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

April 25, 2024

Site Vice President
Entergy Operations, Inc.
Waterford Steam Electric Station, Unit 3
17265 River Road
Killona, LA 70057-3093

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 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 Site Vice President:

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 16, 2013 (ML13137A133), Entergy Operations, Inc. (the licensee) stated that they will pursue Option 2 (deterministic) for the closure of GSI-191 and GL 2004-02 for Waterford Steam Electric Station, Unit 3 (Waterford 3).

On July 23, 2019 (Package 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 NRC staff issued a technical evaluation report on in-vessel downstream effects (IVDEs) (ML19178A252 (publicly available), and ML19073A044 (not publicly available, proprietary information). Following the closure of the GSI-191, the NRC staff also issued review guidance for IVDEs to support review of the GL 2004-02 responses (ML19228A011).

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 Waterford 3.

Enclosed is the summary of the NRC staff's review. If you have any questions, please contact me at 301-415-8378 or via email at Jason.Drake@nrc.gov.

Sincerely,

/RA/

Jason J. Drake, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosure:
NRC Staff Review of GL 2004-02 for
Waterford Steam Electric Station, Unit 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

ENTERGY OPERATIONS, INC.

FOR WATERFORD STEAM ELECTRIC STATION UNIT 3

DOCKET NO. 50-382

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 (LTC) 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 quantity 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 LTC 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 Generic Letter (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 NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," describing 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 (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 (g/FA).

- 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: Used the South Texas Project pilot approach.

- 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 16, 2013 (ML13137A133), Entergy Operations, Inc. (the licensee) stated that they will pursue Option 2 (deterministic) for the closure of GSI-191 and GL 2004-02 for Waterford Steam Electric Station, Unit 3 (Waterford 3).

On July 23, 2019 (Package 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 (publicly available) and ML19073A044 (not publicly available, proprietary information)). 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 of In-Vessel Downstream Effects Supporting Review of Generic Letter 2004-02 Responses” (ML19228A011).

The following is a list of documentation provided by the licensee for Waterford 3 in response to GL 2004-02:

GL 2004-02 CORRESPONDENCE		
DOCUMENT DATE	ACCESSION NUMBER	DOCUMENT
March 3, 2005	ML050660261	Initial Response to GL
May 27, 2005	ML051530547	1st NRC RAI Revision 1
June 2, 2005	ML051660501	1st NRC RAI Revision 2
July 28, 2005	ML052140281	Licensee Response to RAI
September 16, 2005	ML052630441	Supplemental Information
December 19, 2005	ML053550197	Supplemental Information
February 9, 2006	ML060390386	2nd NRC RAI
February 29, 2008	Package ML080650747	Licensee Response to RAI
October 23, 2008	ML083020551	Supplemental Information
September 22, 2009	ML092510124	3rd NRC RAI
June 1, 2010	ML101520476	3rd NRC RAI Revision 2
June 17, 2010	Package ML101730180	Licensee Response to RAI
November 23, 2010	ML103340117	Supplemental Information
May 16, 2013	ML13137A133	Closure Option
March 30, 2023	ML23089A247	Supplemental Information

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

The NRC staff 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 Waterford 3, in support of the resolution of GL 2004-02:

- Evaluation using the guidance of NEI 04-07.
- Downstream effects evaluation using the TR-WCAP-16406-P-A, Revision 1, methodology.
- Containment walkdowns using the guidance of NEI 02-01, "Condition Assessment Guidelines: Debris Sources Inside PWR Containments," dated April 2002 (ML021490241).
- The modification and maintenance programs have been enhanced to assist in controlling the introduction of potential debris sources within containment.
- Installation of new strainer modules with about 3,700 square feet (ft^2) area in place of the original strainers that were about 200 ft^2 in area.
- Replacement of a significant amount of fibrous insulation with reflective metal insulation (RMI) to reduce the potential amount of fiber that could become debris.
- ECCS sump strainer performance was confirmed by performing a prototype chemical precipitates head loss test.

Based on the information provided by the licensee, the NRC staff considers this area 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 zone representing the volume of space affected by the ruptured piping.

NRC Staff Review

The NRC staff review is based on documentation provided by the licensee for Waterford 3, through October 23, 2008.

The licensee provided the information specified in the content guide and stated that the GR/SE discrete approach for selecting potential break locations for evaluation was employed. The break locations selected were chosen to maximize the amounts and types of debris generated. Breaks were postulated near large, insulated equipment; steam generators (SGs); reactor coolant pumps (RCPs); the pressurizer; and near walls and the floor for coating debris. Break locations selected for evaluation were also chosen to maximize potential transport of debris to the sump strainer.

The licensee's October 23, 2008, submittal, describes nine postulated break locations that were evaluated for debris generation and strainer blockage potential. The licensee determined that a break in the 42-inch hot legs result in debris generation amounts greater than other locations and therefore selected this as the limiting break location.

The licensee stated that secondary (main feedwater and main steam) piping breaks were not considered since the associated accident analyses do not credit safety injection or containment spray (CS) recirculation.

NRC Staff Conclusion

The NRC staff concluded that the licensee addressed the break selection area satisfactorily. The staff conclusion is based on the licensee's use of approved guidance and conservative assumptions used in performing the break selection analysis. The NRC staff also noted that an audit was conducted at Waterford 3 (see Audit Report dated January 28, 2008 (ML080140318)), and that there were no audit open items that resulted in the break selection area. There was no information provided in subsequent correspondence that changes the staff conclusions in this area. Therefore, the break selection area was addressed acceptably.

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 is based on documentation provided by the licensee through November 23, 2010.

The Waterford 3 debris generation evaluation was reviewed during the onsite audit and with one exception found to be consistent with the NRC staff SE approved methodology and acceptable. The exception was inadequate justification for the assumed 2D ZOI for Waterford 3's metallic encapsulated fibrous insulation (MEI). This issue became an audit open item.

The licensee's resolution to this open item was to increase the ZOI for MEI from 2D (pipe diameters) to 4D and reevaluate the potential debris loads. Since the NRC staff accepted the 4D ZOI for San Onofre mineral wool encapsulated in the same Transco system (see May 16, 2007, Audit Report (ML070950240)), the staff accepted the 4D ZOI for the Waterford 3 Transco encapsulated fiberglass. The licensee later changed this to a 7D ZOI but changed the debris characteristics from 100 percent fine to 20 percent fines and 80 percent small pieces. The staff finds that the ZOI is acceptable since it is conservative compared to the originally approved ZOI. The issue regarding the change in debris characteristics is discussed in section 5.0, "Debris Characteristic," of this review.

The other licensee assumptions for ZOIs included 7D for stainless steel jacketed Nukon, 17D for unjacketed Nukon, 28.6D for Min-K and Microtherm, and 2D for Transco RMI. Except for the 7D ZOI for jacketed Nukon, these ZOIs are per the NRC staff guidance in the GR/SE.

During the onsite audit, a 17D ZOI was assumed for both jacketed and unjacketed Nukon. Subsequent to the audit, the licensee cited WCAP-16710-P, "Jet Impingement Testing to Determine the ZOI of Min-K and Nukon® Insulation for Wolf Creek and Callaway Nuclear

Operating Plants" (ML100670191; not publicly available, proprietary information), as justification to reduce the ZOI to 7D. With the adoption of a smaller ZOI, the licensee also assumed a different size distribution for the damaged Nukon. The tests used to develop the reduced ZOI were conducted for the jacketing/banding systems used at those specific plants. The 7D ZOI for jacketed Nukon has not been accepted by the NRC, even for the plants specified in the test report. In addition, the NRC staff noted that the licensee did not compare its jacketing system to the one tested so that there was no basis for application of the test results to Waterford 3. In response to NRC RAIs, the licensee changed its assumption for ZOI size for jacketed Nukon to the staff accepted value of 17D. Related to this issue, the licensee replaced a significant amount of jacketed Nukon with RMI when the site's SGs were replaced. This modification resulted in a significant reduction in the amount of potential fibrous debris at Waterford 3.

Ultimately, the licensee adopted ZOIs for materials that meet NRC guidance. The sole exception to this is the 4D ZOI for the MEI. Because the MEI system had not been evaluated when the guidance was developed, the NRC staff evaluated the ZOI and found it appropriate for the plant-specific conditions.

NRC Staff Conclusion

The NRC staff finds that the licensee has appropriately addressed the area of debris generation/ZOI. The licensee used NRC-approved guidance in its evaluation of this area or provided information that adequately justified the ZOI used for the insulating system not covered by NRC guidance. Therefore, the NRC staff considers that this area has been evaluated adequately and can be closed.

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 is based on documentation provided by the licensee through November 23, 2010.

For the Nukon debris size distributions, the licensee originally assumed 8 percent, 25 percent, 32 percent, and 35 percent for fines, small pieces, large pieces, and intact pieces, respectively, for its unjacketed Nukon and 25 percent, 75 percent, 0 percent, and 0 percent, respectively, for its stainless steel jacketed Nukon. The distribution for unjacketed Nukon is in accordance with the NRC SE-approved guidance. The licensee assumed that the distribution for the jacketed Nukon would be either fines or small pieces and then considered 25 percent of that to be fines, which was reasonable for a 7D ZOI.

The licensee originally assumed 100 percent fines for the MEI, Min-K, and Microtherm debris. For the Transco RMI, 75 percent of the debris was assumed to be small pieces and 25 percent large pieces, which is reasonable. After the NRC staff asked several RAIs, the licensee changed the ZOI sizes for jacketed Nukon and MEI, and changed the size distributions for the fibrous debris.

Ultimately, the licensee adopted a size distribution for jacketed and unjacketed Nukon that is in accordance with, or conservative with respect to the NRC SE-approved guidance.

For MEI, the licensee increased the ZOI to 7D, but changed the debris generation from 100 percent fines to 20 percent fines and 80 percent small pieces. The NRC staff requested that the licensee justify the assumption of 20 percent fines and 80 percent smalls. The ZOI increase from 4D to 7D significantly increases the amount of debris that could be generated by a given break while the change from 100 percent fines to 20 percent fines and 80 percent smalls is in the non-conservative direction. However, the debris characterization must be evaluated with respect to the transport evaluation and head loss testing. There were no changes to the transport evaluation due to the fiber size change. The NRC staff reviewed the licensee's response to the RAIs and determined that the 20/80 split was adequately conservative considering the plant-specific conditions, testing conditions, increased ZOI, and effects on the transport evaluation.

The qualified and unqualified zinc coatings debris were assumed to be particulates and treated as 10-micron particles. The unqualified epoxy was assumed to fail as chips.

The licensee's latent debris assessment resulted in a total of 250 pound-mass (lbm). This included both particulate and fiber terms. Per the assessment, 15 percent of the latent debris were considered fibrous. The assessment determined the miscellaneous debris (signs and labels) to be 151 ft^2. The miscellaneous debris were assumed, without any reduction, as strainer sacrificial area.

For Microtherm and Min-K insulation, a conservative size distribution of 100 percent fines was used. Fines are the constituent part of the insulation and are considered 100 percent transportable.

For Transco RMI insulation with a ZOI of 2D, a size distribution of 75 percent small fines and 25 percent large pieces was used.

The licensee assumed that the bulk density of Nukon insulation and the MEI was 2.4 lbm/cubic feet (ft^3). This bulk density used in the sump strainer performance testing was also 2.4 lbm/ft^3. The MEI is Owens-Corning Thermal Insulating Wool Type II. Based on manufacturer data for this insulation the bulk density is 2.4 lbm/ft^3. Nukon has the same density.

As shown in table 3-2 of the guidance in NEI 04-07, the bulk density of Min-K insulation is 8 to 16 lbm/ft^3. The bulk density of the Min-K insulation installed at Waterford 3 is 13 lbm/ft^3. This compares to a bulk density of 14.5 lbm/ft^3 for the insulation used in the sump strainer performance testing.

As shown in table 3-2 of the guidance in NEI 04-07, the bulk density of Microtherm insulation is 5 to 12 lbm/ft^3. This compares to a bulk density of 14.5 lbm/ft^3 of the insulation used in the sump strainer performance testing.

The material density and specific surface area were only used for preliminary analytically determined head loss values across a debris laden sump screen using the correlation given in NUREG/CR-6224, "Parametric Study of the Potential for BWR [Boiling-Water Reactor] ECCS Strainer Blockage Due to LOCA Generated Debris," dated October 1995 (ML083290498). Since the head loss across the installed sump screen is determined via testing, these values are not used in the design basis for Waterford 3. Therefore, these values were not provided as part of this response.

NRC Staff Conclusion

The licensee ultimately performed the debris characteristics evaluation in accordance with the associated NRC staff SE-guidance in the areas where it was available. For the MEI debris characterization, no guidance was available, so the licensee provided an evaluation that the NRC staff found to be adequate. Therefore, the licensee has adequately addressed the area of debris characteristics.

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. The guidance documents used for the review include the revised content guide dated November 2007, the GR/SE, and NEI 02-01.

NRC Staff Review

The NRC staff review is based on documentation provided by the licensee through October 23, 2008. Information provided subsequent to this date had no effect on the NRC review of this area.

The licensee provided estimates of the mass of latent fiber and particulate. A brief description of the methodology used to estimate the quantity of the latent debris was provided. Sufficient detail was presented and was consistent with the information reviewed during the GSI-191 plant audit. An estimate of 239 lbm of latent debris was calculated by the licensee using sampling and extrapolation techniques consistent with NRC-approved guidance. The licensee conservatively assumed 250 lbm of latent debris in its evaluation.

The licensee provided quantitative estimates of the area of tapes, tags, signs and stickers. The licensee provided sufficient detail, largely reviewed during the GSI-191 plant audit, such that the NRC staff finds that the results of the analysis are reasonable and acceptable. The licensee performed a foreign material walkdown and tabulated the areas of labels, tags, stickers, placards and other materials that could block area on the strainer. The licensee determined that there was 151 ft^2 of signs and labels and assigned this value as strainer sacrificial area.#

NRC Staff Conclusion

Since the licensee used conservative assumptions and methods approved by the NRC to determine latent debris amounts, the NRC staff considers this area to be adequately addressed. The latent debris area is considered 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 Waterford 3 is based on documentation provided by the licensee through November 23, 2010.

The NRC staff reviewed the licensee's initial supplemental response in the transport area and concluded that the discussion in the Transport section was consistent in a number of areas with the approved guidance and incorporated conservatism. However, inconsistencies were noted with the approved guidance, significant changes in the analysis were noted from the audit report that were not fully explained, and several open items from the audit report were not fully addressed. The staff proposed RAIs to address the items not fully addressed. There were two rounds of RAIs required for the staff to attain full understanding and acceptance of the transport evaluation for Waterford 3.

The licensee's initial supplemental response summarized the debris transport methodology including the following for each phase:

- Blowdown and washdown: All debris directly to containment pool floor.
- Pool fill: No inactive pool credit with debris holdup; debris transported from break toward perimeter of containment.
- Recirculation: 100 percent transport of fines; more limiting transport metrics for near sump breaks to account for turbulence effects.

The licensee originally used a 90 percent erosion assumption, but later changed to a 10 percent erosion assumption. The NRC staff requested additional information to justify the use of the 10 percent erosion metric. The licensee responded that it was a participant in a proprietary test program that tested the amount of erosion that could occur from low density fiberglass (LDFG). The licensee stated that this testing was reviewed and approved by the NRC for plants that could demonstrate that it applied to its plant-specific conditions. The licensee also stated that the test conditions bound the Waterford 3 plant conditions. The licensee applied the erosion fraction to both Nukon and MEI, which are both forms of LDFG.

The NRC staff did not clearly understand the transport fraction for unqualified coatings and requested additional information about the coating's characteristics and transport metrics. The licensee changed its assumptions to those consistent with NRC-approved guidance.

The NRC staff requested that the licensee clarify the amount of each size category of MEI debris calculated to reach the strainer. The licensee provided the clarification. The licensee also stated that all debris included in the head loss testing was prepared as fines. The use of 100 percent fines in the test makes this issue less important, but the overall debris amount must be accurate.

The NRC staff requested the licensee to provide additional information regarding the velocity and turbulence in the containment pool for the limiting head loss test and for break S7. In a related RAI, the NRC staff requested that the licensee provide information regarding the distance of debris addition from the test strainer. This information was requested because the original head loss testing allowed settling of some debris, and the staff was concerned that some debris allowed to settle in the test would actually transport in the plant. The staff was concerned with break S7 because it is the limiting break that is close to the strainer and could affect the flows in the vicinity of the strainer. This could result in added transport in the plant. The licensee responded by stating that it performed new head loss testing designed to transport all debris to the test strainer in accordance with NRC-approved test methods. The use of this test methodology adequately addresses the staff concerns because all material predicted to reach the strainer transported to the strainer during testing.

The NRC staff requested that the licensee provide information regarding the prototypicality of the test strainer size compared to the plant strainer, and how the scaling of the test parameters was performed. The licensee stated that updated testing was performed and provided information regarding the scaling. The scaling was performed per NRC-approved guidance. The NRC staff also noted from other RAI responses that the testing promoted all debris to reach the strainer making velocities close to the strainer less important from a transport perspective.

The NRC requested information on the potential for debris transport in the vicinity of the strainer via flotation and how this was considered in the head loss tests for Waterford 3. The licensee stated that only a limited number of debris types had the potential to float for a significant period of time following a LOCA. The licensee also stated that the top surfaces of the strainers are not perforated so that flow cannot pass through these surfaces. The licensee stated that any type of debris that may float to the strainer will not result in significant head loss. The NRC staff concluded that the licensee's position is reasonable, and that testing did not need to consider this issue.

The NRC staff requested additional information regarding the assumptions made concerning the settling of particulate down to 100 microns. Specifically, the staff requested that the licensee identify the NRC-sponsored tests being referenced in this discussion. The licensee stated that it used NUREG/CR-6916, "Hydraulic Transport of Coating Debris," dated December 2006 (ML070220061), which is the subject of the transport of coating debris. The licensee stated that particulate settling is only credited in the reactor lower plenum. The staff found that the information was adequate.

The NRC staff requested that the licensee supplement its response to Open Item 4 from the NRC staff audit. Open Item 4 requested that the licensee justify transport percentages from each side of containment and correlate the transport to flow, as well as define the starting point for debris in the recirculation phase. The staff did not consider the licensee's response to be sufficient because (1) the initial containment pool flows during fill up are chaotic and may distribute debris unevenly to the two sides of containment independent of the relative flow split during recirculation, and (2) the response did not appear to define the starting point for the transport paths that had been requested. The licensee stated that the transport analysis assumes that the debris transport begins at the break location and that the flow split from computational fluid dynamics (CFD) models is used to determine the debris transport paths. The licensee stated that the flow split assumption has no impact on the amount of debris that transports to the strainer for most of the breaks. For the single break near the strainer (break S7) a non-conservative effect could occur due to the flow split assumption. Therefore, the licensee revised the debris transport to assume 100 percent transport from break S7. This treatment is conservative and therefore acceptable.

The NRC staff requested that the licensee provide additional information regarding the response to Open Item 7 from the NRC staff audit. The issue concerns the single failure of a low-pressure safety injection (LPSI) pump failing to trip and resulting in excess flow through the strainer. The staff did not consider the licensee's response to have fully addressed the item because (1) the licensee did not provide a technical basis for being capable of stopping flow from an LPSI pump, which has failed to trip in a short period following the switchover to recirculation, and (2) the head loss testing may not have adequately modeled the failure of the LPSI pump to trip. The licensee stated that it was in the process of revising emergency operating procedures to ensure that a running LPSI pump would be stopped within 7 minutes of swapover to sump recirculation. All actions required to stop the flow are performed from the control room. The licensee also stated that the strainer was tested under conditions conservative with respect to the plant to ensure that vortex formation would not result in air ingestion to the pumps. The staff found this information to adequately address the staff concerns.

The NRC staff requested additional information regarding the assumption that 25 percent of small debris is treated as lifting onto the sump strainer for one of the CFD scenarios for which less than 25 percent of the perimeter area around the plenum exceeds the curb lift velocity metric. The licensee stated that the assumption applied to only one case, break S7. For this break the licensee changed the transport assumptions such that 100 percent of the debris generated from this break is assumed to collect on the strainer. The staff found that this assumption is conservative, and therefore acceptable.

The NRC staff requested additional information regarding the settlement of Min-K and Microtherm during head loss testing because of the potential for differences in settling characteristic between plant materials and test materials. The licensee stated that the debris types in question were not observed to settle during the updated testing. Additionally, the updated testing was designed to promote the transport of all debris to the strainer during testing, unlike the testing for which the RAI was written. Since the updated testing promoted the transport of all debris and these debris types were not observed to settle during testing the staff found the response to this RAI acceptable.

NRC Staff Conclusion:

The NRC staff finds that the transport area has been addressed satisfactorily. The licensee used NRC staff approved guidance to perform the evaluation. The licensee decided to make changes to the transport evaluation and reperform head loss testing to ensure that potentially non-conservative methods and assumptions were eliminated. Therefore, the NRC staff considers the debris transport analysis is performed adequately to address issues associated with 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 is based on documentation provided by the licensee through November 23, 2010.

The NRC staff's initial review of the Waterford 3 head loss and vortexing evaluation identified several issues that could have resulted in a non-conservative evaluation. The licensee ultimately performed a second round of head loss testing to resolve the NRC staff concerns. There were several RAIs in addition to audit open items that were addressed during the review.

The licensee's approach to address strainer debris issues was to install a General Electric strainer in place of the original much smaller units. Waterford 3 has a single strainer that serves both trains of the ECCS/CS. The strainer is about 3,699 ft^2. The Waterford 3 containment has a large amount of Nukon and LDFG, which could become debris during a LOCA. Smaller amounts of Min-K and Microtherm are also installed. Other debris includes coatings and latent debris as well as RMI. Testing was conducted using both a full-scale sector and a full-scale module of a strainer. The module consisted of 10 disks instead of the 17 disks per module installed in the plant. The sector represents the gap between two disks. Testing was performed with scaled amounts of debris predicted to arrive at the strainer after accounting for transport.

The NRC staff conducted an audit of the Waterford 3 resolution to GL 2004-02 in 2007. The audit identified several issues with head loss testing that required more detailed responses. The open items resulting from the audit were not all fully addressed in the licensee's first supplemental response. Those items are discussed here along with other staff identified open issues.

The descriptions in the licensee's initial submittal did not provide information that showed the head loss test methods were prototypical or conservative. Some of the licensee's strainer testing was conducted before clear testing guidance was available to the industry. The strainer vendor made improvements to the test procedures, but some of the licensee's testing was completed prior to the implementation of upgraded procedures. The licensee stated that the thin-bed testing was reperformed in response to the audit findings. However, the information provided regarding the testing did not allow the NRC staff to determine whether the testing was adequate. The module testing, which is the limiting case for head loss for Waterford 3 was not initially reperformed. In response to additional questions, the licensee reperformed the head loss testing, both thin-bed and full load module testing, using staff approved guidance. The issues that were addressed by the licensee by performing new testing and performing other evaluations are discussed below. It is also noted that the licensee significantly decreased the fibrous debris loading on the strainer by replacing Nukon insulation on the SGs with RMI during a SG replacement.

The maximum design flow rate through the strainer is 6,470 gallons per minute (gpm). Operation with an LPSI pump failure to trip would result in 12,120 gpm flow. The high-pressure safety injection (HPSI), LPSI, and CS pumps can all take suction from the strainer in the recirculation mode. The strainer area is 3,699 ft^2 resulting in an approach velocity of 0.0039 feet per second (ft/sec) for dual train operation with no LPSI pump operating (design flow rate).

For large-break LOCAs (LBLOCAs), the strainer modules are expected to be fully submerged at the initiation of recirculation with at least 8 inches of submergence. For small-break LOCAs (SBLOCAs) the submergence was calculated to be at least 2 inches.

Several of the NRC staff questions were centered on the methods used to perform head loss testing, the applicability of each test to specific plant conditions, and the evaluation of the applicability of the test results to the plant. The licensee performed updated testing and analysis in accordance with staff approved guidance and replaced the results of all previous testing with these new results. The performance of new testing resolved many of the issues identified by the NRC staff. Updated testing included a thin-bed test, a full load test (maximum postulated debris load), and a test that included 10 percent excess fiber to quantify available margins. The new testing included chemical effects and surrogates for all predicted debris sources within containment. The licensee provided a description of the testing and analysis as a response to several of the staff's questions. The testing was performed at a facility at which the staff had witnessed testing previously and found the labs methods to be acceptable. Only significant issues with the testing and evaluation are documented in this section.

The NRC staff questioned how the testing accounted for the potential of a failure of an LPSI pump to trip. The licensee's initial response to this issue was not accepted by the staff. The final response that was acceptable, is discussed in section 7.0, "Debris Transport" of this staff review.

The methodology used to revise the plenum portion of the clean strainer head loss to 0.063 ft from 0.41 ft was not provided. The licensee stated that it was calculated using standard head loss methods found in Crane Technical Paper No. 410. The NRC staff found this to be an acceptable methodology.

Several questions in the transport and head loss areas were prompted because the licensee credited near field settling during the initial head loss testing. The licensee stated that the updated testing did not credit near field settling, but instead was designed to promote transport of all debris to the strainer. This test feature resolved several NRC staff issues.

In the supplemental response dated October 23, 2008, the licensee identified that flashing at the strainer would not occur because the strainer submergence is 8 inches, and the maximum head loss is about 6 inches. The NRC staff noted that this is true for the LBLOCA but does not address the SBLOCA which has a submergence of about 2 inches. The supplemental response did not address the potential maximum head loss for a SBLOCA. The licensee stated that it determined that it could credit a small amount of overpressure to prevent flashing of the fluid as it passes through the debris bed and strainer. The licensee provided plots of containment pressure at both maximum and minimum sump temperatures that demonstrated that the containment pressure is much greater than that required to suppress flashing. In addition, the licensee demonstrated that containment pressure credit would only be required for a short period of time during the system response under the most conservative conditions. The NRC staff found the response acceptable.

The NRC staff also questioned whether excessive degasification of the sump fluid could occur as it passes through the debris bed and strainer such that the gasses could be transported to the pump suctions resulting in increased required NPSH ($NPSH_R$). The licensee calculated the amount of degasification that could occur under high and low temperature conditions and calculated the adjustment in $NPSH_R$. The amount of degasification was shown to be minimal and to have a small impact on $NPSH_R$. The NRC staff concluded that the licensee's response shows that the impact of degasification on pump NPSH margins are negligible.

NRC Staff Conclusion

The NRC staff finds that the head loss area was adequately addressed. The licensee used staff approved guidance to perform the evaluation, including reperformance of the head loss testing used to determine the strainer performance under debris loaded conditions. The head loss testing was based on a reduced debris load because the licensee removed a considerable amount of fibrous debris from the containment to assist in assuring adequate sump performance. Therefore, the NRC staff considers that the head loss and vortexing evaluations were performed adequately to address issues associated with GL 2004-02.

9.0 NET POSITIVE SUCTION HEAD

The objective of the NPSH section is to calculate the NPSH margin 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 is based on documentation provided by the licensee through November 23, 2010.

The licensee's approach to showing adequate NPSH margin was incomplete with the original supplemental response, but the October 23, 2008, response included much information that had been omitted in this area. The final RAI response included adequate information for the NRC staff to complete its review of this area.

Waterford 3 has a single safety injection system (SIS) strainer that supplies suction to two trains of ECCS when recirculation is required. The HPSI, LPSI, and CS pumps can take suction from the SIS strainer when in recirculation. The LPSI pumps are designed to trip at the initiation of recirculation, but a single failure is postulated where one LPSI pump fails to trip. When the refueling water storage pool (RWSP) is depleted to a predetermined level (10 percent), a recirculation actuation signal (RAS) is generated. The RAS results in swapping the LPSI, HPSI, and CS pump suctions from the RWSP to the SIS sump. The RAS also trips the LPSI pumps.

The minimum sump level calculation was revised to include holdup volumes that were identified during the NRC staff audit of Waterford 3. All of the necessary holdups were included in the revision to the calculation with the exception of potential holdup in the refueling canal (upstream issue) and the potential for RCS volume shrinkage due to cooling of the liquid during the LOCA response. The licensee's calculation for minimum level currently credits only RWSP volume for sump inventory adding safety injection tanks for LBLOCAs. This provides some conservatism in the calculation for many break scenarios.

In accordance with NRC staff guidance, the licensee's calculation does not credit containment accident pressure in the NPSH evaluation. The pressure term is conservatively taken as zero. As the water cools below saturation, credit is taken for the reduced vapor pressure of the fluid.

In the NPSH evaluation, the licensee stated that the methodology used to calculate the pump suction side head losses was a computer code called Pipe-FLO. This is an acceptable standard engineering methodology.

The NRC staff questioned the methodology used to calculate the minimum sump water level because it did not expressly include the potential RCS volume reduction due to cooling of the fluid (part of Audit Open Item 13). The licensee stated that it revised the sump water level calculation to account for a break high in the RCS and also for the volume reduction that occurs when the coolant is reduced in temperature. In making the revision to the level calculation, the licensee also removed some conservatism from the calculation. The containment atmosphere temperature was reduced from 260 to 250 degrees Fahrenheit (°F). Also, the calculation takes into account realistic communication between the containment normal sump and the SIS. The updated calculation appropriately accounts for coolant shrinkage due to RCS cooling.

Another issue identified by the NRC staff is that the sump level calculation assumed that no holdup occurs in the refueling canal. This was also identified as Audit Open Item 16. The licensee stated that a majority of the breaks considered for the GSI-191 evaluation are located below the 14 ft elevation, and that physical structures prevent large debris from being ejected to upper containment where it could reach the refueling canal and potentially block the drain preventing water from reaching the sump. The licensee also noted that much of the potential debris (insulation on the SGs) was being removed during SG replacement and replaced with non-problematic insulation types. The licensee stated that any debris that entered the canal would not transport because there would be inadequate water level to push the debris toward the drains. This is because the two independent drains are large enough to prevent water level in the canal from increasing. The licensee also demonstrated that debris that is not blown directly to the refueling canal cannot be washed into the refueling canal due to trenches and curbs around the canal. The licensee concluded that debris large enough to block the drains cannot transport and block the drains.

NRC Staff Conclusion

The NRC staff determined that the NPSH area was addressed satisfactorily for Waterford 3. The licensee used staff approved guidance to perform the evaluation, including recalculation of the sump level to account for phenomena not previously included. The NRC staff concludes that the NPSH evaluation was performed acceptably, and the issues associated with GL 2004-02 have been addressed.

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 is based on documentation provided by the licensee through November 23, 2010.

The ZOI for epoxy coatings was reduced to 4D per WCAP-16568-P, “Jet Impingement Testing to Determine the Zone of Influence (ZOI) for DBA-Qualified/Acceptable Coatings” (ML061990594 (not publicly available, proprietary information)). The licensee originally used a 4D ZOI to determine the amount of qualified inorganic zinc (IOZ) coatings debris generated. The NRC staff guidance on the use of WCAP-16568-P recommended using a 10D ZOI for un-topcoated IOZ. The licensee stated, in an RAI response, that it verified that all IOZ coating within 10D of any break location is top coated with epoxy. Therefore, the 4D ZOI is acceptable for the Waterford 3 coatings systems.

The licensee assumed 50 percent of the qualified coating in the containment liner dome above 112 feet (above the polar crane rails) would fail since these coatings are inaccessible for a thorough examination or remediation.

All qualified coatings in the ZOI failed as fine particulate. Unqualified coatings, with the exception of degraded qualified epoxies, failed as particulate. This is acceptable in accordance with the NRC SE approved methodology.

Degraded qualified epoxy coatings failed as chips with a distribution in accordance with ALION-REP-TXU-4464-02 and a letter by Cavallo dated September 20, 2007. The NRC staff found this acceptable. The licensee assumed that 100 percent of the particulate transports to the strainer and 20 percent of the chips fall on or near the strainer and are assumed to transport to the strainer. The rest of the chips transport fractions are determined by CFD.

The NRC staff questioned how the degraded epoxy coatings were treated during head loss testing. The licensee stated that testing resulted in a filtering bed and all coatings debris was assumed to be particulate in the testing. The staff found this acceptable.

The NRC staff found the particulate surrogate material, silicon carbide, used for testing acceptable.

The licensee's coating assessment program met expectations.

Audit Open Item 14 identified that the licensee was treating unqualified coatings debris characteristics in the same manner as qualified coatings without any justification. The licensee updated the methodology to assume that all unqualified coatings fail as particulate. This assumption meets NRC staff guidance, especially when the strainer has a fibrous bed fully covering the strainer.

The NRC staff had a question regarding the total volume of qualified and unqualified coating debris, including the treatment of degraded qualified epoxy. The supplemental response by letter dated November 23, 2010 corrected this issue.

NRC Staff Conclusion

The licensee used NRC staff guidance to perform the evaluation of coatings for the resolution of GL 2004-02. Therefore, the staff finds the response acceptable in this area and considers it closed.

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 is based on documentation provided by the licensee through October 23, 2008.

The licensee has implemented administrative controls to ensure that the debris source term that could affect the recirculation sump following a LOCA is bounded by the existing analysis. The NRC staff reviewed the administrative controls for plant modifications and containment materials during the audit and verified that the controls were adequately described in the licensee's October 23, 2008 supplemental response. The licensee stated that materials introduced into the reactor building as part of the plant modification process (e.g., insulation materials, equipment signs and tags) would be reviewed to ensure that adverse interactions with the emergency sump would not occur.

The licensee proceduralized the foreign material exclusion (FME) program to prevent inadvertent introduction of foreign materials into plant systems and components. The most stringent FME controls are applied to the ECCS sump and refueling cavity. Provisions are in place, per site procedures, to ensure that the post-outage conditions exist (containment verified to be free of foreign material) in the event that a containment building entry be required while at power or during a brief forced outage.

The NRC staff review for this area agrees with the audit conclusion, which found that the licensee's housekeeping and FME programs adequately control its respective processes for control of the debris source term as needed to maintain adequate ECCS strainer functionality .

NRC Staff Conclusion

The NRC staff finds that the licensee's housekeeping and FME programs adequately control their respective processes for maintenance of the debris source term as needed to maintain adequate ECCS strainer functionality. The licensee used approved guidance for evaluation of the debris source term area. Therefore, 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 sump screen modification.

NRC Staff Review

The NRC staff review is based on documentation provided by the licensee through November 23, 2008.

The licensee's supplemental response provided an adequate basic description of the major features of the new sump strainers. They are General Electric energy modularized stacked disk strainers. There are 11 modules that are mounted above the containment floor and over the existing recirculation sump pit. Each module is constructed of 17 rectangular, horizontally oriented disks arranged in a vertical stack. Water enters the top and bottom disk surfaces and travels to the center of the disk, down to the plenum, and down into the sump pit. The two trains of recirculation suction are separated by grating and remain hydraulically connected. The 11 modules provide a total strainer surface area of 3,700 ft^2. The sump pit is totally enclosed and covered by the plenum structure. Maintenance access hatches can be opened for inspection and testing on either side of the sump. The strainer modules have an inspection port on top that can be opened for viewing the internals of the module. The new strainers are constructed of stainless steel.

The supplemental response also provided an adequate basic description of the associated modifications to remove or relocate interferences and protect the new sump strainers. The previously existing screens were removed. A sump low level switch was relocated within the sump pit to eliminate interference with the new strainer structures. Capillary tubing and tube track from the sump level transmitters were rerouted. The licensee relocated 19 tri-sodium phosphate (TSP) recirculation pool pH buffer baskets away from the sump pit area to eliminate interference with the new strainer structures.

NRC Staff Conclusion

The licensee provided an adequate description of the modifications performed to implement changes for GL 2004-02. The information provided met the appropriate guidance. Therefore, the screen modification package area has been adequately addressed.

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's review is based on section 3.k, Sump Structural Analysis, of the licensee's initial February 29, 2008, submittal, as well as its final supplemental submittal dated October 23, 2008.

Based on the review of the licensee's initial and final supplemental responses to GL 2004-02, regarding the structural analyses for Waterford 3 replacement recirculation sump strainers, the NRC staff concludes that the licensee adequately addressed the information requested by the revised Content Guide for GL 2004-02 Item 2(d)(vii). The licensee's initial, supplemental response was reviewed by the staff to determine whether Open Item 15 from the NRC staff audit report dated January 28, 2008, had been sufficiently addressed. It was determined during this review that the licensee had provided a sufficient amount of information to conclude that the replacement sump strainers were structurally adequate. To resolve the open issue, the licensee made corrections to its structural analysis and also improved the documentation associated with the analysis.

The licensee used Analysis System (ANSYS) computer software to perform a series of finite element analyses to qualify each of the strainer components. The models were subjected to loading combinations associated with dead weight, differential crush pressure, seismic loads (including hydrodynamic mass), and thermal expansion effects. Each of these loads is consistent with the guidance of NEI 04-07. The induced stress values were then compared to the allowable stress values of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). All resultant stresses were determined to be within the allowable limits.

To address potential dynamic effects associated with a HELB on the replacement sump strainers, the licensee stated that evaluation GENE-0000-0048-9192 concluded that the strainer assembly is protected from pipe whip, jet impingement, or missile impact associated with a HELB. The NRC staff reviewed the evaluation and associated drawings and verified that based on location and containment geometry, the strainers were protected from dynamic effects of a HELB. This conclusion was confirmed in the staff's Audit Report dated January 28, 2008.

The licensee's submittal stated that a backflushing strategy was not credited in the overall resolution of GSI-191. For this reason, a reverse flow condition was not required to be analyzed in the structural evaluation.

In addition to the licensee's initial supplement, the final supplement contained additional information to demonstrate the structural integrity of the licensee's replacement sump strainers.

Included in the final supplement were details surrounding the various loads and loading combinations used to satisfy the code requirements of the ASME Code, Section III, subsection NC and ND. The licensee provided a quantitative summary of its finite element analyses of the strainer models, which indicated that the maximum stresses on the strainer assembly resulting from the bounding loading combinations meet the requirements of the ASME Code used to qualify the strainer components. Based on the NRC staff's detailed audit, previous review, and comments listed above, the staff concludes that the replacement strainers at Waterford 3 will maintain their structural integrity during the recirculation phase of a DBA, and the concerns outlined in GL 2004-02 Item 2(d)(vii) have been sufficiently addressed.

NRC Staff Conclusion

Based on the NRC staff review of section 3.k of the licensee's supplemental response dated February 29, 2008, the staff concludes that the licensee has 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 Code, Section III, subsection NC and ND. The staff's Audit Report dated January 28, 2008, for Waterford 3 also confirmed this fact and stated that all stresses were below code allowable limits. Therefore, the NRC staff concludes that the sump structural analysis for Waterford 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 is based upon documentation provided by the licensee through November 23, 2008.

The NRC audit of upstream effects focused on the drainage flowpaths through the refueling canal and reactor cavity because of the potential for the drains for these large volumes to retain substantial quantities of water if the drains were to become blocked by debris.

An audit open item was generated requesting additional information on how the licensee concluded that the refueling cavity drains would not become blocked with debris following a postulated LOCA. The licensee stated that the refueling cavity has two 6-inch drain lines (without screens) that drain to the containment floor, and one 4-inch line that drains to the containment sump. In the event that large debris is propelled over the SG cavity walls into the refueling cavity, the debris must land on a drain in order to clog it because the velocities in the cavity are too low to transport a large piece of debris to a drain. During plant operations, the upper guide structure lift rig is stored directly above one of the 6-inch drains. The diver stairs are located above the other 6-inch drain and are permanently mounted in the refueling cavity. The licensee concluded that debris larger than the drain lines could not reach the drains due to the obstructing equipment and that any smaller debris that transports to these two 6-inch drains will pass through the drains since they do not contain screens. Therefore, there will always be drainage available from the refueling cavity to the active pool. The NRC staff reviewed the configuration of the refueling cavity drains and equipment that could prevent large debris from reaching them. The NRC staff agreed that it is not credible for large pieces of debris to reach the drains. Therefore, they cannot become blocked.

The Waterford 3 containment is mostly uncompartmentalized with the exception of the pressurizer room. There are no structures totally surrounding the major components (SG, RCP, etc.) of the RCS. All RCS components with the exception of the pressurizer are within the SG compartments, but the compartments are open to each other, open to the annulus below elevation -1 ft, and open to the upper containment dome above elevation +62.25 ft. The pressurizer is located in a separate room with an opening in the floor that connects to the containment annulus.

Waterford 3 does not have any significant potentially inactive volumes other than the reactor cavity and containment normal sump. For significant quantities of debris to be trapped in the reactor cavity or containment normal sump, the break location would have to be within the reactor cavity. For breaks outside the reactor cavity, the flow of water and debris into these volumes is limited by the size of the drain lines connecting the volumes to the active pool. Other connections between the volumes are at elevations above the minimum sump level.

NRC Staff Conclusion

The NRC staff concluded that the licensee evaluation of upstream effects was conducted using staff approved guidance and that there was no potential for significant amounts of water to be held up such that it could not reach the active sump pool. Therefore, the staff considers that the licensee adequately addressed the area of upstream effects for GL 2004-02.

15.0 DOWNSTREAM EFFECTS - COMPONENTS AND SYSTEMS

The objective of the downstream effects, components and systems 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 is based on documentation provided by the licensee through November 23, 2010.

The licensee's GL 2004-02 supplemental responses include a detailed description of the evaluations performed to assess the effect of sump-strainer bypassed debris on components and systems. The licensee stated that it had performed the evaluations per the industry guidance contained in WCAP-16406-P-A, Revision 1 and the NRC Final SE of that document.

The dimensions of close-tolerance components in the ECCS were evaluated against the acceptance criteria of 1.1 times and 2 times the screen opening size. The HPSI pumps, CS pumps, CS header solenoid operated valves, HPSI pump recirculation flow orifices, HPSI header throttle valves, and reactor coolant (RC) loop throttle valves were determined to have minimum flow clearances small enough to require blockage evaluations. The evaluation determined that most components in the system would not be blocked by debris and that potential for mechanical seal failure of the HPSI pump is low. The potential for significant debris-induced wear of the seal faces due to the tight running gap was determined to be low because the HPSI seals use a priming ring to circulate water through a heat exchanger and the seal. An evaluation of the CS pump cyclone separators, based on a comparison of separator test data to the Waterford 3 separators and debris loading, demonstrated that the separators will continue to provide clean water to the CS pump mechanical seals following a DBA.

Wear evaluations of close tolerance components (e. g., HPSI pumps, CS pumps, CS header solenoid operated valves, HPSI pump recirculation flow orifices, HPSI header throttle valves and RC loop throttle valves, instrument lines, relief valves, piston check valves and post-accident sampling system components) by circulated debris concluded that wear was acceptable as it resulted in negligible flow effects. For the HPSI pumps, the analysis determined the worn condition of the pumps and generated performance curves at various time points during the 30-day mission time. These performance curves were used as input to the Waterford 3 LTC analysis of record (AOR) to determine acceptability. The result from the LTC AOR evaluation is that adequate core cooling can be maintained and the AOR remains valid. The worn condition wear ring clearances of the pump were input to a rotor-dynamic analysis to confirm that the pump remains dynamically stable through the mission time.

The licensee stated that the evaluations confirmed that the ECCS equipment at Waterford 3 would remain capable of passing sufficient flow to the reactor to adequately cool the core during the recirculation phase of a postulated LOCA.

The licensee stated that, at this time, no operational changes have been made nor have any been identified for Waterford 3.

NRC Staff Conclusion

The licensee used NRC staff approved guidance to evaluate the effects of debris on components downstream of the strainer. The evaluation determined that any effects of wear do not affect the ability of the components to perform the required post-accident functions. The evaluation also demonstrated that components would not be blocked by debris that passes through the strainer. Therefore, the NRC staff concludes that the licensee's evaluation was performed adequately and demonstrates that the plant configuration is acceptable for the downstream effects area. This area is considered to be adequately addressed 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 review of the downstream effects, fuel and vessel area was performed based on responses from the licensee received through March 30, 2023.

The licensee stated that in-vessel effects had been evaluated using NRC staff review guidance for IVDE (ML19228A011) and associated industry guidance. The referenced industry guidance is TR WCAP-17788-P, Revision 1, "Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090)," dated December 2019 (Package ML20010F181) and TR PWROG-16073-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 [Final Safety Analysis Report] Changes," dated February 28, 2020.

The licensee stated that all fiber that reaches the reactor vessel is conservatively assumed to collect at the core inlet. In addition, the test conditions were designed to conservatively maximize the amount of fiber bypass.

The licensee also stated, as part of its in-vessel submittal, that additional modifications to the plant are not required to address the effects of debris on GL 2004-02 issues and that the plant licensing basis would be updated to incorporate the in-vessel downstream effects analysis and conclusions.

The licensee performed plant-specific penetration (bypass) testing to determine the amount of fiber that can pass through the strainer and potentially affect LTCC. The testing used a prototypical strainer section, the plant-specific strainer approach velocity, and fiber which was added incrementally until the maximum potential fiber bed thickness was reached.

The licensee described the methods and test facility used for the testing. The testing was conducted using expected methods that are similar to those employed for strainer head loss testing except that the methods are designed to maximize fiber penetration and the fiber that passes through the strainer is captured and quantified. The NRC staff has witnessed several penetration tests and found that these methods assure adequate conservatism in the amount of penetrated fiber. The NRC staff checked that the strainer area, incremental debris loading, total debris load, scaling, and approach velocity were determined correctly. The NRC staff determined that the testing is acceptable to provide a basis for the fiber amount that may penetrate the Waterford 3 strainer.

The maximum amount of calculated fiber that can penetrate the strainer was 27.34 lbm. The licensee stated that this is equal to 57.1 g/FA. Based on the Waterford 3 updated FSAR the reactor contains 217 fuel assemblies which validates the amount of fiber per fuel assembly assertion. The NRC staff could not determine the basis for the maximum amount of fibrous debris assumed to be transported to the strainer. The in-vessel submittal stated that the maximum amount of fiber that can transport to the strainer is 244 ft^3. The licensee's previous submittal, dated November 23, 2010, stated that the full load test for the strainer included 253.1 ft^3 of fiber fines and 37.5 ft^3 of latent fiber. This totals 290.6 ft^3. The licensee referred to a calculation that was performed after the November 2010 submittal, so the transported amount of fiber may have been revised. However, the bypass testing was completed in 2010. The licensee confirmed that the penetration testing and analysis were both based on transportable fiber amounts of 244 ft^3, which was developed assuming that the amount of fiber would be reduced when the SGs were replaced. The SGs were replaced in 2012 so the 244 ft^3 value is valid for the current plant configuration.

Based on the above, the NRC staff concluded that 57.1 g/FA is an acceptable fiber load on which to base the Waterford 3 in-vessel evaluation.

The licensee compared the plant-specific key parameters against the key parameters used in WCAP-17788-P, Revision 1. The key parameters were presented in table 5 of WCAP-17788-P and discussed further in the text of the March 30, 2023, submittal. The licensee demonstrated that all of the key parameters except for the maximum core inlet debris amount, and the maximum rated thermal power were within the assumptions from the WCAP-17788-P, Revision 1 analysis and were therefore acceptable according to the NRC staff review guidance for in-vessel effects. The two values that were not bounded by the analysis were evaluated by the licensee as discussed below.

The plant-specific hot-leg switchover (HLSO) and T_{block} times are compared against the times at which chemical precipitation may occur in the post-LOCA environment under Waterford 3 plant-specific conditions. The T_{block} is the earliest time after a LOCA when the core inlet may be fully blocked, but alternate flowpaths (AFPs) will maintain adequate flow to the core to assure LTCC. Chemical effects testing conducted as part of WCAP-17788-P, Revision 1 demonstrated that chemical precipitation will not occur before 24 hours following a LOCA. The licensee stated that WCAP-17788-P, Revision 1, Volume 5, Test Group 33 parameters are representative of the plant-specific conditions. The NRC staff reviewed the autoclave and precipitation test results contained in WCAP-17788-P, Revision 1, Volume 5, and determined that Test Group 33 reasonably represents the postulated post-LOCA conditions for Waterford 3. A comparison of the plant and test parameters was provided in table 6 in WCAP-17788-P. The testing demonstrated that T_{chem} for Waterford 3 is 24 hours. The T_{block} for Waterford 3 was calculated to be 333 minutes (5.55 hours). The HLSO for Waterford 3 occurs within 3 hours. The NRC staff concluded that without chemical precipitates present, the core inlet flow would be adequate to maintain LTCC. If adequate AFPs are available prior to precipitation, LTCC is assured. HLSO provides injection that bypasses the core inlet and AFPs allow flow around the core inlet and to the fuel. Since both HLSO and adequate AFP flow occur prior to T_{chem}, LTCC is assured for Waterford 3.

The licensee's evaluation of the plant-specific core inlet load exceeding the analyzed limit was consistent with the NRC staff review guidance for IVDE. The limit was scaled to the Waterford 3 fuel pitch, which differs from the test and analysis conditions discussed in WCAP-17788-P, Revision 1. The maximum possible core inlet fiber amount is significantly less than the total core fiber limit. Therefore, the licensee cited the NRC staff review guidance, which concluded that nonuniform debris deposition at the core inlet results in conservatism in the core inlet fiber limit and that core inlet amounts up to the total core inlet limit would not inhibit LTCC.

The licensee stated that the earliest possible plant sump switchover time of 26.9 minutes is greater than the analyzed time of 20 minutes and that the analysis is bounding for this value.

The licensee stated that the Waterford 3 AFP resistance is less than the analyzed value and is therefore bounded by the analysis.

The rated thermal power for Waterford 3 is greater than the thermal power assumed in the analysis. The licensee stated that even though the rated power is higher than the analyzed power the AFP resistance at Waterford 3 is low enough to allow adequate flow to assure LTCC even though decay heat will be higher due to the plant's higher thermal power. The NRC staff reviewed the licensee's evaluation and found it acceptable. The NRC staff also noted that the Waterford 3 sump switchover time is greater than the analyzed time. This allows decay heat additional time to decrease because debris introduction to the core inlet is delayed compared to the analysis. In addition, Waterford 3 has significant margin to the total core fiber limit. The staff concluded that debris in the reactor will not inhibit LTCC.

The licensee stated that the plant ECCS limiting flow rate is within the assumptions used in the WCAP-17788-P, Revision 1, analysis.

<u>NRC Staff Conclusion</u>

The licensee used NRC staff approved guidance to evaluate the effects of debris arriving at the reactor on LTCC. Not all of the key parameters in the guidance were bounded by the analysis values, but the licensee demonstrated that margins in other areas were adequate to account for these issues. Therefore, the NRC staff concludes that the downstream in-vessel evaluation was performed acceptably and demonstrates that debris arriving at the vessel will not inhibit LTCC. This area has been adequately addressed for GL 2004-02.

17.0 CHEMICAL EFFECTS:

The purpose of the chemical effects section is to evaluate the effects that chemical precipitates have on head loss across the sump strainer and core cooling. Chemical effects within the reactor vessel were evaluated in section 16.0, “Downstream – Fuel and Vessel,” of this staff review.

NRC Staff Review

The NRC staff review is based on documentation provided by the licensee on October 23, 2008 and November 23, 2010, as well as the updated plant-specific path and schedule for resolution of GL 2004-02 supplement dated May 16, 2013. The reference documents used for this review include “NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Plant Specific Chemical Effects Evaluations,” dated March 2008 (ML080380214).

Waterford 3 uses TSP for pH control of the post-LOCA pool. The licensee’s plant-specific debris generation and transport analyses determined that the debris sources include fiberglass insulation (MEI/Nukon/Latent), Min-K/Microtherm, concrete, aluminum silicate and calcium silicate.

The NRC staff developed questions in the chemical effects area based on the licensee’s initial supplemental responses. In its supplement dated February 29, 2008, the licensee stated that a 30-day integrated chemical effects test was being performed by Alion Science & Technology (Alion) and was ongoing at that time. In its October 23, 2008, supplement, the licensee stated that its 30-day integrated testing and analyses concluded that no aluminum-based precipitates would form in the Waterford 3 environmental conditions with a pH less than 8.1; therefore, any reduction in the aluminum oxy-hydroxide precipitate was reasonable. The licensee based this assertion on aluminum-based precipitates not forming until the post-LOCA environment had cooled to below 140°F. The licensee’s responses provided insufficient information for the NRC staff to conclude that chemical effects had been satisfactorily addressed. At the time, the licensee did not provide sufficient information to demonstrate that aluminum hydroxide will not precipitate. Therefore, on September 22, 2009, the NRC staff requested additional information (RAIs 39-44) with the purpose of helping NRC staff evaluate the licensee’s conclusions based on the expected pH range, the projected aluminum concentration, and the post-LOCA temperature profile used to reach the conclusion that aluminum-based precipitates would not form.

The licensee submitted draft RAI responses for NRC staff information in April 2010 followed by a public teleconference on June 1, 2010 (ML101530232). During the public teleconference, the NRC staff stated that the baseline approach of large-scale head loss testing with full-load WCAP-16530-NP-A precipitates has been accepted in the past; however, the staff needed to review the plant-specific analysis. The NRC staff also stated they required additional information to validate the licensee’s method using either the Argonne National Laboratory (ANL) solubility correlation or a vertical loop methodology. As a result, the licensee submitted revised responses on June 17, 2010, wherein the licensee provided a chemical effects approach summary.

On November 23, 2010, the licensee provided a response to address chemical effects RAIs 39-44, which included a detailed description of the chemical effects head loss testing and evaluation for Waterford 3.

The licensee chose the Alion facility in Warrenville, Illinois, to perform its chemical effects head loss testing and evaluation for Waterford 3. The licensee also used Alion to analyze chemical effects precipitate loading using the guidance provided in WCAP-16530-NP-A and the associated spreadsheet tool (Version 1.1). The WCAP-16530-NP-A methodology involves the determination of the chemical precipitate load, the preparation of the calculated amount of precipitates, and procedures for the addition of the premixed precipitates to the test loop after a debris bed is formed on the test strainer. The WCAP-16530-NP-A analysis of the Waterford 3 debris load and post-LOCA containment sump chemistry predicted that aluminum was the limiting reactant for sodium aluminum silicate and that no aluminum oxy-hydroxide was expected to be formed. Therefore, aluminum oxy-hydroxide was not included in Waterford 3 testing. The corrosion/dissolution sources contributing to elemental release quantities and precipitates for the limiting case (maximum volume, maximum pH) are concrete, silica powder, E-glass, and unsubmerged and submerged aluminum. Nukon and MEI, both composed of E-glass, are the primary sources of calcium and silicon. The TSP buffer reacts with the dissolved calcium release from E-glass, as well as from concrete, to form calcium phosphate precipitate. Unsubmerged aluminum is the primary source of dissolved aluminum, followed by E-glass, and submerged aluminum.

The licensee conducted three chemical effects head loss tests that consisted of full load, thin bed, and additional margin tests. The tests used design-basis loadings for fiber, particulate, and chemicals. For all three tests, a fiber and particulate debris bed was formed on a representative plant strainer module section and a stabilized head loss was determined. Calcium phosphate was then added in three batches and the stabilized head loss values were determined after each batch. Then, sodium aluminum silicate was added in three batches with stabilized head loss determined after each batch.

In addition, the licensee used the ANL empirical equation as referenced in ANL technical letter report, “Aluminum Solubility in Boron Containing Solutions as a Function of pH and Temperature,” dated September 19, 2008 (ML091610696), to predict aluminum solubility as a function of pH and temperature following a LOCA. The licensee takes credit for aluminum solubility at temperatures above 200°F. The licensee compared the predicted WCAP-16530-NP-A released aluminum concentration to the aluminum solubility as predicted by the ANL solubility correlation. The comparison indicated that the aluminum solubility limit exceeds the dissolved aluminum concentration for the entire accident duration for both the minimum and maximum pH cases (7.0-8.1). The licensee also calculated a second aluminum solubility curve at 20°F below the aluminum release rate temperature to show margin. The licensee also compared WCAP-16530-NP-A results to aluminum solubility vs temperature profiles. The comparison was conducted at the minimum pH and minimum water volume to minimize aluminum solubility while maximizing concentration. The comparison identified that the aluminum solubility limit is higher than the maximum aluminum concentration until the temperature falls to 145°F. Therefore, the ANL equation predicted that no aluminum would precipitate at temperatures above 145°F at the sump and maximum aluminum concentration.

After evaluating the licensee’s November 23, 2010, response, the NRC staff concludes that the licensee chemical effects test results have provided sufficient information to resolve chemical effects RAIs 39-44. The licensee performed three different head loss tests with plant-specific chemical precipitate loading as predicted by the WCAP-16530-NP-A methodology to demonstrate that aluminum precipitation was addressed with acceptable head loss results for Waterford 3. The NRC staff has previously reviewed and accepted the WCAP-16530-NP-A methodology.

In addition, the licensee incorporated the following conservatisms into the chemical effects evaluation:

- During testing, a full 30-day chemical precipitate load was assumed to arrive at the strainer at the earliest possible time with no credit for settling or nucleation on containment surfaces.

- Strainer head loss testing used chemical precipitate amounts and types according to the WCAP-16530-NP-A methodology.

- One hundred percent of dissolved aluminum and calcium was assumed to precipitate at the strainer and cause head loss. The licensee did not take credit for a long-term solubility limit, even though some aluminum can remain in solution long-term and not form precipitates.

- The maximum sump pH profile was used for aluminum corrosion predictions to maximize the aluminum dissolution rate and cumulative amount. The minimum sump pH profile was used to determine the aluminum precipitation temperature, which resulted in a maximized precipitation temperature.

NRC Staff Conclusion

For the chemical effects 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 Waterford 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 committed to change the FSAR for Waterford 3 in accordance with 10 CFR 50.71(e) to reflect the changes to the plant in support of the resolution to GL 2004-02. In addition, the licensee stated that changes would be made to the FSAR describing the new licensing basis to reflect the revised debris loading as it affects ECCS sump strainer performance and in-vessel effects.

NRC Staff Conclusion

Based on the licensee's commitment, the NRC has confidence that the licensee will affect the appropriate changes to the Waterford 3 Updated 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 staff 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 Waterford 3.

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SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 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: APRIL 25, 2024

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