design Options for Resolving GSI-191, July 28, 2004 [67]
Engineering Change PCHG-DESG 58982R0, RB Sump Strainer, RB Flow Distribution and RB Debris Interception Modifications [69]
Calculation M04-0005, CR3 Reactor Building GSI-191 Debris Transport Calculation, Rev. 1 [4]
Calculation M04-0016, RB Sump Screen - Downstream Effects Evaluation, Rev. 0 [7]
Calculation M90-0021, Building Spray and Decay Heat NPSRa/r, Rev. 13 [10]
Enercon Report No. FPC112-RPT-001, Vortex Evaluation for Decay Heat Suction Lines within the Reactor Building Sump [18]
Crystal River Unit 3 Plant Operating Manual EM-225E, Guidelines for Long Term Cooling, Rev. 6 [68]
Crystal River Unit 3 Final Safety Analysis Report, Chapter 6, Engineered Safeguards, Rev. 28
Crystal River Unit 3 Final Safety Analysis Report, Chapter 14, Safety Analysis, Rev. 28

NRC Staff Audit:

The NRC staff reviewed design and license mission times and system lineups to support mission critical systems. The material appeared reasonable based on a review of the Updated Final Safety Analysis Report and Emergency Operating procedures.

The staff also noted in Section 3.6.4 of this report the need for the licensee to address the issue of possible pump performance degradation from the potential transport of large air slugs into pump intakes.

The NRC staff examined the presence and evaluation of equipment strainers, cyclone separators, and other components. CR3 is equipped with cyclone separators. The licensee evaluation of the separators did not identify any problems, however, the CR3 evaluations to support that conclusion are incomplete. CR3 recognizes that further evaluation is necessary to support its conclusion.

The licensee for CR3 was awaiting the results of the NRC-sponsored throttle valve tests, NRC-sponsored bypass tests, and the Westinghouse Owners Group (WOG) downstream evaluation to complete its portions of their downstream effects evaluation. Therefore, although the NRC audit team had plans to examine the following areas, no licensee evaluation had been performed as of the audit:

- Review the vulnerability of the high-pressure safety injection (HPSI) throttle valves to clogging.

- Review all LOCA scenarios (i.e., small-break LOCA, medium-break LOCA, and large-break LOCA) to assess system operation.

- Review the characterization and properties of ECCS post-LOCA fluid (abrasiveness, solids content, and debris characterization).

- Review the materials of all wetted downstream surfaces (wear rings, pump internals, bearings, throttle valve plug, and seat materials).

- Review the range of fluid velocities within piping systems. What is the minimum possible, or allowed, to assess settling and what is the maximum possible to assess wear?

- Review the assessment of changes in system or equipment operation caused by wear (i.e., pump vibration and rotor dynamics). Assess whether the internal bypass flow increased, thereby decreasing performance or accelerating internal wear. The current evaluation performed by the pump manufacturer was not included in the submitted package and was therefore not reviewed.

- Assess whether the system, piping, or component flow resistance changed, altering flow balances.

- Assess whether the system piping vibration response changed for any of the above reasons.

The NRC staff examined the CR3 assessment of the following related issues, finding that the licensee has an “unverified assumption” that there is no issue with downstream effects. This is an incomplete evaluation the licensee should address.

- The list of components and flowpaths considered to determine the scope of the licensee’s downstream evaluation (pumps, valves, instruments, and heat exchangers, etc.). CR3 characterized the list as an “un-verified assumption” based on no walkdown information.

- Whether the leakage through seals, etc., increased local dose rates so that credited operator actions, if any, cannot be met.

- Evaluation of the extent of air entrainment (apart from vortexing, this involves ongoing questions about ECCS and incident report evaluation on the significance of ECCS gas intrusion).

- The opening sizes and running clearances in pumps and valves.

- Review the list of system low points and low-flow areas.

- Review the listing and evaluation of instrument tubing connections.

- Review the heat exchangers with small (i.e., 3/8" or less) tubes or whether the ECCS is on the shell side

A comprehensive evaluation of downstream effects responsive to the request in GL 2004-02 [1] had not been performed by the licensee. Ongoing testing and methodology development efforts are expected to provide additional inputs needed to complete this evaluation.

5.3.2 Downstream Effects - Fuel Only

Guidance for considerations when evaluating downstream effects on vessel internals and reactor fuel are also contained in Section 7.3 of the NEI GR [66] and the approved SE [2]. General items identified for consideration include flow blockage associated with core grid supports, mixing vanes, and debris filters; and impact on flowpaths between the downcomer and upper plenum.

Unit 3 Calculation No. M04-0016 “Reactor Building Sump Screen- Downstream Effects Evaluation” [7] documents the methodology the licensee applied to analyze the potential impact of downstream effects on reactor fuel and vessel internal components.

NRC Staff Audit:

Unit 3 Calculation M04-0016, Section 1.0-Methodology, item 4, stated “Blockage of the CR3 reactor internals and the flow passageways in the nuclear fuel assemblies is evaluated in Reference 60.” Section 3.4 of this calculation covers “Reactor Internals and Fuel Blockage.” This section consists of a single paragraph that only provides a summary of the issue, and refers to References 52 and 60. Audit team review of these references found that these evaluations were simplified and were not detailed evaluations of the complex issues associated with injection of debris-laden water into the reactor vessel and core region. For example, in the evaluation of the potential for core blockage, the licensee concluded that “in the unlikely event that the lower end fittings of the fuel were to become completely blocked, there would be adequate flow to the active region of the fuel through the LOCA holes and slots from the ECCS during post-accident mitigation to satisfy the long-term core cooling requirement.” The licensee has not provided calculations showing the adequacy of core cooling using the alternate flow paths. Fine debris transported to the core may be deposited in different areas by the continued boiling process. The licensee has not described any evaluation of the effect of core boiling on debris deposition, such as an evaluation of the potential of blockage of internal core passageways including core grid supports and mixing vanes, or the possibility of binding of the reactor core barrel vent valves. The NRC staff recognizes that if sufficient flow through the core could be provided so that long term core boiling is prevented, or single phase heat removal is established, debris deposition as a result of boiling would not be a problem.

The NRC staff recognizes that both; (1) the results of ongoing NRC-sponsored testing (Note: Test report LA-UR-04-5416 “Screen Penetration Test Report” was publically issued on April 11, 2005, [ADAMS accession number ML051020162]; after the CR3 information was drafted), and (2) the Westinghouse Owners Group (WOG) downstream evaluation methodology were not available at the time for the licensee to complete their evaluation of the potential for downstream effects on reactor fuel and vessel internal components for CR3. Issues that the NRC staff expects to be addressed in the final analysis include:

- Volume of debris injected into the reactor vessel and core region
- Debris types and properties
- Contribution of in-vessel velocity profile to the formation of a debris bed or clog
- Fluid and metal component temperature impact
- Gravitational and temperature gradients
- Debris and boron precipitation effects
- ECCS injection paths
- Core bypass design features
- Radiation and chemical considerations
- Debris adhesion to solid surfaces
- Thermodynamic properties of the coolant

Therefore, during the audit, the NRC staff was unable to completely assess the ability of the CR3 ECCS sump screen to perform as required with the proposed modifications to prevent undesirable downstream effects from impacting reactor fuel or vessel internals performance.

At this time, the NRC staff considers that the following items were not addressed:

- M04-0016, page 1, stated “The process fluid downstream of the reactor building sump screens is estimated to contain volumetric concentrations of less than 1% abrasive and soft, fibrous material.”

- This is listed under “Assumptions Requiring Validation” as item 2.3 “Based on NUREG-0897 [40] and NUREG/CR-2792 [43], volumetric concentrations of less than 1% abrasive and soft, fibrous material will be contained in the process fluid downstream of the reactor building sump screens. This assumption will require validation when the NRC-sponsored strainer bypass test results are made public.” (Note: Test report LA-UR-04-5416 “Screen Penetration Test Report” was publically issued on April 11, 2005, [ADAMS accession number ML051020162]; after the CR3 information was drafted). As noted above, licensee procedures require verification of this assumption.

- A comprehensive tool for evaluation of downstream effects on reactor fuel was not used, and a number of technical issues relevant to injection of debris-laden water into the reactor vessel and core region are not addressed. Ongoing testing and methodology development efforts are expected to provide additional inputs needed to complete this evaluation.

5.4 Chemical Effects

The NRC staff reviewed the licensee’s chemical effects evaluation consistent with the information outlined in Section 7.4 of the GSI-191 SE (Reference 2). This review focused on two areas: (1) the licensee’s plant-specific materials-reactor building pool environment assessment, and (2) the licensee’s evaluation of any potential integrated sump screen head loss consequences related to chemical effects.

For the first area, the NRC staff reviewed the licensee's assessment on whether the joint NRC-EPRI integrated chemical effects test (ICET) parameters (conducted by LANL at the University of New Mexico) are sufficiently bounding for its plant-specific conditions. If the chemical effects test conditions (e.g., pH, types and amounts of materials) did not bound the plant-specific conditions, the NRC staff would expect a technical justification to use the results from ICET in a plant-specific evaluation.

CR3 did not perform an assessment of its plant-specific materials and expected reactor building pool environment relative to the materials and environments of the ICET at the University of New Mexico. CR3 personnel recognize this will be necessary for future submittals.

Since CR3 did not perform a plant-specific materials-reactor building pool environment assessment, the NRC staff cannot conclude that CR3 adequately addressed chemical effects. However, the following discussion is intended to provide some examples of items that should be considered as part of a plant-specific chemical effects evaluation. CR3 uses trisodium phosphate (TSP) as a buffer for the reactor building pool, which is expected to maintain a reactor building pool pH between 7 and 8, similar to the ICET tests that used TSP as the pH control agent. The types and amounts of CR3 plant-specific materials (e.g., aluminum, zinc, carbon steel, etc.) should be compared to those used in the ICET tests. It is important to consider all potential sources of materials during the plant-specific assessment. For example, scaffolding can be a source of significant amounts of aluminum or zinc. CR3 provided documentation indicating a Hi-Heat Aluminum unqualified coating exists within the ZOI. This should be considered as an additional source of aluminum. CR3 specific data from the limiting debris source term for the large break LOCA indicates the following insulation materials would be present in the sump environment: RMI, Nukon® and mineral wool. Any RMI material (e.g., stainless steel, aluminum) should be evaluated based on the behavior of these materials in the appropriate ICET environment. The NRC staff notes that there is no visible corrosion on the stainless steel in the initial ICET tests.

Licensee evaluations indicate an order of magnitude more mineral wool insulation than Nukon® insulation would be transported to the CR3 sump. Since mineral wool insulation is not included in the ICET tests, the NRC staff expects a plant-specific evaluation would consider how the composition and leaching behavior of mineral wool insulation affects the reactor building pool environment relative to the Nukon® insulation used in the ICET tests. If this information is not readily available, bench-top testing could be one approach to evaluating the effects of mineral wool (relative to fiberglass) in the simulated reactor building pool environment. For example, tests to evaluate the effects of insulation materials may consider: the amounts and timing of any precipitants, characterization of deposits that form within the insulation, and comparison of test solution chemistry as a function of time.

The second area of the chemical effects audit plan involved assessing the licensee's evaluation of any potential integrated sump screen head loss consequences related to chemical effects. As part of this area, the NRC staff would review any licensee plans to develop the technical justification necessary to address chemical effects (e.g., testing, analysis, etc.).

Based on information gained from informal testing in Europe, CR3 provided margin for an increase of approximately 30-50% head loss across the new sump strainer. A detailed report from the testing in Europe was not available for NRC staff review. Any technical basis, such as test results, used to support a chemical effects evaluation should be submitted for NRC review.

The chemical effects margin is an unconfirmed assumption in the CR3 sump screen head loss calculation. The licensee understands that this is an inadequate evaluation. Although CR3 applied a 50% increase in head loss at the point of minimum pump NPSH margin (recirculation initiation), an evaluation of head loss consequences should consider the entire ECCS mission time. Results from ICET Tests 1-3 indicated that the time to formation of various chemical species and the amount of deposits within the insulation varied as a function of time in the specific test environments. In some environments, it is possible that the point of minimum NPSH without consideration of chemical effects is different than the point of minimum NPSH margin when chemical effects are included. Since ICET results do not provide head loss associated with any observed chemical effect, additional head loss testing may be necessary as part of some plant-specific evaluations. During the audit process, CR3 personnel described additional plant-specific features such as the ability to measure the differential pressure across the sump screen and a capability of back flush to the sump. A discussion of potential mitigative factors should be included in the chemical effects evaluation section.

6.0 ALION HEAD LOSS TEST FACILITY AUDIT

Crystal River Unit 3 (CR3) sponsored head loss testing performed by Alion Science and Technology at the Alion test facility in its vertical head loss test loop located in Chicago, IL. The purpose of the Alion head loss tests was: (1) to benchmark the NUREG/CR-6224 correlation for application to CR3 debris, and (2) to determine an appropriate equivalent fiber diameter for mineral wool to ascertain its specific surface area for input into the correlation. Since this plant-specific test provides the basis for evaluating debris bed head loss, a team of NRC staff and contractors visited the Alion test facility on May 18, 2005. The team examined the Alion Vertical Loop Test Facility and the associated test procedures used to obtain the CR3 head loss data.

Overall, the NRC staff observed the apparatus to be well designed and constructed. The test crew appeared to be well trained and conscientious in performing their duties. The NRC staff considers that the Alion head loss test apparatus has the ability to meet all testing criteria, but has questions in two test procedural areas. The first question regards the procedure of excluding the floating debris at the debris intake. The second issue is the quality of debris bed uniformity and homogeneity. A detailed report concerning the head loss testing supporting the CR3 sump performance evaluation is appended to this audit report (Appendix III).

7.0 CONCLUSIONS

The pilot audit outcome goal was to provide a report that would inform external stakeholders, CR3, and the NRC review team on lessons learned during implementation of the approved methodology to aid in resolving the PWR sump performance issue.

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

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

The following list reflects the NRC staff technical judgments of the aforementioned key decision points:

3.1 Break Selection

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

The licensee's evaluation of break selection appears to be reasonable. The evaluation was generally performed in a manner consistent with the approved SE methodology. Deviations from the staff-approved methodology appear reasonable based on the technical basis provided by the licensee. The NRC staff and licensee agreed that further clarification of the technical basis for not performing the SBLOCA head loss calculation needs to be documented.

3.2 Debris Generation/Zone of Influence (Excluding Coatings)

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

The licensee's ZOI evaluation appears to be reasonable. The evaluation was performed in a manner generally consistent with the approved SE methodology. The licensee applied the ZOI refinement discussed in Section 4.2.2.1.1 of the SE, which allows the use of debris-specific spherical ZOIs. The licensee applied material-specific damage pressures and corresponding ZOI radius/break diameter ratios as shown in Table 3-2 of the SE, with exceptions. One exception involves application of the Nukon® ZOI for mineral wool type insulation. The licensee also credited robust barriers and "shadowing" effects in its ZOI evaluation, and credits the North D-Ring compartment wall and the primary shield wall as robust barriers in the analysis. The licensee generally provided an adequate level of technical justification with respect to ZOI analyses. Coatings debris generation/zone-of-influence is considered Level 3, as discussed in the conclusion for Section 3.7.1, below.

3.3 Debris Characteristics (Excluding Coatings)

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

The licensee's level of effort for the containment walkdowns for identifying and verifying debris types appears reasonable. Additionally, the NRC staff's review of the licensee's debris size distribution assumptions, which deviate from that approved in the SE for each of the debris types, appear reasonable, except for mineral wool insulation. At this time, the licensee has not provided adequate information for the NRC staff to conclude that mineral wool behaves similarly to Nukon®, and that the debris size distribution applied for mineral wool is appropriate. This issue is also identified in the debris transport section of this report. Coatings debris characteristics is considered Level 2, as discussed in the conclusion for Section 3.7.2, below.

3.4 Latent Debris

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

The licensee's evaluation of latent debris appears reasonable. The evaluation for latent debris was performed in a manner significantly different than the SE approved methodology. However, the large margins of conservatism applied in the final evaluation of sump screen performance, and the evaluation showing low sensitivity to the overall head loss across the sump screen to the amount of latent debris indicate that the evaluation can still provide an acceptable level of confidence. Additional samples for latent debris would be advantageous for strengthening current estimates and evaluation parameters. The licensee is planning for collection of additional confirmatory samples during the next refueling outage.

3.5 Debris Transport

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

The licensee analyzed debris transport in a manner generally consistent with the approved SE methodology. The licensee's debris transport analyses appear reasonable provided that the licensee adequately addresses the issues raised during the audit. The licensee used the following assumptions leading to a non-conservatively lower amount of debris being transported to the sump screen: an upper containment debris retention fraction for individual fibers which is excessive (25%), no floating of mineral wool (ignoring a source of debris transporting to the sump screen), higher incipient transport velocities causing lower amount of transport of debris on the floor, and no erosion of debris (neglecting a source of debris to the sump screen). The licensee used (1) an unverified assumption that the mineral wool debris size distribution would be similar to that of Nukon® and (2) a turbulent kinetic energy model to calculate settling of debris on the reactor building pool floor without addressing model uncertainties. The licensee did not address the effect of debris buildup upstream of the debris interceptor on the pool flow field, and thus the effect on debris transport. The licensee agreed to address these.

3.6.1 Head Loss

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

The licensee's evaluation of head loss appears reasonable if additional information is provided and determined to be sufficient. The licensee performed a high quality evaluation of the head loss across the top-hat strainer array. The NRC staff audited the vendor (Alion) head loss test facility, reviewed fifteen reports submitted in the head loss calculation area, and performed three sets of confirmatory calculations. Based on these audit activities, the licensee's sump strainer head loss methodology appears largely in compliance with the GR and the SE. However, several technical issues remain to be resolved and require additional information from the licensee. First, the uniform debris bed assumption is used as the basis of not conducting scaled top-hat strainer array tests. Additional information is requested for the licensee to provide a sound basis for this assumption. Second, the HLOSS code is the baseline computer code used by the licensee's vendor to perform the head loss calculation. Although the computer code has not been audited by the NRC staff, sample independent confirmatory calculations by the NRC staff were in good agreement for this application. In addition, other minor issues, e.g, air trapping, mineral wool Scanning Electron Microscope testing and head loss testing procedures, are expected to be resolved when the licensee provides additional information.

3.6.4 Net Positive Suction Head

Level of information - Level 1: Robust

The NRC staff reviewed the calculations for NPSH. The calculations were done using generally acceptable methods. The assumptions were generally conservative. However, several aspects reduced conservatism. One of these was the use of a required NPSH value for the building spray pumps based on a 5% head loss rather than the standard 3% value. The other was the use of the methods of Miller for frictional and fitting head loss calculations. While these methods are more realistic in the sense that they account for expected effects, they are evidently used to reduce head loss and therefore increase NPSH margin. The licensee's NPSH calculation methods and results appear generally acceptable and in the robust category, considering the overall conservatism in the calculation.

3.7.1 Coatings Debris Generation/Zone-of-Influence

Level of information - Level 3: Limited information

The staff-approved approach for determining the radius of the ZOI for coatings is by plant-specific experimentation or by use of the SE default value. The licensee assumes a coatings ZOI radius of 4D is conservative and has left this as an unverified assumption in its analyses. The NRC staff expects that the licensee will verify the validity of the coatings ZOI assumption and the NRC staff will review this issue further based on the licensee's September 2005 GL response.

3.7.2 Coatings Debris Characteristics

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

The licensee's coatings debris characteristics evaluations appear reasonable. The licensee's treatment of qualified and unqualified coatings within the coatings ZOI is generally acceptable because this methodology is consistent with the SE. The licensee assumes that all unqualified coatings that are outside the ZOI will fail as 25 micron particulates and transport to the sump. The use of a 25 micron particulate size is generally acceptable because there is not expected to be a significant difference in head loss between 10 and 25 micron particles. This is also discussed in the head loss section of this report. The licensee assumes that degraded qualified coatings outside of the ZOI fail as chips with a failed size equivalent to their applied thickness and are considered in the transport analysis. The licensee's assumption that degraded qualified coatings outside the ZOI will fail as chips appears reasonable. As a result of the audit, the licensee stated that they need to correct the appropriate calculations to reflect the coatings assumed to be failing as chip sizes.

The NRC staff notes that the licensee did not conduct a thorough assessment of the qualified coatings inside containment. This is needed to demonstrate reasonable assurance that the coating will remain intact and not create a debris source in the event of a DBA. Based on visual inspection, the licensee assumes that 30% of the D-ring coatings are degraded, however no testing was performed on the remaining 70% of coating (the coating that is assumed to remain qualified) to demonstrate that it is not degraded. A visual inspection does not sufficiently assess the extent of degradation to the coatings. The coatings that are assumed to remain qualified should be assessed to ensure that they meet their original qualification criteria. The licensee will need to determine the condition of the qualified coatings in containment by more rigorous methods than visual inspection in order to assure that the coatings will remain intact in the event of a DBA

4.3 Screen Modifications

Level of information - Level 3: Limited information

The licensee's overall screen modification approach appears reasonable. However, because the adequacy of the new screen design and other proposed modifications is highly dependent on the acceptability of the various analyses that drive the screen design (i.e., debris generation, debris transport, debris accumulation and head loss), further design changes could be necessary as the licensee finalizes the various ongoing aspects of the sump performance evaluation (e.g., coatings ZOI of 4D, chemical effects, etc.). The licensee stated that they will revise the 10 CFR 50.59 evaluation once the design is finalized and will revise the current licensing basis assumption of 50% blockage to reflect its new mechanistic analyses at that time.

5.1 Sump Structural Analysis

Level of information - Level 1: Not Applicable

The staff review in this area is ongoing, and the staff will inform the licensee of any issues identified upon completion of the review.

5.2 Upstream effects

Level of information - Level 1: Robust

The NRC staff audited CR3 information for appropriate consideration of upstream effects. This included ensuring that the licensee reviewed the flowpaths leading to the sump screen and evaluated them for potential choke points, that they have considered its plant-specific insulation and the unique geometric features of its containment, and that they have considered the effect that the placement of curbs and debris racks intended to trap debris may have on the holdup of water inventory.

Based on discussions with the licensee and review of the pertinent documents, the licensee appears to have adequately reviewed the flow paths leading to the emergency sump screen for choke points, has considered the entrapment of debris upstream of the sump screen with regard to the holdup of water and has considered the effect of holdup in planned modifications. Accordingly, the licensee's treatment of upstream effects appears to be acceptable and the NRC staff considers this a robust conclusion.

5.3.1 Downstream Effects

Level of information - Level 3: Limited information

The licensee was not ready for a detailed review or audit of its assessment of GSI-191 downstream effects. The results of the NRC-sponsored throttle valve tests, NRC-sponsored bypass tests, and the Westinghouse Owners Group (WOG) downstream evaluation were not available at the time the CR3 assessment was performed. In the meantime, the licensee has used a series of “unverified assumptions” as a placeholder. Based upon these “un-verified assumptions,” the licensee conclusion is that there was “no issue.” This is an incomplete evaluation. The licensee recognizes that its plan to address downstream effects should be satisfactorily addressed in the September 2005 submittal. Therefore, the NRC staff is unable, at this time, to assess the ability of the licensee ECCS system to perform as required with the proposed modifications to the licensee containment and containment sumps.

5.3.2 Downstream Effects related to fuel only

Level of information - Level 3: Limited information

For evaluation of downstream effects as they apply to reactor fuel, the licensee evaluation only provides a summary of the issue, and refers to references whose evaluations are over-simplified. The results of the NRC-sponsored throttle valve tests, NRC-sponsored bypass tests, and the Westinghouse Owners Group (WOG) downstream evaluation were not available at the time the CR3 assessment was performed. Therefore, the NRC staff is unable to completely assess the ability of the CR3 ECCS sump screen to perform as required with the proposed modifications to prevent undesirable downstream effects from impacting the reactor fuel or vessel internals performance at this time.

5.4 Chemical Effects

Level of information - Level 3: Limited information

CR3 did not perform an assessment of its plant-specific materials and expected sump environment relative to the materials and environments of the integrated chemical effects tests (ICET). Based on information gained from informal testing in Europe, CR3 provided margin for an increase of approximately 30-50% head loss across the new sump strainer. A detailed report from the testing in Europe was not available for NRC staff review. The licensee understands that this is an inadequate evaluation. Although CR3 applied a 50% increase in head loss at the point of minimum pump NPSH margin (recirculation initiation), an evaluation of head loss consequences should consider the entire ECCS mission time. In some environments, it is possible that the point of minimum NPSH without consideration of chemical effects is different than the point of minimum NPSH margin when chemical effects are included. Since ICET results do not provide head loss associated with any observed chemical effect, additional head loss testing may be necessary as part of some plant-specific evaluations (e.g. if CR3 finds that its plant materials and chemistry match test results that show formation of substances that could contribute to head loss).

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