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

July 26, 2023

Mr. Eric S. Carr
President – Nuclear Operations and
Chief Nuclear Officer
Dominion Energy Nuclear Connecticut, Inc.
Millstone Power Station
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION, UNITS 2 AND 3 – CLOSEOUT OF GENERIC LETTER 2004-02, "POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING DESIGN BASIS ACCIDENTS AT PRESSURIZED-WATER REACTORS" (EPID L-2017-LRC-0000)

Dear Mr. Carr:

The U.S. Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML042360586), dated September 13, 2004, requesting that licensees address the issues raised by Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR [Pressurized-Water Reactor] Sump Performance."

By letter dated May 15, 2013 (ML13141A277), Dominion Energy Nuclear Connecticut, Inc. (the licensee) stated that they will pursue Option 2 (deterministic) for the closure of GSI-191 and GL 2004-02 for Millstone Power Station, Units 2 and 3 (Millstone 2 and 3).

On July 23, 2019 (ML19203A303), GSI-191 was closed. It was determined that the technical issues identified in GSI-191 were now well understood and therefore GSI-191 could be closed. Prior to and in support of closing GSI-191, the NRC staff issued a technical evaluation report on in-vessel downstream effects (ML19178A252 and ML19073A044 (not publicly available, proprietary information)). Following the closure of GSI-191, the NRC staff also issued review guidance for in-vessel downstream effects, "NRC Staff Review Guidance for In-Vessel Downstream Effects Supporting Review of GL 2004-02 Responses" (ML19228A011), to support review of the GL 2004-02 responses.

The NRC staff has reviewed the licensee's responses and request for additional information supplements associated with GL 2004-02. Based on the evaluations, the NRC staff finds the licensee has provided adequate information as requested by GL 2004-02.

The stated purpose of GL 2004-02 was focused on demonstrating compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.46. Specifically, GL 2004-02 requested addressees to perform an evaluation of the emergency core cooling system and containment spray system recirculation and, if necessary, take additional action to ensure system function in light of the potential for debris to adversely affect long-term core cooling. The NRC staff finds the information provided by the licensee demonstrates that debris will not inhibit the emergency core cooling system or containment spray system performance following a postulated loss-of-coolant accident. Therefore, the ability of the systems to perform their safety functions, to assure adequate long-term core cooling following a design basis accident, as required by 10 CFR 50.46, has been demonstrated.

Therefore, the NRC staff finds the licensee's responses to GL 2004-02 are adequate and considers GL 2004-02 closed for Millstone 2 and 3.

Enclosed is the summary of the NRC staff's review. If you have any questions, please contact me at (301) 415-1030 or by email at Richard.Guzman@nrc.gov.

Sincerely,

/RA/

Richard V. Guzman, Senior Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-336 and 50-423

Enclosure:
NRC Staff Review of GL 2004-02 for
Millstone Power Station, Units 2 and 3

cc: Listserv

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

U.S. NUCLEAR REGULATORY COMMISSION STAFF REVIEW

OF THE DOCUMENTATION PROVIDED BY

DOMINION ENERGY NUCLEAR CONNECTICUT, INC.

FOR MILLSTONE POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-336 AND 50-423

CONCERNING RESOLUTION OF GENERIC LETTER 2004-02

POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING

DESIGN-BASIS ACCIDENTS AT PRESSURIZED-WATER REACTORS

1.0 INTRODUCTION

A fundamental function of the emergency core cooling system (ECCS) is to recirculate water that has collected at the bottom of the containment through the reactor core following a break in the reactor coolant system (RCS) piping to ensure long-term removal of decay heat from the reactor fuel. Leaks from the RCS, hypothetical scenarios known as loss-of-coolant accidents (LOCAs), are part of every plant's design basis. Hence, nuclear plants are designed and licensed with the expectation that they are able to remove reactor decay heat following a LOCA to prevent core damage. Long-term cooling following a LOCA is a basic safety function for nuclear reactors. The recirculation sump provides a water source to the ECCS in a pressurized-water reactor (PWR) once the primary water source has been depleted.

If a LOCA occurs, piping thermal insulation and other materials may be dislodged by the two-phase coolant jet emanating from the broken RCS pipe. This debris may transport, via flows coming from the RCS break or from the containment spray system (CSS), to the pool of water that collects at the bottom of containment following a LOCA. Once transported to the sump pool, the debris could be drawn toward the ECCS sump strainers, which are designed to prevent debris from entering the ECCS and the reactor core. If this debris were to clog the strainers and prevent coolant from entering the reactor core, containment cooling could be lost and result in core damage and containment failure.

It is also possible that some debris would pass through (termed "bypass") the sump strainer and lodge in the reactor core. This could result in reduced core cooling and potential core damage. If the ECCS strainer were to remain functional, even with core cooling reduced, containment cooling would be maintained, and the containment function would not be adversely affected.

Findings from research and industry operating experience raised questions concerning the adequacy of PWR sump designs. Research findings demonstrated that, compared to other

LOCAs, the amount of debris generated by a high-energy line break (HELB) could be greater. The debris from a HELB could also be finer (and thus more easily transportable) and could be comprised of certain combinations of debris (i.e., fibrous material plus particulate material) that could result in a substantially greater flow restriction than an equivalent amount of either type of debris alone. These research findings prompted the U.S. Nuclear Regulatory Commission (NRC) to open Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR Sump Performance," in 1996. This resulted in new research for PWRs in the late 1990s.

GSI-191 focuses on reasonable assurance that the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.46(b)(5) are met. This deterministic rule requires maintaining long-term core cooling (LTCC) after initiation of the ECCS. The objective of GSI-191 is to ensure that post-accident debris blockage will not impede or prevent the operation of the ECCS and CSS in recirculation mode at PWRs during LOCAs or other HELB accidents for which sump recirculation is required. The NRC completed its review of GSI-191 in 2002 and documented the results in a parametric study that concluded that sump clogging at PWRs was a credible concern.

GSI-191 concluded that debris clogging of sump strainers could lead to recirculation system ineffectiveness as a result of a loss of net positive suction head (NPSH) for the ECCS and CSS recirculation pumps. Resolution of GSI-191 involves two distinct but related safety concerns: (1) potential clogging of the sump strainers that results in ECCS and/or CSS pump failure; and (2) potential clogging of flow channels within the reactor vessel because of debris bypass of the sump strainer (in-vessel effects). Clogging at either the strainer or in-vessel channels can result in loss of the long-term cooling safety function.

After completing the technical assessment of GSI-191, the NRC issued Bulletin 03-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML031600259), on June 9, 2003. The Office of Nuclear Reactor Regulation (NRR) requested and obtained the review and endorsement of the bulletin from the Committee to Review Generic Requirements (CRGR) (ML031210035). As a result of the emergent issues discussed in Bulletin 03-01, the NRC staff requested an expedited response from PWR licensees on the status of their compliance of regulatory requirements concerning the ECCS and CSS recirculation functions based on a mechanistic analysis. The NRC staff asked licensees who chose not to confirm regulatory compliance, to describe any interim compensatory measures that they had implemented or will implement to reduce risk until the analysis could be completed. All PWR licensees responded to Bulletin 03-01. The NRC staff reviewed all licensees' Bulletin 03-01 responses and found them acceptable.

In developing Bulletin 03-01, the NRC staff recognized that it might be necessary for licensees to undertake complex evaluations to determine whether regulatory compliance exists in light of the concerns identified in the bulletin and that the methodology needed to perform these evaluations was not currently available. As a result, that information was not requested in Bulletin 03-01, but licensees were informed that the NRC staff was preparing a GL that would request this information. GL 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," dated September 13, 2004 (ML042360586), was the follow-on information request referenced in Bulletin 03-01. This document set the expectations for resolution of PWR sump performance issues identified in GSI-191, to ensure the reliability of the ECCS and CSS at PWRs. NRR requested and obtained the review and endorsement of the GL from the CRGR (ML040840034).

GL 2004-02 requested that addressees perform an evaluation of the ECCS and CSS recirculation functions in light of the information provided in the letter and, if appropriate, take additional actions to ensure system function. Additionally, addressees were requested to submit the information specified in GL 2004-02 to the NRC. The request was based on the identified potential susceptibility of PWR recirculation sump screens to debris blockage during design-basis accidents (DBAs) requiring recirculation operation of ECCS or CSS and on the potential for additional adverse effects due to debris blockage of flow paths necessary for ECCS and CSS recirculation and containment drainage. GL 2004-02 required addressees to provide the NRC a written response in accordance with 10 CFR 50.54(f).

By letter dated May 28, 2004 (ML041550661), the Nuclear Energy Institute (NEI) submitted a report (NEI 04-07), "Pressurized Water Reactor Sump Performance Evaluation Methodology" (ML050550138 and ML050550156), which describes a methodology for use by PWR licensees in the evaluation of containment sump performance. This is also called the Guidance Report (GR). NEI requested that the NRC review the methodology. The methodology was intended to allow licensees to address and resolve GSI-191 issues in an expeditious manner through a process that starts with a conservative baseline evaluation. The baseline evaluation serves to guide the analyst and provide a method for quick identification and evaluation of design features and processes that significantly affect the potential for adverse containment sump blockage for a given plant design. The baseline evaluation also facilitates the evaluation of potential modifications that can enhance the capability of the design to address sump debris blockage concerns and uncertainties and supports resolution of GSI-191. The report offers additional guidance that can be used to modify the conservative baseline evaluation results through revision to analytical methods or through modification to the plant design or operation.

By letter dated December 6, 2004 (ML043280641), the NRC issued an evaluation of the NEI methodology. The NRC staff concluded that the methodology, as approved in accordance with the NRC staff safety evaluation (SE), provides an acceptable overall guidance methodology for the plant-specific evaluation of the ECCS or CSS sump performance following postulated DBAs. Taken together NEI 04-07 and the associated NRC staff SE are often referred to as the GR/SE.

In response to the NRC staff SE conclusions on NEI 04-07 "Pressurized Water Reactor Sump Performance Evaluation Methodology" (ML050550138 and ML050550156), the Pressurized Water Reactor Owners Group (PWROG) sponsored the development of the following Westinghouse Commercial Atomic Power (WCAP) Topical Reports (TRs):

- TR-WCAP-16406-P-A, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," Revision 1 (not publicly available, proprietary information), to address the effects of debris on piping systems and components (NRC Final SE at ML073520295).

- TR-WCAP-16530-NP-A, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," dated March 2008 (ML081150379), to provide a consistent approach for plants to evaluate the chemical effects that may occur post-accident in containment sump fluids (NRC Final SE at ML073521072).

- TR-WCAP-16793-NP-A, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," Revision 2, dated July 2013 (ML13239A114), to address the effects of debris on the reactor core (NRC Final SE at ML13084A154).

The NRC staff reviewed the TRs and found them acceptable to use (as qualified by the limitations and conditions stated in the respective SEs). A more detailed evaluation of how the TRs were used by the licensee is contained in the evaluations below.

After the NRC staff evaluated licensee responses to GL 2004-02, the staff found that there was a misunderstanding between the industry and the NRC on the level of detail necessary to respond to GL 2004-02. The NRC staff, in concert with stakeholders, developed a content guide for responding to requests for additional information (RAIs) concerning GL 2004-02. By letter dated August 15, 2007 (ML071060091), the NRC issued the content guide describing the necessary information to be submitted to allow the NRC staff to verify that each licensee's analyses, testing, and corrective actions associated with GL 2004-02 are adequate to demonstrate that the ECCS and CSS will perform their intended function following any DBA. By letter dated November 21, 2007 (ML073110389), the NRC issued a revised content guide (hereafter referred to as the content guide).

The content guide described the following information needed to be submitted to the NRC:

- corrective actions for GL 2004-02,
- break selection,
- debris generation/zone of influence (ZOI) (excluding coatings),
- debris characteristics,
- latent debris,
- debris transport,
- head loss and vortexing,
- NPSH,
- coatings evaluation,
- debris source term,
- screen modification package,
- sump structural analysis,
- upstream effects,
- downstream effects – components and systems,
- downstream effects – fuel and vessel,
- chemical effects, and
- licensing basis.

Based on the interactions with stakeholders and the results of the industry testing, the NRC staff, in 2012, developed three options to resolve GSI-191. These options were documented and proposed to the Commission in SECY-12-0093, "Closure Options for Generic Safety Issue-191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance," dated July 9, 2012 (ML121320270). The options are summarized as follows:

- Option 1 would require licensees to demonstrate compliance with 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," through approved models and test methods. These will be low fiber plants with less than 15 grams of fiber per fuel assembly.

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

- Option 2 Deterministic: Industry to perform more testing and analysis and submit the results for NRC review and approval (in-vessel only).

- Option 2 Risk Informed: Use the South Texas Project pilot approach currently under review with NRR staff.

- Option 3 involves separating the regulatory treatment of the sump strainer and in-vessel effects.

The options allowed industry alternative approaches for resolving GSI-191. The Commission issued a Staff Requirements Memorandum on December 14, 2012 (ML12349A378), approving all three options for closure of GSI-191.

By letter dated May 15, 2013 (ML13141A277), Dominion Energy Nuclear Connecticut, Inc. (DENC or the licensee) stated that they will pursue Option 2 (deterministic) for the closure of GSI-191 and GL 2004-02 for Millstone Power Station, Units 2 and 3 (Millstone 2 and Millstone 3, respectively or Millstone when referring to both units).

On July 23, 2019 (ML19203A303), GSI-191 was closed. It was determined that the technical issues identified in GSI-191 were now well understood and therefore, GSI-191 could be closed. Prior to and in support of closing the GSI, the NRR staff issued a technical evaluation report on in-vessel downstream effects (IVDEs) (ML19178A252 and ML19073A044 (non-public version). Following the closure of GSI-191, the NRR staff also issued review guidance for IVDEs to support review of the GL 2004-02 responses, “NRC Staff Review Guidance for In-Vessel Downstream Effects Supporting Review of Generic Letter 2004-02 Responses” (ML19228A011).

The following is a list of documentation provided by the licensee in response to GL 2004-02 for Millstone:

GL 2004-02 CORRESPONDENCE		
DOCUMENT DATE	ACCESSION NUMBER	DOCUMENT
March 4, 2005	ML050630559	Initial Response to GL
September 1, 2005	ML052500378	Supplemental Information
February 9, 2006	ML060380188	1st NRC RAI (Millstone 2)
February 9, 2006	ML060380199	1st NRC RAI (Millstone 3)
February 29, 2008	ML080650561	Licensee Response to RAI
December 17, 2008	ML083230469	2nd NRC RAI
December 18, 2008	ML083650005	Supplemental Information
March 13, 2009	ML090750436	Licensee Response to RAI
February 4, 2010	ML100070068	3rd NRC RAI
July 8, 2010	ML102010413	Licensee Response to RAI (Millstone 2)
August 10, 2010	ML102140437	NRC Partial Closure Letter (Millstone 2)
September 16, 2010	ML102640210	Licensee Response to RAI (Millstone 3)
December 20, 2010	ML103620562	Licensee Response to RAI (Millstone 3)

May 15, 2013	ML13141A277	Closure Option
April 15, 2021	ML21105A433	Final Supplemental Response (Millstone 3)
May 27, 2021	ML21147A477	Final Response (Millstone 2)
September 9, 2022	ML22251A129	4^{th} NRC RAI
November 7. 2022	ML22312A443	Licensee Response to RAI
May 8, 2023	ML23128A162	Licensee Response to RAI

The NRC staff reviewed the information provided by the licensee in response to GL 2004-02 and all RAIs. The following is a summary of the NRC staff review.

2.0 GENERAL DESCRIPTION OF CORRECTIVE ACTIONS FOR THE RESOLUTION OF GL-2004-02

GL 2004-02 Requested Information Item 2(b) requested a general description of, and implementation schedule for all corrective actions. The following is a list of corrective actions completed by the licensee at Millstone, in support of the resolution of GL 2004-02.

- Replaced trash rack and fine mesh screen (surface area approximately 100 square feet (ft^2)) with new ECCS strainer (with corrugated, perforated stainless-steel fins) with a total surface area of approximately 6,120 ft^2. The replacement strainer is designed to withstand up to approximately one atmosphere (atm) of differential pressure and has a strainer hole size of 1/16 inch (in. or ”), which is smaller than the previous screen hole size of 3/32 inch.

- Removed calcium silicate (cal-sil) insulation, which could become dislodged by any break that could require recirculation such that no cal-sil insulation could become part of the ECCS strainer debris load. The remaining cal-sil insulation in containment is jacketed with stainless-steel and is not susceptible to being dislodged by any break that would require ECCS recirculation (for Millstone 2).

- Changed the start signal for the recirculation spray system (RSS) pumps from approximately 660 seconds following the postulated accident to an automatic start when the refueling water storage tank (RWST) level reaches the low-low level setpoint. This ensures that the replacement strainer is fully submerged prior to drawing water through the strainer for coolant recirculation (for Millstone 3).

- Installed safety related cover plates over the strainer to minimize the potential of air ingestion from falling water entraining air into the strainer (for Millstone 2).

- Defined and detailed containment cleanliness standards in a station housekeeping procedure.

- Established design controls to require evaluation of potential debris sources in containment created by or adversely affected by design changes.

- Completed insulation specification changes to ensure changes to insulation in containment can be performed only after the impact on containment strainer debris loading is considered.

- Performed detailed analyses of debris generation and transport to ensure a bounding quantity and a limiting mix of debris are assumed at the ECCS containment sump strainer. The results were used in performing conservative head loss testing to determine worst-case strainer head loss and downstream effects (for Millstone 3).

- Performed chemical effects bench-top tests, which conservatively demonstrated the solubility and behaviors of precipitates, and applicability of industry data on the dissolution and precipitation tests of station-specific conditions and materials (for Millstone 3).

- Performed reduced-scale testing to establish the influence of chemical products on head loss across the strainer surfaces by simulating the plant-specific chemical environment present in the water of the containment sump after a LOCA (for Millstone 3).

- Performed downstream effects analyses for clogging/wear of components in flow streams downstream of the strainers (for Millstone 3).

- The licensee provided a comprehensive list of conservatisms that were included in its evaluation of ECCS and RSS system performance considering the effects of debris during recirculation.

Based on the information provided by the licensee, the NRC staff considers this item closed for GL 2004-02.

3.0 BREAK SELECTION

The objective of the break selection process is to identify the break sizes and locations that present the greatest challenge to post-accident sump performance. The term ZOI used in this section refers to the spherical zone representing the volume of space affected by the ruptured piping.

NRC Staff Review

The NRC staff review for Millstone 2 is based on documentation provided by the licensee through March 13, 2009. The NRC staff review for Millstone 3 is based on documentation provided by the licensee through February 29, 2008.

The NRC staff review is based on documentation provided by the licensee through February 29, 2008.

Millstone 2

During a GL 2004-02 audit of Millstone 2, the NRC staff found that the licensee met the intent of the NEI 04-07 guidance for break selection. The audit resulted in one open item in the area of break selection. The staff found that a potential break in the RCS loop seal piping had not been evaluated. In its initial supplemental response, the licensee provided information that justified that the potential loop seal piping break was bounded by the breaks that had been previously analyzed. Therefore, the NRC staff considered this item adequately addressed.

In addition to the above, the licensee considered secondary breaks. During the audit, the NRC staff found that secondary breaks do not require recirculation and can therefore be excluded from consideration in the break selection process.

Millstone 3

As stated above, during a GL 2004-02 audit of Millstone 2, the staff found that the licensee met the intent of the NEI 04-07 guidance for break selection. The Millstone 3 break selection process used a similar approach. The ZOIs selected were 17D for fibrous insulation and 28.6D for Microtherm. Because these large ZOIs were used for material destruction zones, the breaks were not moved along the RCS piping incrementally. The break selection process identified two breaks that generate limiting amounts of debris. These are crossover leg breaks at Steam Generators A and B. Breaks at the reactor vessel nozzles were also considered but result in significantly lower amounts of debris that would also be less problematic from a head loss perspective. Because the ZOIs used for the debris generation evaluation are large and most debris within any cubicle is destroyed by any break within that compartment, the NRC staff found that the methodology used is acceptable. The break selection process maximized the amount of Microtherm destroyed, which is a problematic debris type.

In addition to the above, secondary breaks were considered. During the audit, the NRC staff found that secondary breaks do not require recirculation and can therefore be excluded from consideration in the break selection process.

NRC Staff Conclusion

For this review area, the licensee has provided sufficient information such that the NRC staff has reasonable assurance that the subject review area has been addressed conservatively or prototypically. Therefore, the NRC staff concludes that the break selection evaluation for Millstone 2 and 3 is acceptable. Based on the information provided by the licensee, the NRC staff considers this area closed for GL 2004-02 for Millstone 2 and 3.

4.0 DEBRIS GENERATION/ZOI (EXCLUDING COATINGS)

The objective of the debris generation/ZOI evaluation is to determine the limiting amounts and combinations of debris that can occur from the postulated breaks in the RCS.

NRC Staff Review

The NRC staff review for Millstone 2 and 3 is based on documentation provided by the licensee through February 29, 2008.

Millstone 2

The licensee used the GR Section 4.2.2.1.1, ZOI refinement of debris-specific spherical ZOIs. The supplemental response, with reference to the August 30, 2007, audit report (ML072290550) for content, indicated that the licensee assumed the default ZOI of 17D for NUKON, Transco metal cassette encapsulated mineral wool, mineral fiber with stainless-steel jacketing, and Claremont fiberglass insulation. This was deemed acceptable treatment given that the 17D ZOI includes nearly the entire steam generator compartment where the limiting LOCA breaks would occur, and that the large majority of the fibrous insulation was of the Transco cassette construction which afforded more protection than the GR/SE default tested Jacketed NUKON.

The licensee used the GR/SE default ZOI of 2D for Transco Reflective Metallic Insulation (RMI). The originally installed cal-sil insulation was removed from the steam generator compartments and replaced with NUKON. The cal-sil insulation remaining in containment is outside any LOCA break ZOI and is jacketed and not susceptible to erosion. The licensee identified a sacrificial strainer surface area of 150 ft^2 to account for blockage from tags, labels, tape, and other miscellaneous debris.

Millstone 3

The licensee used debris-specific spherical ZOIs to determine the amount of each material in containment that could be damaged by a LOCA jet. The licensee assumed the NUKON default ZOI of 17D for Transco encapsulated fiberglass, fiberglass blankets, and spiral wrap fiberglass. The licensee stated that the large majority (90 percent) of the low-density fiberglass was encapsulated in a more robust configuration, consisting of seal welded stainless-steel cassettes, than the configurations in the tests that determined the approved ZOIs. The smaller quantities (10 percent) of blanket and spiral wrap fiberglass are jacketed/sheathed in stainless-steel similar to the tested configuration. The limiting breaks were determined to be in the steam generator compartments and the 17D ZOI resulted in destruction of most of the insulation in the compartments. The licensee also assumed a ZOI of 28.6D for Microtherm insulation. This was justified as it was comparable with the Min-K granular/particulate insulation with a GR/SE default ZOI of 28.6D and that a 28.6D ZOI included all of the relatively small quantity of such insulation in the steam generator compartments where the limiting breaks would occur. The licensee identified an estimated 872 ft^2 total for miscellaneous debris of signs, placards, tags, stickers, and glass.

NRC Staff Conclusion

For the debris generation/ZOI review area, the licensee provided information such that the NRC staff has reasonable assurance that the subject review area has been addressed conservatively or prototypically. Therefore, the NRC staff concludes that the debris generation/ZOI evaluation for Millstone 2 and 3 is acceptable. The NRC staff considers this item closed for GL 2004-02.

5.0 DEBRIS CHARACTERISTICS

The objective of the debris characteristics determination process is to establish a conservative debris characteristics profile for use in determining the transportability of debris and its contribution to strainer head loss.

NRC Staff Review

The NRC staff review for Millstone 2 and 3 is based on documentation provided by the licensee through February 29, 2008.

Millstone 2

Millstone 2 used a 4-category methodology (described in Appendix II in the SE for NEI 04-07) for defining the size distribution of LOCA generated fibrous insulation debris. The licensee assumed that the debris from NUKON and Claremont low-density fiberglass insulation was 8 percent fines, 25 percent small pieces, 32 percent large pieces, and 35 percent intact pieces. For mineral wool, the licensee assumed that debris generated would be 100 percent fines. The licensee conservatively assumed RMI debris to be 75 percent small pieces and 25 percent large pieces.

The NRC staff considered the size distributions for the insulations acceptable in the audit because they are based on the SE methodology. Acceptable properties were also assumed for latent debris (consistent with SE guidance) and foreign materials (150 ft^2 sacrificial strainer area assumed). This size distribution for fiberglass was considered acceptable in the audit because it is based on the methodology in Appendix II to the GR/SE. The licensee assumed Min-K to be rendered to 100 percent fines.

Millstone 3

The licensee stated that debris types include Transco fiberglass (and other types of low-density fiberglass), Microtherm, latent debris, coatings, and foreign materials. The number of distinct debris types is smaller than that of most plants, which simplifies the characteristics review as well as other areas such as the break selection process.

For Transco fiberglass, which includes small amounts of NUKON (a similar low-density fiberglass), the licensee assumed a size distribution of 8 percent fines, 25 percent small pieces, 32 percent large pieces, and 35 percent intact pieces. This size distribution was based on the methodology in Appendix II of the staff's SE for NEI 04-07 and is identical to the methodology used by Millstone 2 that was accepted during this plant's audit. The licensee's material properties for Transco fiber (fiber size, bulk and material densities) are also acceptable.

For Microtherm, the licensee conservatively assumed 100 percent small fines, with 100 percent transport to the strainer.

For latent debris, the licensee followed SE guidance in assuming 100 percent small fines and full transport to the strainers. The licensee stated that the assumed material properties of latent debris were based on the SE and NUREG/CR-6877. A staff review confirmed this statement. The staff noted that a large quantity of latent debris was assumed for Millstone 3.

For foreign materials, the licensee assumed a very large quantity of debris, and accounted for 75 percent of this surface area (655 ft^2) by designating sacrificial strainer area, which is consistent with SE guidance.

NRC Staff Conclusion

For the debris characteristics review area, the licensee provided information such that the NRC staff has reasonable assurance that the subject review area has been addressed conservatively or prototypically. Therefore, the NRC staff concludes that the debris characteristics evaluation for Millstone 2 and 3 is acceptable. The NRC staff considers this item closed for GL 2004-02.

6.0 LATENT DEBRIS

The objective of the latent debris evaluation process is to provide a reasonable approximation of the amount and types of latent debris (e.g., miscellaneous fiber, dust, dirt) existing within the containment and its potential impact on sump screen head loss.

NRC Staff Review

The NRC staff review for Millstone 2 and 3 is based on documentation provided by the licensee through February 29, 2008.

Millstone 2

Due to the large fibrous debris load in containment, latent debris is a relatively small contributor to strainer head loss. A thorough latent debris inventory was performed, and a conservative bounding number was chosen for the debris calculations. A thin-bed debris load is postulated to form on the containment sump strainer and has been shown, in head loss testing, to produce the worst-case debris bed head loss. The licensee made changes to the plant housekeeping and the containment closeout procedures to explicitly describe containment housekeeping expectations for worksites and the general containment area. Training was provided to plant staff and supplemental staff to emphasize the need for and awareness of the importance of maintaining a clean containment.

The licensee sampled latent debris twice in containment and may sample again if deemed necessary to ensure that the total amount of latent debris in containment remains below the amount used in the strainer design. This sampling is not necessary on a regular basis because the sampling done has demonstrated a significant margin to the acceptance criteria. Latent fiber is an insignificant fraction of the total fiber load and can effectively be ignored. Latent particulate is only a small fraction (approximately 5 percent by volume) of the total particulate load used in the strainer design and so is likewise not expected to be a significant contributor to strainer head loss.

The licensee took two latent debris surveys during two Millstone 2 outages. Based on the surveys and the calculated masses of debris, the licensee selected a conservative estimate for its final total mass of latent debris that allows some margin. The licensee adopted the NEI 04-07 SE value of 200 pound-mass (lbm) latent debris fiber plus particulate, of which 15 percent is assumed to be fiber. The licensee assumed 150 ft^2 sacrificial area of tags and labels.

Millstone 3

Due to the large fibrous debris load in the containment, latent debris is a relatively small contributor to strainer head loss. A thorough latent debris inventory was done, and a conservative bounding number was chosen for the debris calculations. It is expected that further latent debris inventories will not be required, and that latent debris will be adequately controlled through housekeeping and containment cleanup.

The latent debris mass in containment was determined from collection of debris samples from multiple locations in the containment. The methodology followed the guidance in the SE for NEI 04-07. Ten surface categories were identified and a minimum of three samples for each category were collected. The mass per unit area of each sample was calculated. A calculation that extrapolated the mass per unit area to containment scale provided an estimate of the total amount of latent debris in containment. Latent debris surveys were made during two outages, and the larger of the two masses was taken as the conservative estimate. The latent debris mass was taken as 567 pound-mass (lbm). The second survey, from a later outage, yielded an estimate of 344 lbm. Of the 567 lbm, 15 percent was assumed to be fiber. This is consistent with NRC SE guidance.

Strainer sacrificial area due to tags, labels and stickers was computed based upon a plant walkdown that quantified the area associated with these items. A total of 872 ft^2 of tags, labels and stickers was identified, and 75 percent of this area was used to estimate the sacrificial area, as allowed per SE guidance. An area of 655 ft^2 was used as the sacrificial area estimate.

NRC Staff Conclusion

For the latent debris review area, the licensee provided information such that the NRC staff has reasonable assurance that the subject review area has been addressed conservatively or prototypically. Therefore, the NRC staff concludes that the latent debris evaluation for Millstone 2 and 3 is acceptable. The NRC staff considers this item closed for GL 2004-02.

7.0 DEBRIS TRANSPORT

The objective of the debris transport evaluation process is to estimate the fraction of debris that would be transported from debris sources within containment to the sump suction strainers.

NRC Staff Review

The NRC staff review for Millstone 2 is based on documentation provided by the licensee through July 8,2010. The NRC staff review for Millstone 3 is based on documentation provided by the licensee through June 13, 2017.

Millstone 2

The licensee computed the transport of all categories of debris to the sump. The licensee assumed that all debris would initially be swept down into the containment pool. Debris categorized as fines were assumed transported to the sump without settling. Mineral wool and mineral fiber were assumed as 100 percent fines, and fibrous debris settled in the containment pool was assumed to undergo 90 percent erosion. These assumptions provided the input for the head loss testing.

The NRC staff considered the licensee’s transport calculation to be conservative based upon the licensee’s assumptions. However, the NRC staff identified two RAIs. One RAI questioned the basis for the particulate filtration fraction at the strainer for downstream wear evaluations. The licensee stated that no filtration was assumed. This was accepted as conservative by the NRC staff. The second RAI raised the scenario of air ingestion into the strainer by entrainment of air resulting from break flow impacting the pool above the strainer and causing ingestion of the air into the strainer. The staff noted that a cover plate was not installed above the strainer. The licensee responded that the streaming liquid would be broken up by intervening equipment and that air could not be entrained because the governing Froude number would be too low. The staff did not accept the licensee’s response because a significant portion of one of the strainer arrays is located in an area that would be subject to liquid falling from the break. Without a cover plate, the staff concluded that the potential for air entrainment could not be ruled out. The NRC staff requested additional information regarding the outstanding air entrainment question. The licensee ultimately installed a cover plate over the strainer to prevent air entrainment into the structure. The staff found this response acceptable.

Millstone 3

The licensee provided information that was consistent with the content guide. The licensee's approach to addressing GSI-191 was similar to its approach used for Millstone 2, which the staff had previously reviewed and found acceptable (see Millstone 2 audit report for more information at ML072290550). The licensee assumed 90 percent of all small and large fiber debris pieces eroded into fines and that all fines and particulate debris transport to the strainer. The licensee used the FLUENT computational fluid dynamic (CFD) code to compute recirculation in the containment pool. The licensee considered four main transport modes: blowdown, washdown, pool fill-up, and recirculation.

For blowdown, all debris is assumed to blow to the containment floor. The licensee did not complete a washdown analysis because all debris was assumed to transport directly to the containment floor during blowdown, thereby eliminating a need for a washdown analysis. For pool fill-up, the licensee did not credit any inactive pool volumes or debris holdup in its analysis. The licensee assumed that the maximum possible amount of debris available in the pool transported to the strainer.

For recirculation, the licensee used the FLUENT CFD code. The CFD was run for four separate pool recirculation transport scenarios. CFD calculations for large-break LOCA (LBLOCA) cases were performed at minimum water levels at recirculation corresponding to a small-break LOCA (SBLOCA). For a LBLOCA, the water level would be slightly higher at this time due to contribution from safety injection (SI) tanks and several other sources. The licensee assumed 90 percent erosion of settled debris in the containment pool. Due to the large volume of fluid in the RWST, the water level increases significantly following switchover of RSS pumps to recirculation. After 3 hours, the licensee stated that the containment pool depth would increase from 4.3 ft at switchover to 9.3 ft for a LBLOCA. This increase in water level would be expected to significantly reduce flow velocities and turbulence in the containment pool, as well as limiting debris erosion well below 90 percent.

The licensee assumed the following conservatisms with regard to debris transport analysis: 100 percent of fibrous debris transports to the containment floor, 59 percent fibrous debris transport and 100 percent particulate transport to the sump screen, no inactive pool credit, 100 percent fines for Microtherm, and conservative flowrates used in the CFD analysis.

NRC Staff Conclusion

For this review area, the licensee provided information such that the NRC staff has reasonable assurance that the debris transport has been addressed conservatively or prototypically. Therefore, the NRC staff concludes that the debris transport evaluation for Millstone 2 and 3 is acceptable. The NRC staff considers this area closed for GL 2004-02.

8.0 HEAD LOSS AND VORTEXING

The objectives of the head loss and vortexing evaluations are to calculate head loss across the sump strainer and to evaluate the susceptibility of the strainer to vortex formation.

NRC Staff Review

The NRC staff review for Millstone 2 is based on documentation provided by the licensee through July 8, 2010. The NRC staff review for Millstone 3 is based on documentation provided by the licensee through June 13, 2017.

Millstone 2

Based on the staff audit of Millstone, and the licensee's original submittal and initial supplemental response (September 1, 2005, and February 29, 2008, respectively), the licensee's approach to reduce strainer head loss was to install an array of Atomic Energy of Canada Limited (AECL) strainer modules in place of the original sump strainer. Scaled testing of the strainer modules was conducted at AECL. The testing did not initially address chemical effects but concentrated on fibrous and particulate debris predicted to reach the strainer following a postulated LOCA. The final chemical effects head loss testing had not been completed by the licensee at the time the February 29, 2008, supplemental response was sent.

On December 17, 2008, the NRC staff requested additional information (four RAIs) regarding this area. The first RAI requested a complete submittal on head loss and vortex formation. This was provided in the licensee's March 13, 2009, RAI response. The NRC staff found the response acceptable. The second RAI requested the results of a flashing evaluation for the strainers after the final head loss evaluations were completed. This flashing evaluation was provided by the licensee in its March 13, 2009, supplemental response and was confirmed by NRC staff using a debris bed head loss correlation. This confirmed that downstream voiding would not occur for a LBLOCA, and the issue was considered closed. The third RAI requested information showing that the stirring of the test tank did not adversely affect formation of the debris bed on the strainer during thin-bed testing. The licensee stated that all fibrous debris for the testing was prepared as 100 percent fines, with no larger fibrous shreds. The NRC staff accepted this response based on the licensee's answer and other observations regarding the testing. The fourth RAI requested justification that the final chemically laden head loss value used in the strainer evaluation was adequate considering the previous non-chemical head losses were significantly higher than the non-chemical head losses determined in association with the recent chemical testing. The licensee provided additional detail in this area in its December 18, 2009, supplemental response but there were outstanding issues associated with this RAI. An additional question regarding water falling near the strainer and causing air entrainment is discussed in the Debris Transport Section.

On February 4, 2010, the NRC staff asked two additional RAIs. The first RAI requested information that justified that the most recent (Rig 89) head loss test was conducted with a fibrous debris load that maximized non-chemical debris head loss. The second RAI requested information that the debris bed contained adequate fiber to ensure that a maximum head loss was attained without bed disturbances limiting the head loss.

On July 8, 2010, the licensee responded to the RAIs. In response to the first RAI, the licensee stated that early testing (Rig 33 and Rig 42) had established that the limiting thin bed for the Millstone 2 plant-specific debris combination occurred at 1/8 inch. The licensee also stated that due to a miscalculation of test strainer area, the fiber amount was 13 percent greater than that required to form the limiting 1/8-inch thin bed. The licensee also stated that the amount of debris that reached the strainer during the various tests was similar. The licensee provided a photograph of the debris bed formed in the Rig 89 test. The NRC staff reviewed the licensee's response and determined that the additional 13 percent fiber load would not significantly affect the morphology of the thin-bed. The NRC staff was more concerned that the fibrous debris load was too small to result in a fully formed filtering debris bed. The photograph of the debris bed showed that it covered the strainer completely and uniformly with no discontinuities. The NRC staff found the response acceptable based on the early testing, which validated the thin bed fiber amount, and the conclusion that a uniform filtering bed was formed. In response to the second RAI, the licensee referenced its response to the first RAI as relevant. The licensee emphasized that the photograph of the debris bed showed uniformity across the entire surface of the strainer and that the bed contained no cracks, holes, or other discontinuities. The licensee noted that the debris bed was about 1/4 inch thick which provided adequate structural integrity to support chemical effects head losses without bed degradation. The licensee also noted that flow sweeps were conducted at the end of the test and that the flow sweeps showed no signs of hysteresis and that head loss returned to the previous value when flow was restored. A post-test debris bed inspection did not find any flaws in the debris bed that would result in lower head losses. The NRC staff reviewed the licensee’s response and found it acceptable. The photograph of the debris bed, the description of the post-test debris bed inspection, and the results of the flow sweeps all indicate that the debris bed was structurally intact at the end of the test such that head loss was not limited by any flaw in the bed.

Prior to asking the last two RAIs, the NRC staff had identified significant differences between the non-chemical head losses in the Rig 33 and Rig 89 tests for all Dominion strainer tests conducted at AECL. For Millstone 2, the Rig 89 test provides the design basis head loss for the strainer. The Rig 33 tests were performed without chemicals to assist in the strainer evaluation. The Rig 89 tests were performed later and contained chemical debris. For Millstone 2, the final Rig 89 test head loss maximum was 1.04 pounds per square inch (psi) at 40 degrees Celsius (°C). When corrected, this is very close to the strainer design limit of 1.05 ft. at 210 degrees Fahrenheit (°F). The earlier Rig 33 test maximum was about 0.81 psi at 40 ˚C. The Rig 33 test did not include chemical effects. The Rig 89 test, prior to the injection of chemicals was only 0.26 psi, significantly lower than the Rig 33 result. However, the Rig 89 test assumptions were based on a reduced debris source term, so some reduction in head loss would be expected. The issue regarding the differences between the non-chemical head losses among the test facilities was initially addressed for North Anna in a chemical effects audit report (ML090410618). It was also noted that the tested head loss for Millstone 2 slightly exceeded the short-term design limit during the Rig 89 test chemical portion of the test. The Dominion plants that used AECL did not perform time or temperature based NPSH evaluations or attempt to analytically credit delayed precipitation of chemicals. For North Anna, the NRC staff was able to conclude that the conservatism inherent in the analysis and testing, along with the consideration of the timing associated with NPSH margins, debris bed formation, and chemical precipitation, provided reasonable assurance that adequate NPSH margins would be maintained. The NRC staff made a similar determination for Millstone 2. This reasoning was also applied to Surry. A major consideration is that the non-chemical head loss from both Rig 33 and Rig 89 tests was less than the minimum allowable. Combined with the delayed arrival of chemicals, and the increasing subcooling resulting in NPSH margin increases, there is reasonable assurance that adequate NPSH will be available for the Millstone 2 pumps throughout the post-LOCA period. The NRC staff also noted that although the Rig 89 final head loss was higher than Rig 33, the final chemical additions did not result in increases in head loss. Therefore, it is unlikely that the Rig 33 head loss would have increased significantly beyond the maximum in the Rig 89 test Therefore, the NRC staff concluded that the strainer head loss developed by the licensee is acceptable.

Millstone 3

The review of the head loss and vortexing area for Millstone 3 elicited significant questions from the NRC staff. The licensee provided additional information in response to the staff questions. The review will be described chronologically.

The licensee installed an array of AECL strainer modules in place of the original sump strainer. Scaled testing of the strainer modules was conducted at AECL. The new strainer modules increase the area of the strainer significantly (about 5,041 ft^2). The AECL strainer design incorporates internal orifices to promote uniform flow through all strainer perforated surfaces. However, the flow for a clean strainer may not be fully uniform. The testing completed up to the time that the licensee's original submittal was received had not addressed chemical effects. The testing concentrated on determining the effects of fibrous and particulate debris predicted to reach the strainer following a postulated LOCA.

All evaluated breaks generate significant fibrous debris. The licensee assumed 567 lbs. of latent material within containment, of which 85 lbs. would be fine fibers. The licensee assumed 655 ft^2 of sacrificial area for labels, tape, and other miscellaneous debris. Because the final chemical effects head loss testing had not been completed when the supplemental response was sent in, no detailed review of this area was performed.

A preliminary evaluation of the head loss and vortex evaluation was performed based on the information that was included in the initial submittal and NRC staff observations of the AECL chemical effects and non-chemical effects testing made during trips to the AECL test facility. The Millstone testing was conducted by AECL at their labs in Chalk River, Ontario, Canada. The Millstone testing was conducted using various test set ups. AECL has performed large scale testing (Rig 85), reduced-scale testing (Rig 33), and, at the time of the initial supplemental response was performing chemical testing (Rig 89). In general, the NRC staff has found the testing procedures used by AECL for non-chemical debris to result in prototypical or conservative head loss predictions for the strainer. Testing of AECL strainers has found that, in general, thin beds (vs. full loads) present the most challenging head losses. NRC staff observed AECL's debris preparation and introduction practices and found that they result in prototypical or conservative debris bed formation.

The Millstone 3 strainer is submerged by at least 8 inches during a SBLOCA at the onset of recirculation. For a LBLOCA, submergence is greater than 8 inches and increases to about 5 ft by the time the RWST is emptied. The maximum time to empty the RWST is 3 hours. Clean strainer head loss (CSHL) (calculated) is 0.382 ft.

Flow through the strainer is predicted to be about 8,220 gallons per minute (gpm) resulting in a strainer approach velocity of 0.0036 ft/sec. Testing was conducted with a slightly higher approach velocity of 0.0043 ft/sec. The test velocities were therefore conservative.

Vortex testing was performed at below zero submergence and also with the strainer just covered. This testing was conducted at double the scaled design flow rate. No vortex formation occurred, and no air ingestion was observed under these conditions.

Prior to the NRC staff issuance of RAIs, the licensee developed a detailed description of its chemical effects testing and provided it to the NRC in an updated response dated December 18, 2008.

The staff reviewed the updated response and developed and issued RAIs on December 17, 2008, requesting that the licensee provide additional information regarding the strainer testing and evaluation. Some of the questions were based on staff observations of head loss testing for Millstone 3. Only the most significant questions will be discussed in detail in this summary. On March 12, 2009, the licensee provided responses to the RAIs.

The NRC staff requested that the licensee provide a flashing evaluation for the strainer after the final design head loss values had been determined. The licensee provided the requested information, and the staff found the response acceptable.

One of the major head loss issues identified by the NRC staff was that the non-chemical effects tests sometimes incurred higher head losses than the chemical effects tests prior to the addition of chemicals. This issue had been identified previously by the NRC staff and addressed for other plants. For example, refer to the North Anna Chemical Effects Audit (ML090410618), which discusses the issue and the sub-issues identified below in greater detail. Each plant affected by this issue was requested to demonstrate that it had margins adequate to ensure successful strainer performance, even considering the results of the non-chemical tests that the test vendor and licensee considered to be unrealistic. The NRC staff requested a similar evaluation from Millstone 3.

The NRC staff asked several questions regarding this issue to understand the significance of its effect on the Millstone 3 strainer performance. The NRC staff was concerned that the final chemical effects tests may not identify the limiting head loss for the plant conditions if the head losses incurred prior to the addition of chemicals were non-prototypically low. Some of the reasons provided by the licensee for the excessively high non-chemical effects test head loss are as follows:

- The particulate debris loading for the non-chemical effects tests was high due to a recalculation of debris source term. More particulate in the test would result in a higher head loss.
- Biological growth in the debris bed caused by organisms from river water used in the test caused the non-chemical head loss tests to incur higher than typical head losses.
- Fine particulate contained in river water in the non-chemical tests caused a less porous bed and increased the head losses in the non-chemical tests.
- Air entrainment in the debris bed resulted in higher head losses for the non-chemical tests.

The licensee also provided a list of significant conservatisms that were included in the head loss evaluation.

The NRC staff have observed many tests conducted under similar conditions but has not observed such a large difference in test results. Therefore, the staff found that the issue was not adequately addressed in the initial RAI response. The NRC staff determined that additional information was required from the licensee for resolution of the issue.

The staff restated the questions regarding the differences in head losses between chemical and non-chemical tests including more detail in an additional round of RAIs. These RAIs were transmitted by letter dated February 4, 2010. The licensee responded in a letter dated September 16, 2010. The licensee’s response contained additional details regarding the testing and the potential reasons for the differences in head loss. The staff considered the response but were unable to conclude that the licensee had provided definitive information for the cause of the significant differences in the test results. The licensee provided additional information regarding the margins and conservatisms included in the head loss evaluation by letter dated December 20, 2010. The NRC staff reviewed the information and found inconsistencies in the information provided regarding two non-chemical effects head loss tests conducted in the Rig 33 facility, and information regarding the strainer submergence. The licensee provided clarifications on these issues in a letter dated June 13, 2017. The staff reviewed all of the information provided by the licensee that is pertinent to strainer performance under accident conditions including the effects of debris. The staff performed confirmatory calculations and evaluations to verify that the strainer would perform acceptably to support the recirculation function. The staff verified that NPSH limitations, flashing, vortexing, structural collapse, or deaeration of the fluid passing though the strainer would not result in failure. The staff considered the reduction in coatings source term that was calculated after the non-chemical testing that reduced the particulate load in the chemical effects testing. The staff also evaluated the assumptions used regarding the timing of head loss increases and found them to be acceptable. The NRC staff found that early chemical precipitation would not occur based on the plant-specific Rig 89 testing and autoclave testing performed in support of WCAP-17788-P. (Refer to Section 16, Downstream Fuel and Vessel section of this summary.) Delay in the development of chemical effects allows the sump pool temperature to decrease resulting in significantly greater NPSH margins due to subcooling of the fluid. The staff judgment of adequate strainer performance did not rely on crediting the potential reasons for higher head losses in the non-chemical testing in its evaluation. These phenomena may have had some effect on the tested head loss, but the staff could not quantify their magnitude. The NRC staff did consider the margins and conservatism included in the Millstone 3 strainer evaluation. Even though the effects of these margins and conservativisms cannot be quantified, it is likely that head loss would be lower than that determined via testing. The NRC evaluations were reviewed by senior technical staff and determined to be acceptable.

NRC Staff Conclusion

For the head loss and vortexing area, the licensee provided information such that the NRC staff has reasonable assurance that the strainer head loss and potential for air ingestion has been addressed conservatively or prototypically. Therefore, the NRC staff concludes that the head loss and vortexing evaluation for Millstone 2 and 3 is acceptable. The NRC staff considers this area closed for GL 2004-02.

9.0 NPSH

The objective of the NPSH section is to calculate the respective NPSH margins for the ECCS and CSS pumps that would exist during a LOCA, considering a spectrum of break sizes.

NRC Staff Review

The NRC staff review for Millstone 2 is based on documentation provided by the licensee through March 13, 2009. The NRC staff review for Millstone 3 is based on documentation provided by the licensee through December 20, 2010.

Millstone 2

The NRC staff reviewed the licensee’s submittal and supplemental information and determined additional information was needed. In the first RAI, the NRC staff noted that the supplemental response led to an increase in NPSH margin from that prior to GL 2004-02 evaluations. The assumptions that led to this increase were not identified. The licensee provided additional information explaining that the difference was due to changes in water level calculations and primary system flow rates. The NRC staff reviewed the changes and found them acceptable. The second RAI was related to the potential for a single failure of a low-pressure safety injection (LPSI) pump to trip and the decrease in NPSH margin that would result from the increased flow rate. The licensee stated that procedural changes had been implemented that would limit the flow rate to less than the strainer design flow rate. The NRC staff accepted this response based on the licensee's emergency operating procedure changes, which would involve closing LPSI system injection valves, if necessary.

Audit Open Item 6 was generated during the audit because it was not clear that the NPSH margins calculated for the LBLOCA bounded the SBLOCA case. The licensee provided the differences between the cases to demonstrate that the SBLOCA case has margin greater than the LBLOCA case. Although the sump level is lower for the SBLOCA case, the NPSH requirements for the high-pressure safety injection (HPSI) pump is significantly lower due to lower flow during the SBLOCA scenario. The NPSH required difference more than offsets the difference in sump levels.

During the audit, the NRC staff determined that the sump level calculation did not account for some potential holdups of water from the sump pool (Audit Open Item 7). The licensee stated that the holdup calculation was revised to account for the items identified during the audit, where appropriate. The holdups include water droplets in transit, refueling cavity holdup, and condensate films on structures and heat sinks. The sump level was also revised to account for a slope in the containment floor and to remove a credit for ductwork displacing water volume. The staff found the revision to the water level calculation acceptable.

The NRC staff considered the NPSH calculation to have been performed using conservative methods but did not consider it to be overly conservative. The final NPSH margin for the HPSI pumps calculated by the licensee was 1.05 ft. at 210 °F. This margin is relatively small and represents the margin for debris bed head loss. The calculational methodology used to derive this value of 1.05 ft was considered to contain sufficient conservatism; therefore, the NPSH area was considered to be addressed sufficiently.

Millstone 3

The licensee presented a high-level summary of its NPSH analysis, providing an overview of the methodology that was used and the assumptions that were made; the basic results of the analysis are presented. The information requested in the GL content guidance document is presented by the licensee.

In general, the methodology used is standard industry practice for calculation of NPSH available and NPSH margin. Each of the terms of the NPSH margin equation is calculated using standard engineering models. The NPSH analysis has been performed with a combination of conservative and realistic assumptions.

The licensee provided NPSH margin analyses and results for its RSS pumps for the recirculation phase of a LOCA. The RSS pumps are the only pumps that take suction through the sump strainer. This recirculation system consists of two trains, each containing two RSS pumps taking suction from the containment sump. The RSS pumps provide both core cooling (including piggyback flow to the SI and charging pumps) and flow to the spray headers.

The licensee calculated a minimum NPSH margin at RSS pump start in excess of 19 feet. The NRC staff noted that it appeared that this margin was computed including suction line losses, but not sump screen and debris losses and determined that this must be clarified by the licensee.

The licensee discussed suction line flashing and stated that it is more limiting than NPSH margin for the RSS pumps. The reported margin to flashing is 6.49 ft for a pump flow rate of 2,450 gpm, while the maximum debris bed head loss is 5.8 ft. The margin to flashing in the presence of a design basis debris bed, therefore, is 0.6 ft. According to the licensee, if the pump flow rate were 3,000 gpm, the margin would be negative, indicating that flashing could occur in the suction lines.

The licensee assumed that water vapor pressure is equal to containment atmospheric pressure and that Millstone 3 does not credit containment accident-induced overpressure for NPSH calculations for the ECCS pumps.

The static head term, or sump level, was computed using only the minimum volume of water in the RWST that will be injected by switchover, which is presented as 597,593 gallons. Water holdup was analyzed and includes containment spray water condensation on heat sink surfaces, mass required to fill the RCS, filling of spray headers, and filling of the containment sump and suction piping. The volumes of some large equipment were conservatively neglected in computing displaced water volume when calculating sump level. It was not clear to the NRC staff whether holdup of water as steam in the containment atmosphere was accounted for.

The RSS pump flow rate is used in the estimation of the head loss term and the required net positive suction head ($NPSH_r$). The licensee stated that orifices limit RSS pump flow to 3,000 gpm +5 percent uncertainty. These orifices limit pump flow and therefore prevent RSS pump suction line flashing. The licensee stated that the only time this flow would be approached is during startup until the discharge header is full. Once the header is filled, flows are predicted to drop to a steady state maximum of 2,450 gpm per pump.

The head loss term was modeled using an industry-standard hydraulic program that makes use of data from the “Flow of Fluids Through Valves, Fittings, and Pipe,” Crane Technical Paper No. 410.

The $NPSH_r$ was computed using information provided by the pump vendors. These data are obtained from vendor tests which generally use the 3 percent flow reduction approach. In this case, the licensee stated that a 1 percent criterion was used. Use of a 1 percent criteria is conservative when compared to the typical use of 3 percent. No reduction in $NPSH_r$ was taken for elevated fluid temperatures. This is conservative and follows NRC guidance.

The licensee identified two modes of recirculation heat removal, cold-leg recirculation and two-path recirculation. The differences between these two modes with respect to NPSH margin was not discussed.

The licensee does not distinguish between small-break and large-break LOCA analysis in the discussion of NPSH.

Because flashing margins were identified as being limiting, the licensee provided a more detailed discussion for this topic. The licensee computed the margin to flashing including CSHL. The margin to flashing was calculated for two RSS pump flow rates: 3,000 gpm and 2,450 gpm per pump. The value of 3,000 gpm is viewed as overly conservative by the licensee. The 2,450 gpm is viewed as a more reasonable flow rate that is still conservative. The computed margins are 5.2 ft for 3,000 gpm and 6.49 ft for 2,450 gpm. These compare with a maximum design basis debris bed head loss of 5.8 ft. The smaller flow rate can accommodate the debris bed head loss, and the larger flow rate cannot.

Based on the initial review, the NRC staff issued RAIs for the NPSH area on December 17, 2008. These RAIs and the licensee responses are discussed below.

In RAI 7, the NRC staff requested that the licensee provide information that demonstrates that adequate NPSH margins are maintained throughout the post-loss of coolant accident mission time. The licensee stated that the maximum $NPSH_r$ for the design-basis RSS lineup after RSS system fill-up is complete is 4 ft at a flowrate of 3,000 gpm. During fill-up, the licensee stated that the $NPSH_r$ could be 4.5 ft at a flowrate of 3,150 gpm. The licensee stated that the minimum available NPSH ($NPSH_a$) for the RSS pumps is 17.4 ft calculated using conservative inputs. As a result, the licensee stated that the minimum NPSH margin is 13.4 ft for a flowrate of 3,000 gpm. The licensee stated that the NPSH margin will increase from this value as more water is added to the sump from the RWST and as the sump water cools over time. The licensee stated that suction line flashing is more limiting than NPSH for the RSS system. For evaluating flashing, the licensee used conservative inputs and calculated a flashing margin of just over 2 inches for the SBLOCA case and about 10 inches for the LBLOCA. The licensee noted that the calculation was done without fully adding the RWST inventory (for a LBLOCA, sump level will reach 9 ft after 3 hours) and assuming the debris bed formation at time zero. Based upon the information provided by the licensee, the NRC staff found that adequate margin is available for the RSS pumps for NPSH and flashing based on the information provided and the conservatism contained in the calculations.

In RAI 8, the NRC staff requested that the licensee clarify whether water holdup due to steam in the containment atmosphere was included in the minimum water level calculation. The licensee stated this holdup had been included in the calculation. The NRC staff found that this response was acceptable since steam in the atmosphere is an expected holdup, and it was appropriately accounted for by the licensee.

In RAI 9, the NRC staff requested the licensee to clarify the holdup assumptions for fluid in the refueling cavity, especially long-term holdup because spray water could continue to fill the cavity and be prevented from reaching the sump. The licensee stated that in the minimum water level calculation, the refueling cavity saddle volume is 99 percent full prior to RSS pump start. Spillover from the refueling cavity saddle volume fills the reactor cavity and instrumentation tunnel prior to spilling onto the containment floor. The NRC staff found that the information provided by the licensee provided a partial answer to the question. The NRC staff did not understand how water drains from the refueling cavity into the reactor cavity, and whether this drainage path is large enough to ensure that debris blockage would not occur, thus preventing water from reaching the sump. A follow up to RAI 9 was transmitted to the licensee via letter dated February 4, 2010. The follow up to RAI 9 requested the licensee to provide additional information regarding how potential holdups in the refueling cavity were considered in the sump level calculation, including the concerns discussed above. In a letter dated September 16, 2010, the licensee stated that the water level calculation considered the maximum potential holdup volume of the refueling cavity. Any water volume greater than that considered to hold up in the refueling cavity would spill into the reactor cavity and the instrument tunnel. The reactor cavity was assumed to hold up 99 percent of the maximum possible volume and the instrument tunnel was considered to hold up 100 percent of the maximum volume possible. The licensee stated that water that overflows the refueling cavity spills into the reactor cavity by way of eight seal ring hatches that are about 24 inches in diameter. The licensee also stated that the hatches have protective covers installed about 8.5 inches above the openings and that the 8.5-inch clearance is adequate to allow unimpeded flow of water through the hatches. Based on the information provided by the licensee, the NRC staff was able to conclude that the water level calculation contained conservative assumptions regarding potential holdup in the refueling cavity. Therefore, the NRC staff found the response to RAI 9 acceptable.

In RAI 10, the NRC staff requested that the results of the NPSH margin calculation, in the absence of the strainer or debris bed, be clearly presented. The NRC staff requested that the response include an explanation of the assumptions and plant conditions and operating states that are relevant to each NPSH margin case. The licensee stated that the RSS pumps are the only pumps that draw suction from the sump, summarized the assumptions made for the NPSH calculation, and provided the results of the calculations. The licensee showed that a large NPSH margin exists for the RSS pumps during recirculation, even allowing for the increased flowrate during the RSS piping fill-up. With regard to suction line flashing, the licensee stated that during system fill-up (which takes about 2 minutes), the margin available to suction line flashing is about 4.1 ft. Afterward, the licensee stated the margin is 5.7 ft, based on a reduced flowrate from 3,150 gpm to 2,500 gpm. The NRC staff considered the response to be acceptable because the requested information was provided and found to adequately address the concern identified in the RAI.

In RAI 11, the NRC staff requested that the licensee describe the basic assumptions and methodology used in the analysis for computing the strainer flashing margins so that the NRC staff could validate that the results are conservative. The licensee stated the flashing analysis used a value of 8,220 gpm for the largest steady-state value of RSS combined pump flow. The response tabulated the locations of the dominant CSHL contributors, and the licensee stated over 90 percent of the CSHL would be at the containment floor level, submerged by 44 inches. The licensee also stated that the CSHL was calculated with 100 °F water, whereas the lowest saturation temperature of the sump fluid is 195 °F. Any subcooling credited in the flashing calculation would improve the calculated margin. The NRC staff did not identify any concerns with the licensee’s response and found the issue to be addressed acceptably.

The licensee provided additional information that could impact the NPSH evaluation in a letter dated December 20, 2010. The information was in response to questions regarding the strainer head loss evaluation. The review of that information is included in the Head Loss and Vortexing section. The staff evaluation of that information found that the NPSH evaluation, as reviewed in this section, remains valid.

NRC Staff Conclusion

For the NPSH area, the licensee provided information such that the NRC staff has reasonable assurance that it has been addressed conservatively or prototypically. Therefore, the NRC staff concludes that the NPSH evaluation for Millstone 2 and 3 is acceptable. The NRC staff considers this area closed for GL 2004-02.

10.0 COATINGS EVALUATION

The objective of the coatings evaluation section is to determine the plant-specific ZOI and debris characteristics for coatings for use in determining the eventual contribution of coatings to overall head loss at the sump screen.

NRC Staff Review

The NRC staff review for Millstone 2 and 3 is based on documentation provided by the licensee through February 29, 2008.

The NRC staff noted that although the review guidance calls for the dry film thicknesses, this information was not listed in the Millstone 2 and 3 responses. Although missing, the staff did not consider this a significant gap in information since conservative quantities of coating debris were provided.

For Millstone 2, the licensee used a ZOI of 10D for strainer sizing and head loss testing (acceptable by the NRC SE). Subsequently, a ZOI of 5D, which is acceptable per WCAP-16568-P for epoxy or epoxy top coated qualified coating systems, was used for the Millstone 2 chemical effects and downstream evaluations. A ZOI of 5D was also used for the chemical effects and downstream testing for Millstone 3.

All qualified coatings in the ZOI are assumed to fail as fine particulate. All unqualified coatings in containment are assumed to fail as fine particulate to maximize transport. Debris transport assumptions are that 100 percent of the coating debris transports to the sump. This assumption was also used in development of coating debris values for head loss testing. The surrogate material used for testing was walnut shell flour.

The licensee’s coating assessment program meets expectations.

NRC Staff Conclusion

For this review area, the licensee provided information such that the NRC staff has reasonable assurance that the subject review area has been addressed conservatively or prototypically. Therefore, the NRC staff concludes that the coatings evaluation for Millstone 2 and 3 is acceptable. The NRC staff considers this item closed for GL 2004-02.

11.0 DEBRIS SOURCE TERM

The objective of the debris source term section is to identify any significant design and operational measures taken to control or reduce the plant debris source term to prevent potential adverse effects on the ECCS and CSS recirculation functions.

NRC Staff Review

The NRC staff review for Millstone 2 and 3 is based on documentation provided by the licensee through February 29, 2008.

The licensee provided a description of significant design and operational measures taken to control or reduce the plant debris source term to prevent potential adverse effects on the ECCS and CSS recirculation functions. The program establishes overall standards for containment conditions relative to containment recirculation sump performance.

Housekeeping and Foreign Material Exclusion programs are in place to maintain cleanliness of containment and to protect plant equipment by preventing entry of foreign material. Tags and stickers are controlled by procedure and the strainer is designed for bounding amounts of such items. A conservatively large surface area was added to the strainer to account for blockage by foreign material, including stickers, labels, and tags. Sufficient guidance exists to prevent addition of significant quantities of additional tags or stickers to containment.

The licensee sampled latent debris twice in containment and may sample again if deemed necessary to ensure that the total amount of latent debris in containment remains below the amount used in the strainer design. This sampling is not necessary on a regular basis because the sampling that has been accomplished has demonstrated a significant margin to the value used in design of the strainer. Latent fiber is an insignificant fraction of the total fiber load and so can effectively be ignored. Latent particulate is only a small fraction (approximately 10 percent by volume) of the total particulate load used in the strainer design and so is likewise not expected to be a significant contributor to strainer head loss.

The containment housekeeping procedure has been updated to ensure that material that could impact strainer performance is not left in containment. This procedure includes direction to set up debris barriers at work areas, perform cleanup of work areas at the end of each shift, remove barriers at the end of work, and to leave the work area cleaner than it was found at the start of the work.

The containment closeout procedure has been updated to require an inspection of containment for loose debris that could block the containment sump strainer and an inspection of the strainer for debris, damage, or blockage.

For permanent plant changes, the design review process was updated to require that all design changes be reviewed using a series of detailed questions designed to determine whether any potential debris source is being added to containment. If a debris source is introduced, the process requires that a detailed review be conducted to review the potential impact on ECCS sump strainer head loss.

Unqualified coating systems are not allowed to be applied to the inside of containment buildings. Small quantities of unqualified coatings on vendor supplied equipment may be allowed if added to the unqualified coating total maintained in a calculation by engineering. Coating systems used inside containment are required to comply with containment coatings specification.

A coatings inspection and remediation procedure are also in place to ensure that the coatings inside containment remain within the requirements of the analysis.

Insulation inside the containment is controlled by insulation specifications as well as plant drawings. Any deviations from these specifications are subject to the design review process described above. Signs and labels are controlled by procedure that requires engineering approval for the use of labels inside containment.

The licensee controls on materials within containment described above provide assurance that maintenance and modification activities will not introduce unevaluated materials into containment. As an added measure of assurance, maintenance activities in containment are controlled through work order screening before each outage to determine whether debris sources may be introduced to containment.

NRC Staff Conclusion

For this review area, the licensee provided information such that the NRC staff has reasonable assurance that the subject review area has been addressed conservatively or prototypically. Therefore, the NRC staff concludes that the debris source term evaluation for Millstone 2 and 3 is acceptable. The NRC staff considers this item closed for GL 2004-02.

12.0 SCREEN MODIFICATION PACKAGE

The objective of the screen modification package section is to provide a basic description of the information provided for the sump screen modification.

NRC Staff Review

The NRC staff review for Millstone 2 and 3 is based on documentation provided by the licensee through February 29, 2008.

Millstone 2

The licensee provided a basic description of the major features of the new sump strainers. They are AECL design with two containment floor mounted horizontal headers of six modules extending about 40 feet from the existing original ECCS suction pipes in two directions. The modules extend underneath the steam generator compartments and refueling cavity. The collection headers are rectangular boxes with vertically mounted fins on the sides of corrugated plates with 1/16-inch perforations. The fins are 10 inches apart center to center. The total strainer surface area is approximately 6,120 ft^2. At the maximum assumed flow rate of 6,800 gpm, the average strainer surface approach velocity would be about 0.0025 feet per second. The design structural differential pressure is 15 psid. The top of the strainers will be submerged about 9 inches for a SBLOCA and about 22 inches for a LBLOCA. The new strainer is constructed of stainless-steel materials.

The licensee provided the revised content guide specified information. The description provided a basic understanding of the new sump strainers location, configuration, and construction details.

Millstone 3

The screen modification description provided the revised content guide specified information. The description provided a basic understanding of the new sump strainers location, configuration and construction, and identified associated modifications necessitated by the sump/strainer modification.

The licensee provided a basic description of the major features of the new sump strainers. They are AECL design with 17 modules mounted over the existing sump pit at the containment east side wall. Each module has five vertical fins of corrugated plate spaced 8 inches center to center with 1/16-inch perforations. The total surface area is 5,041 ft^2. The modules are mounted 7 inches above the containment floor. Water enters the fins and drops down into a channel that leads over/down into the sump pit. Minimum containment pool level at the start of recirculation is

8 inches above the strainer. The design strainer differential pressure is 10 psid. There are no vents that connect the strainer to the containment atmosphere once it is submerged.

The new strainer construction is of stainless-steel materials.

The only associated modifications required for the new sump strainers was the removal of the original vertical screen surrounding the sump pit and perforated divider plate between train suctions in the sump pit.

NRC Staff Conclusion

For the screen modification package review area, the licensee provided screen location, configuration, and construction information such that the NRC staff has confidence in the design of the strainer. Therefore, the NRC staff concludes that the screen modification package information provided for Millstone 2 and 3 is acceptable. The NRC staff considers this item closed for GL 2004-02.

13.0 SUMP STRUCTURAL ANALYSIS

The objective of the sump structural analysis section is to verify the structural adequacy of the sump strainer including seismic loads and loads due to differential pressure, missiles, and jet forces.

NRC Staff Review

The NRC staff review is based on documentation provided by the licensee through February 29, 2008.

Millstone 2

The licensee stated that the maximum stress induced in the components associated with the replacement sump strainer were shown to be within the allowable stress limits of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. The licensee stated that the Analysis System (ANSYS) computer program was used to create finite element analysis models for the qualification of the replacement sump strainers. The models were subjected to dead weight, live loads, seismic loads, suction pressure loads, hydrodynamic loads, and thermal loads. These loads are consistent with the guidance of NEI 04-07. The licensee stated that the governing load combination consisted of dead weight, suction pressure, safe shutdown earthquake (SSE) inertia, and SSE hydrodynamic loads. The seismic responses were stated to be combined using the absolute sum method in accordance with the plant’s FSAR requirements. The maximum stresses which were extracted from this governing load combination were then compared with the allowable stress values from the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. All reported values were within the allowable limits.

Concerning the potential for pipe whip, jet impingement, and missile loading, the licensee stated that the new strainer was positioned outside the range of any postulated pipe breaks. Also, potential missile loadings were determined to not be credible based on physical separation and/or the physical aspects of the strainer. The leak-before-break analysis of the updated FSAR (UFSAR) was relied upon to eliminate potential dynamic effects due to the rupture of reactor coolant piping.

The information provided by the licensee shows that the sump structural evaluation contains inherent conservatism by complying with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. All stresses are lower than the allowable limits. The licensee has provided sufficient information to show that a level of conservatism exists, and the intent of the Revised Content Guide for GL 2004-02 Item 2(d)(vii) has been met.

An audit report associated with Millstone 2 focused exclusively on the potential for dynamic effects due to a high-energy line break. While on-site, the staff informed the licensee that additional detail was needed to demonstrate a clear basis for the acceptability of the new strainer design in terms of pipe whip and jet impingement (see Audit Open Item 11). This was primarily due to the larger dimensions of the new sump strainers in comparison to the existing design. The February 29, 2008, submittal provides information from the latest revision of the subject pipe whip and jet impingement calculation. In this calculation, it is concluded that the replacement strainer assembly is outside the potential range of postulated pipe breaks. The leak-before-break analysis method, consistent with General Design Criteria 4 (GDC 4), is also employed from the UFSAR to eliminate potential dynamic effects associated with the reactor coolant piping. The transmittal of this information is sufficient to close Open Item 11. The response also answers RAI 33.

<u>Millstone 3</u>

The staff review of Section 3k concluded that the licensee adequately addressed the information requested by the Revised Content Guide for GL 2004-02 Item 2(d)(vii). The licensee's submittal stated that the maximum stress induced in the components associated with the replacement sump strainer were shown to be within the allowable stress limits of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF.

The licensee stated that the ANSYS computer program was used to create finite element analysis models for the qualification of the replacement sump strainers. The models were subjected to dead weight, live loads, seismic loads, suction pressure loads, hydrodynamic loads, and thermal loads. These loads are consistent with the guidance in NEI 04-07 and accepted by the staff SE. The licensee stated that the governing load combination consisted of dead weight, suction pressure, SSE inertia, and SSE hydrodynamic loads. The seismic responses were stated to be combined using the square root of the sum of the squares method in accordance with the plant's FSAR requirements. The maximum stresses, which were extracted from this governing load combination, were then compared with the allowable stress values from the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. All reported values were within the allowable limits.

The licensee evaluated the potential for pipe whip, jet impingement, and missile loading. The licensee stated that the new strainer was positioned outside the range of any postulated pipe breaks and that missile loadings are not credible based on physical separation from the strainer. The leak-before-break analysis of the UFSAR was not relied upon to eliminate potential dynamic effects due to the rupture of reactor coolant piping. The licensee also noted in Section 3J, Screen Modification Package, that the strainer was completely passive, and no active approach (including backflushing) was credited in the design of the strainer.

The information provided by the licensee demonstrates that the sump structural evaluation contains inherent conservatism by complying with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. All interaction ratios were stated to be within the allowable limits with the maximum value being 0.95. The licensee has provided sufficient information to show that a level of conservatism exists, and the intent of the Revised Content Guide for GL 2004-02 Item 2(d)(vii) has been met.

NRC Staff Conclusion

For this review area, the licensee provided information such that the NRC staff has reasonable assurance that the subject review area has overall been addressed conservatively or prototypically. Therefore, the NRC staff concludes that the sump structural analysis evaluation for Millstone 2 and 3 is acceptable. The NRC staff considers this item closed for GL 2004-02.

14.0 UPSTREAM EFFECTS

The objective of the upstream effects assessment is to evaluate the flow paths upstream of the containment sump for holdup of inventory, which could reduce flow to the sump.

NRC Staff Review

The NRC staff review for Millstone 2 is based on documentation provided by the licensee through February 29, 2008. The NRC staff review for Millstone 3 is based upon documentation provided by the licensee through March 12, 2009.

Millstone 2

The staff audit report of Millstone 2 corrective actions report (ML072290550) describes the licensee's upstream effects evaluation as summarized below.

The Millstone 2 containment is designed such that potential for upstream holdup and flow choke points is very limited. The upper and intermediate containment floors contain large areas of grating, and numerous flow-paths to allow water to flow to the basement elevation. The reactor cavity shield has drains that are open to the containment basement during reactor operation. The reactor cavity seal has large access covers that are removed for normal reactor operations. These also provide an alternate drain path for the refueling cavity. To minimize water hold-up in the refueling canal, screen enclosures were provided over the refuel pool drain to prevent blockage by debris. The screen enclosures consisted of 18-inch boxes with a 2-inch hole mesh. The base of the enclosures is made of two triangular pieces of ½-inch thick stainless-steel plate. This arrangement provides a ½-inch gap at the bottom of the enclosure to allow water to drain to the floor level without leaving a 2-inch deep pool in the canal. The licensee concluded that the clogging of the drains during a LOCA is not a concern because the locations of the breaks are remote from the refueling cavities and therefore, it is not likely for large pieces of insulation debris to fall into the reactor cavity due to a break in one of the coolant loops.

As discussed in the NPSH section, the licensee revised its water holdup and minimum water level calculations in a conservative manner to address NRC staff comments.

Millstone 3

The licensee initially provided the basic information requested by the NRC in GL 2004-02 and the associated content guide. However, the initial submittal accounting of water holdup mechanisms was incomplete. RAIs in this area are discussed below.

The licensee stated that postulated worst-case break locations include breaks in cubicles 1 and 4. These locations were selected due to their close proximity to the sump. A break in cubicle 2 was also evaluated because it resulted in the worst-case debris load. Walkdowns were performed to identify flowpaths and chokepoints that could prevent water from reaching the sump pool and revealed no significant choke points between the postulated break locations and the containment sump. In addition, a detailed calculation was performed to identify flow paths from spray headers to the containment floor and to quantify the flow rate of each path as a function of time. This calculation determined the total mass of water in the sump and is used in determining minimum water level when the RSS pumps start.

The licensee stated that much of the floor space in containment is grating which will not holdup significant volumes of water. Water is assumed to be held up in the following areas:

- Loop cubicles at elevation 3’8” due to kick plate curbs.
- Liquid films on solid floors and heat sink surfaces.
- The refueling cavity is assumed to collect quench spray and recirculation spray flow. Drainage of the refueling cavity is not credited.
- The reactor cavity and incore instrumentation tunnel is assumed to fill with water prior to overflowing and reaching the containment floor.
- Water vapor is assumed to be held up in the containment atmosphere.
- Curbs on floors with the potential for water holdup were assumed to retain water.

After its initial review, the NRC staff determined that two questions should be asked in the upstream effects area. First, the NRC staff requested the licensee to discuss whether the calculation accounted for water holdup due to steam in the containment atmosphere or to justify any assumptions that result in the omission of this holdup from the water level calculation. The licensee stated that the holdup of water as steam was included in the calculation. The staff found this clarification acceptable.

The second question asked by the staff was related to holdup in the refueling canal. The question requested details of how the calculation accounted for water holdup in the refueling canal or justification of the omission of the hold-up. The licensee responded that it does credit spillover from the refueling canal and reactor cavity. However, the refueling volume is assumed to be 99 percent full prior to RSS pumps start. This maximizes holdup in this volume. The reactor cavity and instrument tunnel are assumed to be filled prior to water spilling to the containment floor. The staff found this response acceptable.

<u>NRC Staff Conclusion</u>

For this review area, the licensee provided information such that the NRC staff has reasonable assurance that the subject review area has overall been addressed conservatively or prototypically. Therefore, the NRC staff concludes that the upstream effects evaluation for Millstone 2 and 3 is acceptable. The NRC staff considers this item closed for GL 2004-02.

15.0 DOWNSTREAM EFFECTS - COMPONENTS AND SYSTEMS

The objective of the downstream effects, components and systems (ex-vessel) section is to evaluate the effects of debris carried downstream of the containment sump screen on the function of the ECCS and CSS in terms of potential wear of components and blockage of flow streams.

NRC Staff Review

The NRC staff review for Millstone 2 is based on documentation provided by the licensee through March 13, 2009. The NRC staff review for Millstone 3 is based on documentation provided by the licensee through December 18, 2008.

Millstone 2

The licensee used a methodology for downstream effects analysis that is consistent with PWROG TR-WCAP-16406-P, Revision 1, “Evaluation of Downstream Sump Debris Effects in Support of GSI-191,” and the NRC limitations and conditions described in the NRC SE regarding the performance of this analysis, dated December 20, 2007 (ML073520295).
The combined supplemental responses provide a detailed description of the downstream ex-vessel evaluation. The licensee concluded that the ECCS and CSS components subjected to debris-laden sump fluid would perform their safety related function for the duration of their required mission time. Because the licensee followed the NRC accepted guidance in WCAP-16406-P, Revision 1, and the NRC limitations and conditions described in the NRC SE of that document, the NRC staff finds that the downstream effects of recirculated debris-laden sump fluid has been adequately addressed for this plant.

Millstone 3

The licensee’s initial response did not include a final evaluation of Downstream Effects – Components and Systems. On December 18, 2008, the licensee provided an updated response. This response contained the information requested by the NRC in the content guide. The licensee used a methodology for downstream effects analysis that is consistent with PWROG TR, WCAP-16406-P, Revision 1, “Evaluation of Downstream Sump Debris Effects in Support of GSI-191,” and the NRC limitations and conditions described in the NRC SE regarding the performance of this analysis, dated December 20, 2007 (ML073520295).

The licensee stated that it evaluated wear in the SI, charging, and RSS pumps, manually throttled valves, motor operated valves, orifices, and heat exchangers. The licensee also evaluated the wear effects on the performance of these components. The evaluation found that the hydraulic performance and seal performance of the pumps would not be adversely affected. The licensee also found that wear of heat exchangers, orifices, and containment spray nozzles would not inhibit their performance.

The licensee also evaluated downstream instrumentation, including temperature indicators, pressure indicators, and flow indicators for potential blockage. The blockage evaluation found that instrumentation was installed in such a manner that it was not susceptible to blockage.

NRC Staff Conclusion

For the ex-vessel downstream effects review area, the licensee has provided sufficient information such that the NRC staff has reasonable assurance that the subject review area has been addressed conservatively or prototypically. Therefore, the NRC staff concludes that the licensee's evaluation of this area for Millstone 2 and 3 is acceptable. Based on the information provided by the licensee, the NRC staff considers this area closed for GL 2004-02.

16.0 DOWNSTREAM EFFECTS - FUEL AND VESSEL

The objective of the downstream effects, fuel, and vessel section is to evaluate the effects that debris carried downstream of the containment sump screen and into the reactor vessel has on LTCC.

NRC Staff Review

The NRC staff review for Millstone 2 and 3 is based on documentation provided by the licensee through May 8, 2023.

Millstone 2

In its supplemental response (ML080650561), the licensee stated that the evaluations completed to date followed the guidance in WCAP-16793-NP, Rev 0. Because later strainer penetration evaluations found that the fiber limits associated with WCAP-16793-NP could be exceeded, the licensee reperformed its in-vessel evaluation based on WCAP-17788 and the associated NRC staff guidance.

In a May 15, 2013, letter, the licensee stated that testing had been conducted to determine the amount of fiber that could penetrate the strainer. Testing was conducted for North Anna, Surry, and Millstone 2. The licensee stated that 99.7 percent of the fiber is captured by the strainer on its first pass. The testing used grab samples to determine the penetration amounts. The NRC staff has recommended that penetration testing use full flow filtering to capture fiber that bypasses the strainer.

By letter dated August 13, 2015 (ML15232A026) the licensee informed the NRC that Millstone 2 would demonstrate compliance with the WCAP-17788-P reactor vessel debris acceptance criteria, instead of relying on WCAP-16793-NP to evaluate reactor vessel debris.

In its May 27, 2021, response, the licensee stated that the total fibrous debris quantity from fiberglass and latent fiber that could potentially reach the sump strainer was conservatively calculated to be approximately 5,429 lbm. The licensee stated that because the strainer fiber bypass testing performed by AECL for the strainer design installed at Millstone 2 used a "grab sample" method there is no data for the quantity of bypassed fiber as the debris bed is forming and therefore, cumulative fiber bypass fractions could not be determined. However, because other plants performed this testing and determined cumulative fiber bypass fractions for various debris bed thicknesses, the licensee was able to perform an evaluation to develop an engineering basis for the use of cumulative fiber bypass data from other plants to apply to the AECL strainer at Millstone 2.

The licensee applied Point Beach test results to the Millstone 2 strainer. The licensee assumed that all of the strainer sacrificial area would be available for formation of the fibrous debris bed to minimize the thickness of the calculated theoretical debris bed, which results in larger cumulative bypass fractions. The licensee considered the slightly larger strainer perforation size for Point Beach (0.066”) to Millstone 2 (0.0625”) and determined it has a conservative influence on cumulative bypass fractions when applying the Point Beach test results to Millstone 2.

The licensee discussed conservatisms used in determining the cumulative bypass fraction for Millstone 2 and provided a table listing the critical parameter comparison for sump strainer bypass testing. An evaluation of these parameters determined that the Point Beach testing could be applied to Millstone 2. The cumulative bypass fraction at the theoretical debris bed thickness of 2.655” was calculated as 1.9 percent. Because the Millstone 2 approach velocity of 0.00248 ft/s is bounded by the Point Beach approach velocity of 0.0027 ft/s, no additional correction factors were applied to the 1.9 percent cumulative fiber bypass. The NRC staff drafted a question regarding the methodology used to calculate the fiber amount that would penetrate the strainer. The NRC staff did not understand how the bed thickness correction factor was implemented to scale Point Beach test results to the Millstone 2 plant conditions. The licensee responded on November 7, 2022. Based on the response, the NRC staff developed an additional clarification question on the issue that requested the licensee to justify the method used. The licensee responded in letter dated May 8, 2023 (ML23128A162). The response recalculated the amount of debris penetrating the strainer and the amounts of debris that can arrive at the core. The response also required the licensee to revise the assumptions and methods used to evaluate the in-vessel debris load acceptability because the in-vessel debris amounts increased. The following discussion reflects the final debris values.

Millstone 2 is a Combustion Engineering (CE) PWR design that uses Framatome (AREVA) CE 14 HTP fuel assemblies. The licensee stated that the proprietary total in-vessel (core inlet and heated core) fibrous debris limit contained in Section 6.5 of WCAP-17788-P, Volume 1, Revision 1, “Comprehensive Analysis and Test Program for GSI-191 Closure” (ML20010F181), applies to Millstone 2.

The licensee calculated the fibrous debris amounts using the methodology from WCAP-17788-P. The licensee calculated that the maximum amount of fiber to potentially reach the reactor vessel is 20.67 grams per fuel assembly (g/FA), which is less than the proprietary in-vessel fibrous debris limit in Section 6.5 of WCAP-17788, Volume 1. Based on the NRC staff questions (cited above) regarding the bed thickness correction factor the licensee recalculated the amounts of fiber reaching the reactor vessel. The updated amount of debris was calculated to be 89.9 g/FA. This is greater than the core inlet debris limit, but less than the total core debris limit.

The licensee stated that chemical effects would not occur before 24 hours and referred to Test Group 38 of WCAP-17788-P. The NRC staff reviewed the Test Group 38 data in Volume 5 of WCAP-17788-P and confirmed that the autoclave test conditions represented the projected post-LOCA chemical environment for Millstone 2. The staff also confirmed the test results demonstrate chemical precipitates did not form during the 24-hour duration test. The licensee stated that it performs injection realignment to mitigate the potential for boric acid precipitation no later than 10 hours, which is less than 24 hours. The licensee stated that the chemical effects timing (t_{chem}) of 24 hours is greater than the t_{block} of 333 minutes, which is the earliest time that complete fuel inlet blockage can occur while not compromising LTCC (WCAP-17788-P) nor inhibiting LTCC.

The licensee also stated that its sump switchover (SSO) time is 33 minutes, which is greater than the SSO time assumed in the analysis.

The licensee stated that Millstone 2 rated thermal power (2,754 megawatts thermal (MWt)) is less than the applicable analyzed thermal power in the WCAP (3,458 MWt).

The licensee stated that the Millstone 2 specific alternate flow path (AFP) resistance (WCAP-17788, Volume 4, RAI Table 4.3-7), is less than the proprietary analyzed AFP resistance (Table 6-3 of WCAP-17788, Volume 4) and is therefore bounded by the resistance applied to the AFP analysis.

The licensee stated that because the range of ECCS recirculation flow rates at Millstone 2 is not bounded by the range analyzed as part of the WCAP-17788 thermohydraulic analysis, the unbounded flow rates are dispositioned pursuant to guidance in PWROG-16703-P, Revision 0, “TSTF-567 Implementation Guidance, Evaluation of In-Vessel Debris Effects, Submittal Template for Final Response to Generic Letter 2004-02 and FSAR Changes,” February 2020. This is an industry guidance document that was not provided to the NRC. However, the licensee’s response has been reviewed by the NRC staff as discussed below.

The licensee stated that the minimum plant-specific ECCS recirculation flow rate is less than the minimum analyzed ECCS flow rate used to develop K_{max} in WCAP-17788-P, Rev. 1. Kmax is the maximum resistance due to fibrous and particulate debris bed formation that can be tolerated at the core inlet. Because the maximum ECCS flow rate at Millstone 2 creates the most limiting case, which has margin to the WCAP-17788 core inlet fiber limit, the unbounded minimum ECCS flow rate is acceptable because it does not create the limiting fiber load at the core inlet, and K_{max} is valid for the limiting fiber load case.

The licensee stated that the maximum plant-specific ECCS recirculation flow rate is higher than the analyzed ECCS flow rate in WCAP-17788-P. Based on discussion in RAI 4.26 of WCAP-17788-P, a higher than analyzed flow rate will be conservative with respect to K_{max} (i.e., more water being delivered to the core thereby increasing the tolerance for debris accumulation and K_{max}/fiber limit) but non-conservative with respect to debris arrival timing. The maximum ECCS flow rate has been evaluated as acceptable because the limiting calculated cumulative fiber load remains below the fiber limit. The licensee stated that there is also margin available in multiple parameters that ensure the calculated fiber load is conservative, including thermal power, SSO time, and AFP resistance. Finally, AFPs, while they are not credited in the Millstone 2 thermohydraulic analysis, would exist and provide core cooling.

Because the updated fiber amounts for the core increased to a value greater than the core inlet debris limit, the licensee cited NRC staff guidance that the debris bed is realistically expected to collect non-uniformly. As a result, the licensee concluded that the amount of debris required to completely block the core inlet would be greater than that assumed in the analyses and the updated amount of debris would not result in blockage at the core inlet, and the current LTCC analyses remain applicable.

The NRC staff reviewed the licensee’s in-vessel evaluation and found that it followed staff guidance. All of the key parameters were bounded by the WCAP-17788-P analyses except for the minimum and maximum ECCS flow rates and the core inlet fiber limit. For chemical effects, the licensee demonstrated that precipitation would not occur before 24 hours and confirmed that this is longer than the analyzed t_{block} time for the plant and also longer than the hot-leg switchover (HLSO) time for the plant. Therefore, AFPs are available for coolant to reach the core should the core inlet be blocked by chemical precipitates combined with other debris.

For the issue of the Millstone 2 flow rates being outside the range of analyzed values, the licensee stated that both the minimum flow rate of 2.6 gpm and the maximum flow rate of 18.9 gpm were outside the analyzed range. The higher flow rate was used to calculate a conservative in-vessel fibrous debris load. The licensee stated that this high flow would result in lower head losses at the core inlet as was demonstrated in the WCAP-17788-P analyses. The staff agrees that a higher ECCS flow rate will result in a higher in-vessel fiber value and that it will result in reduced head loss at the core inlet. Therefore, the higher flow rate is acceptable.

The lower flow rate of 2.6 gpm is below the analyzed range and lower flow rates resulted in higher debris head losses in the analysis. However, the debris amount for the low flow case would be less than the higher flow case. The NRC staff agrees that the lower flow case would result in less debris at the core inlet. Since the total debris amount that could reach the core is less than the total core fiber limit, the NRC staff concluded that LTCC would be maintained.

During its review of the April 15, 2021, submittal, the NRC staff also identified additional information required to ensure that the in-vessel evaluation was performed acceptably. In letter dated September 9, 2022 (ML22251A129), the NRC staff provided its questions to the licensee. The staff requested additional information regarding the licensee's method for correcting the penetration values for fiber bed thickness (discussed above), clarifications on the references used, clarification regarding the t_{block} value reported in the submittal, and confirmation that the CSS would start and run at the flow rate assumed for the period during which penetration is calculated. The NRC staff also requested additional information on the flow rates used in the in-vessel analysis.

The licensee responded in letter dated November 7, 2022 (ML22312A443). The NRC concluded that the references used were correct. The NRC staff also found that the CSS would operate, as assumed, for the entire time that penetration of the strainer is calculated. The licensee also confirmed that the t_{block} time for the plant is 333 minutes and that the table contained a typographical error. The licensee stated that the correct time was used for the calculations. The licensee also stated that the maximum ECCS flow rate of 4,100 gpm is based on the maximum ECCS flow rate for all scenarios. The NRC staff found that these responses are consistent with staff guidance and the licensee's evaluation. The issues regarding the fiber penetration methodology are discussed above.

The NRC staff recognizes that the licensee is citing NRC staff guidance that allows crediting non-uniform debris bed distribution at the core inlet combined with maintaining the total debris amount reaching the reactor less than the total core fiber limit as the primary criteria for maintaining adequate LTCC. The licensee also provided information that indicates that the AFPs will be available to provide cooling to the core should the core inlet become blocked. This is an acceptable evaluation methodology to ensure LTCC will not be compromised by debris entering the core.

Millstone 3

In a 2008 supplemental response (ML080650561), the licensee stated that the evaluations completed to date followed the guidance in WCAP-16793-NP, Rev 0. Because later strainer penetration evaluations found that the fiber limits associated with WCAP-16793-NP could be exceeded, the licensee reperformed its in-vessel evaluation based on WCAP-17788 and associated NRC staff guidance.

In a May 15, 2013 (ML13141A277), letter, the licensee stated that testing had been conducted to determine the amount of fiber that could penetrate the strainer. Testing was conducted for North Anna, Surry, and Millstone 2, but the licensee stated that the bypass amounts for Millstone 3 were expected to be similar. The licensee stated that 99.7 percent of the fiber is captured by the strainer on its first pass. The testing used grab samples to determine the penetration amounts. The staff has recommended that penetration testing use full flow filtering to capture fiber that bypasses the strainer.

The test method provided no data for the quantity of bypassed fiber as the debris bed is forming and cumulative fiber bypass fractions could not be determined. However, the licensee stated other plants performed strainer bypass testing with downstream continuous in-line filters that were able to determine cumulative fiber bypass fractions for various debris bed thicknesses. Therefore, the licensee performed an evaluation to develop an engineering basis for the use of cumulative fiber bypass data from other plants to apply to the AECL strainer installed at Millstone 3. The licensee also stated that Millstone 3 would reference a new topical report that is being prepared to justify greater in-vessel fiber limits for US PWRs instead of referring to WCAP-16793-NP-A.

By letter dated August 13, 2015 (ML15232A026) the licensee informed the NRC that Millstone 3 would demonstrate compliance with the WCAP-17788-P reactor vessel debris acceptance criteria, instead of relying on WCAP-16793-NP to evaluate reactor vessel debris.

In its April 15, 2021, response, the licensee stated that based on the debris generation and transport analyses, it determined the types of quantities of fibrous debris that could be transported to the strainers. The fibrous debris sources considered include fiberglass and latent fiber. The licensee calculated the total fibrous debris quantity from these sources that could potentially reach the sump strainer as approximately 2,053 lbm. Because the grab sample test method was not accepted by the NRC, the licensee used Point Beach and Vogtle fiber penetration test results to calculate a cumulative strainer bypass fraction for Millstone 3.

The licensee stated that the geometry for the PCI furnished Point Beach strainer was compared with the AECL furnished Millstone 3 strainer and assessed to be conceptually equivalent in its hydraulic performance characteristics. Regarding sacrificial area for the Millstone 3 strainer, the licensee assumed that the entire strainer area would be available for formation of the fibrous debris beds, which would minimize the thickness of the calculated theoretical debris bed, thereby resulting in larger cumulative bypass fractions.

The licensee selected cases that result in the maximum design flow rates to provide the highest approach velocity. The licensee stated that the strainer perforation size for Point Beach (0.066”) being slightly larger than the Millstone 3 (0.0625”) has a conservative influence on cumulative bypass fractions when applying Point Beach test results to Millstone 3.

The licensee provided a list of conservatisms applied when determining cumulative bypass fractions for Millstone 3 and provided a table providing the critical parameter comparison for sump strainer bypass testing. An evaluation of these parameters determined that the Point Beach testing could be applied to Millstone 3. The cumulative bypass fraction at the theoretical debris bed thickness of 2.037” was calculated as 2.3 percent. Since the Millstone 3 strainer approach velocity is higher than the Point Beach test velocity, the licensee applied a correction factor to scale the Point Beach data to the higher velocity. The correction factor was derived based on the Vogtle plant tests for bypass fractions at various velocities. The licensee stated that this methodology is based on the premise that the impact of approach velocity on the filtering efficiency of a debris bed is not strongly dependent on the specific strainer design. Correcting for the velocity difference, the licensee determined that the Millstone 3 overall bypass fraction would be 3.5 percent. The NRC staff drafted a question regarding the methodology used to calculate the fiber amount that would penetrate the strainer. In particular, the NRC staff did not understand how the bed thickness correction factor was implemented to scale Point Beach test results to the Millstone 3 plant conditions. The licensee responded to the NRC staff questions in a November 7, 2022, RAI response. Based on the response, the NRC staff developed an additional clarification question on the issue. The licensee responded in letter dated May 8, 2023 (ML23128A162). The response recalculated the amount of debris penetrating the strainer and the amounts of debris that can arrive at the core. The response also required the licensee to revise the assumptions and methods used to evaluate the in-vessel debris load acceptability because the in-vessel debris amounts increased. The following discussion reflects the final debris values.

The licensee stated that Millstone 3 is a Westinghouse 4-loop PWR with an upflow barrel/baffle configuration. Millstone 3 uses Westinghouse 17x17 Robust Fuel Assembly 2 (RFA-2) fuel. The licensee stated that the proprietary total in-vessel fibrous debris limits contained in WCAP-17788-P, Revision 1, Volume 1, “Comprehensive Analysis and Test Program for GSI-191 Closure” (ML20010F181) apply to Millstone 3. The licensee determined the quantity of fibrous debris calculated to arrive at the reactor vessel following the method described in WCAP-17788-P, Rev. 1, Volume 1, Section 6.5.

The licensee calculated the maximum amount of fiber to potentially reach the reactor vessel as 9.6 grams per fuel assembly (g/FA), which is less than the proprietary in-vessel fibrous debris limit provided in Section 6.5 of WCAP-17788-P, Rev. 1, Volume 1 and less than the core inlet fiber threshold limit for Westinghouse fuel provided in the WCAP. In its RAI response dated November 7, 2022, the licensee provided updated flow rates that caused the core inlet fiber value to increase to 12.74 g/FA. This value is also below the acceptance limit. The total fiber reaching the vessel was calculated as 15.42 g/FA. Based on the NRC questions (cited above) regarding the bed thickness correction factor the licensee recalculated the amounts of fiber reaching the vessel. The updated amount of debris was calculated as 56.7 g/FA. This is greater than the core inlet debris limit, but less than the total core debris limit.

The licensee confirmed that the earliest SSO time was 33 minutes. The licensee stated that chemical precipitation timing is dependent on the plant buffer, sump pool pH, volume and temperature, and debris types and quantities. The licensee identified Test Group 35 from WCAP-17788-P as representative of Millstone 3 and determined the predicted chemical effects timing (t_{chem}) is 24 hours. The NRC staff reviewed the Test Group 35 data contained in Volume 5 of WCAP-17788-P and confirmed that the autoclave environment is representative of the projected post-LOCA environment. The staff also confirmed that the data shows no chemical precipitation occurred during the 24-hour test duration. Millstone 3 performs injection realignment to mitigate the potential for boric acid precipitation no later than 5 hours, which is less than 24 hours. Based on WCAP-17788-P, Rev. 1, Volume 1, Table 6-1, t_{block} for Millstone 3 is 143 minutes, which is the earliest time that complete fuel inlet blockage can occur while not compromising LTCC (WCAP-17788-P) nor inhibiting LTCC. The earliest time of chemical precipitation for Millstone 3 is 24 hours, which is greater than the applicable t_{block} of 143 minutes.

The licensee stated that because Millstone 3 analyzed thermal power (3.658 megawatts thermal (MWt)) is greater than the WCAP-17788-P analyzed value, it is necessary to demonstrate the decay heat for Millstone 3 at the time of SSO is less than the decay heat in the WCAP thermal hydraulic analysis at the time of SSO. The decay heat at the time of SSO (20 minutes) in the

WCAP-17788-P thermohydraulic analysis is 87.4 MWt, which is based on the 1971 ANS Infinite Standard plus 20 percent uncertainty. Since the earliest time of SSO for Millstone 3 is 33 minutes, this will result in additional time for the core heat to decay and thus result in a lower normalized core power than that resulting at 20 minutes. The licensee stated that the Millstone 3 plant-specific post-LOCA decay heat fractions is 0.0216 at 30 minutes and includes a 20 percent uncertainty and is conservative for a SSO time of 33 minutes. This is less than the WCAP-17788 decay heat at SSO. Therefore, the licensee stated that the WCAP-17788-P thermohydraulic analysis and fuel limits are still bounding for Millstone 3.

The licensee stated that the specific AFP resistance for Millstone 3 (Table RAI-4.2-24 of WCAP-17788-P, Rev. 1, Volume 4) is less than the analyzed value (Table 6-1 of WCAP-17788-P, Rev. 1, Volume 4) and therefore, the Millstone 3 resistance is bounded by the resistance applied to the AFP analysis.

The licensee stated that the AFP analysis for Westinghouse upflow plants analyzed a range of ECCS recirculation flow rates from 8-40 gpm/FA (Table 6-1 of WCAP-17788-P, Rev. 1, Volume 4). The Millstone 3 flow rate corresponding to the worst-case GSI-191 hot-leg break scenario is 9 gpm/FA, which is within the range considered in the AFP analysis. The minimum plant-specific ECCS flow rate of 5 gpm/FA is less than the minimum analyzed ECCS flow rate used to develop K_{max}. K_{max} is the maximum resistance due to fibrous and particulate debris bed formation that can be tolerated at the core inlet. Since debris bed resistance increases as ECCS flow rate decreases, an unbounded low flow has potential to cause the K_{max} used in the calculation to be non-conservative. The licensee stated that the maximum ECCS flow rate at Millstone 3 creates the most limiting fiber case, which has significant margin to the WCAP-17788 total core fiber limit. In its RAI response of November 7, 2022, the licensee changed the limiting flow rate from 9.0 to 20.1 gpm. This resulted in a slight increase in the amount of fiber reaching the core compared to the previous calculations. However, this amount was superseded by the recalculated amount of 56.7 g/FA provided above. The NRC determined that the discussion above continues to apply to the Millstone 3 plant-specific condition based on the significant margin to the total core fiber limit which is the acceptance criterion applied to Millstone 3 as discussed below.

Based on the above information in this section, the licensee stated that Millstone 3 is bounded by the key parameters, and the WCAP-17788 methods and results are applicable.

Because the updated fiber amounts for the core increased to a value greater than the core inlet debris limit, the licensee cited NRC staff guidance that the debris bed is realistically expected to collect non-uniformly. As a result, the licensee concluded that the amount of debris required to completely block the core inlet would be greater than that assumed in the analyses and the updated amount of debris would not result in blockage at the core inlet, and the current LTCC analyses remain applicable.

The NRC staff reviewed the licensee's information and found that it had generally followed staff guidance in the in-vessel evaluation. All of the key parameters were bounded by the WCAP-17788-P analyses except for the plant rated thermal power and the minimum ECCS flow rate.

For chemical effects, plant-specific testing demonstrated that precipitation would not occur before 24 hours and stated that 24 hours is longer than the analyzed t_{block} time for the plant and also longer than the hot-leg switchover (HLSO) time for the plant. Therefore, AFPs are available for coolant to reach the core should the core inlet be blocked by chemical precipitates combined with other debris.

In its evaluation of the Millstone 3 unbounded rated thermal power, the licensee stated that although its rated thermal power is higher than the analyzed value, the earliest SSO time for Millstone 3 is 33 minutes which is later than the WCAP-17788-P analyzed time of 20 minutes. The additional time before SSO ensures the plant decay heat is lower than the analyzed value when debris may be introduced into the reactor. Therefore, the NRC staff agrees that the higher thermal power is acceptable for Millstone 3.

For the issue of the Millstone 3 minimum flow rate being lower than the range of analyzed values, the licensee stated that the expected flow rate of 9 gpm, which was later revised to 20.1 gpm, is within the analyzed range and that it represents the highest in-vessel debris loading case. The minimum flow rate of 5 gpm is below the analyzed range and lower flow rates resulted in higher debris head loss in the analysis. However, the debris amount for the low flow case would be less than the higher flow case. The NRC staff agrees that the lower flow case would result in less debris at the core inlet. The NRC staff also noted that the calculated amount of debris that can reach the core for Millstone 3 has significant margin to the total core limit.

During its review of the April 15, 2021, submittal, the NRC staff identified additional information required to ensure that the in-vessel evaluation was performed acceptably. In letter dated September 9, 2022 (ML22251A129), the NRC staff requested additional information regarding the licensee’s method for correcting the penetration values for fiber bed thickness, clarifications on the references used, and confirmation that the RSS would start and run in the spray mode at the flow rate assumed for the period during which penetration is calculated.

The licensee responded in letter dated November 7, 2022 (ML22312A442). The NRC staff concluded that the references used were correct. The NRC staff also found that the RSS would operate, as assumed, for the entire time that penetration of the strainer is calculated. The NRC staff found that these responses are consistent with staff guidance and the licensee’s evaluation. The issues regarding the fiber penetration methodology are discussed above.

The NRC staff recognizes that the licensee is citing NRC staff guidance that allows crediting non-uniform debris bed distribution at the core inlet combined with maintaining the total debris amount reaching the reactor less than the total core fiber limit as the primary criteria for maintaining adequate LTCC. The licensee also provided information that indicates that the AFPs will be available to provide cooling to the core should the core inlet become blocked. This is an acceptable evaluation methodology to ensure LTCC will not be compromised by debris entering the core.

<u>NRC Staff Conclusions</u>

For the in-vessel downstream effects review area, the licensee has provided sufficient information such that the NRC staff has reasonable assurance that the subject review area has been addressed conservatively or prototypically. Therefore, the NRC staff concludes that the licensee’s evaluation of this area for Millstone 2 and 3 is acceptable. Based on the information provided by the licensee, the NRC staff considers this area closed for GL 2004-02.

17.0 CHEMICAL EFFECTS

The objective of the chemical effects section is to evaluate the effect that chemical precipitates have on strainer head loss. The evaluation of chemical effects on the reactor vessel is contained in section 16.0 above.

NRC Staff Review - Millstone 2

The NRC staff review is based on the following documentation provided by the licensee (ML083650005 and ML102010413), as well as a May 15, 2013, updated plant-specific path and schedule for resolution of GL 2004-02 supplement (ML13141A277). The reference documents used for this review include the March 31, 2008, NRC staff SE of WCAP-16530-NP-A, “WCAP-16530 NP-A, Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191” (ML081150379) and the March 28, 2008, review guidance “NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Plant-Specific Chemical Effects Evaluations” (ML080380214).

The licensee uses tri-sodium phosphate (TSP) for post-LOCA sump pool pH control in Millstone 2. The long-term minimum value of the sump pool pH is 7.1. The licensee replaced the original ECCS sump strainer that had an area of approximately 110 ft^2 with a new AECL stacked disk strainer design with an area of 6,120 ft^2.

The licensee’s plant-specific debris generation and transport analysis determined that the debris sources for Millstone 2 included NUKON, Claremont fiberglass, mineral fiber, Transco encapsulated mineral wool, Transco RMI, latent debris (fiber, particulate), coatings, and foreign materials (tags and labels).

The licensee performed chemical effects testing for Millstone 2 at the AECL Chalk River facility. Chemical effects testing consisted of bench-top testing and a reduced scale debris bed test (Test Rig 89).

Background

The licensee provided initial responses to GL 2004-02 in letters dated March 4, September 1 and November 29, 2005, December 19, 2007, and February 29, 2008 (ML050630559, ML052500378, ML053330540, ML090860438, and ML080650561). The initial NRC staff RAIs are not discussed in this review, since at the time, the licensee was still performing testing for the chemical effects evaluation.

The NRC staff visited AECL’s Chalk River Facility from May 5-9, 2008, to observe integrated chemical effects head loss testing for the Dominion plants. The purpose of the NRC staff trip was to observe chemical effects head loss testing in the recently constructed multi-loop test rig and to discuss the bench-top testing results used to justify the chemical effects testing procedure. For more details, refer to the February 9, 2009, NRC trip report (ML090400786). The licensee’s testing program for chemical effects consisted of protocols developed by AECL. The AECL protocol was different from the approaches used by many other test vendors (testing approaches were typically based on the WCAP-16530-NP-A methodology). The NRC staff held follow on phone calls with the licensee and AECL (June 25 and August 8, 2008) to discuss results of the multi-loop test rig and critical aspects of the testing program, such as the behavior of the measured head loss when significant quantities of chemicals were added to the test loops.

In addition to observing testing, the NRC staff also visited Dominion's Innsbrook facility from November 12-14, 2008, to perform a chemical effects audit for another Dominion plant (North Anna). The main purpose of the audit was to allow the NRC staff to perform a detailed review of the chemical effects evaluation. The February 9, 2009, audit report (ML090400742), includes detailed descriptions and evaluations of the head loss testing facilities, a detailed review of head loss testing results, review of conservative assumptions incorporated into the sump strainer performance analysis, and an assessment of the post-LOCA NPSH margins.

On December 18, 2008, the licensee provided updated information about the chemical effects testing for Millstone 2 (ML083650005). The letter provided additional detail regarding the licensee's bench-top and reduced scale testing (Rig 89) used for the chemical assessment.

Testing

The AECL methodology for the evaluation of chemical effects impact on sump strainer performance consisted of three elements: an assessment of potential chemical precipitates, bench-top testing, and reduced-scale testing.

Assessment of Potential Chemical Precipitates

The AECL methodology consisted of an initial assessment of potential chemical precipitates, including determination of reactive material amounts present in the containment sump pool, pH and temperature profiles in containment, and a review of existing test and scientific literature data. The chemical precipitates of concern for Millstone 2 are aluminum hydroxide or oxyhydroxide and calcium phosphate.

The chemical effects testing methodology developed by AECL was different from the approaches used by other vendors, which were based on the methodology described in WCAP-16530-NP-A. The WCAP methodology involves determining the chemical precipitate load (based on plant-specific data), preparing the calculated amount of precipitates, and adding the pre-mixed precipitates to the test loop after a debris bed is formed on the test strainer.

The AECL methodology used a similar approach as outlined in WCAP-16530-NP-A to calculate the mass of the chemical element released. However, rather than using the WCAP-16530-NP-A release rate equation, AECL used data from the Integrated Chemical effects test (ICET) and other sources (e.g., scientific literature) to develop a semi-empirical release rate equation. To model the aluminum release rate, the pH and temperature dependencies of the corrosion rates were evaluated separately.

The licensee compared the results of the application of the AECL release rate model to the WCAP-16530-NP-A model results using Millstone 2 aluminum inventories and found they predicted a very similar 30-day release of aluminum. The total aluminum mass released to the sump water (plant-specific dissolved aluminum concentration) was calculated using the AECL aluminum release rate equation along with Millstone 2 aluminum inventories based on exposure category, sump and spray water pH, and sump/spray water temperatures for specific time intervals following a LOCA.

Bench-top Testing

The licensee conducted bench-top testing to gain understanding of the chemistry expected in the strainer chemical effects testing (reduced-scale testing). The bench-top testing for Millstone 2 consisted of:

- Calcium dissolution and calcium phosphate precipitation testing to determine the amount of calcium released from concrete and other insulations. The licensee used this data to determine the amount of calcium to be added to the test tank for the reduced scale tests.

- Precipitation testing of aluminum hydroxide to determine aluminum solubility under the worst-case conditions expected in the post-LOCA sump water. The predicted 30-day aluminum release was used for the determination of aluminum precipitation.

- Determination of dependence of walnut shell (paint surrogate) properties on chemistry. Tests were carried out as part of bench-top testing to determine whether exposure to chemicals would dissolve or alter the walnut shell particulate (paint surrogate) used in the debris head loss test to simulate qualified and unqualified coatings (anticipated to be transported to the strainer post-LOCA).

Reduced Scale Testing

Reduced scale testing for Millstone 2 was conducted to evaluate the effects that chemical precipitates have on head loss. The licensee performed chemical effects testing using a newly constructed multi-loop test facility (Test Rig 89) at the Chalk River Ontario Laboratories (run by AECL). The Rig 89 facility was initially filled with buffered borated water. After a debris bed was formed, chemicals based on Millstone 2 plant conditions and pH predictions, were added to the test loop.

The effects of calcium and aluminum precipitate formation on strainer debris bed head loss were evaluated by adding calcium chloride and sodium aluminate to the test in the already established chemical environment of TSP buffer, boric acid, and lithium hydroxide at a pH of 7.0. The amount of calcium chloride to be added was calculated from a concrete dissolution rate obtained from bench-top testing in boric acid solution at a pH of 7.0 (without TSP). The bench-top test showed that concrete coupons exposed for 30-days to a solution of boric acid and TSP at a pH of 7.0 had no significant mass change. The amount of sodium aluminate to be added was calculated from the AECL aluminum corrosion rate model. The calculations of calcium and aluminum release ignored the inhibitory effect of TSP. Multiple test programs (ICET and testing for WCAP-16785-NP, among others) have shown that phosphate from the TSP buffer will significantly reduce aluminum release by phosphate inhibition of aluminum corrosion.

After establishing a stable baseline head loss (from non-chemical debris) across the test strainer section, small batches of chemicals were added to maintain the dissolved aluminum and calcium concentrations in the Rig 89 test loop equal to the predicted plant-specific aluminum and calcium concentrations. Chemical additions were made in small batches throughout the test to simulate the gradual corrosion of aluminum and degradation of concrete and insulation fiber. Rig 89 test loop chemical additions were scaled according to the post-LOCA pool concentration (instead of scaling the chemical precipitate mass to the strainer area). Precipitation of aluminum or calcium during the test could result in a non-conservative dissolved chemical concentration in the test loop. Therefore, if dissolved chemical measurements indicated that aluminum or calcium precipitation had occurred during the test, additional aluminum or calcium was added to the test loop. These additions continued up to an amount representing the maximum amount of chemical precipitate mass per strainer area predicted for the plant. The total head loss measured in the Rig 89 test loop represents the plant-specific, integrated head loss across the sump strainer for plant debris and chemical effects.

Final Review – Millstone 2

Following the review of the chemical effects evaluation detailed in the licensee's December 18, 2008, response update letter (ML083650005), the NRC staff identified that additional information was needed to determine whether the testing was performed in an acceptable manner. Therefore, on February 4, 2010, the NRC staff issued additional RAIs (RAIs 12-15) (ML100070068). On June 8, 2010, the licensee responded to the RAIs (ML102010413). The RAIs and the licensee's responses to the RAIs are detailed below:

For RAI 12, the NRC staff noted that the Millstone 2 calcium dissolution test at a pH of 7.0 resulted in a 30-day calcium concentration of 126 mg/L. The licensee's December 18, 2008, letter stated that the pH 7.0 case (without TSP present) was used to determine the concentration of calcium in the Rig 89 test. However, the calcium concentration used for Rig 89 testing was 40.4 mg/L. The NRC staff requested justification of 40.4 mg/L as a representative value in the Rig 89 testing when the dissolution testing conducted with scaled quantities of concrete resulted in a calcium concentration of 126 mg/L.

In response to RAI 12, the licensee stated that the concrete surface area to volume ratio used in bench-top testing was approximately 3 times greater than the actual plant-specific concrete surface area to volume ratio. This explains the apparent discrepancy between the calcium concentration from bench-top testing and the calcium concentration used in Rig 89 head loss testing. In addition, the licensee stated that a WCAP-16530 spreadsheet calculation for the actual plant debris amounts predicted less dissolved calcium than that used during Rig 89 testing.

The NRC staff evaluated the licensee's response to RAI 12 and found it acceptable since the dissolved calcium concentration in the Rig 89 ICET was appropriate for the revised plant-specific amount of calcium.

For RAI 13, the NRC staff noted that in Attachment 1, Table 0-2, of the licensee's December 18, 2008, letter, the calcium concentration for time infinity is shown as 117 mg/L for the pH 7.0 case. The NRC staff requested the licensee to explain why this concentration for time infinity is appropriate, given the 30-day bench-top test calcium concentration at a pH of 7.0 was 126 mg/L.

In response to RAI 13, the licensee stated that the "infinity" value for calcium concentration was determined using a Table Curve 2D data fitting program to fit all experimental data. The licensee also stated that the difference between the two values in question was within the experimental error involved with measuring the dissolved calcium concentrations during bench-top testing.

The NRC staff evaluated the licensee response to RAI 13 and found it acceptable since the relative difference between the two concentration values is not statistically significant and the licensee subsequently revised the calcium calculation that reduced the concrete surface area to volume ratio by a factor of 3 (see licensee's response to RAI 12 discussed above).

For RAI 14, the NRC staff noted that the licensee's testing was performed at 104 °F, which is well below early post-loss-of-coolant accident pool temperatures. The solubility of calcium phosphate (hydroxyapatite) decreases as the temperature increases. The NRC staff requested the licensee to discuss whether more calcium phosphate precipitate would have formed in the Rig 89 tests if they had been performed at a higher temperature. If more calcium phosphate precipitate would be expected at a higher temperature, when the NPSH margin is more limiting, the staff requested the licensee to justify why the overall Rig 89 test results provide for an adequate evaluation of chemical effects.

In response to RAI 14, the licensee stated that a potentially decreased calcium phosphate solubility at higher temperatures does not significantly impact the Millstone 2 test results due to significant conservatisms built into the testing program. Specific examples of conservatisms provided in the licensee's RAI response included:

1. There is no significant source of calcium in the Millstone 2 containment. The only potential calcium sources for the containment are uncoated concrete and dislodged fibrous insulation. By design, there is no uncoated concrete in the containment. For the Rig 89 testing, a total of 1,325 ft^2 of concrete is assumed to be uncoated in containment. Of that total, 825 ft^2 is considered uncoated due to the break jet impacting coated walls. The remaining 500 ft^2 is margin for damaged concrete coating in containment. No cal-sil insulation exists within the loop rooms and cal-sil insulation is not a part of any debris load since it is a small amount outside the loop rooms and is steel-jacketed and not subject to dissolution. Calcium releases due to degradation of other dislodged insulation are included in the total calcium release used in the testing. Based on the conservative estimates of existing uncoated concrete, there will be significantly less calcium released into the containment sump water than was tested.

2. In the bench-top testing, TSP inhibited calcium release from uncoated concrete. Identical tests were run in the bench-top testing to determine the effect of TSP on calcium concentration. Both sets of tests were conducted with scaled amounts of concrete and fibrous insulation. In one set of tests, no TSP was used. In an identical set of tests, a representative concentration of TSP was established in the test water. At a pH of 7.0, the expected calcium concentration in containment in the absence of TSP is 40.4 mg/L based on tests. In the presence of TSP, the 30-day calcium concentration in bench-top testing is less than 10 mg/L. In the absence of TSP, the concrete coupons in the test showed significant dissolution. In the tests with TSP present, concrete coupons showed no evidence of dissolution and experienced less than a 1 percent loss in mass. For conservatism, the results from calcium dissolution tests without TSP present were used to determine the amount of calcium to add to the Rig 89 (chemical effects) test tank.

3. Concrete used in testing was not safety-related concrete and thus was more likely to degrade in the bench-top testing than the safety-related concrete installed in containment.

4. Concrete dissolution data for a pH of 7.0 was used in the testing to determine the amount of calcium released and the amount of calcium used in chemical effects testing. The Millstone 2 containment water is expected to have a pH above 8.0 following the LOCA resulting in much less calcium release. Concrete dissolution is much lower at

higher pH values as was demonstrated in the response to RAI 12. Expected long-term calcium concentration at a pH of 8.0 (without TSP) is 23.7 mg/L as compared to the expected (and tested) calcium concentration at a pH of 7.0 (without TSP) of 40.4 mg/L. Thus, the calcium concentration in containment is likely to be as much as 40 percent lower than the tested value due only to the pH in containment.

5. A total of 15 calcium additions were made to the Millstone 2 Rig 89 test. These additions had a minimal impact on head loss although TSP was present in the test tank at the expected concentration in containment. This TSP concentration far exceeded the amount needed to precipitate all of the available calcium in the test.

The NRC staff evaluated the licensee’s response to RAI 14 and finds it acceptable since the conservatisms identified by the licensee are sufficient to offset the potential for additional precipitates from potential retrograde solubility behavior of calcium phosphate.

For RAI 15, the NRC staff requested the licensee to compare the total amount of aluminum that is predicted to be released by the AECL model with that predicted by the WCAP-16530 base model (i.e., no refinements for silicate or phosphate inhibition) and to discuss any significant differences between the plant-specific predictions for the two methods, including the acceptability of these differences.

In response to RAI 15, the licensee compared the AECL aluminum release predictions to the WCAP-16530 aluminum release predictions. The licensee stated that the WCAP model predicts more aluminum release at moderate pH values (7.0 to 9.5) while the AECL model predicts greater aluminum release at high pH values. Both models, however, predicted a conservative 30-day aluminum release for ICET 5, which had a similar pH value to Millstone 2. The licensee also stated that the last five additions of aluminum did not produce a head loss response during integrated head loss testing.

The NRC staff evaluated the licensee’s response to RAI 15 and finds it acceptable since the AECL model conservatively predicted the amount of aluminum released during the ICET 5 test. In addition, the plant-specific head loss test showed little to no effect for the approximately last 60 percent of aluminum added to the test.

The NRC staff reviewed the licensee’s chemical effects evaluation for Millstone 2 and finds it acceptable since the chemical effects testing performed by the licensee has provided an adequate simulation of post-LOCA chemical effects. The licensee made several conservative assumptions when evaluating the precipitation of aluminum. For example, for calculating the aluminum release, the licensee ignored the inhibition of corrosion by phosphate from the TSP and calculated the aluminum release at a pH of 8.0. From a solubility perspective, the licensee tested at a pH of 7.0 to minimize aluminum solubility and promote precipitation. The NRC staff finds that all outstanding RAIs have been addressed.

NRC Staff Review – Millstone 3

Although the overall NRC staff review of chemical effects is based on documentation provided by the licensee through April 15, 2021, the licensee's analysis of chemical effects related to the sump strainer has not changed since the December 20, 2010, RAI response. The Millstone 3 sump strainer chemical effects evaluation was performed by AECL. The NRC staff visited the AECL Chalk River facility from May 5-9, 2008, to observe integrated chemical effects testing for the Dominion plants. The purpose of the NRC staff trip was to observe chemical effects head loss testing in the recently constructed multi-loop test rig and to discuss the bench-top testing results used to justify the chemical effects testing procedure. For more details, refer to the February 9, 2009, NRC trip report (ML090400786).

Assessment of Potential Chemical Precipitates:

The chemical effects testing methodology developed by AECL was different from the approaches used by most other vendors, which were based on the methodology described in WCAP-16530-NP-A. This WCAP methodology involves determining the chemical precipitate load, preparing the calculated amount of precipitates, and adding the pre-mixed precipitates to the test loop after a plant-specific debris bed is formed and stabilized on the test strainer.

The AECL methodology used a similar approach as outlined in WCAP-16530-NP-A to calculate the mass of aluminum released. However, rather than using the WCAP-16530-NP-A release equation, AECL used the data from WCAP-16530-NP-A and other sources (e.g., available experimental data including ICETs and data from scientific literature) to develop a semi-empirical release equation. In order to model the aluminum release rate, the pH and temperature dependencies of the corrosion rates were evaluated separately.

Since the Millstone 3 post-LOCA buffer is trisodium phosphate (TSP), the chemical precipitates considered in the licensee's evaluation included aluminum hydroxide/oxyhydroxide and calcium phosphate. The NRC staff review of chemical effects included questions related to the calcium and aluminum release predicted by the AECL methodology. The licensee's September 16, 2010, RAI response letter addressed the staff questions. For calcium, the projected post-LOCA term calcium concentration at a pH greater than 8 is expected to be as much as 30 percent lower than the Millstone 3 Rig 89 chemical effects test. The licensee's response also discussed the AECL aluminum release equation and compared the predicted AECL aluminum release to the predicted WCAP-16530-NP aluminum release for Millstone 3. The WCAP predicted more aluminum release than the AECL method for the Millstone specific conditions. Both models conservatively predicted the aluminum released during ICET No. 5 at the University of New Mexico. ICET Test No. 5 had a long-term pH that was most representative of the Millstone 3 projected post-LOCA pH.

Testing:

The AECL methodology for the chemical effects on sump strainer performance consisted of three elements:

1. An assessment of potential chemical precipitates, including determination of reactive material amounts present in the containment sump pool, pH and temperature profiles in containment, and a review of existing test and scientific literature data. The Millstone 3 chemical precipitates of concern were aluminum oxyhydroxide and calcium phosphate.

2. Bench-top testing to demonstrate that the solubility behavior of potential precipitates determined from literature is reproducible under plant conditions and to confirm that precipitates can be produced, if required, for reduced-scale testing.

3. Reduced-scale testing to determine that any chemical products formed in the post-LOCA containment sump pool would not produce unacceptable head loss across the ECCS strainer debris bed. These tests verified that adequate NPSH was available to support the operation of the low-head SI and recirculation spray pumps during the post-LOCA recirculation mode.

The initial goal of the bench-top test program was to show that no precipitation would occur in the projected plant-specific post-LOCA environment. Since the bench-top test results indicated precipitation could occur, the licensee concluded that additional chemical effects testing was needed. Therefore, the licensee performed integrated chemical effects head loss testing at the AECL Chalk River facility.

In particular, a multi-loop test facility identified as Rig 89 was fabricated to perform these tests. Rig 89 integrated chemical effects tests were performed in a simulated post-LOCA pool environment containing representative amounts of boron and scaled amounts of plant-specific debris. TSP was added to the test loop to adjust the pH to 7. The test loop temperature was held constant at 104 °F (40 °C). Plant-specific particulate debris quantities and the quantity of fiber needed to develop a debris bed were added in increments to the test loop. After a stable baseline head loss was established across the test strainer section, sodium aluminate and dissolved calcium were added in small batches with the objective of having the dissolved aluminum concentration and calcium concentrations in the Rig 89 test loop be representative of the predicted plant-specific calculated dissolved aluminum concentration.

The NRC staff has reviewed all documentation submitted from the licensee with regards to chemical effects. Although the AECL method calculated aluminum release is less than that calculated with WCAP-16530-NP method, the plant-specific aluminum release was calculated at a pH greater than 8, and the Rig 89 multi-loop rig tests were performed at a pH of 7. The AECL method also did not attempt to credit aluminum passivation from phosphate. These conservatisms result in the staff having confidence that the aluminum released into the test loop, was either conservative or representative of the projected post-LOCA plant-specific environment. In addition, the concrete in the Millstone 3 containment is coated, and the dissolved calcium added to the Rig 89 test loop was conservative to the projected post-LOCA calcium. In addition, although Millstone 3 plant-specific autoclave testing in WCAP-17788-P was performed to support the evaluation of chemical effects in the reactor vessel, these results were consistent with the AECL Rig 89 test results that demonstrated there were no early chemical effects. After reviewing all the licensee’s chemical effects related submittals and considering the conservatisms incorporated into the licensee’s sump performance analysis, the NRC staff concludes that the Millstone 3 chemical effects evaluation bounds any uncertainties associated with the Rig 89 testing and is therefore acceptable.

<u>NRC Staff Conclusion</u>

For the chemical effects review area, the licensee provided sufficient information such that the NRC staff has reasonable assurance that the subject review area has overall been addressed conservatively or prototypically. Therefore, the NRC staff concludes that the licensee’s evaluation of this area for Millstone 2 and 3 is acceptable. Based on the information provided by the licensee, the NRC staff considers this area closed for GL 2004-02.

18.0 LICENSING BASIS

The objective of the licensing basis section is to provide information regarding any changes to the plant licensing basis due to the changes associated with GL 2004-02.

The licensee stated that a license amendment was approved and implemented for an administrative change in Technical Specifications Section 4.5.2.j to replace text in a surveillance requirement of “screen and trash rack” with the word “strainer.” Amendment No. 300 was approved in NRC letter dated September 18, 2007 (ML072290132). This change was implemented within 30 days of receipt of the amendment.

The licensee stated that changes to the FSAR will be made consistent with the description of the modifications and analyses described in the supplemental response. The licensee stated that no other changes to plant licensing bases were identified.

NRC Staff Conclusion

For this review area the licensee has provided information, such that the NRC staff has reasonable assurance that the subject review area has overall been addressed conservatively or prototypically. Based on the licensee’s commitment, the NRC staff has confidence that the licensee will make the appropriate changes to the Millstone 2 and 3 FSAR in accordance with 10 CFR 50.71(e), that will reflect the changes to the licensing basis as a result of corrective actions made to address GL 2004-02. Therefore, the NRC considers this item closed for GL 2004-02.

19.0 CONCLUSION

The NRC staff performed a thorough review of the licensee’s responses and RAI supplements to GL 2004-02. The NRC staff conclusions are documented above. Based on the above evaluations the NRC staff finds the licensee has provided adequate information as requested by GL 2004-02.

The stated purpose of GL 2004-02 was focused on demonstrating compliance with 10 CFR 50.46. Specifically, the GL requested addressees to perform an evaluation of the ECCS and CSS recirculation and, if necessary, take additional action to ensure system function in light of the potential for debris to adversely affect LTCC. The NRC staff finds that the information provided by the licensee demonstrates that debris will not inhibit the ECCS or CSS performance following a postulated LOCA. Therefore, the ability of the systems to perform their safety functions, to assure adequate LTCC following a DBA, as required by 10 CFR 50.46, has been demonstrated.

Therefore, the NRC staff finds that the licensee’s responses to GL 2004-02 are adequate and considers GL 2004-02 closed for Millstone 2 and 3.

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SUBJECT: MILLSTONE POWER STATION, UNITS 2 AND 3 – CLOSEOUT OF GENERIC LETTER 2004-02, “POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING DESIGN BASIS ACCIDENTS AT PRESSURIZED-WATER REACTORS” (EPID L-2017-LRC-0000) DATED JULY 26, 2023

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