Wolf Creek
Nuclear Operating Corporation

Stephen L. Smith
Engineering Vice President

August 12, 2021
ET 21-0005

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Docket No. 50-482: License Amendment Request for a Risk-Informed Resolution to GSI-191

Commissioners and Staff:

In accordance with the provisions of Title 10 of the Code of Federal Regulations (10 CFR Part 50.90), "Application for amendment of license, construction permit, or early site permit," Wolf Creek Nuclear Operating Corporation (WCNOC) is submitting a request for an amendment to Operating License NPF-42 for the Wolf Creek Generating Station (WCGS). The proposed amendment would revise the licensing basis as described in the Wolf Creek Updated Safety Analysis Report (USAR) to allow the use of a risk-informed approach to address safety issues discussed in Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance."

The proposed changes would allow the use of a risk-informed approach to address Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2004-02 for WCGS. The risk-informed approach is consistent with the guidance of NRC Regulatory Guide (RG) 1.174 and SECY-12-0093. The proposed changes will apply only for the effects of accident-generated debris as described in GSI-191 and GL 2004-02.

In addition, WCNOC proposes to revise the WCGS Technical Specifications (TS) to address containment accident generated and transported debris and the potential impact on the containment sumps.

Attachment I provides a description and technical basis for the proposed change. Attachment II contains a request for an exemption from certain requirements in 10 CFR 50.46. Attachment III provides the existing TS pages marked up to show the proposed change. Attachment IV provides revised (clean) TS pages. Attachment V provides the proposed TS Bases changes for information only. Attachment VI provides the proposed USAR changes for information only. Attachment VII provides an overview of the risk-informed approach to resolve GSI-191.

Attachment VIII provides WCNOC's updated response to NRC Generic Letter 2004-02. Attachment IX provides a discussion regarding the defense-in-depth approach and safety margin involved in the risk-informed approach. Attachment X provides WCNOC's planned approach to maintain GSI-191 compliance for information only.

WCNOC requests approval of this license amendment request by October 1, 2022. The license amendment, as approved, will be effective upon issuance and will be implemented within 90 days from the date of issuance.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," Section (b)(1), a copy of this amendment application, with Attachments, is being provided to the designated Kansas State official.

If you have any questions concerning this matter, please contact me at (620) 364-4156, or Ron Benham at (620) 364-4204.

Sincerely,

Stephen L. Smith

SLS/rlt

Attachments:

Attachments:	I	Evaluation of Proposed Change
	II	Request for Exemption
	III	Proposed Technical Specification Changes (Mark-Up)
	IV	Revised Technical Specification Pages
	V	Proposed Technical Specification Bases Changes (Mark-Up) for Information Only
	VI	Proposed USAR Changes (Mark-up) for Information Only
	VII	Overview of Risk-Informed Approach
	VIII	Updated Response to NRC Generic Letter 2004-02
	IX	Defense-in-Depth and Safety Margin
	X	Maintain GSI-191 Compliance for Information Only

cc: S. S. Lee (NRC), w/a
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Senior Resident Inspector (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Stephen L. Smith, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the contents thereof; that he has executed the same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By:______________________________
Stephen L. Smith
Vice President Engineering

SUBSCRIBED and sworn to before me this 12th day of August, 2021.

Rhonda L. Tiemeyer
Notary Public

RHONDA L. TIEMEYER
Notary Public, State of Kansas
My Appointment Expires
January 11, 2022

Expiration Date January 11, 2022

Wolf Creek Generating Station
Licensing Submittal for a Risk-Informed Resolution of Generic Letter 2004-02

Attachment I

Evaluation of Proposed Change

Table of Contents

Evaluation of Proposed Change

1.0 Implementation of a Risk-Informed Approach for Addressing GSI-191

1.1 Summary Description

Pursuant to 10 CFR 50.90, Wolf Creek Nuclear Operating Company (WCNOC) requests an amendment to Operating License NPF-42 for the Wolf Creek Generating Station (WCGS). The proposed amendment will revise the licensing basis as described in the Wolf Creek Updated Safety Analysis Report (USAR) to allow the use of a risk-informed approach to address safety issues discussed in Generic Safety Issue (GSI)-191, “Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance.”

The proposed changes would allow the use of a risk-informed approach to address Generic Letter (GL) 2004-02 (Reference 1) for WCGS. The risk-informed approach is consistent with the guidance of Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.174 (Reference 2) and SECY-12-0093 (Reference 3). The proposed changes will apply only for the effects of accident-generated debris as described in GSI-191 and GL 2004-02 (Reference 1).

In addition, as described in this Attachment, and Attachments III and IV of this submittal, WCNOC proposes to amend the WCGS operating license to revise the Technical Specifications (TS) to address containment accident generated and transported debris and the potential impact on the containment sumps (Reference 4).

1.2 Detailed Description

GSI-191 identifies the possibility that debris generated during a loss of coolant accident (LOCA) could clog the containment recirculation sump strainers in pressurized water reactors (PWRs) and result in loss of net positive suction head (NPSH) for the emergency core cooling system (ECCS) and containment spray system (CSS) pumps, impeding the flow of water from the sump. Additionally, debris that passes through the strainer could affect safety functions of the components downstream of the strainer or challenge long-term core cooling due to debris accumulation in the reactor core. GL 2004-02 requested licensees to address GSI-191 issues and demonstrate compliance with the ECCS acceptance criteria in 10 CFR 50.46 by performing analyses using an NRC-approved methodology. WCNOC’s response to GL 2004-02 is provided in Attachment VIII.

1.2.1 System Design and Operation

A fundamental function of the ECCS is to recirculate water that has collected in the containment sump 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 in excess of the plant's normal makeup capability (scenarios known as LOCAs), are part of a nuclear power plant's design bases. Hence, nuclear plants are designed and licensed with the expectation that they are able to remove reactor decay heat following a LOCA to prevent

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core damage. Long-term cooling following a LOCA is also a basic safety function for nuclear reactors. The recirculation sump located in the lower areas of the reactor containment structure provides a water source to the ECCS in a PWR once the initial water source has been depleted and the systems are switched over to recirculation mode for extended cooling of the core.

<u>ECCS</u>

As stated in the Wolf Creek USAR (Reference 5), the ECCS is designed to cool the reactor core and provide shutdown capability following initiation of the following accident conditions:

a. LOCA, including a pipe break or a spurious relief or safety valve opening in the RCS which would result in a discharge larger than that which could be made up by the normal makeup system.

b. Rupture of a control rod drive mechanism, causing a rod cluster control assembly ejection accident.

c. Steam or feedwater system break accident, including a pipe break or a spurious relief or safety valve opening in the secondary steam system which would result in an uncontrolled steam release or a loss of feedwater.

d. A steam generator tube failure.

The primary function of the ECCS is to provide emergency core cooling in the event of a LOCA resulting from a break in the primary RCS or to provide emergency boration in the event of a steam/or feedwater break accident.

Two residual heat removal (RHR) pumps are provided. Each pump is a single stage, vertical, centrifugal pump. In the event of a LOCA, the RHR pumps are started automatically on receipt of a safety injection signal (SIS). The RHR pumps take suction from the refueling water storage tank (RWST) during the injection phase of an accident and from the containment sump during the recirculation phase of an accident.

Two centrifugal charging pumps (CCPs) are provided. Each pump is a multistage diffuser design, barrel-type casing with vertical suction and discharge nozzles. In the event of an accident, the CCPs are started automatically on receipt of an SIS and are automatically aligned to take suction from the RWST during the injection phase of the accident. These high head pumps deliver flow through the boron injection tank to the RCS at the prevailing RCS pressure. During the recirculation phase of the accident, suction is provided from the RHR pump discharge.

Two safety injection pumps are provided. Each pump is a multi-stage, diffuser design, split-case centrifugal pump with side suction and side discharge. In the event of an accident, the safety injection pumps are started automatically on receipt of an SIS. The pumps take suction from the RWST via normally open, motor-operated valves, and deliver water to the RCS during the injection phase of the accident. During the recirculation phase of the accident, the pumps take suction from the containment sump via the RHR pumps.

The RWST serves as a source of emergency borated cooling water for injection and containment spray.

<u>Containment Spray System</u>

The CSS consists of two separate trains of equal capacity, each independently capable of meeting the design bases. Each train includes a containment spray pump, spray header and nozzles, spray additive eductor, valves, and the necessary piping, instrumentation, flushing connections, and controls. The RWST supplies borated injection water to the CSS. Each train takes suction from separate containment recirculation sumps during the recirculation phase of the accident.

1.2.2 Current Licensing Basis Requirements

10 CFR 50.46(a)(1) requires that the criteria set forth in 10 CFR 50.46 paragraph (b), be demonstrated with ECCS cooling performance calculated in accordance with an acceptable evaluation model and includes a requirement for "other properties" with regard to the methodology for showing those requirements are met. The methodology is governed by 10 CFR 50.46(a)(1) and is deterministic with no provision for a risk-informed approach. This license amendment request (LAR) supports an exemption to 10 CFR 50.46(a)(1) as described in Attachment II.

1.2.3 Reason for the Proposed Change

In order to meet a deterministic threshold value for sump debris loads, the debris sources in containment would need to be significantly reduced. The amount of radiological exposure received during the removal and/or modification of insulation from the WCGS containment is dependent on the scope of the changes. As discussed in Attachment II of this submittal, the expected dose for replacing insulation in the WCGS containment is estimated to be about 100 rem. This estimate does not include dose associated with disposal of the removed insulation or dose associated with insulation modifications for small-bore piping.

The dose considerations discussed above demonstrate that compliance would result in substantial personnel exposure due to insulation modifications in containment, which is not commensurate with the expected safety benefit based on the risk evaluation results showing that the risk associated with post-accident debris effects is less than the threshold for Region III in RG 1.174 (Reference 2).

1.2.4 Description of the Proposed Change

The proposed change in methodology in this LAR is to use a risk-informed approach to address the effects of accident-generated and transported debris on the containment emergency sumps instead of a deterministic approach. The details of the risk-informed approach are provided in Attachment VII. The debris analysis covers a full spectrum of postulated LOCAs, including double-ended guillotine breaks (DEGBs), to provide assurance that the most severe postulated LOCAs are evaluated.

Attachment VI provides markups to the USAR, which includes revision of applicable USAR safety system and design bases descriptions that take credit for the risk-informed evaluation that will be described in USAR Appendix 6A. The USAR markups are provided for information only.

1.3 Technical Evaluation

The methodology change affects the analysis of systems and functions that are susceptible to the effects of accident-generated debris. The affected systems are those supported by the containment recirculation sumps and strainers during the recirculation phase of LOCA mitigation. These include the ECCS and CSS.

The risk-informed approach identifies LOCA break scenarios that fail any one of the following GSI-191 acceptance criteria, as determined by break-specific analysis: blockage of flow paths upstream of the strainer, pump NPSH margin, strainer structural margin (i.e., differential pressure limit), void fraction limit, flashing limit, un-submerged strainer head loss limit (not applicable for WCGS because the strainer is fully submerged even for a small break LOCA), air ingestion due to vortexing limit, ex-vessel blockage and wear limits (downstream effects), and in-vessel effects limits.

The bounding threshold break size and the relevant LOCA frequency information were used to calculate the change in core damage frequency (ΔCDF) associated with the effects of debris. In addition, the conditional large early release probability (CLERP) determined from the WCGS probabilistic risk assessment (PRA) model was used to calculate the change in large early release frequency (ΔLERF). These values, along with the base CDF and LERF, were used to determine the risk region based on the acceptance guidelines in RG 1.174 (Reference 2). The results of the evaluation show that the risk from the proposed change is "very small" in that it is in Region III of RG 1.174. These results meet the requirement for the risk from debris to be small in paragraph (e) of the proposed 10 CFR 50.46c rule change (Reference 6) and associated draft RG 1.229 (Reference 7). See Attachment VII for a more detailed description of the risk quantification.

The proposed Wolf Creek USAR Appendix 6A (see Attachment VI) describes the risk-informed approach used to confirm that the ECCS and CSS will operate with a high probability following a LOCA when considering the impacts of accident-generated debris. This new appendix identifies the key elements of the risk-informed analyses. Future changes to the key elements are to be evaluated as a potential departure from a method of evaluation described in the USAR in accordance with 10CFR50.59(c)(2)(viii). The key elements include:

1. The methodology used to quantify the amount of debris generated at each break location, including the assumed zone of influence (ZOI) size based on the target destruction pressure and break size, and the assumed ZOI shape (spherical or

hemispherical) based on whether the break is a DEGB or partial break (see the Responses to 3.a and 3.b in Attachment VIII).

2. The methodology used to evaluate debris transport to the containment sump recirculation strainers (see the Response to 3.e in Attachment VIII).

3. The methodology used to quantify chemical precipitates, including the refinements to WCAP-16530-P-A, application of the solubility correlation, and application of the WCAP-17788-P autoclave testing (see the Responses to 3.n and 3.o in Attachment VIII).

4. The strainer debris limits shown in TS Bases Table B 3.6.8-1, which are based on tested and analyzed debris quantities (see Attachment V). Any changes to these debris limits are subject to 10 CFR 50.71(e) reporting requirements.

5. The methodology and acceptance criteria used to assess ex-vessel component blockage and wear (see the Response to 3.m in Attachment VIII).

6. The methodology used to assess in-vessel fiber accumulation and the associated limits (see the Response to 3.n in Attachment VIII).

7. The methodology used to quantify ΔCDF and ΔLERF (see Attachment VII).

The performance evaluations for accidents requiring ECCS operation are described in USAR Chapters 6 and 15 (Reference 5), based on the WCGS Appendix K large-break LOCA analysis. System redundancy, independence, and diversity features are not changed for those safety systems credited in the accident analyses. No new programmatic compensatory activities or reliance on manual operator actions are required to implement this change.

1.3.1 Engineering Analysis Overview

The design and licensing basis descriptions of accidents requiring ECCS and CSS operation, including analysis methods, assumptions, and results provided in USAR Chapters 6 and 15 remain unchanged. This is based on the functionality of the ECCS and CSS during design basis accidents being confirmed by demonstrating that safety margin and defense-in-depth (DID) are maintained with high probability (Attachment IX).

The methodology for calculating the risk associated with GSI-191 concerns evaluates a full spectrum of breaks up to and including DEGBs for all RCS pipe sizes. The results show that the risk associated with GSI-191 concerns for WCGS is "very small" as defined by Region III in RG 1.174 (Reference 2). The detailed technical description of the risk quantification process is presented in Attachment VII.

This LAR is requesting a change to the licensing basis such that the effects of LOCA generated and transported debris can be evaluated using a risk-informed methodology.

Detailed evaluations of DID and safety margin are presented in Attachment IX. The evaluations determined that there is substantial DID and safety margin that provides a high level of confidence that the calculated risk is conservative and that the actual risk is likely much lower.

1.3.2 Conclusion for Technical Evaluation

The technical evaluation shows that the functionality of the ECCS and CSS during design basis accidents is confirmed by demonstrating that safety margin and DID are maintained with high probability.

2.0 Implementation of TSTF-567

2.1 Description

WCNOC requests revision to the WCGS TS to address post-accident debris effects on the containment sump. The selected TS changes follow the model application in TSTF-567, "Add Containment Sump TS to Address GSI-191 Issues," Revision 1 (Reference 4).

The proposed amendment adds a new TS 3.6.8, "Containment Sump," and adds an Action to address the condition of the containment sump made inoperable due to containment accident generated and transported debris exceeding the analyzed limits. The Action provides time to correct or evaluate the condition in lieu of an immediate plant shutdown. This Action is placed in a new specification on the containment sump that otherwise retains the existing TS requirements. An existing Surveillance Requirement (SR) is moved from TS 3.5.2 to the new specification. The requirement to perform the SR in TS 3.5.3 is deleted.

The regulatory evaluation and environmental consideration for the proposed TS changes are addressed in Sections 3.0 and 4.0 of this Attachment, respectively.

2.2 Assessment

2.2.1 Applicability of Safety Evaluation

WCNOC has reviewed the safety evaluation for TSTF-567 provided to the Technical Specifications Task Force in a letter dated July 3, 2018 (Reference 8). This review included the NRC staff's safety evaluation, as well as the information provided in TSTF-567. As described herein, WCNOC has concluded that the justifications presented in TSTF-567 and the safety evaluation prepared by the NRC staff are applicable to WCGS and justify this amendment for the incorporation of the changes to the WCGS TS.

2.2.2 Variations

WCNOC is proposing the following variations from the TS changes described in TSTF-567 or the applicable parts of the NRC staff's safety evaluation. These variations do not

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affect the applicability of TSTF-567 or the NRC staff's safety evaluation to the proposed license amendment.

1. The WCGS TS utilize different numbering than the Standard TS on which TSTF-567 was based. Specifically, the new containment sump specification, TS 3.6.19, in TSTF-567 (Reference 4) is TS 3.6.8 in the WCGS TS. This difference is administrative and does not affect the applicability of TSTF-567 to the WCGS TS.

2. The required action and notes of proposed Condition B in TSTF-567, Revision 1, are revised to require declaring the affected ECCS and CSS trains inoperable immediately. The proposed Condition B in TSTF-567 requires that the containment sump be restored to operable status within a specific completion time and has two notes requiring entry into the associated ECCS and CSS TS actions. In addition, the proposed Condition C in TSTF-567 is revised to state, "Required Action and associated Completion Time of Condition A not met," since the proposed required actions of Condition B to "declare" affected ECCS and CSS trains inoperable are immediate and can easily be accomplished. The TS Bases markups have also been revised to reflect the changes to the actions. These changes have minimal impact and are further explained, as follows.

   When the containment sump is inoperable for reasons other than Condition A, such as blockage, structural damage, or abnormal corrosion that could prevent recirculation of coolant, one or more ECCS or CSS trains are rendered inoperable. Therefore, declaring the affected trains inoperable immediately will ensure appropriate restrictions are implemented in accordance with the required actions of the ECCS and CSS TS.

   As indicated in TSTF-567, Revision 1, the completion time of TSTF-567 Required Action B.1 is specified as either 72 hours or depending on the completion time established for a single inoperable ECCS or CSS train. This action is redundant to those required when one ECCS or CSS train is inoperable since the proposed Required Action B.1 in TSTF-567 requires the actions of TS 3.5.2, TS 3.5.3 and TS 3.6.6 to be applied. The required actions proposed for WCGS (see Required Actions B.1 and B.2 in Attachment III) achieve the same goal while providing simplified action requirements.

   This variation is considered administrative since the proposed requirements will result in equivalent action taken for the condition and, therefore, does not affect the applicability of the TSTF-567 model application or the NRC staff's model safety evaluation for the proposed TS changes.

3. The containment sump debris limits are provided in the TS Bases, Table B 3.6.8-1, instead of the USAR. This is an administrative change to place the debris limits in a more convenient location for the operators. Any changes to these debris limits are subject to review under 10 CFR 50.59 and 10 CFR 50.71(e) reporting requirements. The USAR markup in Attachment VI refers the USAR reader to TS Bases, Table B 3.6.8-1, for the analyzed debris limits.

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3.0 Regulatory Evaluation

3.1 Applicable Regulatory Requirements/Criteria

Approval of the proposed amendment is contingent upon approval of the request for exemption from certain aspects of 10 CFR 50.46(a)(1) as provided and justified in Attachment II of this submittal.

<u>Regulatory Guide 1.174</u>

NRC RG 1.174 (Reference 2) provides the NRC staff's recommendations for using risk information in support of licensee-initiated licensing basis changes to a nuclear power plant that require NRC review and approval. This RG describes an acceptable approach for assessing the nature and impact of proposed licensing basis changes by considering engineering issues and applying risk insights.

In implementing risk-informed decision-making, licensing basis changes are expected to meet a set of key principles. These principles include the following:

(1) <u>The proposed change meets the current regulations unless it is explicitly related to a requested exemption (i.e., a specific exemption under 10 CFR 50.12, "Specific Exemptions").</u>

The exemption requested in Attachment II of this submittal complies with this requirement.

(2) <u>The proposed change is consistent with a DID philosophy.</u>

Defense-in-depth is presented in detail in Attachment IX of this submittal. The proposed change is consistent with the DID philosophy in that the following aspects of the facility design and operation are unaffected:

- Functional requirements and the design configuration of systems
- Existing plant barriers to the release of fission products
- Design provisions for redundancy, diversity, and independence
- Plant response to transients or other initiating events
- Preventive and mitigative capabilities of plant design features

The WCGS risk-informed approach analyzes a full spectrum of postulated LOCAs, including DEGBs for all piping sizes up to and including the largest pipe in the RCS. By requiring that mitigative capability be maintained in a risk-informed evaluation of GSI-191 for a full spectrum of LOCAs, the approach ensures that DID is maintained.

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(3) The proposed change maintains sufficient safety margins.

As described in Attachment IX of this submittal, sufficient safety margins associated with the design will be maintained by the proposed change.

(4) When proposed changes result in an increase in CDF or risk, the increases should be small and consistent with the intent of the Commission's safety goal policy statement.

The proposed change involves evaluation of the risk associated with effects of accident-generated and transported debris using a risk-informed methodology. Using engineering analysis and the PRA, this risk has been calculated and shown to be "very small" as defined by Region III in RG 1.174 (Reference 2) and is therefore consistent with the Commission's safety goal policy statement.

(5) The impact of the proposed change should be monitored using performance measurement strategies.

WCGS has implemented procedures and programs for monitoring, controlling, and assessing changes to the plant that have a potential impact on plant performance related to GSI-191 concerns. The following procedures and programs provide the capability to monitor the performance of the sump strainers and assess impacts to the inputs and assumptions used in the PRA and associated engineering analyses that support the proposed change.

- The WCGS TS surveillance procedures ensure that the sump strainers do not have openings in excess of the strainer's maximum designed opening. Additionally, the TS surveillance requires visual inspection to ensure each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet strainers show no evidence of structural distress or abnormal corrosion.
- The WCGS design change process requires the use of Design Attribute Review (DAR) forms to identify potential impact on long term core cooling by the proposed modification. Questions in the DAR forms that are related to GSI-191 include:
    - Does the modification affect insulation?
    - Does the modification add or remove components in containment?
    - Does the modification change the amount of exposed aluminum and/or zinc in containment?
    - Does the modification introduce materials that could affect sump performance or lead to equipment degradation?
    - Does the modification repair, replace, or install coatings inside containment, including installing coated equipment?

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   - Does the modification affect installation, replacement, or storage of any structure, system, component or other items in containment that has vendor applied or site applied protective coatings?
   - Does the modification affect high/moderate energy line break analysis?
   - Does the modification affect the design, performance or operation of pumps?
   - Does the modification affect foreign material that would require cleaning to prevent degradation of downstream components?
- A 10 CFR 50.59 screening or evaluation is required to be completed for all design changes.
- The WCGS containment coatings program monitors and assesses the containment building coatings, and establishes administrative controls for conducting coating examinations, deficiency reporting, and documentation.
- As part of the WCGS condition reporting process, condition reports are written when adverse conditions are identified during containment inspections or during surveillances of the containment emergency sumps and strainers. Documentation and evaluation of nonconformances are discussed in Attachment VII.
- The WCGS Maintenance Rule program includes performance monitoring of the high safety significant functions associated with the ECCS and CSS. The Maintenance Rule program provides continued assurance of the availability and reliability for performance of the required functions.
- The on-line configuration risk management procedure establishes the administrative controls for performing on-line maintenance of structures, systems, and components (SSCs) to enhance overall plant safety and reliability.
- The WCGS quality assurance (QA) program is implemented and controlled in accordance with the Quality Assurance Topical Report (QATR) and is applicable to SSCs to an extent consistent with their importance to safety. The QA program complies with the requirements of 10 CFR 50, Appendix B and other program commitments as appropriate.

The proposed change does not alter the ASME Section XI inspection programs or mitigation strategies that have been shown to be effective in early detection and mitigation of weld and material degradation in Class I piping applications.

Periodic updates to the risk-informed GSI-191 analysis will be performed to capture the effects of any plant changes, procedure changes, or new information on the risk-informed analysis and to confirm that the results are still within the established acceptance criteria (see Attachment VII).

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Regulatory Guide 1.200

NRC RG 1.200 (Reference 9) describes one acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for light water reactors. As described in Attachment VII of this submittal, the WCGS PRA model used for the risk-informed GSI-191 evaluation complies with RG 1.200 (Reference 9).

3.2 Precedent

The proposed licensing change for WCGS is very similar to the license amendment and 10 CFR 50.46(a)(1) exemption granted to South Texas Project Nuclear Operating Company for implementation of the risk-informed approach to address GSI-191 concerns at South Texas Project (STP), Units 1 and 2 (References 10 and 11).

WCGS requests implementation of a risk-informed methodology for resolution of GSI-191 that is similar to the approach used by STP. Key similarities include, but are not limited to:

1. Use of RG 1.174 acceptance guidelines and key principles.
2. Identification of key methods and approaches in the risk-informed methodology that, if changed after implementation, are to be evaluated as a potential "departure from a method of evaluation described in the USAR" under 10 CFR 50.59.
3. Associated request for exemption from 10 CFR 50.46(a)(1) "other properties."
4. Technical Specification changes that provide for additional time to address the effects of debris on ECCS and CSS operability.

Key differences include, but are not limited to:

1. No software was used for the WCGS risk analysis, while CASA Grande was used for the STP analysis.
2. The methodology used for the WCGS risk quantification (threshold break approach) differs from the methodology used by STP (risk over deterministic, RoverD, approach).
3. STP requested exemption from 10 CFR 50, Appendix A, General Design Criteria 35, 38 and 41. WCGS determined that an exemption from the general design criteria is not necessary. The approval of WCGS's request for exemption from 10 CFR 50.46(a)(1) "other properties" and approval of this LAR would provide an acceptable alternative to meet the intent of the stated General Design Criteria.
4. The WCGS TS changes follow the TSTF-567, Revision 1 model application, which was not available at the time of the STP LAR submittal.

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3.3 No Significant Hazards Consideration

The proposed amendment implements a risk-informed approach to address the effects of accident-generated and transported debris on the containment emergency sumps.

The proposed amendment also proposes changes to the Wolf Creek Generating Station (WCGS) Technical Specifications (TS). The proposed TS changes add a new TS 3.6.8, "Containment Sump," and add an Action to address the condition of the containment sump made inoperable due to containment accident generated and transported debris exceeding the analyzed limits. The Action provides time to correct or evaluate the condition in lieu of an immediate plant shutdown. This Action is placed in a new specification on the containment sump that otherwise retains the existing TS requirements. An existing Surveillance Requirement (SR) is moved from TS 3.5.2 to the new specification. The requirement to perform the SR in TS 3.5.3 is deleted.

Wolf Creek Nuclear Operating Corporation (WCNOC) has evaluated whether a significant hazards consideration is involved with the proposed change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

(1) <u>Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?</u>

Response: No

The proposed change is a methodology change for assessment of debris effects that adds the results of a risk-informed evaluation to the WCGS licensing basis. This is a viable approach for the resolution of GSI-191 per SECY-12-0093. The analysis that supports the methodology change concludes that the functionality of the emergency core cooling system (ECCS) and containment spray system (CSS) during design basis accidents is confirmed by demonstrating that safety margin and defense-in-depth (DID) are maintained with high probability.

There is no significant increase in the probability of an accident previously evaluated. The proposed change addresses mitigation of loss of coolant accidents (LOCAs) and has no effect on the probability of the occurrence of a LOCA. The proposed methodology change does not implement any changes in the facility or plant operation that could lead to a different kind of accident.

The proposed change does not involve a significant increase in the consequences of an accident previously evaluated. The methodology change confirms that required structures, systems, and components (SSCs) supported by the containment sumps will perform their safety functions with a high probability, as required, and does not alter or prevent the ability of SSCs to perform their intended function to mitigate the consequences of an accident previously evaluated within the acceptance limits. The proposed change has no impact on existing barriers that prevent the release of radioactivity. The safety analysis acceptance criteria in

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the Updated Safety Analysis Report (USAR) continue to be met for the proposed methodology change.

The proposed change also adds a new specification to the TS for the containment sump. An existing SR on the containment sump is moved to the new specification and a duplicative requirement to perform the SR in TS 3.5.3 is removed. The new specification retains the existing requirements on the containment sump and the actions to be taken when the containment sump is inoperable with the exception of adding new actions to be taken when the containment sump is inoperable due to containment accident generated and transported debris exceeding the analyzed limits. The new action provides time to evaluate and correct the condition instead of requiring an immediate plant shutdown.

The containment sump is not an initiator of any accident previously evaluated. The containment sump is a passive component and the proposed change does not increase the likelihood of the malfunction. As a result, the probability of an accident is unaffected by the proposed change.

The containment sump is used to mitigate accidents previously evaluated by providing a borated water source for the ECCS and CSS. The design of the containment sump and the capability of the containment sump assumed in the accident analysis is not changed. The proposed action requires implementation of mitigating actions while the containment sump is inoperable and more frequent monitoring of reactor coolant leakage to detect any increased potential for an accident that would require the containment sump. The consequences of an accident during the proposed action are no different than the current consequences of an accident if the containment sump is inoperable.

Therefore, the proposed methodology and TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) <u>Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?</u>

Response: No

The proposed change is a methodology change for assessment of debris effects from LOCAs and secondary side breaks that are already evaluated in the Wolf Creek USAR. No new or different kind of accident is being evaluated. The proposed change does not install or remove any plant equipment, or alter the design, physical configuration, or mode of operation of any plant SSCs. The proposed change does not introduce any new failure mechanisms or malfunctions that can initiate an accident.

Evaluation of Proposed Change

The proposed change also adds a new specification to the TS for the containment sump. An existing SR on the containment sump is moved to the new specification and a duplicative requirement to perform the SR in TS 3.5.3 is removed. The new specification retains the existing requirements on the containment sump and the actions to be taken when the containment sump is inoperable with the exception of adding new actions to be taken when the containment sump is inoperable due to containment accident generated and transported debris exceeding the analyzed limits. The new action provides time to evaluate and correct the condition instead of requiring an immediate plant shutdown.

The proposed change does not alter the design or design function of the containment sump or the plant. No new systems are installed or removed as part of the proposed change. The containment sump is a passive component and cannot initiate a malfunction or accident. No new credible accident is created that is not encompassed by the existing accident analyses that assume the function of the containment sump.

Therefore, the proposed methodology and TS changes do not create the possibility of a new or different kind of accident from any previously evaluated.

(3) <u>Does the proposed amendment involve a significant reduction in a margin of safety?</u>

Response: No

The proposed change is a methodology change for assessment of debris effects from LOCAs and secondary side breaks that are already evaluated in the Wolf Creek USAR. The effects from a full spectrum of LOCAs and secondary side breaks inside containment, including double-ended guillotine breaks, are analyzed. Appropriate redundancy and consideration of loss of offsite power and worst-case single failure are retained, such that DID is maintained.

Application of the risk-informed methodology showed that the increase in risk from the contribution of debris effects is very small as defined by RG 1.174 and that there is adequate DID and safety margin. DID and safety margin were extensively evaluated. This evaluation showed that there is substantial DID and safety margin that provide a high level of confidence that the calculated risk for the effects of debris is conservative and that the actual risk is likely much lower. Consequently, WCGS determined that the risk-informed method demonstrates the containment sumps will continue to support the ability of safety-related components to perform their design functions when the effects of debris are considered. Note that the risk-informed approach was identified as viable for the resolution of GSI-191 per SECY-12-0093.

Evaluation of Proposed Change

The proposed change does not alter the manner in which safety limits are determined or the acceptance criteria associated with a safety limit. The proposed change does not implement any changes to plant operation and does not significantly affect SSCs that respond to safely shutdown the plant and to maintain the plant in a safe shutdown condition. The proposed change does not significantly affect the existing safety margins in the barriers for the release of radioactivity. There are no changes to any of the safety analyses in the USAR.

The proposed change also adds a new specification to the TS for the containment sump. An existing SR on the containment sump is moved to the new specification and a duplicative requirement to perform the SR in TS 3.5.3 is removed. The new specification retains the existing requirements on the containment sump and the actions to be taken when the containment sump is inoperable with the exception of adding new actions to be taken when the containment sump is inoperable due to containment accident generated and transported debris exceeding the analyzed limits. The new action provides time to evaluate and correct the condition instead of requiring an immediate plant shutdown.

The proposed change does not affect the controlling values of parameters used to avoid exceeding regulatory or licensing limits. No Safety Limits are affected by the proposed change. The proposed change does not affect any assumptions in the accident analyses that demonstrate compliance with regulatory and licensing requirements.

Therefore, the proposed methodology and TS changes do not involve a significant reduction in a margin of safety.

Based on the above, WCNOC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

3.4 Conclusions for Regulatory Evaluation

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the license amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.0 Environmental Consideration

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or SR. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

5.0 References

1. NRC Generic Letter (GL) 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents for Pressurized-Water Reactors,” September 13, 2004.
2. Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Revision 3, January 2018.
3. SECY-12-0093 (ML121310648), “Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance,” July 9, 2012.
4. ML17214A813, TSTF-567, “Add Containment Sump TS to Address GSI-191 Issues,” Revision 1, August 2, 2017.
5. Wolf Creek Updated Safety Analysis Report (USAR), Revision 33, March 2020.
6. ML15238A947, “SECY-16-0033: Final Draft Rulemaking - 10 CFR 50.46c: Emergency Core Cooling Systems Performance During Loss-of-Coolant Accidents,” April 4, 2016.
7. Draft Regulatory Guide 1.229 (ML16062A016), “Risk-Informed Approach for Addressing the Effects of Debris on Post-Accident Long-Term Core Cooling,” March 2016.
8. ML18109A071, “Final Safety Evaluations of Technical Specifications Task Force Traveler TSTF-567, Revision 1, “Add Containment Sump TS to Address GSI-191 Issues” (EPID: L-2017-PMP-0005),” July 3, 2018; Enclosure 1 (ML18116A606), Enclosure 2 (ML18116A599), Enclosure 3 (ML18117A453).
9. Regulatory Guide 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Revision 2, March 2009.
10. ML17019A001, “South Texas Project, Units 1 and 2 – Issuance of Amendment Nos. 212 and 198 – Risk-Informed Approach to Resolve Generic Safety Issue 191 (CAC Nos. MF2400 and MF2401),” July 12, 2017.
11. ML17037C871, “South Texas Project, Units 1 and 2, Exemptions from the Requirements of 10 CFR Part 50, Section 50.46 and 10 CFR Part 50, Appendix A, General Design Criteria 35, 38, and 41 (CAC Nos. MF2402-MF2409),” July 11, 2017.

Wolf Creek Generating Station
Licensing Submittal for a Risk-Informed Resolution of Generic Letter 2004-02

Attachment II

Request for Exemption

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Request for Exemption

1.0 GENERAL

1.1 Introduction

In support of the Wolf Creek Generating Station (WCGS) risk-informed approach to address Generic Safety Issue 191 (GSI-191) and response to the United States Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2004-02 (Reference 1), this Attachment provides Wolf Creek Nuclear Operating Corporation's (WCNOC) request for exemption under Title 10 of the Code of Federal Regulations (CFR) Section 50.12 (10 CFR 50.12) from certain requirements in 10 CFR 50.46. This exemption request complements a license amendment request (LAR) provided in Attachment I of this submittal for adopting a risk-informed approach for addressing GSI-191. Changes to the WCGS Technical Specifications (TS) and Updated Safety Analysis Report (USAR) will be made as described in Attachments III and VI in this submittal. Attachment VII discusses the risk quantification, and Attachment VIII contains the updated GL 2004-02 response. Attachment IX discusses defense-in-depth and safety margins, and Attachment X discusses maintaining GSI-191 compliance.

The specific exemption request pertains to requirements associated with the emergency core cooling system (ECCS) function for core cooling following a postulated loss of cooling accident (LOCA). The scope and key elements of the requested exemption are described in Section 2.0.

Approval of the exemption will allow the use of a risk-informed method to account for the probabilities and uncertainties associated with mitigation of the effects of debris following postulated LOCAs. The method evaluates concerns raised by GSI-191 related to the effects of post-accident debris on the containment sump recirculation strainers and reactor core blockage due to debris in the recirculating fluid. To confirm acceptable sump design, the risk associated with loss of core cooling due to the effects of debris is evaluated. The risk-informed approach is designed to be consistent with the guidance in Regulatory Guide (RG) 1.174 (Reference 2).

The WCGS approach is the risk-informed part of an overall graded approach that is based on the amount of debris in the plant, as discussed in SECY-12-0093 (Reference 3). The WCGS risk-informed approach addresses the five key principles in RG 1.174 (Reference 2). The resulting risk metrics (i.e., CDF, LERF, ΔCDF, and ΔLERF) are used to determine whether plant modifications are warranted to ensure acceptable sump performance. The WCGS risk quantification in Attachment VII of this submittal shows that ΔCDF and ΔLERF are below the threshold for RG 1.174 Region III, "Very Small Changes," without further plant or procedure modifications. Therefore, the risk-informed approach provides an equivalent level of assurance for sump performance without incurring significant cost and occupational dose associated with removing, replacing, or reinforcing insulation in containment. Approval of the requested exemption will support the application of the risk-informed approach.

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1.2 Background and Overview

GSI-191 identifies the possibility that debris generated during a LOCA could clog the containment sump strainers in pressurized water reactors (PWRs) and result in loss of net positive suction head (NPSH) for the ECCS and containment spray system (CSS) pumps, impeding the flow of water from the sump. GL 2004-02 requested licensees to address GSI-191 by demonstrating compliance with the 10 CFR 50.46 ECCS acceptance criteria (Reference 1). In addition, GL 2004-02 required licensees to address downstream effects. As stated in GL 2004-02, licensees were requested to perform analyses using an NRC-approved methodology and to ensure successful operation of the ECCS and CSS during design-basis accidents (DBAs) that require containment sump recirculation (Reference 1):

> *"Although not traditionally considered as a component of the 10 CFR 50.46 ECCS evaluation model, the calculation of sump performance is necessary to determine if the sump and the ECCS are predicted to provide enough flow to ensure long-term cooling.*
>
> *Based on the new information identified during the efforts to resolve GSI-191, the staff has determined that the previous guidance used to develop current licensing basis analyses does not adequately and completely model sump screen debris blockage and related effects. As a result, due to the deficiencies in the previous guidance, an analytical error could be introduced which results in ECCS and CSS performance that does not conform to the existing applicable regulatory requirements outlined in this generic letter. Therefore, the staff is revising the guidance for determining the susceptibility of PWR recirculation sump screens to the adverse effects of debris blockage during design basis accidents requiring recirculation operation of the ECCS or CSS. In light of this revised staff guidance, it is appropriate to request that addressees perform new, more realistic analyses and submit information to confirm the functionality of the ECCS and CSS during design basis accidents requiring recirculation operations."*

In addition, GL 2004-02 identified the following regulatory requirement (Reference 1):

> *"NRC regulations in Title 10, of the Code of Federal Regulations Section 50.46, 10 CFR 50.46, require that the ECCS has the capability to provide long-term cooling of the reactor core following a LOCA. That is, the ECCS must be able to remove decay heat, so that the core temperature is maintained at an acceptably low value for the extended period of time required by the long-lived radioactivity remaining in the core."*

Compensatory and mitigative measures have been implemented in response to Bulletin 2003-01 (Reference 4) and GL 2004-02 (Reference 1) to address the potential for sump strainer clogging and related GSI-191 concerns. This included installation of larger containment sump strainers that greatly reduce the potential for loss of NPSH for the RHR

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and CSS pumps. The station design and compliance with GL 2004-02 are addressed in Attachment VIII. Defense-in-depth measures are described in Attachment IX.

The Commission issued Staff Requirements Memorandum (SRM) SECY-10-0113 directing the staff to consider alternative options for resolving GSI-191 (Reference 5). Subsequently, SECY-12-0093 outlined a few different options that PWR licensees can use to address GSI-191, including deterministic and risk-informed approaches (References 3 and 6). In a letter to the NRC on May 16, 2013 (Reference 7), WCNOC selected Option 2, the full risk-informed resolution path.

Based on the guidance in RG 1.174, the risk-informed approach requires an exemption from certain requirements of 10 CFR 50.46 in accordance with 10 CFR 50.12.

2.0 EXEMPTION REQUEST

Pursuant to 10 CFR 50.12, WCNOC is submitting this request for exemption from certain requirements of 10 CFR 50.46(a)(1), “other properties,” as it relates to using specific deterministic methodology to evaluate the effects of debris on long-term core cooling.

10 CFR 50.46(a)(1) is shown below with the “other properties” portion for which exemption is requested in bold.

> *“(a)(1)(i) Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations,* ***and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated. Except as provided in paragraph (a)(1)(ii) of this section, the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident.*** *Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. Appendix K, Part II Required Documentation, sets forth the documentation requirements for each evaluation model. This section does not apply to a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted.*

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(ii) Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of appendix K ECCS Evaluation Models."

The scope of the exemption applies to all debris effects addressed in the risk-informed element of the WCGS methodology described in Attachment VII. The debris effects are associated with those breaks that potentially generate and transport debris that exceeds the analyzed debris limits. The key elements of the exemption request are listed as follows. WCNOC is requesting exemption for these breaks to allow evaluation of the debris effects using a risk-informed methodology.

1. The exemption will apply only to the effects of debris as described in Attachment VII.
2. The exemption will apply to any breaks that can generate and transport debris that is not bounded by WCGS-specific analyzed limits, provided that the ΔCDF and ΔLERF remain in RG 1.174 Region III (Reference 2).

This exemption request is complemented by the accompanying LAR in Attachment I, which seeks NRC approval to amend the licensing basis based on acceptable design of the containment sump. The risk-informed method provides a high probability of assurance for acceptable sump performance and debris mitigation as assumed in the ECCS evaluation model.

The WCGS risk-informed approach to addressing GSI-191 and responding to GL 2004-02 is consistent with the NRC safety evaluation (SE) for NEI 04-07 that discussed the modeling of sump performance as follows (Reference 9):

"While not a component of the 10 CFR 50.46 ECCS evaluation model, the calculation of sump performance is necessary to determine if the sump and the residual heat removal system are configured properly to provide enough flow to ensure long-term cooling, which is an acceptance criterion of 10 CFR 50.46. Therefore, the staff considers the modeling of sump performance as the validation of assumptions made in the ECCS evaluation model. Since the modeling of sump performance is a boundary calculation for the ECCS evaluation model, and acceptable sump performance is necessary for demonstrating long-term core cooling capability (10 CFR 50.46(b) (5)), the requirements of 10 CFR 50.46 are applicable."

This exemption request is consistent with the provisions of the proposed ECCS rule change. The following statement, found on Page 85 of the proposed 10 CFR 50.46c final rule change package attached to SECY-16-0033 (Reference 10), applies to the new 10 CFR 50.46c(d), which will replace the current 10 CFR 50.46(a)(1):

"Demonstration of consideration of such factors may also be achieved through analytical models that adequately represent the empirical data obtained regarding debris deposition. The final rule alternatively allows the use of risk-informed

approaches to evaluate the effects of debris on localized coolant flow and delivery of coolant to the core during the long-term cooling (post-accident recovery) period."

The proposed ECCS rule change will allow use of a risk-informed approach, addressed in 10 CFR 50.46c(e), in lieu of a deterministic evaluation. Similar to the proposed new rule change, WCNOC's risk-informed approach is an alternative to the current deterministic evaluation required by 10 CFR 50.46(a)(1). WCNOC requires exemption from 10 CFR 50.46(a)(1) "other properties" since there currently is no risk-informed evaluation alternative. WCNOC requests an exemption from those deterministic requirements in order to enable the use of a risk-informed method to demonstrate acceptable sump performance and debris mitigation, and to validate assumptions in the ECCS evaluation model.

3.0 REGULATORY REQUIREMENTS INVOLVED

By regulatory precedent, licensees are required to demonstrate compliance with the relevant regulations by the use of a bounding calculation or other deterministic method. WCNOC seeks exemption to the extent that 10 CFR 50.46(a)(1) "other properties" requires deterministic calculations or other analyses to address the concerns raised by GSI-191 related to acceptable plant performance during the recirculation mode following a LOCA. The proposed changes to the licensing basis and technical specifications, submitted for NRC approval through an LAR (see Attachment I), address GSI-191 and GL 2004-02 for WCGS on the basis that the associated risk is shown to meet the acceptance guidelines in RG 1.174 and adequate defense-in-depth and safety margin are demonstrated.

This exemption request is to allow the use of a risk-informed method to demonstrate acceptable mitigation of the effects of debris following postulated LOCAs. Prior to the risk-informed approach, deterministic methods were used to evaluate the effects of accident-generated and transported debris in order to meet the current licensing basis assumptions for analyzing the effects of post-accident debris blockage in the sump and in-vessel. However, these evaluations did not address debris effects fully for the as-built, as-operated plant conditions. The risk-informed approach evaluates the debris effects as part of the assessment of the residual risk associated with GSI-191 concerns. Based on confirmation of acceptable ECCS design as determined by the resulting risk meeting the acceptance guidelines in RG 1.174, the licensing basis for ECCS compliance with 10 CFR 50.46(a)(1) can be amended.

The exemption request to support closure of GL 2004-02 for WCGS is intended to address ECCS cooling performance design as presented in 10 CFR 50.46(a)(1) as it relates to imposing the deterministic requirements in "other properties." For the purposes of demonstrating the balance of the acceptance criteria of 10 CFR 50.46, the design and licensing basis descriptions of accidents requiring ECCS operation remain unchanged, as documented in WCGS USAR Chapters 6 and 15, including analysis methods, assumptions, and results. The performance evaluations for accidents requiring ECCS operation described in USAR Chapters 6 and 15 are based on the Appendix K large-

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break loss-of-coolant accident (LBLOCA) analysis and demonstrate that, for breaks up to and including a double ended guillotine break (DEGB) of a reactor coolant pipe, the ECCS will limit the clad temperature to below the limit specified in 10 CFR 50.46 and the core will remain in place and substantially intact with its essential heat transfer geometry preserved.

The requirements of 10 CFR 50.46(a)(1) remain applicable to the model of record that meets the required features of Appendix K. Approval of the requested exemption does not impact the current ECCS evaluation. The ECCS evaluation model remains the licensing basis for demonstrating that the ECCS calculated cooling performance following postulated LOCAs meets the acceptance criteria.

The WCGS risk-informed approach determines strainer and core blockage conditional failure probabilities that are input into the plant PRA model to determine the ΔCDF and ΔLERF associated with debris-related failures, as described in Attachment VII of this submittal. The results show that WCGS meets the acceptance guidelines defined in RG 1.174 (Reference 2). The exemption request is specific to the requirement for demonstrating ECCS cooling performance design as required by 10 CFR 50.46(a)(1) as it pertains to the requirements for deterministic analyses described in “other properties.” It is not intended to be applicable to other requirements provided in 10 CFR 50.46 or Appendix K to 10 CFR 50.

As noted in Section 1.2, the NRC staff considers the modeling of sump performance to be an input to the ECCS evaluation model, and therefore the requirements of 10 CFR 50.46 are applicable. Consistent with this, the requirements and attributes for the proposed WCGS risk-informed method include a full spectrum of postulated breaks, up to and including DEGBs on the largest reactor coolant system (RCS) pipes in containment, as described in Attachment VII of this submittal.

Engineering analyses and evaluations used to perform plant-specific prototypical testing consider a wide range of effects, including those addressed in NEI 04-07 (Reference 8) and its associated NRC SE (Reference 9) for evaluation of sump performance. The requested exemption does not affect any of the 10 CFR 50.46(a)(1) or Appendix K requirements for an acceptable ECCS evaluation model and does not change the ECCS acceptance criteria in 50.46(b) as it applies to the calculated results. Application of the exemption request allows the use of a risk-informed approach to evaluate the effects of debris. The results of the risk-informed method demonstrate that the risk associated with GSI-191 meets the acceptance guidelines of RG 1.174 (Reference 2). The current licensing basis for addressing the adequacy of the ECCS to meet the criteria of 10 CFR 50.46, including the Appendix K large-break LOCA analysis and the associated Chapter 15 accident analysis for LOCA, remains in place.

4.0 BASIS FOR THE EXEMPTION REQUEST

Under 10 CFR 50.12, a licensee may request and the NRC may grant exemptions from the requirements of 10 CFR 50 that are authorized by law, will not present an undue risk to the public health and safety, are consistent with the common defense and security, and when special circumstances are present.

The exemption request meets a key principle of RG 1.174, which states, “The proposed licensing basis change meets the current regulations unless it is explicitly related to a requested exemption” (Reference 2). This exemption request is provided in conjunction with the proposed changes provided in the risk-informed LAR (see Attachment I).

As required by 10 CFR 50.12(a)(2), the Commission will not consider granting an exemption unless special circumstances are present. Special circumstances are present whenever one of the listed items (i through vi) under 10 CFR 50.12(a)(2) are applicable.

WCNOC has evaluated the requested exemption against the conditions specified in 10 CFR 50.12(a) and determined that this requested exemption meets the requirements for granting an exemption from the regulation, and that special circumstances are present. The information supporting the determination is provided below.

4.1 Applicability of 10 CFR 50.12(a)(1)

Pursuant to 10 CFR 50.12, “Specific exemptions,” the NRC may grant exemptions from the requirements of this part provided the following three conditions are met as required by 10 CFR 50.12(a)(1):

1. The exemption is authorized by law.

   The NRC has authority under the Atomic Energy Act of 1954, as amended, to grant exemptions from its regulations if doing so would not violate the requirements of law. This exemption is authorized by law as 10 CFR 50.12 provides the NRC authority to grant exemptions from 10 CFR 50 requirements with provision of proper justification. Approval of the exemption from 10 CFR 50.46(a)(1), “other properties,” would not conflict with any provisions of the Atomic Energy Act of 1954, as amended, any of the Commission’s regulations, or any other law.

2. The exemption does not present an undue risk to the public health and safety.

   The purpose of 10 CFR 50.46 is to establish acceptance criteria for ECCS performance to provide a high confidence that the system will perform its required functions. The requested exemption does not involve any modifications to the plant that could introduce a new accident precursor or affect the probability of postulated accidents, and therefore the probability of postulated initiating events is not increased. The PRA and engineering analysis demonstrate that the calculated risk is very small (Attachment VII) and consistent with the intent of the Commission’s safety goal policy statement, which defines an acceptable level of risk that is a small fraction of other risks to which the public is exposed.

   As discussed in previous 10 CFR 50.46 rulemaking, the probability of a large break LOCA is sufficiently low. Application of a risk-informed approach shows a high

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probability with low uncertainty that the ECCS will meet 10 CFR 50.46 requirements (Attachment VII), rather than using deterministic methods to achieve a similar understanding. This is applicable to evaluating acceptable containment sump design in support of ECCS and CSS recirculation modes.

The proposed change is to apply a risk-informed method rather than a deterministic method to establish a high probability of success for performance of ECCS in accordance with the requirements in 10 CFR 50.46(a)(1). The risk-informed approach involves a complete evaluation of the spectrum of LOCAs up to and including DEGBs on the largest pipe in the reactor coolant system, as described in Attachment VII of this submittal.

The risk-informed approach analyzes LOCAs, regardless of break size, using the same methods, assumptions, and criteria in order to quantify the uncertainties and overall risk metrics (Attachment VII). This ensures that large break LOCAs with a low probability of occurrence and smaller break LOCAs with higher probability of occurrence are both considered in the results. Because the design-basis requirement for consideration of a DEGB of the largest pipe in the reactor coolant system is retained, the existing defense-in-depth and safety margin established for the design of the facility are not reduced.

This exemption only affects 10 CFR 50.46(a)(1), “other properties,” requirements that a licensee is able to demonstrate, using a bounding calculation or other deterministic method, that the ECCS and CSS are capable of functioning during a design basis event. This exemption does not impact the adequacy of the acceptance criteria for cladding performance, which is important to maintain adequate safety margins.

3. The exemption is consistent with the common defense and security.

This exemption involves a change to the licensing basis for the plant that has no relation to the control of licensed material or any security requirements that apply to WCGS. Therefore, the exemption is consistent with the common defense and security.

4.2 Applicability of 10 CFR 50.12(a)(2)

This section discusses the presence of special circumstances as related to 10 CFR 50.12(a). 10 CFR 50.12(a)(2) states that the NRC will not consider granting an exemption to the regulations unless special circumstances are present. Special circumstances are present whenever one of the listed items (i through vi) under 10 CFR 50.12(a)(2) are applicable.

Such special circumstances are present in this instance to warrant exemption from the requirements in 10 CFR 50.46(a)(1) “other properties,” which use deterministic calculation methods as the design basis for acceptable sump performance to validate the results of the ECCS evaluation model. Approval of this exemption request would allow the use of a risk-informed method to amend the design basis for acceptable performance of the containment emergency sump, as a validation of inputs in the ECCS evaluation model, and in support of the existing licensing bases for compliance with 10 CFR 50.46.

As described below, special circumstances in 10 CFR 50.12(a)(2)(ii) and 10 CFR 50.12(a)(2)(iii) are present as required by 10 CFR 50.12(a)(2) for consideration of the request for exemption.

4.2.1 Applicability of 10 CFR 50.12(a)(2)(ii)

10 CFR 50.12(a)(2)(ii) applies:

> *"Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule."*

The intent of 10 CFR 50.46(a)(1) is to ensure ECCS cooling performance design requirements imposed by 10 CFR 50.46 are determined to be met by a rigorous method that provides a high level of confidence in ECCS performance. This exemption request is consistent with that purpose because use of the proposed risk-informed approach accounts for the effect of debris on the ECCS cooling performance and supports a high probability of successful ECCS performance, based on the risk results meeting the acceptance guidelines of RG 1.174 (Attachment VII and Reference 2).

The need for this exemption is based on the requirements in the regulations for using deterministic methods to demonstrate acceptable design. Regulatory requirements are largely based on a deterministic framework, and are established for DBAs, such as the LOCA, with specific acceptance criteria that must be satisfied. Licensed facilities must be provided with safety systems capable of preventing and mitigating the consequences of DBAs to protect public health and safety. The deterministic regulatory requirements were designed to ensure that these systems are highly reliable. The LOCA analysis was established as part of this deterministic regulatory framework.

In comparison, the risk-informed approach considers nuclear safety in a more comprehensive way by examining the likelihood of a broad spectrum of initiating events and potential challenges, considering a wide range of credible events and assessing the risk based on mitigating system reliability.

An objective of 10 CFR 50.46 is to maintain low risk to the public health and safety through a reliable ECCS, as supported by the containment sump. The supporting analysis demonstrates that using a risk-informed approach to evaluate sump performance is consistent with the Commission's Safety Goals for nuclear power plants and supports ECCS operation with a high degree of reliability. Consequently, the special circumstances described in 10 CFR 50.12(a)(2)(ii) apply.

4.2.2 Applicability of 10 CFR 50.12(a)(2)(iii)

10 CFR 50.12(a)(2)(iii) applies:

> *"Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated."*

In order to meet a deterministic threshold value for sump debris loads, the debris sources in containment would need to be significantly reduced. The amount of radiological exposure received during the removal and/or modification of insulation from the WCGS containment is dependent on the scope of the changes.

Due to uncertainties in radiation levels, contamination levels, and the required modification scope, it is difficult to predict the total occupational dose associated with insulation removal and/or modifications. Dose estimates for removal of insulation from South Texas Project (STP) are described in some detail in the STP pilot submittal (Reference 11). STP is considered representative of WCGS since both plants are four-loop Westinghouse designs and use low-density fiberglass insulation on major equipment in containment. Based on the STP calculations, the total volume of fiberglass insulation replacement for WCGS is estimated to be over 4000 ft^3. The expected total dose for replacing insulation in the WCGS containment is estimated generically to be about 100 rem.

For the above estimates, the highest dose contributor is personnel work hours in close proximity to high dose sources (Reference 11). The estimate considered person-hours required to erect and remove scaffolding and the dose associated with removal of insulation. However, the estimate did not consider dose associated with disposal of the removed insulation or dose associated with insulation modifications for small-bore piping (Reference 11).

The dose considerations discussed above demonstrate that compliance would result in substantial personnel exposure due to insulation modifications in containment, which is not commensurate with the expected safety benefit based on the risk evaluation results showing that the risk associated with post-accident debris effects is less than the threshold for Region III in RG 1.174 (Attachment VII). Consequently, the special circumstances described in 10 CFR 50.12(a)(2)(iii) apply to the exemption requested by WCNOC.

4.3 Environmental Consideration

Pursuant to the requirements of 10 CFR 51.41, "Requirement to submit environmental information," and 10 CFR 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments," the following information is provided. As demonstrated below, WCNOC has determined that this exemption is eligible for categorical exclusion as set forth in 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," paragraph (c)(9).

Request for Exemption

4.3.1 Environmental Impacts Consideration

The requested exemption has been evaluated and determined to result in no significant radiological environmental impacts. This conclusion is based on the following.

The requested exemption is to allow the use of a risk-informed method to demonstrate that the design and licensing bases for the ECCS are not significantly affected by accident-generated and transported debris. The intent of the proposed change is to quantify the risk associated with GSI-191 concerns. This quantification, provided in the form of risk metrics using the guidance in RG 1.174 (Reference 2), demonstrates that the risk is less than the threshold for Region III, “Very Small Changes” (Attachment VII). Therefore, the requested exemption supports a change that represents a very small ΔCDF and ΔLERF, which is consistent with the Commission’s Safety Goals for public health and safety.

Since the risk-informed analysis demonstrated that the increases in risk are very small, the requested exemption has a negligible effect on the consequences of an accident, and adequate assurance of public health and safety is maintained. The requested exemption does not involve any changes to the facility or facility operations that could create a new accident or release path, or significantly affect a previously analyzed accident or release path. Therefore, the requested exemption would not cause changes in the types or quantities of radiological effluents, or the permitted effluent release paths.

The requested exemption does not impact the release of radiological effluents during and following a postulated LOCA. The design-basis LOCA radiological consequence analysis in the current licensing basis is a deterministic evaluation based on the assumption of a major rupture of the reactor coolant system piping and a significant amount of core damage as specified in RG 1.4 (Reference 12). The current licensing basis analysis shows the resulting doses to the public and control room are acceptable. The requested exemption does not change the radiological analysis for a LOCA. Therefore, the requested exemption does not affect the amount of radiation exposure resulting from a postulated LOCA.

The requested exemption does not involve any changes to non-radiological plant effluents or any activities that would adversely affect the environment. The requested exemption only pertains to the licensing basis for components located within the restricted area of the facility, to which access is limited to authorized personnel. Therefore, the requested exemption would not create any significant non-radiological impacts on the environment in the vicinity of the plant.

The requested exemption does not involve the use of any resources not previously considered by the NRC in its past environmental statements for issuance of the facility operating licenses or other licensing actions for the facility. Therefore, the requested exemption does not involve any unresolved conflicts concerning alternative uses of available resources.

Request for Exemption

4.3.2 Categorical Exclusion Consideration

WCNOC has evaluated the requested exemption against the criteria for identification of licensing and regulatory actions requiring environmental assessments in accordance with 10 CFR 51.21. It was determined that the requested exemption meets the criteria and is eligible for categorical exclusion as set forth in 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," paragraph (c)(9).

This determination is based on the fact that this exemption request is from requirements under 10 CFR 50 with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, specifically to authorize a change to the licensing basis for ECCS as it relates to acceptable containment sump performance in the recirculation mode following a postulated LOCA. The requested exemption has been evaluated to meet the following criteria under 10 CFR 51.22(c)(9).

<u>(i) The exemption involves no significant hazards consideration.</u>

An evaluation of the three criteria set forth in 10 CFR 50.92(c) as applied to the exemption is provided below. The evaluation is consistent with the no significant hazards consideration determination provided in the LAR (see Attachment I).

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change to allow an exemption from 10 CFR 50.46(a)(1) to implement a risk-informed evaluation methodology does not initiate an accident and therefore, the proposed change does not increase the probability of an accident occurring.

   Approval of the requested exemption and accompanying LAR would allow the results of a risk-informed evaluation to be included in the USAR. The evaluation concludes that the ECCS and CSS will serve their safety functions with a high probability following a LOCA. The evaluation considers the impacts of accident-generated and transported debris on the containment emergency sump strainers in recirculation mode, as well as core blockage due to in-vessel effects.

   The risk evaluation concludes that the risk associated with the proposed change is very small and within Region III as defined by RG 1.174, for both CDF and LERF (Attachment VII). As a result, the required systems, structures, and components (SSCs) supported by the containment sumps will perform their safety functions with a high probability, and the proposed change does not alter or prevent the ability of SSCs to perform their intended function to mitigate the consequences of an accident previously evaluated within the acceptance limits. The safety analysis acceptance criteria in the USAR continue to be met for the proposed change. Additionally, in accordance with the guidance of RG 1.174, there is substantial safety margin and defense-in-depth (Attachment IX) that provide additional confidence that the design-basis functions are maintained.

Request for Exemption

Therefore, the proposed change does not involve a significant increase in the probability or consequences of any of the accidents previously evaluated in the USAR.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change is allowance of a risk-informed analysis of debris effects from accidents that are already evaluated in the WCGS USAR. No new or different kind of accident is created by the proposed change. No new failure mechanisms or malfunctions that can initiate an accident are created by the proposed change. Therefore, the proposed change does not create the possibility for a new or different kind of accident from any accident previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

The proposed change does not modify any functional requirements or method of performing functions of plant SSCs. The effects of debris are analyzed for a full spectrum of LOCAs, including DEGBs and partial breaks for all RCS piping sizes. Appropriate redundancy, consideration of loss of offsite power, and worst-case single failure are retained, such that defense-in-depth is maintained.

Application of the risk-informed methodology concludes that the increase in risk from the contribution of debris effects is very small as defined by RG 1.174 (Reference 2) and that there is adequate defense-in-depth and safety margin (Attachment IX). Consequently, WCNOC determined that the containment sumps would continue to support the safety-related components to perform their design functions when the effects of debris are considered.

The proposed change does not alter the manner in which safety limits are determined or the acceptance criteria associated with a safety limit. The proposed change does not implement any significant changes to plant operation that can challenge an SSC's capability to safely shut down the plant or maintain the plant in a safe shutdown condition. The proposed change does not affect the existing safety margins in the barriers for the release of radioactivity. There are no changes to any of the safety analyses in the USAR. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

(ii) <u>The exemption involves no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.</u>

No physical modifications or changes to operating requirements are proposed for the facility, including any SSCs relied upon to mitigate the consequences of a LOCA. No changes are made to the safety analyses in the USAR. Approval of the exemption will require the calculated risk associated with post-accident debris effects to meet the Region III acceptance guidelines in RG 1.174 (Reference 2), thereby maintaining public health and safety. Therefore, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

(iii) The proposed exemption involves no significant increase in individual or cumulative occupational radiation exposure.

No new operator actions are implemented that could affect occupational radiation exposure. No physical modifications or changes to operating requirements are proposed for the facility, including any SSCs relied upon to mitigate the consequences of a LOCA. No changes are made to the safety analyses in the USAR. Therefore, with respect to installation or use of a facility component located within the restricted area, approval of this exemption request will not result in a significant increase in individual or cumulative occupational radiation exposure.

Based on the above, WCNOC concludes that the requested exemption meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

5.0 TECHNICAL JUSTIFICATION FOR THE EXEMPTION

Technical justification for the risk-informed method is provided in the accompanying LAR (Attachment I) and in Attachments VII through IX.

The proposed risk-informed approach meets the key principles in RG 1.174 (Reference 2) in that it is consistent with the defense-in-depth philosophy, maintains sufficient safety margins, results in a very small increase in risk, and is monitored using performance measurement strategies. The requested exemption to allow use of the risk-informed method is consistent with the key principle in RG 1.174 that requires the proposed change to meet current regulations unless explicitly related to a requested exemption.

The WCGS risk evaluation results (Attachment VII) show that the risk associated with post-accident debris effects is within RG 1.174 Region III acceptance guidelines as a “Very Small Change,” and, therefore, is consistent with the Commission’s Safety Goals for public health and safety.

6.0 CONCLUSION

Approval of the requested exemption to allow the use of a risk-informed approach will not present an undue risk to the public health and safety and is consistent with the common defense and security as required by 10 CFR 50.12(a)(1). Furthermore, special circumstances required by 10 CFR 50.12(a)(2) are present for 10 CFR 50.12(a)(2)(ii) and 10 CFR 50.12(a)(2)(iii). The requested exemption has been evaluated and determined to result in no significant radiological environmental impacts. Based on the determination that the risk of the exemption meets the acceptance guidelines of RG 1.174 (Reference 2), the results demonstrate reasonable assurance that the ECCS will function in the recirculation mode and that the public health and safety will be protected.

Request for Exemption

7.0 REFERENCES

1. NRC Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," September 13, 2004.
2. Regulatory Guide 1.174, "An Approach for Using Probabilistic Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, January 2018.
3. SECY-12-0093 (ML121310648), "Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance," July 9, 2012.
4. NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors," June 9, 2003.
5. ML103570354, "Staff Requirements – SECY-10-0113 – Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance," December 23, 2010.
6. ML12349A378, "Staff Requirements – SECY-12-0093 – Closure Options for Generic Safety Issue – 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance," December 14, 2012.
7. WCNOC Letter ET 13-0017 (ML13148A272), "Wolf Creek Nuclear Operating Corporation Proposed Path to Closure of Generic Safety Issue-191, 'Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance,'" May 16, 2013.
8. NEI 04-07 Volume 1, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Revision 0, December 2004.
9. NEI 04-07 Volume 2, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02," Revision 0, December 2004.
10. SECY-16-0033 Enclosure 1 (ML15238B016), "Federal Register Notice, Draft Final Rulemaking: Performance-Based Emergency Core Cooling Systems Cladding Acceptance Criteria," February 16, 2016.
11. STP Letter NOC-AE-13002986 (ML13175A211), "South Texas Project, Units 1 and 2, Docket Nos. STN 50-498 and STN 50-499, Revised STP Pilot Submittal and Requests for Exemptions and License Amendment for a Risk-Informed Approach to Resolving Generic Safety Issue (GSI)-191 (TAC Nos. MF0613 and MF0614)", June 19, 2013.
12. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Revision 2, June 1974.

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Attachment III

Proposed Technical Specification Changes (Mark-Up)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.5	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.5.2.7	Verify, for each ECCS throttle valve listed below, each mechanical position stop is in the correct position. Valve Number EM-V0095 EM-V0107 EM-V0089 EM-V0096 EM-V0108 EM-V0090 EM-V0097 EM-V0109 EM-V0091 EM-V0098 EM-V0110 EM-V0092	18 months
~~SR 3.5.2.8~~	~~Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet strainers show no evidence of structural distress or abnormal corrosion~~.	~~18 months~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	The following SRs are applicable for all equipment required to be OPERABLE: SR 3.5.2.1 SR 3.5.2.7 SR 3.5.2.3 ~~SR 3.5.2.8~~ SR 3.5.2.4	In accordance with applicable SRs

3.6 CONTAINMENT SYSTEMS

3.6.8 Containment Sump

LCO 3.6.8 Two containment sumps shall be OPERABLE.

APPLICABILITY MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment sumps inoperable due to containment accident generated and transported debris exceeding the analyzed limits.	A.1 Initiate action to mitigate containment accident generated and transported debris.	Immediately
	AND	
	A.2 Perform SR 3.4.13.1.	Once per 24 hours
	AND	
	A.3 Restore the containment sumps to OPERABLE status.	90 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more containment sumps inoperable for reasons other than Condition A.	B.1 Declare affected Emergency Core Cooling System train(s) inoperable.	Immediately
	AND	
	B.2 Declare affected containment spray train(s) inoperable.	Immediately
C. Required Action and associated Completion Time of Condition A not met.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.8.1	Verify by visual inspection, the containment sumps do not show structural damage, abnormal corrosion, or debris blockage.	18 months

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Attachment IV

Revised Technical Specification Pages

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.5	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.5.2.7	Verify, for each ECCS throttle valve listed below, each mechanical position stop is in the correct position. <u>Valve Number</u> EM-V0095 EM-V0107 EM-V0089 EM-V0096 EM-V0108 EM-V0090 EM-V0097 EM-V0109 EM-V0091 EM-V0098 EM-V0110 EM-V0092	18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	The following SRs are applicable for all equipment required to be OPERABLE: SR 3.5.2.1 SR 3.5.2.7 SR 3.5.2.3 SR 3.5.2.4	In accordance with applicable SRs

3.6 CONTAINMENT SYSTEMS

3.6.8 Containment Sump

LCO 3.6.8 Two containment sumps shall be OPERABLE.

APPLICABILITY MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment sumps inoperable due to containment accident generated and transported debris exceeding the analyzed limits.	A.1 Initiate action to mitigate containment accident generated and transported debris.	Immediately
	AND	
	A.2 Perform SR 3.4.13.1.	Once per 24 hours
	AND	
	A.3 Restore the containment sumps to OPERABLE status.	90 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more containment sumps inoperable for reasons other than Condition A.	B.1 Declare affected Emergency Core Cooling System train(s) inoperable.	Immediately
	AND	
	B.2 Declare affected containment spray train(s) inoperable.	Immediately
C. Required Action and associated Completion Time of Condition A not met.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.8.1	Verify by visual inspection, the containment sumps do not show structural damage, abnormal corrosion, or debris blockage.	18 months

Wolf Creek Generating Station
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Attachment V

Proposed Technical Specifications Bases Changes (Mark-Up) for Information Only

TABLE OF CONTENTS

BACKGROUND (continued)

sequence. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive accumulators, ~~and~~ the RWST, and the containment sump, covered in LCO 3.5.1, "Accumulators," ~~and~~ LCO 3.5.4, "Refueling Water Storage Tank (RWST)," and LCO 3.6.8, "Containment Sump," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature is $\leq$ 2200°F;

b. Maximum cladding oxidation is $\leq$ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is $\leq$ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps and SI pumps are credited in a small break LOCA event. This event establishes the flow and discharge head at the design point for the centrifugal charging pumps. The SGTR and MSLB events also credit the centrifugal charging pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.7

The position of throttle valves in the flow path is necessary for proper ECCS performance. These valves are necessary to restrict flow to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. The 18 month Frequency is based on the same reasons as those stated in SR 3.5.2.5 and SR 3.5.2.6. The ECCS throttle valves are set to ensure proper flow resistance and pressure drop in the piping to each injection point in the event of a LOCA. Once set, these throttle valves are secured with locking devices and mechanical position stops. These devices help to ensure that the following safety analyses assumptions remain valid: (1) both the maximum and minimum total system resistance; (2) both the maximum and minimum branch injection line resistance; and (3) the maximum and minimum ranges of potential pump performance. These resistances and pump performance ranges are used to calculate the maximum and minimum ECCS flows assumed in the LOCA analyses of Reference 3.

~~SR 3.5.2.8~~

~~This SR requires verification that each ECCS train containment sump inlet is not restricted by debris and the suction inlet strainers show no evidence of structural distress or abnormal corrosion. A visual inspection of the suction inlet piping verifies the piping is unrestricted. A visual inspection of the accessible portion of the containment sump strainer assembly verifies no evidence of structural distress or abnormal corrosion. Verification of no evidence of structural distress ensures there are no openings in excess of the maximum designed strainer opening. The 18 month Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. USAR, Sections 6.3 and 15.6.
4. USAR, Chapter 15, "Accident Analysis."
5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, “Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
6. IE Information Notice No. 87-01.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS - Shutdown

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS - Operating," is applicable to these Bases, with the following modifications.

In MODE 4, the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and residual heat removal (RHR) (low head).

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) and the containment sump can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that certain automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

For MODE 3, with the accumulators blocked, and MODE 4, the parameters assumed in the generic bounding thermal hydraulic analysis for the limiting DBA (Reference 1) are based on a combination of limiting parameters for MODE 3, with the accumulators blocked, and parameters for MODE 4. However, assumed ECCS availability is based on MODE 4 conditions; the minimum available ECCS flow is calculated assuming only one OPERABLE ECCS train.

Only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation. The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

In MODE 4, one of the two independent (and redundant) ECCS trains is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Containment Sump

BASES

BACKGROUND

The containment sumps provide a borated water source to support recirculation of coolant from the containment sumps for residual heat removal, emergency core cooling, containment cooling, and containment atmosphere cleanup during accident conditions.

The containment sumps supply both trains of the Emergency Core Cooling System (ECCS) and the Containment Spray System (CSS) during any accident that requires recirculation of coolant from the containment sumps. The recirculation mode is initiated when the pump suction is transferred to the containment sumps on low Refueling Water Storage Tank (RWST) level, which ensures the containment sumps have enough water to supply the net positive suction head (NPSH) to the ECCS and CSS pumps. There are two containment sumps, each providing suction for one train of the ECCS and CSS.

The containment sumps contain strainers to limit the quantity of the debris materials from entering the sump suction piping. Debris accumulation on the strainers can lead to undesirable hydraulic effects including air ingestion through vortexing or deaeration, and reduced NPSH at pump suction.

While the majority of debris accumulates on the strainers, some fraction penetrates the strainers and is transported to downstream components in the ECCS, CSS, and the Reactor Coolant System (RCS). Debris that penetrates the strainer can result in wear to the downstream components, blockages, or reduced heat transfer across the fuel cladding. Excessive debris in the containment sump water source could result in insufficient recirculation of coolant during the accident, or insufficient heat removal from the core during the accident.

APPLICABLE SAFETY ANALYSIS

During all accidents that require recirculation, the containment sumps provide a source of borated water to the ECCS and CSS pumps. As such, it supports residual heat removal, emergency core cooling, containment cooling, and containment atmosphere cleanup during an accident. It also provides a source of negative reactivity (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "ECCS - Operating," B 3.5.3, "ECCS - Shutdown," and B 3.6.6, "Containment Spray and Cooling Systems."

BASES

APPLICABLE SAFETY ANALYSIS (continued)

USAR Appendix 6A (Ref. 2) describes evaluations that confirm long-term core cooling is assured following any accident that requires recirculation from the containment sump.

The containment sumps satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two containment sumps are required to ensure a source of borated water to support ECCS and CSS OPERABILITY. A containment sump consists of the containment drainage flow paths, the associated containment sump strainer, and the inlet to the ECCS and CSS piping. An OPERABLE containment sump has no structural damage or abnormal corrosion that could prevent recirculation of coolant and will not be restricted by containment accident generated and transported debris.

Containment accident generated and transported debris consists of the following:

a. Accident generated debris sources - Insulation, coatings, and other materials which are damaged by the high-energy line break (HELB) and transported to the containment sump. This includes materials within the HELB zone of influence and other materials (e.g., unqualified coatings) that fail due to the post-accident containment environment following the accident;

b. Latent debris sources – Pre-existing dirt, dust, paint chips, fines or shards of insulation, and other materials inside containment that do not have to be damaged by the HELB to be transported to the containment sump; and

c. Chemical product debris sources – Aluminum and non-metallic materials such as paints, thermal insulation, and concrete that are susceptible to chemical reactions within the post-accident containment environment leading to corrosion products that are generated within the containment sump pool or are generated within containment and transported to the containment sump.

Containment debris limits are defined in Table B 3.6.8-1, and additional discussion is provided in USAR Appendix 6A (Ref. 2).

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, containment sump OPERABILITY requirements are dictated by the ECCS and CSS OPERABILITY requirements. Since both the ECCS and the CSS must be OPERABLE in MODES 1, 2, 3, and 4, the containment sumps must also be OPERABLE to support their operation.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the containment sumps are not required to be OPERABLE in MODES 5 or 6.

ACTIONS

A.1, A.2, and A.3

Condition A is applicable when there is a condition which results in containment accident generated and transported debris exceeding the analyzed limits. Containment debris limits are defined in Table B 3.6.8-1 and additional discussion is provided in USAR Appendix 6A (Ref. 2).

Immediate action must be initiated to mitigate the condition. Examples of mitigating actions are:

- Removing the debris source from containment or preventing the debris from being transported to the containment sump;
- Evaluating the debris source against the assumptions in the analysis;
- Deferring maintenance that would affect availability of the affected systems and other LOCA mitigating equipment;
- Deferring maintenance that would affect availability of primary defense-in-depth systems, such as containment coolers;
- Briefing operators on LOCA debris management actions; or
- Applying an alternative method to establish new limits.

While in this condition, the RCS water inventory balance, SR 3.4.13.1, must be performed at an increased Frequency of once per 24 hours. An unexpected increase in RCS leakage could be indicative of an increased potential for an RCS pipe break, which could result in debris being generated and transported to the containment sump. The more frequent monitoring allows operators to act in a timely fashion to minimize the potential for an RCS pipe break while the containment sump is inoperable.

BASES

ACTIONS (continued)

For the purposes of applying LCO 3.0.6 and the Safety Function Determination Program while in Condition A, the two containment sumps are considered a single support system for all ECCS and CSS trains because containment accident generated and transported debris issues that would render one sump inoperable would render all of the sumps inoperable.

The inoperable containment sump must be restored to OPERABLE status in 90 days. A 90-day Completion Time is reasonable for emergent conditions that involve debris in excess of the analyzed limits that could be generated and transported to the containment sump under accident conditions. The likelihood of an initiating event in the 90-day Completion Time is very small and there is margin in the associated analyses. The mitigating actions of Required Action A.1 provide additional assurance that the effects of debris in excess of the analyzed limits will be mitigated during the Completion Time.

B.1

When one or more containment sumps are inoperable for reasons other than Condition A, such as blockage, structural damage, or abnormal corrosion that could prevent recirculation of coolant, the affected ECCS and CSS trains are rendered inoperable; therefore, the affected ECCS and CSS trains must be immediately declared inoperable. Declaring the affected trains inoperable ensures appropriate restrictions are implemented in accordance with the Required Actions of the ECCS and CSS Specifications.

C.1 and C.2

If the containment sump cannot be restored to OPERABLE status within the associated Completion Time for Condition A, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

Periodic inspections are performed to verify the containment sumps do not show current or potential debris blockage, structural damage, or abnormal corrosion to ensure the operability and structural integrity of the containment sumps (Ref. 1).

The 18-month Frequency is based on the need to perform this Surveillance during a refueling outage, because of the need to enter containment. This Frequency is sufficient to detect any indication of structural damage, abnormal corrosion, or debris blockage of the containment sump.

REFERENCES

1. USAR, Chapter 6 and Chapter 15.

2. USAR Appendix 6A, Resolution of NRC Generic Letter 2004-02.

Table B 3.6.8-1
Containment Sump Debris Limits for Breaks ≤ 10 inches

Debris Type	Debris Limit
Fiber Fines*	144.1 lb_m
Total Fiber Fines, Small Pieces, and Large Pieces**	322.5 lb_m
Latent Particulate**	122.2 lb_m
ThermoLag Particulate**	0.50 ft^3
Coatings Particulate**	2.43 ft^3
Degraded Paint Chips**	158.4 ft^2
Miscellaneous Debris (Tags, Labels, etc.)**	20.0 ft^2

*Transportable fine fiber debris in the pool that is available to transport to either strainer during single train operation or split between both strainers during two train operation.
**Maximum quantity of debris allowed to transport to a single strainer.

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Attachment VI

Proposed USAR Changes (Mark-Up) for Information Only

In order to keep materials within the containment that are subject to corrosion to a minimum, the following restrictions are placed on the use of zinc, aluminum, and mercury in the containment:

a. Aluminum is severely attacked by the alkaline containment spray solution. This reaction may result in the loss of structural integrity and the generation of gaseous hydrogen. Aluminum exposed to containment spray or submerged in the post-LOCA sump pool may also contribute to sump strainer chemical effects (GSI-191). The use of aluminum in the containment is minimized.

b. Boric acid reacts with zinc, oxidizing it and liberating hydrogen gas. The use of zinc (galvanized materials and paint) in the containment is minimized to reduce the generation of hydrogen.

c. The use of mercury and mercuric compounds is minimized inside the containment because of its corrosive effects on stainless steel, NiCrFe alloy 600, and alloys containing copper. The amount of mercury associated with plant lighting and control switches, etc., is negligible.

Table 6.2.5-3 is a list of the amounts of aluminum and zinc which are in the containment and which could potentially be exposed to a corrosive environment. These materials are listed by the system or component in which they are used, and an estimate of their expected corrosion rate is given. Aluminum or zinc is not used in any safety-related item where exposure to the spray solution is possible.

For other materials which could come in contact with containment sprays, tests have been performed and are detailed in Reference 2. These tests have shown that no significant amount of corrosion products is produced from these materials.

Many coatings which are in common industrial use may deteriorate in the post-accident environment and contribute substantial quantities of foreign solids and residue to the containment sump. Consequently, protective coatings used inside the containment in significant quantities are demonstrated to withstand the design basis accident conditions and are designed to meet the criteria given in ANSI N101.2 (1972), "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities," and are in compliance with Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," as indicated in Table 6.1-2. Some small items may be painted or coated using common industrial practice but the paint/coating is not in sufficient quantity to cause any

6.1.3 POST-ACCIDENT CHEMISTRY

Following a main steam line break or design basis LOCA, sodium hydroxide and boric acid solutions will be present in the containment sumps. Figure 6.5-5 represents the time-history of the pH of the aqueous phase in the containment sump. Table 6.5-5 indicates the quantities of sodium hydroxide and boric acid that will be present in the containment after an accident. The pH control reduces the probability of chloride stress corrosion cracking on stainless steel and attack on aluminum fittings.

GSI-191 chemical effects on the containment sump recirculation system are addressed in USAR Appendix 6A.

6.1.4 REFERENCES

1. Whyte, D. D. and Picone, L. F., "Behavior of Austenitic Stainless Steel in Post Hypothetical Loss-of-Coolant Environment," WCAP-7798-L (Proprietary), November 1971 and WCAP-7803 (Non-Proprietary), December 1971.

2. Picone, L. F., "Evaluation of Protective Coatings for use in Reactor Containment," WCAP-7198-L (Proprietary), April 1968 and WCAP-7825 (Non-Proprietary), December 1971.

3 Caplan, J. S., "The Application of Preheat Temperatures after Welding Pressure Vessel Steels," WCAP-8577 (Non-Proprietary), September 1975.

SAFETY DESIGN BASIS EIGHT - The CSS, in conjunction with the containment fan cooler system and the emergency core cooling system, is designed to be capable of removing sufficient heat and subsequent decay heat from the containment atmosphere following the hypothesized LOCA or MSLB to maintain the containment pressure below the containment design pressure. Section 6.2.1 provides the assumptions as to sources and amounts of energy considered and the analysis of the containment pressure transient following a LOCA or MSLB accident inside the containment (GDC-38).

SAFETY DESIGN BASIS NINE - The CSS remains operable in the accident environment.

SAFETY DESIGN BASIS TEN - The containment spray water does not contain substances which would be unstable in the thermal or radiolytic environment of the LOCA or cause extensive corrosive attack on equipment.

SAFETY DESIGN BASIS ELEVEN - The CSS is designed so that adequate net positive suction head (NPSH) exists at the suction of the containment spray pumps ~~during all operating phases~~, in accordance with Regulatory Guide 1.1. NPSH during the containment sump recirculation phase of an accident is addressed in USAR Appendix 6A.

SAFETY DESIGN BASIS TWELVE - The CSS is designed to prevent debris which could impair the performance of the containment spray pumps, valves, eductors, or spray nozzles from entering the recirculation piping. Design is in accordance with Regulatory Guide 1.82, as discussed in Table 6.2.2-1. The design basis for the CSS with regard to the effects of debris on the recirculation sump strainers is a risk-informed analysis, which shows the risk associated with the effects of debris is very small as defined by Regulatory Guide 1.174. The conclusion is based on plant-specific testing and analyses using inputs and assumptions that provide safety margin and defense-in-depth. Details of the design basis for the effects of debris on the function of the emergency sump strainers are provided in USAR Appendix 6A.

6.2.2.1.1.2 Power Generation Design Bases

The CSS has no power generation design bases.

6.2.2.1.2 System Design

6.2.2.1.2.1 General Description

The CSS, shown schematically in Figure 6.2.2-1, consists of two separate trains of equal capacity, each independently capable of meeting the design bases. Each train includes a containment spray pump, spray header and nozzles, spray additive eductor, valves, and the necessary piping, instrumentation, flushing connections, and controls. The containment spray additive tank supplies 30 weight percent (nominal) sodium hydroxide to both trains. The refueling water storage tank supplies borated injection water to the containment spray system. Each train takes suction from separate containment recirculation sumps during the recirculation phase.

The CSS provides a spray of cold or subcooled borated water, adjusted with NaOH, from the upper regions of the containment to reduce the containment pressure and temperature during either a LOCA or MSLB inside the containment.

Each CSS pump discharges into the containment atmosphere through an independent spray header. The spray headers are located in the upper part of the reactor building to allow maximum time for the falling spray droplets to reach thermal equilibrium with the steam-air atmosphere. The condensation of the steam by the falling spray results in a reduction in containment pressure and temperature. Each spray train provides adequate coverage to meet the design requirements with respect to both containment heat removal and iodine removal. Further discussion of the iodine removal function of the CSS is provided in Section 6.5.2.

In the CSS, only the containment recirculation sumps and the spray headers, nozzles, and associated piping and valves are located within the containment. The remainder of the system is located within the auxiliary building, separated from that portion in the containment by motor-operated isolation valves.

During the recirculation phase, leakage outside of the containment is detected with the auxiliary building radiation indicators and alarms, temperature alarms, and auxiliary building sump alarms. The motor-operated isolation valves in each train assure train isolation capability in the event of leakage during the recirculation phase. Leakage detection within the auxiliary building is discussed in Section 9.3.3.

6.2.2.1.2.2 Component Description

Mechanical components of the CSS, except those in the spray additive subsystem, are described in this section. Description of the mechanical components in the spray additive subsystem is provided in Section 6.5.2. Component design parameters are given in Table 6.2.2-2.

Each component in the CSS is designed and manufactured to withstand the environmental effects, including radiation, found in Table 3.11(B)-2.

CONTAINMENT SPRAY PUMPS - The two CS pumps are the vertical centrifugal type, driven by electric induction motors. The motors have open drip-proof enclosures and are provided with adequate insulation which allows continuous operation of a 100-percent-rated load at 50 C ambient. Power for these motors is supplied from the Class IE 4,160-Volt busses. Power supply availability is discussed in Section 8.3.

The pump motors are specified to have the capability of starting and accelerating the driven equipment, under load, to a design point running speed within 4 seconds, based on 75 percent of the rated motor voltage. The pumps are designed to withstand a thermal transient from 37°F to 300°F occurring in 10 seconds, which exceeds the severity of the transient occurring when pump suction is switched from the RWST to the containment sump.

The shaft seals on the pumps are reliable, easy to maintain, and compatible with the fluids to be circulated. They are designed to operate at a temperature of 300°F, which exceeds the maximum temperature to which they will be exposed following an accident.

The containment spray pumps are designed to handle the runout flow associated with the startup transient, when minimal discharge head is applied.

CONTAINMENT SPRAY HEADER AND NOZZLES - Each containment spray header contains 197 hollow cone nozzles, each capable of the design flow and differential pressure given in Table 6.2.2-2. These nozzles have a 7/16-inch spray orifice. The nozzles produce a drop size distribution, as described in Figure 6.5-2, at system design conditions. Special tests performed on the spray nozzles are discussed in Section 6.5.2.2.2. The spray solution is completely stable and soluble at temperatures of interest in the containment and, therefore, does not precipitate or otherwise interfere with nozzle performance. The nozzles of each header are oriented to provide greater than 90-percent area coverage at the operating deck of the reactor building. The area coverage at the operating deck (based on the calculated post-LOCA containment saturation temperature) is provided in Table 6.5-2 for various nozzle orientations. The containment spray envelope reduction factor as a function of post-LOCA containment saturation temperature is provided in Figure 6.5-4. The spray header design, nozzle spacing, and orientation are shown in Figure 6.2.2-2. The containment spray header and nozzles are designed to withstand the impulse of a water hammer at the commencement of flow.

CONTAINMENT RECIRCULATION SUMPS - The two containment recirculation sumps are collecting reservoirs from which the containment spray pumps and the residual heat removal pumps separately take suction after the contents of the refueling water storage tank have been expended. The sumps are located as far as feasible from the reactor coolant system piping and components which could become sources of debris. Thermal insulation used inside containment will be a significant source of debris. The majority of insulation is removable fiberglass blanket type enclosed in a stainless steel jacket with quick-release latches. Limited quantities of other types of insulation are used in widely dispersed locations. Insulation other than removable fiberglass blanket type has been evaluated to ensure that it will not be subject to degradation under a design basis accident or, if in a few dispersed locations the insulation should degrade under DBA conditions, the debris generated as a result of the degradation is trapped by the building components so that the debris will not adversely affect the performance of the sump. The strainer arrangement consisting of stacked modules with fine mesh perforated plates completely surrounds the inlet piping to prevent floating debris and high-density particles from entering. Sources of debris, as indicated above, are physically remote from the recirculation sumps. Debris generated as a result of a LOCA will either be retained in an area such as the reactor cavity or refueling pool or must follow a tortuous path to reach the recirculation sump strainers. Figure 6.2.2-3 shows the stacked module arrangement.

A risk-informed analysis was performed to evaluate containment recirculation sump performance with the effects of post-accident debris. The analysis concluded that the risk associated with the effects of debris is very small as defined by Regulatory Guide 1.174. The risk-informed analysis is described in USAR Appendix 6A.

~~However, t~~The strainers have been evaluated to meet the intent of Regulatory Guide 1.82. To limit any possible vortexing, vortex breakers are placed in the suction lines from containment sumps to the containment spray pumps. Additionally, the strainers have been evaluated for the possibility of vortexing and found to be acceptable, as discussed in USAR Appendix 6A. The suction lines from the containment sumps to the containment spray pumps are sloped to assure switchover capability. These lines, up to and including the isolation valve, are encased in guard piping.

SAFETY EVALUATION ELEVEN - System piping size and layout provides adequate NPSH to the containment spray pump during all anticipated operating conditions, in accordance with Regulatory Guide 1.1. ~~In calculating available NPSH, the conservative assumption has been made that the water in the containment sump after a design basis LOCA is a saturated liquid, and no credit has been taken for anticipated subcooling. That is, although NPSH = elevation head + (containment pressure - liquid vapor pressure) - suction line losses, the (containment pressure - liquid vapor pressure) term has been assumed to be zero. Calculated NPSH exceeds required NPSH by at least 10 percent~~The calculation of NPSH during the recirculation phase of an accident is discussed in USAR Appendix 6A. The recirculation piping penetrating the containment sumps is nearly horizontal to minimize vortexing. In addition, a vortex breaker is provided in the inlet of the piping from the sump.

In calculating the water level within the reactor building which contributes to the NPSH available to the containment spray pumps at the beginning of its recirculation phase, consideration has been given to the potential mechanisms of water loss within the reactor building. These water loss mechanisms include water present in the vapor phase, water loss to compartments below El. 2,000, water loss above El. 2,000, and water loss due to wetted surfaces. Tables 6.2.2-6 and 6.2.2-6a identify each water source which releases water to the reactor building and its associated mass and each potential water loss mechanism and the volume of water not assumed to contribute to the water level within the containment for a large LOCA and a MSLB, respectively. ~~The static head available to contribute to the NPSH of the pump, suction line losses, and the minimum NPSH available are also given in Table 6.2.2-7.~~ The CSS pump NPSH required versus flow is shown in Figure 6.2.2-5. The reduction in water level due to potential water loss mechanisms is considered in the calculated NPSH available.

SAFETY EVALUATION TWELVE - Recirculation sump construction provides straining down to 0.045-inch strainer hole size to prevent entrained particles in excess of that size from entering the containment recirculation sump and containment spray system suction piping. Restrictions in the reactor core channels and ECCS throttle valves are the minimum restrictions and, therefore, the basis of the strainer hole opening size.

Since the containment spray pumps are designed to operate with entrained particles up to 1/4 inch in diameter and the minimum constriction size in the spray nozzles is 7/16 inch, this strainer hole size is adequate to assure proper system operability.

Each strainer provides sufficient NPSH to the ECCS pumps to maintain recirculation cooling during an event.

The sump curb does not allow flow into the sump below 6 inches above the concrete floor level surrounding the sump. This arrangement leaves ample depth for buildup of high-density debris without affecting sump performance. Additionally, the velocity of recirculated fluids approaching the curb will be between 0.01 and 0.08 fps for all modes of operation following a LOCA or MSLB, and thus a low velocity settling region for high-density particles is provided. Table 6.2.2-9 provides flow velocities at several times and locations for a large LOCA and an MSLB.

~~Any D~~debris which eludes the curb could accumulate on the strainer or pass~~es~~ ~~into the sump~~ through the 0.045 inch perforated plate openings.~~and will be drawn into the suction piping for the containment spray and residual heat removal systems. Such debris is small enough to pass through any restrictions in the ECCS throttle valves, the Containment Spray System, or the reactor vessel channels, and will eventually be pumped back into the containment.~~ The effects of post-accident debris on the containment recirculation sump and containment spray system ~~is~~are addressed by a risk-informed analysis in accordance with Regulatory Guide 1.174 and ~~is~~are presented in USAR Appendix 6A.~~The effects of post-accident debris on the containment recirculation sump and containment spray system is addressed by a risk-informed analysis in accordance with Regulatory Guide 1.174 and is presented in USAR Appendix 6A.~~

A comparison of the containment recirculation sump design features with each of the positions of Regulatory Guide 1.82, "Sump for Emergency Core Cooling and Containment Spray Systems," is provided in Table 6.2.2-1.

6.2.2.1.4 Tests and Inspections

Testing and inspection of components of the CSS, except those in the spray additive subsystem, are discussed in this section. Testing and inspection of components in the spray additive subsystem are discussed in Section 6.5.2.4.

Each containment spray pump has a shop test to generate complete performance curves. The test includes verifying total differential developed head (TDH), efficiency and brake horsepower for various flow rates. An NPSH test for various flow rates was performed on one pump. A shop thermal transient analysis, from ambient temperature to 350 F in 10 seconds, has been performed on the CSS pump. Results of that analysis assure that the design is suitable for the switchover from the injection to the recirculation phase.

TABLE 6.2.2-1 (Sheet 2)

Regulatory Guide 1.82 Position	Recirculation Sump Design
4. The floor level in the vicinity of the coolant sump location should slope gradually down away from the sump.	The floor is level in the vicinity of the sump. However, a 6-inch concrete curb is provided which prevents the high-density particles from reaching the sumps. The intent is met.
5. All drains from the upper regions of the reactor building should terminate in such a manner that direct streams of water, which may contain entrained debris, will not impinge on the filter assemblies.	All drains in the upper regions of the reactor building are terminated in such a manner that direct streams of water which may contain entrained debris will not impinge on the filter assemblies
6. A vertically mounted outer trash rack should be provided to prevent large debris from reaching the fine inner screen. The strength of the trash rack should be considered in protecting the inner screen from missiles and large debris.	Each sump strainer has approximately 3300 ft^2 of effective surface area that can accommodate the amount of debris generated and carried to the sumps following a debris-generating event. The sumps and strainers are located outside the Secondary Shield Wall, which protects them from missiles. The intent is met.
7. A vertically mounted fine inner screen should be provided. The design coolant velocity at the inner screen should be approximately 6 cm/sec (0.2 ft/sec). The available surface area used in determining the design coolant velocity should be based on one-half of the free surface area of the fine inner screen to conservatively account for partial blockage. Only the vertical screens should be considered in determining available surface area.	The strainers are installed in the sump pit with each strainer consisting of 72 modules stacked in a four by four matrix. The approach velocity of the recirculation coolant flow at the sump strainer face is less than 0.006 ft/sec. The intent is met. In addition, in ~~accordance~~response to~~ with~~ Generic Letter 2004-02~~ requirements~~, a ~~mechanistic ~~risk-informed analysis ~~has been~~ was performed to assess the potential adverse effects of post-accident debris blockage and operation with debris-laden fluids to impede or prevent the recirculation functions of the ECCS and CSS following postulated accidents for which the recirculation of these systems is required. The methodology for this analysis is ~~consistent with that documented in NEI 04-07~~discussed in USAR Appendix 6A.

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TABLE 6.2.2-1 (Sheet 3)

Regulatory Guide 1.82 Position	Recirculation Sump Design
8. A solid top deck is preferable, and the top deck should be designed to be fully submerged after a LOCA and completion of the safety injection.	The strainers consist of individual modules stacked on top of each other. The top of each module on the top layer contains a perforated plate. The strainers extend approximately one foot above the Reactor Building floor. Therefore, they will be submerged following a ~~Large Break~~ LOCA. ~~For the small break LOCA, a small portion of the upper modules will not be submerged.~~ The intent is met.
9. The trash rack and screens should be designed to withstand the vibratory motion of seismic events without loss of structural integrity.	The strainers are designed to be seismic Category I.
10. The size of openings in the fine screen should be based on the minimum restrictions found in systems served by the sump. The minimum restriction should take into account the overall operability of the system served.	The strainers have a nominal 0.045” hole size. The strainers protect the downstream equipment by removing material from the flow stream that potentially could cause damage. The perfor~~m~~ated hole size effectively removes particles larger than 0.045” from the fluid stream. This protects the reactor core channels, safety injection valves and other equipment from clogging. The effects of post-accident debris on the recirculation sump performance, including downstream effects, was addressed and is described in USAR Appendix 6A.

TABLE 6.2.2-2

CONTAINMENT HEAT REMOVAL SYSTEMS COMPONENT DESIGN PARAMETERS

Containment Spray Pumps	
Type	Vertical centrifugal
Quantity	2
Design pressure, psig	450
Design temperature, F	300
Motor, hp	500
Service factor	1.15
Start time, sec	4
Design flow rate, gpm (injection/recirculation)	3,165/3,750
Design head, ft (injection/recirculation)	464/400
NPSH available, ft	~~See Table 6.2.2-7~~See USAR Appendix 6A
Material in contact with fluid	Stainless steel
Design codes	
Pump	ASME Section III, Class 2
Motor	NEMA, IEEE 323, 334, 344
Seismic design	Category I
Containment Spray Nozzles	
Type	Whirljet, hollow cone spray nozzles
Design flow per nozzle at 40 psi ΔP	15.2 gpm
Number of nozzles	197/header
Material	Stainless steel

~~TABLE 6.2.2-7~~

~~INPUT AND RESULTS OF NPSH ANALYSIS~~

~~Containment Spray Pumps~~

~~Static head available (MSLB)~~	~~31 ft - 9-3/16 in.~~
~~Pump elevation (discharge centerline)~~	~~1971 ft - 0-3/4 in.~~
~~Suction line losses @ 3,950 gpm~~	~~9.56 ft~~
~~Available NPSH @ 3,950 gpm (3)~~	~~20.1 ft~~
~~Required NPSH @ 3,950 gpm (from Figure 6.2.2-5)~~	~~16.5 ft~~

~~Residual Heat Removal Pumps~~

~~NPSH Reference Elevation (2)~~	~~1972.07 ft.~~
~~Static head available (LOCA)(1)~~	~~30.015 ft~~
~~Suction line losses @ 4,760 gpm~~	~~3.945 ft~~
~~Available NPSH @ 4,760 gpm (3)~~	~~23.79 ft~~
~~Required NPSH @ 4,760 gpm~~	~~21.01 ft~~

~~(1) Large LOCA conditions are provided for the RHR pumps since the flow rates, line losses, and NPSH required are greater than those associated with an MSLB wherein the RCS pressure remains above the RHR shutoff head at switchover to recirculation.~~

~~(2) NPSH reference elevation is 3 3/8 inches above the discharge centerline.~~

~~(3) Includes 1.724 ft. total head loss across the sump strainer with both the Spray Pump and RHR Pump running in Recirculation, and a 0.56 ft. allowance for EDG frequency uncertainities.~~

SAFETY DESIGN BASIS THREE - Safety functions can be performed, assuming a single active component failure coincident with the loss of offsite power (GDC-35).

SAFETY DESIGN BASIS FOUR - The active components are capable of being tested during plant operation. Provisions are made to allow for inservice inspection of components at appropriate times specified in the ASME Boiler and Pressure Vessel Code, Section XI (GDC-36 and 37).

SAFETY DESIGN BASIS FIVE - The ECCS was designed and fabricated to codes consistent with the quality group classification assigned by Regulatory Guide 1.26 and the seismic category assigned by Regulatory Guide 1.29. The power supply and control functions are in accordance with Regulatory Guide 1.32.

SAFETY DESIGN BASIS SIX - The capability to isolate components or piping was provided so that the ECCS safety function is not compromised. This includes isolation of components to deal with leakage or malfunctions and to isolate safety-related portions of the system (GDC-35).

SAFETY DESIGN BASIS SEVEN - The containment isolation valves in the system were selected, tested, and located in accordance with the requirements of GDC-54 and 55 and 10 CFR 50, Appendix J, Type A testing.

SAFETY DESIGN BASIS EIGHT - ECCS equipment design qualifications ensures acceptable performance for all environments anticipated under normal, testing, and design basis accident conditions.

SAFETY DESIGN BASIS NINE - The functional requirements of the ECCS are derived from Appendix K limits for fuel cladding temperature, etc., following any of the above accidents, as delineated in 10 CFR 50.46. The subsystem functional parameters are integrated so that the Appendix K requirements are met over the range of anticipated accidents and single failure assumptions. A risk-informed analysis was performed to analyze the effects of debris on the recirculation sump strainers and the ability to meet required acceptance criteria. The risk-informed analysis concluded that ECCS will perform its design basis functions with high probability and the risk associated with the effects of debris is very small, as defined by Regulatory Guide 1.174. The conclusion was based on plant-specific testing and analyses using inputs and assumptions that provide safety margin and defense-in-depth. Details of the risk-informed analysis are provided in USAR Appendix 6A.

6.3.1.2 Power Generation Design Basis

There are no power generation design bases for the ECCS function. Portions of the ECCS are also portions of the residual heat removal system (RHRS) and chemical and volume control system (CVCS) and are used during normal power operation. Power generation design bases for these portions of the ECCS are discussed in Sections 5.4.7 and 9.3.4, respectively.

Relief Valves

Relief valves are installed in various sections of the ECCS to protect lines which have a lower design pressure than the RCS. The valve stem and spring adjustment assembly are isolated from the system fluids by a bellows seal between the valve disc and spindle. The closed bonnet provides an additional barrier for enclosure of the relief valves. Table 6.3-2 lists the system's relief valves with their capacities and setpoints.

Butterfly Valves

Each main residual heat removal line has an air-operated butterfly valve which is normally open and is designed to fail in the open position. The actuator is arranged so that air pressure on the diaphragm overcomes the spring force, causing the linkage to move the butterfly to the closed position. Upon loss of air pressure, the spring returns the butterfly to the open position. These valves are left in the full-open position during normal operation to maximize flow from this system to the RCS during the injection mode of the ECCS operation. These valves are used during normal RHR system operation to control cooldown flowrate.

Each RHR heat exchanger bypass line has an air-operated butterfly valve, which is normally closed and is designed to fail closed. Those valves are used during normal cooldown to avoid thermal shock to the residual heat removal heat exchanger.

Net Positive Suction Head

Calculation of NPSH during containment sump recirculation is addressed in USAR Appendix 6A. ~~Available and required net positive suction head (NPSH) for ECCS pumps are shown in Table 6.3-1. Table 6.2.2-7 provides the assumptions and results of the NPSH analyses for the RHR pumps.~~ The safety intent of Regulatory Guide 1.1 is met by the design of the ECCS so that adequate NPSH is provided to system pumps. ~~In addition to considering the static head and suction line pressure drop, the calculation of available NPSH in the recirculation mode assumes that the vapor pressure of the liquid in the sump is equal to the containment ambient pressure. This ensures that the actual available NPSH is always greater than the calculated NPSH. To ensure that the required NPSH is available during the recirculation phase of ECCS operation, r~~Restriction orifices are provided in the four discharge lines into the RCS cold legs and in the two discharge lines into the RCS hot legs. The orifices are sized to provide the RHR flow rates specified in the notes to Figure 6.3-2.

TABLE 6.3-1 (Sheet 2)

Safety Injection Pumps	
Number	2
Design pressure, psig	1,750
Design temperature, F	300
Design flow rate, gpm	440
Design head, ft	2,780
Maximum flow rate, gpm	660
Head at maximum flow rate, ft	1,760
Discharge head at shutoff, ft	3,645
Required NPSH	25
Available NPSH	44
Design code	ASME III, Class 2
Seismic design	Category I
Driver:	
Type	Electric motor
Horsepower, hp	450
Rpm	3,600
Power	4,160 V, 60 Hz, 3-phase, Class IE
Start time	<5 sec
Design code	NEMA
Seismic design	Category I
Residual Heat Removal Pumps	
Number	2
Design pressure, psig	600
Design temperature, F	400
Design flow, gpm	3,800
Design head, ft	350
NPSH required~~ at 4,760 gpm, ft~~	~~21.01~~See USAR Appendix 6A
Available NPSH~~ at 4,760 gpm, ft~~	See USAR Appendix 6A~~23.79*~~
Design code	ASME III, Class 2
Seismic design	Category I
Driver:	
Type	Electric motor
Horsepower, hp	500
Rpm	1,800
Power	4,160 V, 60 Hz, 3-phase, Class IE
Start time	<5 sec
Design code	NEMA
Seismic design	Category I
Residual Heat Exchangers	
(See Section 5.4.7 for design parameters)	

~~* Includes 1.724 ft. total head loss across the sump strainer with both the Spray Pump and RHR Pump running in Recirculation, and a 0.56 ft. allowance for EDG frequency uncertainties.~~

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TABLE 7A-3, DATA SHEET 6.6

I. REGULATORY GUIDE 1.97 TABLE 2 RECOMMENDATIONS

VARIABLE IDENT. NO.	VARIABLE	RANGE	CATEGORY	PURPOSE
D.6.4	Containment Sump Water Temperature	50°F to 250°F	2	To monitor operation

II. WCGS DESIGN PROVISIONS

VARIABLE IDENT. NO.	VARIABLE	RANGE	SENSOR/TRANSMITTER		CONTROL ROOM				PLANT COMPUTER
					INDICATOR		RECORDER		
			IDENT. NO.	CL. 1E	PANEL	CL. 1E	PANEL	CL. 1E	
D.6.4	Containment Sump Water Temperature (unnecessary variable)								

III. REMARKS

1. This variable is unnecessary for the WCGS plant. The recommended purpose is to "monitor operation"; however, there is no system on WCGS for it to monitor. Containment cooling is monitored by the air temperature monitors described on data sheet 6.5.
2. Sump temperature is not required for RHR operation or assurance of NPSH available, since NPSH calculations conservatively assume saturated water was present. See Safety Evaluation Eleven of Section 6.2.2.1.3~~ and Table 6.2.2-7~~.
3. Primary system, PRT, and other containment parameters are all available to help determine the plant conditions. Sump level indications indicate the amount of water, and the other parameters indicate its source.
4. Note that proper RHR functions during the recirculation mode are provided by other variables described on data sheet 3.1.
5. The Callaway SER (NUREG-0830) in Section 6.2.1.1 (page 6-4) indicates that the NRC Staff agrees that this variable is not necessary for the SNUPPS plants and finds this exception to the guidelines of Regulatory Guide 1.97 acceptable.
6. The Callaway SER also addresses the containment heat removal systems and similarly finds them acceptable. Page 6-10 indicates that the RHR system serves to remove heat from the containment during the recirculation mode following a LOCA by cooling the containment sump fluid in the RHR heat exchanger. During this mode of operation, the RHR inlet temperature monitors described on Data Sheet 3.1 would provide indication of the containment sump water temperature. As noted on Data Sheet 3.1, the RHR heat exchanger inlet temperature is not considered to be part of the Regulatory Guide 1.97 data base.

APPENDIX 6A

Resolution of NRC Generic Letter 2004-02

6A.1 Introduction and Risk-Informed Approach Summary

NRC Generic Letter (GL) 2004-02 (Reference 1) required licensees to perform an evaluation of the emergency core cooling system (ECCS) and containment spray system (CSS) recirculation functions, and the flow paths necessary to support those functions, based on the potential susceptibility of sump strainers to debris blockage during design basis accidents that require recirculation operation of ECCS or CSS. This generic letter resulted from Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance." As a result of the evaluation required by GL 2004-02, and to ensure system function, sump strainer design modifications were implemented.

The plant licensing basis considers long-term core cooling (LTCC) following a loss of coolant accident (LOCA) as identified in 10 CFR 50.46(b)(5). Long-term cooling is supported by the ECCS, which includes the charging, safety injection (SI), and residual heat removal (RHR) systems. These systems and the CSS are subject to the effects of accident-generated debris because they rely on the containment emergency sumps in the recirculation mode. Debris generated from non-LOCA initiating events are also considered. The risk-informed evaluation analyzes the following events:

1. Small, medium, and large LOCAs due to:
   i. Pipe breaks
   ii. Failure of non-piping components
   iii. Water hammer
2. Secondary side breaks inside containment that result in a consequential LOCA (e.g., due to failure to terminate safety injection, loss of auxiliary feedwater, or a stuck open power operated relief valve) that requires sump recirculation

In addition to the internal events listed above, internal fires, seismic events, and other external events were also considered.

A risk-informed evaluation was performed to respond to GL 2004-02. The evaluation provides confidence that the sump design supports LTCC following a LOCA. The evaluation meets the acceptance guidelines for a very small risk impact as defined in Regulatory Guide (RG) 1.174 (Reference 2).

The licensing basis with regard to effects of debris is determination of a high probability that the ECCS and CSS can perform their design basis functions based on WCGS-specific testing using an NRC-approved methodology. The risk from breaks that could generate debris and that do not meet one (or more) of the GSI-191 acceptance criteria is very small and is, therefore, acceptable in accordance with the RG 1.174 guidelines (Reference 2).

The use of a risk-informed method, rather than the deterministic methods prescribed in the regulation, required an exemption to 10 CFR 50.46(a)(1), which has been granted pursuant to 10 CFR 50.12.

The risk-informed method used to analyze the impact on long-term core cooling due to LOCA-generated debris is the threshold break approach. This approach shows that all breaks that are smaller than the threshold break size would not result in any strainer or reactor core failures due to the effects of debris, while those breaks larger than or equal to the threshold break size are conservatively assumed to fail the acceptance criteria. The acceptance criteria can be divided into two categories – criteria analyzed for all breaks and criteria analyzed for breaks up to the threshold break size:

Criteria analyzed for all breaks:

- Flow paths upstream of the strainer would not be sufficiently blocked to prevent water from reaching the strainer (i.e., upstream effects)
- Blockage and wear of components downstream of the strainer do not exceed the limits given in WCAP-16406-P-A (Reference 7) (i.e., ex-vessel downstream effects)

Criteria analyzed for breaks up to the threshold break size:

- Strainer head loss does not result in negative pump net positive suction head (NPSH) margins
- Strainer head loss does not exceed strainer structural limits
- Strainer head loss does not result in a void fraction at the pump suctions that exceeds the acceptance criteria given in NEI 09-10 (Reference 10)
- Strainer head loss does not result in flashing immediately downstream of the strainer
- Strainer flow conditions do not result in air ingestion due to vortexing
- In-vessel fiber loads and other relevant parameters do not exceed the limits given in WCAP-17788-P (Reference 8)

Note that upstream effects and ex-vessel downstream effects have been evaluated and shown not to cause failures for all postulated breaks.

The minimum water levels for both LBLOCA and SBLOCA result in full strainer submergence. Therefore, no evaluation of a partially submerged strainer is required.

The LOCA frequency of breaks greater than the threshold break size was determined based on generic industry frequency. This was assigned as the change in core damage frequency (ΔCDF) associated with GSI-191 failures due to small, medium, and large break LOCAs. The base CDF, base large early release frequency (LERF) and an estimate of the change in LERF (ΔLERF) were determined using inputs from the Wolf Creek probabilistic risk assessment (PRA) model.

The comparison of the CDF, LERF, ΔCDF, and ΔLERF to the risk acceptance guidelines in RG 1.174 show that the risk from the proposed change is "very small" (i.e., in Region III of RG 1.174). The methodology includes conservatisms in the plant-specific testing and in the assumption that all unbounded breaks result in loss of core cooling.

Key elements of the risk-informed evaluation include:

1. The methodology used to quantify the amount of debris generated at each break location, including the assumed zone of influence (ZOI) size based on the target destruction pressure and break size, and the assumed ZOI shape (spherical or hemispherical) based on whether the break is a double-ended guillotine break or partial break.
2. The methodology used to evaluate debris transport to the containment sump recirculation strainers.
3. The methodology used to quantify chemical precipitates, including the refinements to WCAP-16530-P-A, application of the solubility correlation, and application of the WCAP-17788-P autoclave testing (Reference 6).
4. The strainer debris limits shown in TS Bases Table B 3.6.8-1, which are based on tested and analyzed debris quantities. Changes to these debris limits are also subject to 10 CFR 50.71(e) reporting requirements.
5. The methodology and acceptance criteria used to assess ex-vessel component blockage and wear.
6. The methodology used to assess in-vessel fiber accumulation and the associated limits.
7. The methodology used to quantify ΔCDF and ΔLERF.

6A.2 Debris Generation

Post-accident debris includes insulation, fire barrier, and qualified coatings debris generated within the ZOI of the pipe break, as well as latent debris, unqualified coatings, and miscellaneous debris in containment. The pipe break characterization followed the methodology of NEI 04-07 (Reference 3) and associated NRC safety evaluation (SE) (Reference 4), with the exception that it characterized a full range of breaks rather than just the worst-case breaks as suggested by NEI 04-07. Double-ended guillotine breaks (DEGBs) and partial breaks on every in-service inspection (ISI) weld within the Class 1 pressure boundary, excluding the reactor nozzle breaks, were considered.

In the debris generation calculation, a three-dimensional CAD model of the WCGS containment building was used to model the ZOI for each postulated break. ZOIs representing possible breaks were modeled at each ISI weld inside the first isolation valve and outside of the reactor cavity.

Breaks of sizes ranging from 0.375 inch to 31 inches were modeled at each weld, as applicable. DEGBs are modeled using a spherical ZOI with a radius proportional to the pipe inner diameter. Partial breaks are any breaks smaller than a DEGB and are modeled using a hemispherical ZOI with a radius proportional to the equivalent break size. Because the orientation of partial breaks can have a significant effect on the results, partial breaks were modeled every 45 degrees around the circumference of the pipe at each weld. Credit was taken for shielding by concrete walls. While DEGBs on main loop piping are typically bounding with regard to the volume of debris generated, smaller breaks are more likely to occur.

Although the probability of occurrence is low, a secondary side break inside containment could require ECCS recirculation. A simplified and bounding evaluation was performed to assess the risk contribution from the secondary side breaks. It was assumed that all secondary side breaks that require ECCS recirculation (e.g., due to a feed and bleed scenario) would fail due to the effects of debris. This is a conservative assumption because secondary side breaks would generate less debris than equivalent primary side breaks due to the lower pressure on the secondary side. Also, the flow rate through the strainer required for feed and bleed cooling is significantly lower than the ECCS flow rate for a large break LOCA.

Since different material types have different destruction pressures, a ZOI was determined for each type of material. The quantity of generated debris for each break case was calculated using these material specific ZOI sizes.

Evaluation of failed unqualified coatings and their transport to the recirculation strainers is included in the analysis. Unqualified coatings could include: coatings within containment that do not have a specified preparation, application, or inspection compliant with plant specifications; coatings inaccessible for inspection; and coatings applied by vendors on vendor supplied items that cannot be qualified. There are several types of unqualified coatings applied over numerous substrates within containment outside the primary shield wall, including epoxies, inorganic zincs, and alkyds. It was assumed that unqualified coatings fail at the start of sump recirculation for all postulated breaks.

The total amount of latent debris in containment was calculated based on walkdown data but a higher value was assumed for conservatism, providing operating margin. Per the guidance in NEI 04-07 Volume 2, latent debris is assumed to consist of 15 percent fiber and 85 percent particulate by mass (Reference 4).

A total surface area of foreign materials, such as labels, tape, stickers, placards and other miscellaneous materials, were identified via walkdown. However, a greater surface area was assumed for miscellaneous debris in the strainer evaluation to account for the impact of foreign materials. Per the guidance in NEI 04-07 (Reference 3) and the SE (Reference 4), the total surface area of miscellaneous debris was assumed to block an equivalent surface area of the sump strainers after allowance for 25% overlap.

6A.3 Debris Transport to the Sump Strainers

The debris transport analysis determines the fraction of each type and size of debris that could be transported to the sump strainers. The evaluation considers debris transport during the blowdown, washdown, pool fill, and recirculation phases based on plant-specific layout and flow conditions. For the recirculation phase, computational fluid dynamic (CFD) modeling was used to determine the sump pool flow conditions and transport of various debris types inside the pool for different break locations and pump lineups (e.g., number of ECCS and CSS trains in service).

Potential upstream blockage points in containment were reviewed. Specifically, the doorways through the bioshield wall and the refueling canal drains were evaluated and were shown that blockage would not occur during post-LOCA operation for any size breaks.

6A.4 Chemical Effects

The post-LOCA sump strainer chemical effects analysis methodology includes:

- Quantification of chemical precipitates using the WCAP-16530-NP-A (Reference 5) base methodology.
- Introduction of pre-prepared precipitates in strainer head loss testing.
- Application of an aluminum solubility correlation to determine the maximum precipitation temperature for strainer head loss related evaluations.
- Use of autoclave test results to determine the minimum precipitation timing for in-vessel downstream effects.

Multiple breaks were postulated at every Class 1 ISI pipe weld inside the first isolation valve and outside of the reactor cavity for evaluating debris generation quantities. The amount of chemical precipitate was determined using the WCAP-16530-NP-A (Reference 5) methodology for the bounding quantities of LOCA generated debris. The amount of chemical precipitate was maximized by applying conservative plant-specific inputs, such as pH, temperature, aluminum quantity, and spray times.

The generated amounts of chemical precipitates were used as inputs for strainer head loss testing to determine the head loss across the strainers. Aluminum oxyhydroxide (AlOOH) used during testing was prepared according to the WCAP-16530-NP-A (Reference 5) recipe and settling test criteria.

An aluminum solubility correlation, developed by Argonne National Laboratory, was used to determine a maximum precipitate formation temperature, which effectively delays the onset of aluminum precipitation when analyzing strainer head loss. Additionally, WCAP-17788-P, Volume 5 (Reference 6) autoclave test results were used to determine the minimum precipitation timing for the evaluation of in-vessel downstream effects.

6A.5 Post-LOCA Containment Sump Water Level

The containment sump pool water volume following a LOCA was determined by considering all water sources (i.e., the refueling water storage tank, reactor coolant system (RCS), the safety injection accumulators, and the spray additive tank) and subtracting various holdup volumes. The holdup volumes include steam holdup, filling of empty pipes, water heldup in the RCS, water film on surfaces, water in transit, and miscellaneous holdup volumes throughout containment. The sump pool volume was used to determine the pool water level using a correlation between pool water depth and volume.

6A.6 Blockage and Wear of Downstream Components

An analysis bounding all break sizes was performed to evaluate the impact of debris on the wear or blockage of the ECCS and CSS piping and components downstream of the strainer following a LOCA. This ex-vessel downstream effects evaluation used the methodology presented in WCAP-16406-P-A (Reference 7). The analyzed effects of debris ingested through the containment sump strainers during the recirculation mode include erosive wear, abrasion, and potential blockage of downstream flow paths.

ECCS and CSS system valves, piping, instrument tubing, and heat exchangers were evaluated for susceptibility to blockage from the debris that passes through the sump strainers. It was concluded that the components can accommodate sump bypass particles without blockage.

The heat exchangers, orifices and spray nozzles were evaluated for the effects of erosive wear for a constant debris concentration over the mission time of 30 days. The erosive wear on these components was determined to be insufficient to affect the system performance.

The effects of debris ingestion were evaluated for three aspects of pump performance: hydraulic performance, mechanical shaft seal assembly performance, and mechanical performance (vibration). The evaluation concluded that hydraulic and mechanical performance remain acceptable when operating with debris in the recirculation fluid.

The ECCS system valves were evaluated for erosive wear when operating with debris in the recirculation fluid and were found to have acceptable results. The evaluation was performed using the wear criteria in WCAP-16406-P-A (Reference 7).

6A.7 Strainer Head Loss and Fiber Penetration Testing

Strainer head loss tests were performed to measure the head losses of the conventional debris (fiber and particulate) and chemical precipitate debris generated and transported to the sump strainers following a LOCA. Different test cases were performed with the thin bed and full debris load protocols, following the 2008 NRC staff review guidance (Reference 14). Fiber-only penetration testing was also performed to measure time-dependent fiber penetration through the strainer as fiber debris loads onto the strainer. A correlation was derived from the test data and was applied to determine the amount of fiber that could pass through the strainer and reach the reactor core during the post-LOCA recirculation phase.

Both test programs used a test strainer and flow rates that were prototypical to WCGS. The tested debris quantities bounded those associated with the threshold break size. The head loss and fiber penetration testing results provided basis when determining the threshold break size.

6A.8 Determination of Threshold Break Size

To implement the simplified risk-informed approach, a threshold break size was determined such that breaks of this size and smaller do not fail any GSI-191 acceptance criteria. The strainer failure due to accumulation of debris on the strainer and reactor core failure due to accumulation of debris within the reactor were first analyzed separately to establish their own threshold break sizes. The overall threshold break size was defined as the smaller of the two.

Both the strainer and in-vessel threshold break sizes were calculated based on the most limiting equipment configurations. The bounding equipment configuration for strainer head loss is an assumed single train failure (i.e., all pumps running on only one train), because this maximizes both the flow rate and debris accumulation on the active strainer. The bounding equipment configuration for in-vessel effects is an assumed failure of both containment spray (CS) pumps (i.e., both trains of ECCS pumps running without either train of CS), because this maximizes strainer area for penetration and minimizes the core bypass flow. The most likely scenario in the event of a large LOCA is that there would be no random equipment failures, and both trains of ECCS and CS would be operating. For this scenario, it is expected that the smallest break that fails would be larger than the threshold break size. If all pumps are running, the debris would be spread across both strainers (making it much less likely that either strainer would fail). Also, a large fraction of the debris that penetrates the two strainers would bypass the core with the flow through the two CS pumps (making it less likely for sufficient debris accumulation to result in core blockage). Therefore, the assumption that the threshold break size applies to all equipment configurations provides a significant level of conservatism in the GSI-191 risk quantification.

The following criteria were considered in the determination of the threshold break size: pump NPSH, strainer structural limit, void fraction, flashing, vortexing, and in-vessel fiber load. The breaks that are larger than the threshold break size were assumed to result in strainer and reactor core failures. Based on the results of the evaluation, the threshold break size was found to be 10 inches for both strainer failure and reactor core failure; thus, the overall threshold break size was defined as 10 inches.

Pump NPSH Criteria

The RHR and CS pump NPSH margin was calculated based on the NPSH available minus the NPSH required for the respective pumps. NPSH available was calculated based on containment pressure, sump temperature, water level, losses in the pump suction piping (including the strainer head loss), and vapor pressure. The NPSH required was first determined using pump vendor curves and was increased to account for the impact of void fraction at pump suction using the RG 1.82 methodology.

The most limiting pump NPSH margin was analyzed at a sump temperature of 212°F. The analysis assumed that the containment pressure is equal to water vapor pressure without crediting containment accident pressure.

Because the SI pumps and centrifugal charging pumps (CCPs) take suction from the RHR pumps during recirculation, only the NPSH margins of the RHR and CS pumps are calculated.

Acceptable NPSH margin results were determined for breaks up to the threshold break size of 10 inches. During the recirculation mode, the minimum RHR pump NPSH margin was determined to be 1.2 ft at a sump temperature of 212°F and the maximum RHR pump flow rate of 4,760 gpm. The minimum CS NPSH margin was determined to be 2.0 ft at 212°F and the maximum CS pump flow rate of 3,950 gpm.

Strainer Structural Criteria

The recirculation sump strainers are installed inside the sump pits with approximately one foot extending above the containment floor. The sumps are located outside the secondary shield walls and are also protected by a concrete slab above. The arrangement of the sump strainers protects them from missiles, pipe whip or jet impingement from a LOCA. The strainers are analyzed for seismic loads, live loads, thermal loads, hydrodynamic loads, and applicable load combinations.

The analyzed strainer structural limit for each strainer is 4 ft at a sump temperature of 268°F and 5.5 ft at a sump temperature of 175°F. The head loss across the strainers was compared to this value to ensure that the structural margin is not exceeded for postulated breaks up to the threshold break size of 10 inches.

Void Fraction Criteria

The degasification analysis determined void fraction at pump suction due to flow head loss through the debris-laden strainer. The analysis used post-LOCA containment pressure and sump temperature curves as inputs and considered both minimum and maximum safeguards cases to ensure the bounding scenario was captured. Additionally, a combination of conservative inputs (e.g., strainer submergence and head loss), which do not occur simultaneously, was used in the analysis to ensure bounding results.

The void fraction was first determined downstream of the strainer at the strainer midpoint elevation. The voids formed at the strainer were assumed to transport intact to the pump suction, conservatively neglecting any redissolution of the voids as they transport to the pump suction. The void fraction at the pump suction was determined by accounting for compression of the voids due to pressure increase as they transport downward to the pump suction.

The resulting void fractions at the pump suction are significantly lower than the NEI 09-10 (Reference 10) criterion of 2%. The void fractions at a sump temperature of 212°F were used to make a correction to the pump NPSHr using the RG 1.82 methodology.

Flashing Criteria

The acceptance criterion is zero flashing as the fluid experiences a pressure drop across the debris bed and strainer. Strainer flashing was calculated by comparing the internal strainer pressure at the top of the strainer to the sump water saturation pressure. If the internal strainer pressure is less than the saturation pressure, the water flashes and the strainer is assumed to fail. By crediting a small amount of accident pressure (i.e., above saturation pressure), there will be no strainer failure due to flashing for breaks up to the threshold break size of 10 inches.

Strainer Vortexing Criteria

An evaluation of the potential for air-entraining vortexing during post-LOCA recirculation through the sump strainers was performed based on observations during strainer testing. The vortex testing was conducted using a prototype strainer assembly with a conservatively higher approach velocity and lower strainer submergence. Comparison of test results with plant conditions showed no vortex formation for the plant strainers up to the tested debris limits associated with the threshold break size of 10 inches. Note that the vortex testing conservatively ignored the vortex breaking straightening vanes inside the RHR and CS line suction nozzles.

In-vessel Fiber Load Criteria

During the post-LOCA sump recirculation phase, debris that passes through the ECCS sump strainers could accumulate at the reactor core inlet or inside the reactor vessel, potentially challenging LTCC. In-vessel downstream effects were analyzed in accordance with the NRC review guidance (Reference 9) and WCAP-17788-P (Reference 8), using plant-specific fiber penetration test data.

Fiber penetration testing was performed to collect time-dependent fiber penetration data using a prototypical strainer array. A fiber-only penetration test was conducted with test parameters selected to be representative of the most conservative plant strainer configuration and post-accident conditions (e.g., debris characteristics and composition, flow rate, and water chemistry). The test results were used to derive a curve-fit to quantify fiber penetration at plant conditions.

Methods and acceptance criteria contained in WCAP-17788-P, Revision 1 (Reference 8) were used to evaluate the accumulation of fiber inside the reactor vessel. Applicability of the WCAP-17788-P methods to WCGS was demonstrated in accordance with the NRC staff review guidance for in-vessel effects (Reference 9). The evaluation used time-dependent fiber penetration fractions obtained from the fiber penetration test. The analysis concluded that post-accident LTCC will not be challenged by accumulation of debris within the reactor core for all postulated LOCAs up to the threshold break size of 10 inches.

6A.9 Analyzed Debris Limits

Containment accident generated and transported debris is defined as the quantity of debris calculated to arrive at the containment sump strainers. As described in the previous sections, the evaluation of the effects of debris includes strainer head loss, downstream ex-vessel effects, and downstream in-vessel effects.

Based on the tested and analyzed debris quantities, strainer debris limits were defined as shown in TS Bases Table B 3.6.8-1. These debris limits cannot be exceeded for breaks smaller than or equal to 10 inches. Larger breaks may exceed these debris limits and will not increase the risk beyond the RG 1.174 Region III acceptance guidelines (Reference 2).

If debris quantities greater than the analyzed debris limits are identified, the containment sump LCO (TS 3.6.8) would not be met and Condition A would be entered. Immediate action would be initiated to mitigate the condition and restore the sump to operable status in accordance with the TS and as described in the TS Bases.

6A.10 References

1. NRC Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents for Pressurized-Water Reactors," September 13, 2004.
2. Regulatory Guide 1.174, "An Approach for Using Probabilistic Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, January 2018.
3. NEI 04-07 Volume 1, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Revision 0, December 2004.
4. NEI 04-07 Volume 2, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02, Revision 0, December 6, 2004."
5. WCAP-16530-NP-A, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," March 2008.
6. WCAP-17788-P Volume 5, "Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090) – Autoclave Chemical Effects Testing for GSI-191 Long-Term Cooling," Revision 1, December 2019.
7. WCAP-16406-P-A, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," Revision 1, March 2008.
8. WCAP-17788-P Volume 1, "Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090)," Revision 1, December 2019.
9. ML19228A011, "U.S. Nuclear Regulatory Commission Staff Review Guidance for In-Vessel Downstream Effects Supporting Review of Generic Letter 2004-02 Responses," September 4, 2019.
10. NEI 09-10, Revision 1a-A, "Guidelines for Effective Prevention and Management of System Gas Accumulation," April 2013.
11. NUREG-1829 Volume 1, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," April 2008.
12. Generic Letter 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems," January 11, 2008.
13. Regulatory Guide 1.82, Revision 4, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," March 2012.
14. ML080230038, "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Strainer Head Loss and Vortexing," March 2008.

Wolf Creek Nuclear Operating Corporation
Licensing Submittal for a Risk-Informed Resolution of Generic Letter 2004-02

Attachment VII

Overview of Risk-Informed Approach

Table of Contents

Overview of Risk-Informed Approach

1.0 Introduction

Generic Safety Issue (GSI)-191 was raised by the United States Nuclear Regulatory Commission (NRC) to ensure that post-accident debris blockage will not impede or prevent the operation of the emergency core cooling system (ECCS) or containment spray system (CSS) in recirculation mode at pressurized water reactors (PWRs) during loss of coolant accidents (LOCAs) or other high energy line break (HELB) accidents that would require recirculation (Reference 14). In 2010, due to the ongoing challenges of resolving GSI-191, the NRC commissioners issued a staff requirements memorandum (SRM) directing the NRC staff to consider new and innovative resolution approaches (Reference 1). One of the approaches included in the SRM was the option of addressing GSI-191 using a risk-informed approach. In 2011, South Texas Project (STP) initiated a multi-year effort as a pilot plant to define and implement a risk-informed approach to address the concerns associated with GSI-191. In 2012, the NRC staff issued SRM-SECY-12-0093 (Reference 2) providing recommendations for closure options. Wolf Creek Nuclear Operating Company (WCNOC) selected Option 2b (risk-informed resolution path) for resolution of GSI-191 at the Wolf Creek Generating Station (WCGS) and closure of NRC Generic Letter (GL) 2004-02 (Reference 14). The method that will be used to implement the risk-informed resolution is a simplified threshold break approach. The results of this approach are used to determine the change in core damage frequency (CDF) and large early release frequency (LERF) associated with post-accident debris effects and does not require the use of the WCGS probabilistic risk assessment (PRA) model.

The results of the WCGS evaluation show with high confidence that the risk associated with GSI-191 is very low, as defined by RG 1.174 Region III (Reference 4). The analysis includes significant safety margin and does not affect any of the existing defense-in-depth measures that are in place to protect the public.

2.0 Systematic Risk Assessment of Debris

As described in RG 1.174 (Reference 4), the systematic risk assessment should consider all hazards, initiating events, and plant operating modes. However, a screening process can be used to eliminate scenarios that are not relevant, not affected by debris, or have an insignificant contribution. Acceptance criteria were developed to identify the postulated breaks that contribute to an increase in risk.

The specific GSI-191 acceptance criteria that were considered are:

1. Flow paths upstream of the strainer would not be sufficiently blocked to prevent water from reaching the strainer.
2. Strainer head loss does not result in negative pump net positive suction head (NPSH) margins.
3. Strainer head loss does not exceed the strainer structural limits.
4. Strainer head loss does not result in a void fraction at the pump suctions that exceeds the acceptance criteria given in NEI 09-10 (Reference 22)

Overview of Risk-Informed Approach

5. Strainer head loss does not result in flashing immediately downstream of the strainer.
6. Strainer head loss does not exceed half of the submergence depth of a partially-submerged strainer (not applicable for WCGS because the strainer is fully submerged even for a small break LOCA).
7. Strainer flow conditions do not result in air ingestion due to vortexing.
8. Blockage and wear of components downstream of the strainer do not exceed the limits given in WCAP-16406 (Reference 23).
9. In-vessel fiber loads and other relevant parameters do not exceed the limits given in WCAP-17788 (Reference 24).

The results of the screening criteria and determination of a threshold break size are described in Section 2.4.1.

Figure 1 shows the various calculations and analyses that are used in the risk and uncertainty quantification. Although not shown in the figure, the containment sump water volume and level calculation is also an important input for several aspects of the GSI-191 evaluation.

Overview of Risk-Informed Approach

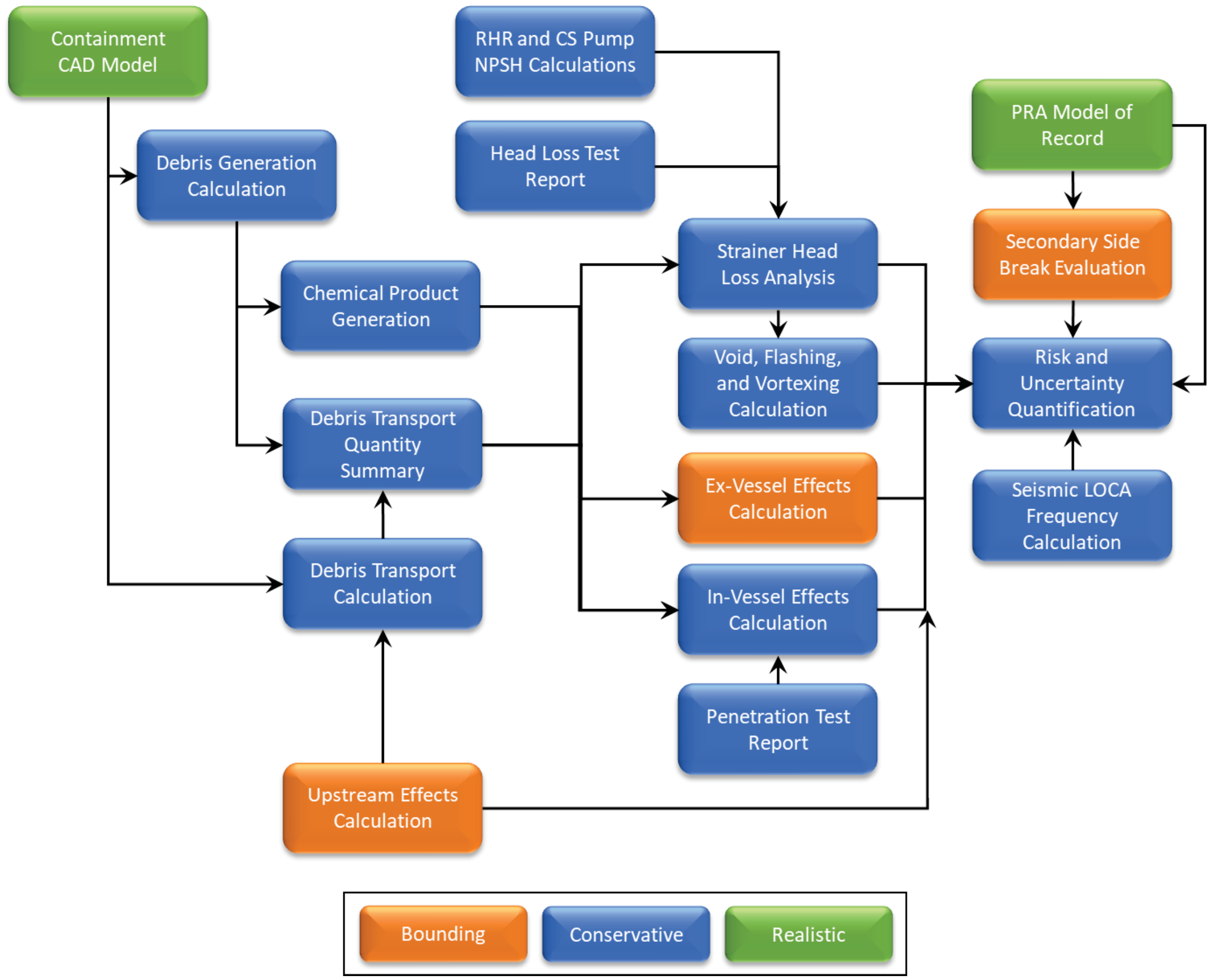

Figure 1: Calculations and Analyses Used in Risk and Uncertainty Quantification

2.1 Hazards, Initiating Events, and Plant Operating Modes

The scenarios that need to be considered for GSI-191 are scenarios that require recirculation through the ECCS and CSS strainers. If recirculation is not required, there is no potential for debris-related failures of the strainers, pumps, downstream components, or reactor core.

A systematic process was used to determine the hazards, initiating events, and operating modes to be addressed in the WCGS GSI-191 analysis. The process was based on the identification of hazards and initiating events with the potential to (1) generate debris inside containment, (2) require sump recirculation for mitigation of the event, and (3) result in debris transport to the containment sump. Hazards or initiating events that do not meet these three criteria were excluded from the analysis.

Overview of Risk-Informed Approach

Among internal plant hazards, the following initiating events do not have the potential to generate debris inside containment and were screened from the analysis:

- Steam generator tube rupture (SGTR)
- Interfacing systems LOCAs (ISLOCAs) that discharge outside containment
- Anticipated transients including inadvertent safety injection (SI), inadvertent or stuck-open power operated relief valves (PORVs) that discharge to the pressurizer relief tank (PRT), and loss of offsite power
- Secondary side breaks outside containment
- Initiating events due to loss of component cooling water, loss of service water, and loss of AC or DC power

The internal initiating events that have the potential to generate debris inside containment are LOCAs (small, medium, and large) due to breaks inside containment, and secondary side breaks inside containment (SSBIs). Although the reactor vessel failure initiating event has the potential to generate a significant quantity of debris, this event is assumed to be a catastrophic failure that exceeds the capacity of the ECCS and results in core damage. Therefore, the effects of debris do not need to be addressed for this event.

Internal flood hazards were evaluated. Pipe breaks outside containment do not have the potential to generate debris inside containment. Pipe breaks that flood inside containment are evaluated as LOCA or secondary side break internal events.

The quantitative risk assessment was performed for LOCAs and SSBIs that occur during full power operation (i.e., Mode 1), which is assumed to be equivalent or bounding compared to the other operating modes. This is a reasonable assumption because the reactor coolant system (RCS) pressure and temperature (key inputs affecting the zone of influence (ZOI) size) would either be approximately the same or significantly lower for Modes 2 through 6. Also, the flow rate required to cool the core (a key input affecting core blockage) would be significantly reduced for low power or shutdown modes.

Consistent with the guidance in NUREG/CR-6850 (Reference 15), internal fire hazards are not assumed to result in pipe breaks. However, fire induced LOCAs can occur, including spurious opening of a pressurizer PORV, spurious reactor head vent, continuous letdown, spurious ISLOCA, or reactor coolant pump (RCP) seal LOCA due to loss of seal cooling. Of these, only an RCP seal LOCA has the potential to generate debris inside containment. A spurious opening of a pressurizer PORV or spurious reactor head vent is discharged to the PRT, which has negligible sources of debris near the rupture disk. Spurious ISLOCAs or continuous letdown would discharge outside containment. Therefore, these scenarios are screened out of the analysis. The quantity of debris generated by an RCP seal LOCA would be equivalent to the quantity generated by a small or medium LOCA, which was found to not challenge the sump strainers; therefore, fire induced RCP seal LOCAs were also screened from the analysis.

Seismic events can result in direct or indirect LOCAs that generate and transport debris similar to a random pipe break LOCA. A direct seismically-induced LOCA occurs when the RCS pressure boundary fails due to seismic forces. An indirect seismically-induced

LOCA occurs when a support or structure fails due to seismic forces, which subsequently causes an RCS pressure boundary failure. The frequency of seismically-induced LOCAs is not included in the LOCA frequency estimates in NUREG-1829 (Reference 6) and therefore, must be considered separately for the GSI-191 risk quantification.

High wind events, including tornados, would not generate debris inside containment and therefore are screened from the GSI-191 risk quantification.

An evaluation of external hazards conducted for WCGS concluded that, in addition to seismic and high wind events, the only potentially significant external hazards applicable to WCGS are:

- Aircraft impacts
- Lightning strikes
- Local intense precipitation

None of these external hazards have the potential to generate debris inside containment and are screened from the GSI-191 analysis.

In summary, the following events are included in the scope of the WCGS GSI-191 analysis and were addressed qualitatively or with a conservative or bounding quantitative assessment:

1. Small, medium, and large LOCAs due to:
    a. Pipe breaks
    b. Failure of non-piping components
    c. Water hammer
2. SSBIs that result in a consequential LOCA (e.g., due to failure to terminate SI, loss of auxiliary feedwater, or a stuck open PORV) that requires sump recirculation
3. Seismically-induced LOCAs

These hazards and initiating events are discussed in more detail in Section 2.3.

2.2 Baseline CDF and LERF

The PRA models for WCGS include internal events, internal flooding, internal fire, and high winds. The baseline CDF and LERF values were taken from the quantification notebooks for each of these models and are shown in Table 1.

Table 1: Baseline CDF and LERF Values

PRA Model	CDF (yr^{-1})	LERF (yr^{-1})
Internal Events	7.25E-06	7.31E-08
Internal Flooding	9.06E-06	3.77E-08
Internal Fire	5.49E-04	1.33E-05
High Winds	3.40E-06	7.98E-09
Total	**5.69E-04**	**1.34E-05**

Overview of Risk-Informed Approach

The total CDF for internal and external events for WCGS is between 1.0E-04 yr^{-1} and 1.0E-03 yr^{-1}, and the total LERF is between 1.0E-05 yr^{-1} and 1.0E-04 yr^{-1}. As shown in Figure 2 and Figure 3, these values exceed the RG 1.174 total risk guidelines for Region II, but are within the guidelines for Region III (Reference 4). This means that the change in CDF (ΔCDF) and change in LERF (ΔLERF) values must fall within the y-axis boundaries for Region III. Note that these CDF and LERF results are based on the current models. The fire PRA model has not yet been finalized or peer reviewed, and the CDF and LERF values are expected to decrease.

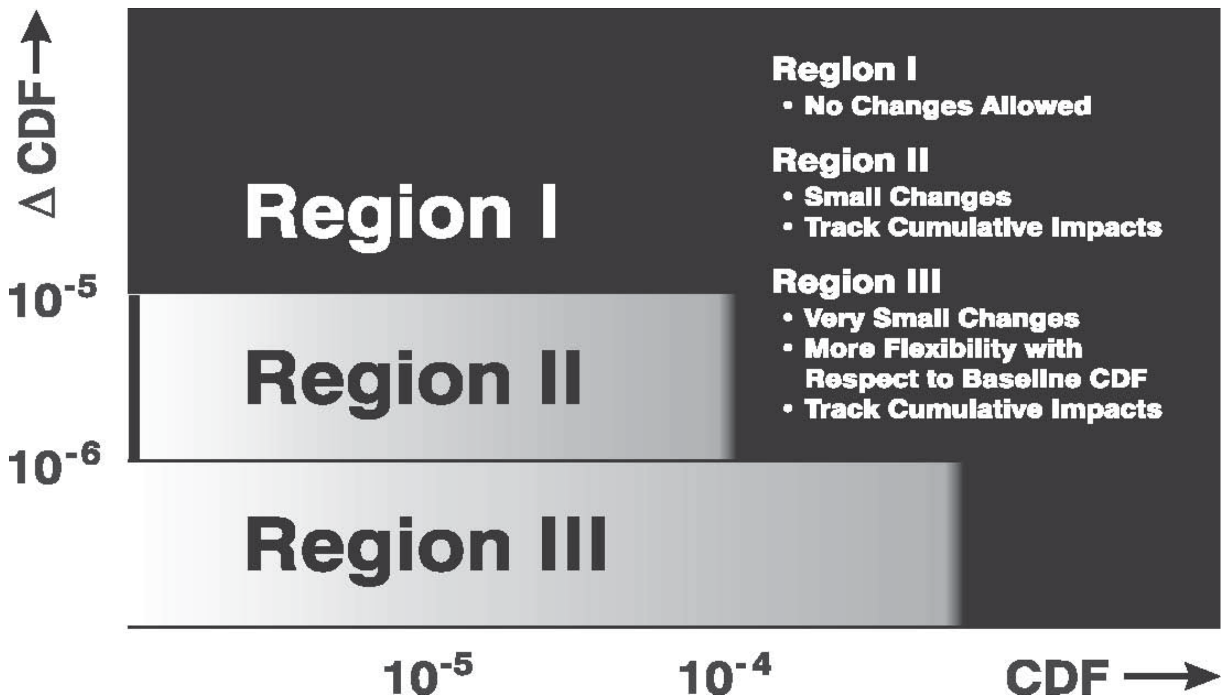

Figure 2: RG 1.174 Risk Acceptance Guidelines for CDF and ΔCDF

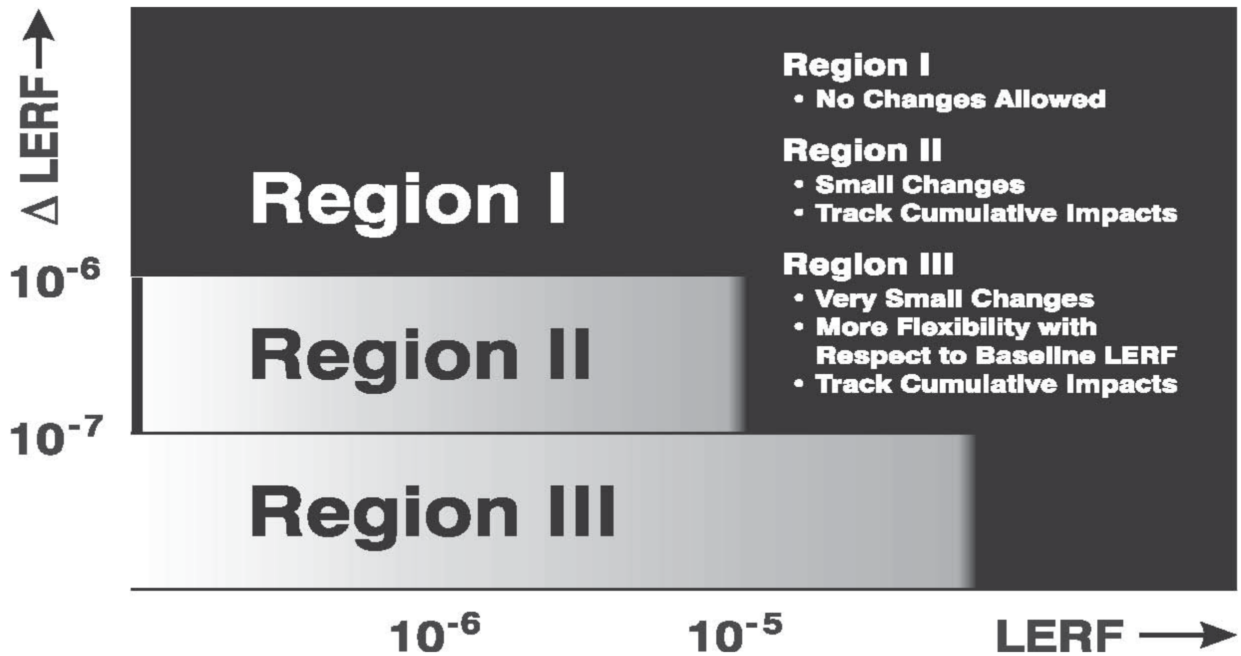

Figure 3: RG 1.174 Risk Acceptance Guidelines for LERF and ΔLERF

2.3 Initiating Event Frequencies

Initiating event frequencies were determined for pipe LOCAs, secondary side breaks, and seismically induced LOCAs. As described in Sections 2.3.1 and 2.3.2, non-pipe LOCAs and water hammer induced LOCAs were qualitatively screened out.

2.3.1 Small, Medium, and Large LOCAs

Small, medium, and large break LOCAs include both pipe breaks and non-pipe breaks. For the WCGS GSI-191 evaluation, breaks were analyzed at pipe welds on the RCS and connected system piping. As described in Section 3.b of Attachment VIII, a range of break sizes and orientations were evaluated at in-service inspection (ISI) welds in the unisolable portion of the Class 1 pressure boundary. The use of Class 1 ISI welds for break locations is both systematic and thorough because there are multiple ISI welds on every RCS pipe and the welds cover the range of possible break locations. In addition, a weld is generally closer to equipment that has a large quantity of insulation compared to a break in the middle of a span of straight pipe (e.g., a break on the hot leg weld at the base of the steam generator (SG) will typically generate more debris than a break halfway between the SG and reactor vessel). Also, “welds are almost universally recognized as likely failure locations because they can have relatively high residual stress, are preferentially attacked by many degradation mechanisms, and are most likely to have preexisting fabrication defects” (Reference 6, p. xviii). Non-pipe LOCAs were not explicitly evaluated (as discussed below), but are implicitly included by using the NUREG-1829 LOCA frequencies, which include non-pipe contributors (Reference 6, p. 1-2).

Overview of Risk-Informed Approach

The WCGS internal events PRA model uses the following definitions for small, medium, and large LOCAs:

- Small LOCA = 0.5 to 2-inch breaks
- Medium LOCA = 2 to 6-inch breaks
- Large LOCA = Greater than 6-inch breaks

Therefore, given a threshold break size of 10 inches (see Section 2.4.1), small and medium break LOCAs do not contribute to the GSI-191 risk quantification.

The large LOCA exceedance frequency can be determined using log-linear interpolation of the geometric mean (GM) LOCA frequencies in NUREG-1829 (Reference 6) Table 7.19 (25-year values) based on the threshold break size. The evaluations used to determine the threshold break size showed that all breaks at weld locations that are smaller than or equal to the threshold break size would not fail due to the effects of debris, while those breaks larger than the threshold break size are conservatively assumed to fail. Log-linear interpolation can be performed using the following equation:

$$y = 10^{\log(y_0)+\left(\frac{\log(y_1)-\log(y_0)}{x_1-x_0}\right)(x-x_0)}$$

Where:

y = LOCA frequency at the threshold break size (x)
y_0 = LOCA frequency at a smaller break size (x_0)
y_1 = LOCA frequency at a larger break size (x_1)

Given a GM LOCA frequency of 1.6E-06 yr^{-1} for 7-inch breaks and a GM LOCA frequency of 2.0E-07 for 14-inch breaks (Reference 6, Table 7.19), the exceedance frequency for a 10-inch break is 6.6E-07 yr^{-1} as shown below:

$$y = 10^{\log\left(1.6\cdot 10^{-6}\right)+\left(\frac{\log\left(2.0\cdot 10^{-7}\right)-\log\left(1.6\cdot 10^{-6}\right)}{14-7}\right)(10-7)} = 6.6\cdot 10^{-7}$$

The NUREG-1829 LOCA frequencies represent generic, or average, estimates for the commercial fleet and are not meant to represent a specific site or design (Reference 6). The experts developed these generic estimates using representative assumptions about important variables such as material conditions, plant geometry, degradation mechanisms, loading, and maintenance practices. The experts also assumed normal plant operational cycles and loading histories (e.g., pressure, thermal, residual). Finally, the experts assumed that plant construction and operation comply with all applicable codes and standards required by regulation and technical specifications.

WCGS is similar to other Westinghouse 4-Loop plants, and the generic LOCA frequencies in NUREG-1829 are applicable. Note that the NUREG-1829 frequencies were used for medium and large LOCAs in the WCGS internal events PRA model.

Overview of Risk-Informed Approach

NUREG-1829 contains “25-year” or “current-day” LOCA frequencies and “40-year” or “end-of-plant-license” LOCA frequencies (Reference 6). As discussed in the safety evaluation (SE) for the STP risk-informed GSI-191 submittal, the NRC staff considers it to be acceptable to use the 25-year LOCA frequencies for plants that have been operating between 25 and 40 years (Reference 20 p. 30, and Reference 29). Because WCGS was commissioned in 1985, it is less than 40 years old and the 25-year LOCA frequencies are applicable.

As described in an evaluation performed for the STP pilot project (Reference 17), the LOCA frequency GM aggregation provides a better representation of the center of the group’s opinion (made up of nine experts) compared to the arithmetic mean (AM) aggregation. Therefore, it is appropriate to use the GM LOCA frequency (Reference 6, Table 7.19) for the GSI-191 risk quantification. However, as described in draft RG 1.229 (Reference 3), the LOCA frequency is a significant source of uncertainty and the AM LOCA frequency (Reference 6, Table 7.13) should also be considered.

As described in draft RG 1.229 (Reference 3), semi-log interpolation (i.e., linear interpolation between break sizes and logarithmic interpolation between frequencies) is acceptable.

Non-pipe LOCAs were not explicitly evaluated. It was reasonably assumed that breaks at non-piping components (including nozzles, component bodies, pressurizer heater sleeves, manways, control rod drive mechanism (CRDM) penetrations, safety relief valves, RCP seals, the reactor vessel, the pressurizer vessel, the SG vessels, welded caps on retired lines, and other components) would be bounded by already-analyzed breaks at pipe weld locations. With the exception of non-pipe components that are located in the reactor cavity, these non-pipe components are located at or near pipe welds. For example, there are many weld locations in lines around the pressurizer vessel including the surge line, spray lines, and the safety and relief valve lines that could be used to estimate debris generated from non-pipe components in that area of containment. In addition, there are many welds distributed along the cold legs, including those near the RCPs, that could be used to estimate debris generated from non-weld locations in those areas. The modeled welds that are located at the safe ends on the nozzles at the pressurizer vessel and the SG vessels are reasonably close to the associated nozzle welds and are close enough to the vessels to produce significant debris from the insulation around those vessels. Breaks at the reactor vessel nozzles do not need to be considered as discussed in Section 3.a.1 of Attachment VIII. CRDM housings extend above the reactor vessel top head insulation. Therefore, if a CRDM ejection were to occur, the fluid issuing from the housing would be expelled above the insulation without damaging it. If a pressure boundary failure were to occur at the instrumentation penetrations on the reactor vessel bottom head, the jet would be directed toward the bottom of the reactor cavity and would not generate significant debris quantities. A rupture of the reactor vessel itself is modeled in the PRA as an excessive LOCA that proceeds directly to core damage.

Note that it was determined that the breaks on piping past the first RCS isolation valve are not risk significant because there would have to be a coincidental failure of the valve along with the pipe break, which is a low probability event. Additionally, there are no localized problematic insulation types or any other factors that are unique to the isolable weld locations that would significantly increase the probability of debris-related failures.

2.3.2 Water Hammer Induced LOCAs

The approach used to demonstrate that the risk of water hammer is acceptably low is to verify that the potential for water hammer is not likely to cause a pipe rupture in the break locations that can produce and transport problematic quantities of debris.

The portions of the WCGS RCS that are subject to a LOCA are designed to the Class 1 requirements of Section III of the ASME Boiler and Pressure Vessel Code, which includes consideration of appropriate transients. The reactor coolant pressure boundary (RCPB) is designed to accommodate the system pressures and temperatures attained under the expected modes of plant operation, including anticipated transients, with stresses within applicable limits. Consideration was given to loadings under normal operating conditions and to abnormal loadings, such as pipe rupture and seismic loadings. Pressurizer piping is a primary area of consideration due to its function during RCS pressure transients. The pressurizer safety valve, including valve supports, is designed for loads due to water relief, including the passage of a water slug and the effects of water hammer.

Because the RCS is kept water-solid during operation, a water hammer event can only be introduced from one of the systems that interact with the primary loop piping. At WCGS, the only systems that flow into the primary loop piping are the SI system, the residual heat removal (RHR) system, and charging from the chemical and volume control system (CVCS).

The potential for gas accumulation in the ECCS, which includes the CVCS, RHR, and SI sub-systems, is addressed under WCNOC's response to GL 2008-01 (Reference 8). To address GL 2008-01, WCNOC performed a review of site documents, procedures, and equipment, and implemented modifications and document revisions as necessary. These changes included adding vent valves, revising procedures to include ultrasonic testing for gas voids, and creating/maintaining an active program to prevent, monitor, and trend gas voids in the ECCS and the CSS (Reference 27). WCNOC's documented resolution of GL 2008-01 was accepted by the NRC and deemed effective in precluding gas accumulation in the ECCS and CSS, and, therefore, preventing a water hammer event that could challenge system integrity or operation (Reference 28).

Lastly, a search of WCNOC's corrective action program data was performed, and no issues were found related to water hammer induced LOCAs. Based on the fact that the piping is designed to ASME III Class 1 standards, the implementation of an approved gas accumulation prevention/monitoring program, and the lack of historical data for water hammer induced LOCA events, the relevance of water hammer events in the context of GSI-191 is deemed insignificant. Therefore, LOCA frequencies are not impacted for WCGS due to water hammer considerations in these systems.

2.3.3 Secondary Side Breaks

Although some SSBIs (e.g., a large break in a main steam line) would be expected to initiate containment spray (CS), the ECCS would not be required for long term core cooling unless there are subsequent failures, such as a stuck open PORV or loss of auxiliary feedwater, that would require feed and bleed cooling.

A simplified, bounding evaluation was performed to assess the risk contribution from SSBIs. For this evaluation, it was assumed that all secondary side breaks that require ECCS recirculation (e.g., in a feed and bleed scenario) would fail due to the effects of debris. This is a conservative assumption because secondary side breaks would generate less debris than equivalent primary side breaks due to the lower pressure on the secondary side. Also, the flow rate through the strainer required for feed and bleed cooling is significantly lower than the ECCS flow rate for a large break LOCA.

Using the PRA model, the frequency of SSBIs that require ECCS recirculation (which is equivalent to the ΔCDF contribution) was calculated to be 6.5E-08 yr^{-1}. In addition, the ΔLERF contribution for SSBIs was calculated to be 1.1E-10 yr^{-1}.

2.3.4 Seismically Induced LOCAs

Seismically induced LOCA frequencies were determined for WCGS using two approaches:

1. Using representative LOCA fragility parameters presented in EPRI 3002000709 (Reference 19).
2. Using site-specific LOCA fragility parameters calculated per the guidelines provided in NUREG-1903 (Reference 21).

Note that the second approach is a confirmatory analysis, which serves to confirm the results of the first approach. For both approaches, the fragility functions were convolved with site-specific seismic hazard curves to determine the annual frequency exceedance of various sized LOCAs.

Representative LOCA fragility parameters are presented in EPRI 3002000709 (Reference 19) Table H-1. Median capacity, randomness variability, and uncertainty are given for small, medium, and large LOCAs and are used to determine the fragility function in the form of a lognormal distribution as discussed below.

Site-specific LOCA fragility parameters were determined by scaling the safe shutdown earthquake (SSE) peak ground acceleration (PGA), consistent with the methodology presented in Reference 21. Indirect LOCAs caused by support failure were further separated into small and large LOCAs, following the approach used in Reference 30. Since both STP and WCGS are Westinghouse 4-loop plants, the conditional probabilities for LOCAs due to support failures used for STP were considered applicable to WCGS.

Following are the general steps used to determine the annual frequency of exceedance for both approaches once the LOCA fragility parameters had been determined.

1. Define fragility curves: The lognormal distribution function for the fragility curves is defined in Reference 31. Equation 2-3 of Reference 31 shows the lognormal distribution equation used to plot the mean fragility curve.

2. Define seismic bins: A seismic hazard curve was plotted as spectral acceleration versus annual frequency of exceedance. The selected hazard curve was divided into several seismic bins based on different spectral acceleration levels. For each bin, the representative magnitude and seismic initiator frequency were determined. The representative magnitude is a GM between the upper bound and lower bound spectral accelerations of each bin. The seismic initiator frequency is defined as the difference between the lower bound and upper bound annual frequency of exceedance of each bin.

3. Determine convolved frequency for individual seismic bins: The convolved frequency for each seismic bin is estimated by the product of the mean fragility failure probability and the seismic initiator mean frequency.

4. Determine total LOCA frequency: The convolved frequencies for different seismic hazard bins were summed up to estimate the total LOCA frequency across the entire hazard curve.

As described in Section 2.3.1, small and medium LOCAs would not cause failures due to the effects of debris. The large break seismic LOCA frequency was determined to be 6.9E-07 yr^{-1} based on representative fragility parameters from EPRI 3002000709 (Reference 19) and 3.9E-07 yr^{-1} based on the site-specific fragility parameters and the guidance in NUREG-1903 (Reference 21).

2.4 Risk Attributable to Debris

Based on the qualitative screening and assessment of relevant hazards and initiating events described in Sections 2.1 and 2.3, the following events are included in the quantitative risk assessment:

- RCS pipe breaks resulting in small, medium, and large LOCAs (including breaks ranging from ½" partial breaks to double-ended guillotine breaks (DEGBs) on every Class 1 ISI weld within the first isolation valve); this includes seismically-induced LOCAs
- SSBIs that result in a consequential LOCA (e.g., due to failure to terminate SI, loss of auxiliary feedwater, or a stuck open PORV) that requires sump recirculation

The specific GSI-191 failure modes that were included in the risk model are discussed in Section 2.0.

2.4.1 Threshold Break Size

The threshold break approach conservatively identifies a threshold break size where breaks smaller than or equal to the threshold would not cause a failure (e.g., strainer blockage or core blockage) due to the effects of debris that would challenge the ability to provide long-term core cooling. With the threshold break methodology, two conservative assumptions are made:

- All breaks larger than the threshold break size will result in core damage. This is a conservative assumption because the quantity of debris generated is dependent on the location and orientation of the break, as well as the break size, and there are many instances where larger breaks generate less debris that would not exceed any of the GSI-191 acceptance criteria.
- The threshold break size determined for the most limiting equipment configuration is applicable to all equipment configurations. This is a conservative assumption because some configurations would be significantly less likely to fail (e.g., the most likely scenario with both trains running would be less likely to have a strainer head loss failure, because the debris would be distributed across both strainers).

The threshold break size was determined based on the evaluations for the GSI-191 acceptance criteria (see Attachment VIII). Table 2 shows the various acceptance criteria and the corresponding threshold break size. Both the strainer head loss analysis and the in-vessel core blockage calculation showed a threshold break size of 10 inches. Therefore, the overall threshold break size for WCGS is 10 inches.

Table 2: Acceptance Criteria and Corresponding Threshold Break Sizes

GSI-191 Acceptance Criteria	Threshold Break Size (in)
Upstream Blockage	N/A
Strainer Head Loss	10
Degasification and Flashing	N/A[1]
Vortexing	N/A
Ex-vessel Blockage and Wear	N/A
Core Blockage	10
Overall Threshold Break Size	**10**

Both the strainer and in-vessel threshold break sizes were calculated based on the most limiting equipment configurations. The bounding equipment configuration for strainer head loss is an assumed single train failure (i.e., all pumps running on only one train), because this maximizes both the flow rate and debris accumulation on the active strainer.

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[1] The degasification and flashing analysis showed that no gas void failures occur at the maximum tested head loss. The head loss testing did not test the maximum debris quantities for all postulated breaks, but because no degasification or flashing failures occurred, the head loss acceptance criteria are more limiting than the degasification and flashing acceptance criteria.

The bounding equipment configuration for in-vessel effects is an assumed failure of both CS pumps (i.e., both trains of ECCS pumps running without either train of CS), because this maximizes strainer area for penetration and minimizes the core bypass flow. The most likely scenario in the event of a large LOCA is that there would be no random equipment failures, and both trains of ECCS and CS would be operating. For this scenario, it is expected that the smallest break that fails would be larger than the 10-inch threshold break size. If all pumps are running, the debris would be spread across both strainers (making it much less likely that either strainer would fail). Also, a large fraction of the debris that penetrates the two strainers would bypass the core with the flow through the two CS pumps (making it less likely for sufficient debris accumulation to result in core blockage). Therefore, the assumption that the threshold break size applies to all equipment configurations provides a significant level of conservatism in the GSI-191 risk quantification.

Note that upstream effects and ex-vessel downstream effects have been evaluated and shown not to cause failures for the bounding break scenarios. Also, degasification, flashing, and vortexing were evaluated and shown not to cause failures at the threshold break size. Therefore, the threshold break size is based on the most limiting failures identified in the strainer head loss analysis and in-vessel core blockage calculation.

2.4.2 Risk Contribution from Relevant Hazards

The risk attributable to debris was quantified in terms of ΔCDF and ΔLERF, where the change in risk is based on a comparison to a hypothetical plant condition without any debris.

The risk evaluation relies on many engineering calculations and tests that have been developed and conducted for WCGS over the last several years to address GSI-191 and GL 2004-02. These calculations and tests are described in detail in Attachment VIII.

As described in Section 2.3.1, the exceedance frequency for breaks larger than 10 inches is 6.6E-07 yr^{-1}. Since all breaks larger than 10 inches are assumed to fail due to the effects of debris, this frequency is equivalent to the ΔCDF contribution from large LOCAs. The ΔLERF contribution was calculated based on the conditional large early release probability (CLERP) for a large LOCA that results in core damage. The relevant LERF sequences for a large LOCA inside containment include failure of containment isolation and failure of containment due to overpressure caused by a hydrogen burn. The CLERP value was calculated to be 2.83E-05, which gives a ΔLERF contribution of 1.9E-11 yr^{-1} for large LOCAs.

As described in Section 2.3.3, the bounding ΔCDF and ΔLERF contributions for secondary side breaks were determined to be 6.5E-08 yr^{-1} and 1.1E-10 yr^{-1}, respectively.

As described in Section 2.3.4, the frequency of a large seismically induced LOCA is 6.9E-07 yr^{-1}. Since all large LOCAs were conservatively assumed to result in core damage due to the effects of debris, this is equivalent to the ΔCDF contribution from seismically induced LOCAs. Using the large LOCA CLERP described above, this gives a ΔLERF contribution of 2.0E-11 yr^{-1}.

All other hazards were determined to have no effect on the GSI-191 risk quantification.

As described in Section 2.2, the base case CDF and LERF for WCGS are 5.69E-04 yr^{-1} and 1.34E-05 yr^{-1}, respectively. A summary of the ΔCDF and ΔLERF associated with the effects of debris are shown in Table 3.

Table 3: GSI-191 Risk Quantification Results

Hazard	ΔCDF (yr^{-1})	ΔLERF (yr^{-1})
Piping and Non-Piping LOCAs	6.6E-07	1.9E-11
Water Hammer Induced LOCAs	0.0	0.0
Secondary Side Breaks	6.5E-08	1.1E-10
Fire Induced LOCAs	0.0	0.0
Seismically Induced LOCAs	6.9E-07	2.0E-11
Other External Hazards	0.0	0.0

Note that the ΔCDF and ΔLERF values from various hazards are not added together because bounding methods were used to calculate the values. Figure 2 and Figure 3 show the RG 1.174 risk guidelines (Reference 4). Based on these guidelines, the risk associated with the effects of debris at WCGS is very small (Region III) for each of the evaluated hazards.

2.5 Technical Adequacy of WCGS PRA Results

This section provides justification for the technical adequacy of the WCGS internal events PRA model for use in the GSI-191 risk assessment. Note that the internal flood, internal fire, and high winds PRA models are only used to determine the total baseline CDF and LERF values and no specific GSI-191 calculations were performed using these models. Therefore, the justification of technical adequacy described in this section is limited to the internal events PRA model.

RG 1.200 (Reference 5, Section 4.2) requires the following information to demonstrate the technical adequacy of the PRA when used in a risk-informed application:

1. How the PRA model represents the as-built, as-operated plant.
2. Identification of permanent plant changes (such as design or operational practices) that have an impact on the PRA but have not been incorporated into the PRA.
3. Documentation that the parts of the PRA used in the application are performed consistently with the PRA standard as endorsed by RG 1.200 (Reference 5).
4. A summary of the risk assessment methodology used to assess the risk of the application, including how the base PRA model was modified to appropriately model the risk impact of the application and results.

5. Identification of the key assumptions and approximations relevant to the results used in the decision-making process, as well as the peer reviewers' assessment of those assumptions.
6. A discussion of the resolution of the peer review findings and observations that are applicable to the parts of the PRA required for the application.
7. Identification of the parts of the PRA used in the application that were assessed to have capability categories lower than required for the application (Reference 7, Section 1-3).

This section provides the information to address these items.

2.5.1 PRA Model of As-Built, As-Operated Plant

The WCGS internal events PRA model was developed based on the as-built, as-operated configuration as of March 31, 2018 and plant specific data as of December 31, 2018. The WCGS PRA uses the CAFTA software with fault tree linking methodology to perform model quantification. The WCGS PRA models are controlled in accordance with guidance documents for PRA generation, maintenance, and updates.

The internal events PRA model was developed and is maintained in accordance with the ASME/ANS PRA standard (Reference 7) and RG 1.200 requirements (Reference 5). Compliance with the standard includes mandatory PRA model updates, reliability data updates, and peer reviews in order to ensure the model reflects the as-built, as-operated plant.

2.5.2 Plant Changes Not Incorporated in PRA

The WCGS PRA model maintenance and update process is governed by a WCGS desktop guidance document. This guidance document ensures that all plant changes are systematically reviewed to identify PRA impact, determine significance, and schedule timely implementation. The guidance document also ensures that model updates occur on a periodic basis to ensure that the PRA model continually represents the as-built, as operated plant. The WCGS PRA has recently undergone a model update and peer review and there are currently no unaddressed plant changes that are significant to the model results.

2.5.3 Parts of the WCGS PRA Used for GSI-191 Risk Assessment

The internal events PRA model was the primary model used in the GSI-191 risk assessment. The internal events model was not directly used to calculate ΔCDF associated with the effects of debris following a LOCA. However, the model was used to calculate ΔLERF using the CLERP (conditional early release probability) and the bounding ΔCDF and ΔLERF associated with the effects of debris following a secondary side break inside containment. In addition, the internal events, internal flooding, internal fire, and high winds PRA models were all used to determine the baseline CDF and LERF values.

2.5.4 Summary of the Risk Assessment Methodology

The WCGS GSI-191 risk assessment methodology involves using the total CDF and LERF from all modeled internal and external hazards, calculating ΔCDF based on LOCA frequency, calculating ΔLERF using the CLERP and ΔCDF results, and performing a bounding analysis of SSBIs to determine the risk impact of strainer clogging. No PRA model modifications were required for this evaluation. The risk assessment methodology is described in more detail in Section 2.4.

2.5.5 Key Assumptions and Sources of Uncertainty in the PRA

The uncertainty associated with PRA modeling is addressed by making assumptions. NUREG-1855 (Reference 9) defines assumptions as follows:

- An assumption is a decision or judgment that is made in the development of the PRA model. An assumption is either related to a source of model uncertainty or related to scope or level of detail.
- An assumption related to a model uncertainty is made with the knowledge that a different reasonable alternative assumption exists. A reasonable alternative assumption is one that has broad acceptance in the technical community and for which the technical basis for consideration is at least as sound as that of the assumption being made.
- By contrast, an assumption that is related to scope or level of detail is one that is made for modeling convenience. Such assumptions result in defining the boundary conditions for the PRA model.
- An assumption is labeled key when it may influence (i.e., have the potential to change) the decision being made. Therefore, a key assumption is identified in the context of an application.

The PRA assumptions and modeling uncertainties that could potentially impact the risk-informed GSI-191 application were identified based on a review of the relevant PRA notebooks.

Overview of Risk-Informed Approach

Table 4: Evaluation of Relevant Assumptions in WCGS Internal Events PRA Model

Relevant Assumption	Impact on GSI-191	Key Assumption?
Hot leg recirculation is not modeled as a requirement for any initiating event. WCAP-15750 (Reference 26) indicates that switchover to hot leg recirculation would not be required for any initiator except at the upper range of the large LOCA. This is a realistic treatment, although slightly non-conservative for large LOCA initiators.	Switchover to hot leg recirculation is explicitly evaluated in the risk-informed GSI-191 evaluation. This assumption could have a conservative effect on the delta risk associated with debris related failures if the risk was quantified using the PRA model. However, because the risk is quantified outside the PRA model (i.e., using the threshold break approach), this assumption does not impact the risk-informed GSI-191 application.	No
It is assumed that the low-pressure system flow path to RCS Loop 2 and 3 hot legs is not a flow diversion path during the injection or cold leg recirculation modes since the flow would still be into the RCS. This path is isolated by normally-closed motor operated valve (MOV) EJHV8840, which has power locked out by an isolation switch in the control room. This is a realistic treatment of the failure impact.	Debris blockage is less likely to occur during hot leg recirculation since the coolant enters at the top of the core and boiling or convective forces would disrupt a debris bed that may form at this location. Therefore, flow diversion to the hot legs during injection or cold leg recirculation would not have a detrimental effect on in-vessel debris effects or a significant impact on the risk-informed GSI-191 application.	No
No credit is taken for low-pressure injection and recirculation flow, or accumulator flow, to the faulted loop for a large or medium break LOCA due to assumed flow diversion. No credit is taken for high-pressure injection or recirculation flow to the faulted loop for a medium break LOCA. This is a slightly conservative treatment.	Depending on the orientation of the break on the pipe (e.g., a partial break at the top of the pipe), it is possible that the faulted loop could still supply low-pressure injection and recirculation flow to the core. However, this is a reasonable assumption for larger break sizes, which are more likely to be affected by debris (up to and including DEGBs).	No

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Relevant Assumption	Impact on GSI-191	Key Assumption?
Containment sumps may experience debris clogging during LOCA scenarios. WCAP-16882 (Reference 25) identifies industry guidance for the probability of clogging for different sizes and types of LOCAs. This guidance is implemented in the logic at gate GLPR-SUMP-FP. This is a realistic treatment of the failure probability.	The risk-informed GSI-191 evaluation provides a more detailed assessment of strainer failures due to the effects of debris, so the WCAP-16882 (Reference 25) failure probabilities are not used for this application. These values were set to 0.0 or 1.0 to quantify the risk contribution from secondary side breaks.	No
Plugging of the CS nozzles is assumed to not occur. The nozzles are not subject to plugging due to their design, and the design and location of the containment sump per the Updated Safety Analysis Report (USAR), pages 6.2-45, 6.2-52, 6.2-53 and Table 6.2.2-1. This is a realistic treatment of the plant design.	This assumption is consistent with the conclusion of the GSI-191 ex-vessel downstream effects evaluation (see Section 3.m of Attachment VIII).	No

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Relevant Assumption	Impact on GSI-191	Key Assumption?
A steam line break (SLB) initiating event with failure to isolate more than one SG is assumed to proceed directly to core damage. The plant design basis assumes only one SG is depressurized, and no analysis exists to identify the impact of two or more SGs being depressurized on 1) RCS temperatures and vessel structural integrity, 2) reactor subcriticality, 3) containment structural integrity. This is a conservative assumption and a source of model uncertainty.	This assumption could have a non-conservative effect on the GSI-191 risk assessment for secondary side breaks because the delta risk from strainer failures could potentially be higher if these scenarios weren't assumed to proceed directly to core damage. However, a bounding approach was used to assess SSBIs by conservatively assuming that the strainers would fail for all SSBI scenarios where they are required. Realistically, strainer clogging is less likely to occur for SSBIs than for LOCAs based on the smaller ZOIs for SSBIs and significantly lower ECCS flow rate required to mitigate an SSBI consequential LOCA. The conservatism associated with the bounding approach for SSBIs outweighs the potential non-conservatism of this assumption and implementing a more realistic analysis would not change the overall conclusion that the effects of debris on secondary side breaks is very small.	No

2.5.6 Assessment of PRA Model Technical Adequacy

An independent peer review of the WCGS internal events PRA model, data, and documentation was performed by Enercon Services, Inc. the week of June 17-21, 2019. The PRA model was reviewed against the Capability Category (CC) II requirements from the ANS/ASME PRA standard (Reference 7), including clarifications imposed by RG 1.200 (Reference 5). The review was conducted by six industry experts who are experienced in performing PRAs. There were a total of 34 finding level facts and observations (F&Os) identified.

Following the peer review, WCNOC incorporated changes in the model and an F&O closure review was performed by the Pressurized Water Reactor Owner's Group (PWROG) between November 2019 and March 2020, including an onsite meeting at WCGS on December 10 and 11, 2019. The F&O closure review was conducted by three industry experts from Westinghouse and NextEra who are experienced in performing PRAs. Out of the 34 F&Os identified, 31 were closed and 3 remain open.

During the F&O closure review, two unique F&Os were judged to be closed with a PRA upgrade, which required a focused scope peer review. Therefore, two supporting requirements (SRs) in Part 2 and one SR in Part 1 of the PRA standard (Reference 7) were re-peer reviewed. Following this focused scope peer review, the SRs were judged to be met at CC II or higher, but a new F&O (AS-B3-01) was assigned and remains open.

Table 5 summarizes the open F&Os and the resolutions with respect to use of the PRA model in the GSI-191 risk assessment.

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Table 5: Resolution of WCGS Internal Events PRA Peer Review Findings

F&O Number	Review Element	CC	F&O Description	F&O Impact on GSI-191 Evaluation
3-8	QU-F4	Not Met	**Original Peer Review:** Identify plant specific sources of uncertainty. This identification can be documented in a manner similar to the tables that characterize the generic sources of model uncertainty and related assumptions. **Utility Resolution:** All documents have been reviewed and updated to ensure each assumption and source of uncertainty is properly identified and characterized (conservative, non-conservative, realistic). **F&O Closure Review:** Individual assumptions in the various notebooks have been reviewed with an initial characterization of the associated uncertainties. There is still a gap in the assessment of the individual assumption and the final uncertainty assessment in the quantification notebook. The section discussing sensitivities (which is where any quantitative assessment on the impact of uncertainties associated with assumption should be discussed) does not allow a clear tracking on which assumption in the rest of the documentation is being addressed. Especially for assumptions that are marked as “non conservative” a closure statement	Assumptions and sources of uncertainty in the internal events PRA model that are relevant to the risk-informed GSI-191 application were identified and specifically dispositioned with respect to this application (see Section 2.5.5). Additional sources of uncertainty in the risk-informed GSI-191 evaluation (i.e., sources of uncertainty outside the PRA model) were assessed as described in Section 2.6.

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F&O Number	Review Element	CC	F&O Description	F&O Impact on GSI-191 Evaluation
			needs to be added somewhere on the importance for the results. It is also recommended that an assessment on some of the most conservative assumptions is made, to ensure that risk insights are not masked.	
4-10	LE-C13	CC I/II/III Met	**Original Peer Review:** The approach to scrubbing of SGTR releases is consistent with the CC-II requirements and, therefore, allows the SR to be considered MET at CC-II. However, the current SGTR documentation does not provide sufficient technical basis to justify the credit taken. Additionally, the simplified approach for ISLOCA releases does not discuss any consideration of potential scrubbing credit. **This F&O was not included in the F&O closure review scope.**	SGTR and ISLOCA events were screened out of the risk-informed GSI-191 evaluation as discussed in Section 2.1.
6-8	SY-C2	CC I/II/III Met	**Original Peer Review:** The notebook states that walkdowns and interviews were performed but not documented. Without the documentation there is no evidence that these tasks were performed and that the walkdown was included the present as built plant. **Utility Resolution:** A new set of System Engineer Interviews have been completed and walkdowns were completed during outage and documented through the system	There is no technical issue associated with this F&O and the lack of documentation for the interview of the Electrical Systems engineer is not significant for the risk-informed GSI-191 application.

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F&O Number	Review Element	CC	F&O Description	F&O Impact on GSI-191 Evaluation
			health reporting process as described by the System Engineer Program, AP 23-006. **F&O Closure Review:** The majority of the interviews with System Engineers were performed and documented in the appropriate system notebooks. The results of the interviews and insights are appropriately captured in both the interview notes as well as the text. The task remains incomplete as one interview is still needed to the Electrical Systems engineer (PSA-05-0011).	
AS-B3-01	AS-B3	CC I/II/III Met	**Focused Scope Peer Review:** Feed and Bleed scenarios involving open PORVs did not consider the potential for sump strainer blockage. The review identified no model logic or a documented basis that would address open PORV transients including considerations of the complications associated with containment sump blockage with the actuation of CS.	The PRA model includes high pressure recirculation and containment sump blockage for feed and bleed scenarios associated with main steam and feedwater line breaks. This is important since these breaks could generate significant quantities of debris. Other feed and bleed scenarios associated with a stuck open PORV would not generate significant quantities of debris and were screened out of the risk-informed GSI-191 evaluation as discussed in Section 2.1.

2.5.7 Capability Categories for Parts of the PRA

The PRA standard has 325 individual SRs; 13 of these SRs were determined to be not applicable to the peer review. Based on the F&O closure review, only one SR is not met. Therefore, over 99% of the SRs satisfy CC II requirements. The following SR was "not met":

> QU - The quantification did not include characterization of the sources of uncertainty (QU-F4).

The number of SRs in each Capability Category are presented in Table 6.

Table 6: Summary of WCGS Internal Events Capability Categories

Capability Category Met	Number of SRs	% of Total SRs
SR Not Met	1	0.3%
CC-I	0	0.0%
CC-I/II	18	5.8%
CC-II	43	13.8%
CC-II/III	27	8.7%
Met All (I/II/III)	223	71.5%
Total	**312**	**100%**

Because approximately 99% of the SRs in the internal events PRA model satisfy Capability Category II requirements and non-conforming aspects of the model were addressed with respect to the GSI-191 risk assessment (see Section 2.5.6), the WCGS internal events PRA meets the requirements of RG 1.200 and is adequate for this risk-informed application.

2.6 Uncertainty Quantification

Uncertainty quantification is a key requirement in RG 1.174 for a risk-informed evaluation (Reference 4). As defined in RG 1.174 and explained in more detail in NUREG-1855 (Reference 9) and two corresponding EPRI reports (References 10 and 11), there are three types of uncertainty that should be addressed:

1. Parametric uncertainty
2. Model uncertainty
3. Completeness uncertainty

Parametric uncertainty is the uncertainty in the value for a specific parameter. Conservative or bounding values have been used for most parameters in the WCGS GSI-191 evaluation, and no uncertainty quantification is required for these parameters. Due to the wide range of plant-specific post-LOCA conditions related to GSI-191 phenomena, understanding which parameters are uncertain along with the level of parametric uncertainty is important for understanding the overall uncertainty in the GSI-191 risk quantification.

Model uncertainty refers to the potential variability in an analytical model when there is no consensus approach. A consensus approach is a model that has been widely adopted or accepted by the NRC for the application for which it is being used (Reference 9). For example, the use of a spherical ZOI to model the debris quantity generated by a HELB is a consensus model that has been widely adopted and accepted by the NRC (References 12 and 13). In general, WCGS is using standard models that have been widely accepted for deterministic evaluations (e.g., accepted insulation and qualified coatings ZOI sizes and prototypical strainer module testing for head loss and penetration). By using these consensus approaches, the effort to address model uncertainty is minimized.

Completeness uncertainty refers to 1) the uncertainty associated with scenarios or phenomena that are excluded from the risk evaluation, and 2) the uncertainty associated with unknown phenomena. Although it may not be practical to quantify the uncertainty associated with factors that are not explicitly evaluated, their potential impact can be qualitatively assessed. Uncertainties associated with unknown phenomena, on the other hand, cannot be directly evaluated (either quantitatively or qualitatively). Uncertainties associated with unknown phenomena are the reason that it is important to maintain defense-in-depth and safety margins (see Attachment IX).

2.6.1 Parametric Uncertainty

Parametric uncertainty is the uncertainty in the value for a specific parameter. Determining whether to use a bounding or realistic value for each input parameter (for the purpose of calculating the mean ΔCDF) is a plant-specific process. This process involves a consideration of the level of conservatism that can be tolerated, the confidence in the test or analysis used to determine the value, how the overall analysis will affect the plant design and licensing basis, and other factors. In general, WCGS has used a set of inputs that are mostly bounding.

If a bounding value is selected for an input parameter, it is essentially equivalent to a consensus model where the uncertainty does not need to be quantified. This is consistent with the guidance in Section C.4 of Draft RG 1.229 (Reference 3). The term “consensus input” is used to refer to conservative or bounding inputs that are consistent with general industry guidance, which has been accepted by the NRC for design basis GSI-191 evaluations.

Depending on the models that are used, the worst-case direction for some input parameters may not be intuitively obvious. For example, a minimum water temperature could be worse with respect to strainer head loss, but a maximum temperature could be worse with respect to degasification. Similarly, a minimum pool volume could be worse with respect to NPSH margin, but either a minimum or maximum pool volume could be worse with respect to the quantity of chemical precipitates predicted using the WCAP-16530 methodology (Reference 18).

Table 7 shows a summary of important input parameters for a GSI-191 evaluation with an indication of which direction is more limiting in terms of strainer failures and core failures for WCGS. This table illustrates the logic for determining the worst-case conditions. Note that the worst case (or best case) set of input parameters is highly dependent on the WCGS configuration as well as the specific GSI-191 models that were implemented for WCGS.

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Table 7: Bounding (Worst Case) Direction for Important Input Parameters

Parameter	Bounding Direction for Strainer Failures	Bounding Direction for Core Failures	Comments
Fiber Insulation Debris Quantity	Maximum	Maximum	The maximum quantity of fiberglass insulation debris that would be generated for a given break was determined in the debris generation calculation and subsequently used in the strainer head loss analysis and in-vessel effects calculation. Therefore, no uncertainty quantification is required for this parameter.
Qualified Coatings Debris Quantity	Maximum	N/A	The maximum quantity of qualified coatings debris that would be generated for a given break was determined in the debris generation calculation and subsequently used in the strainer head loss analysis. The core acceptance criteria are only a function of the fiber debris quantity (Reference 24). Therefore, no uncertainty quantification is required for this parameter.
Unqualified Coatings Debris Quantity	Maximum	N/A	The maximum quantity of unqualified coatings debris that would be generated following a LOCA in containment was determined in the debris generation calculation and subsequently used in the strainer head loss analysis. The core acceptance criteria are only a function of the fiber debris quantity (Reference 24). Therefore, no uncertainty quantification is required for this parameter.
Latent Debris Quantity	Maximum	Maximum	The maximum quantity of latent fiber and particulate debris in containment was determined in the debris generation calculation based on latent debris surveys and subsequently used in the strainer head loss analysis and in-vessel effects calculation. Therefore, no uncertainty quantification is required for this parameter.
Miscellaneous Debris Quantity	Maximum	Minimum	The maximum quantity of miscellaneous debris in containment was determined in the debris generation calculation based on containment walkdowns and subsequently used in the strainer head loss analysis. Miscellaneous debris blocks strainer area, which could be beneficial for reducing penetration. However, the WCGS penetration correlation uses the full strainer area without adjusting for area reductions due to miscellaneous debris. Therefore, no uncertainty quantification is required for this parameter.
Debris Transport Fractions	Maximum	Maximum	The maximum debris transport fractions for each type and size of debris were determined in the debris transport calculation and subsequently used in the strainer head loss analysis and in-vessel effects calculation. Therefore, no uncertainty quantification is required for this parameter.

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Parameter	Bounding Direction for Strainer Failures	Bounding Direction for Core Failures	Comments
Pool Volume/Level	Minimum or Maximum	Minimum or Maximum	The pool volume/level affects many different aspects of the GSI-191 evaluation including NPSH margin, degasification, flashing, vortexing, debris transport, chemical precipitation, and in-vessel effects. The NPSH margin was conservatively calculated based on the minimum large break water level. The degasification and flashing calculation conservatively used a minimum water level. Recirculation pool debris transport was conservatively calculated based on the minimum water level. Chemical precipitation was calculated using both maximum and minimum pool volumes to conservatively determine the quantity and precipitation timing. In-vessel effects were evaluated using both minimum and maximum pool volumes to determine the bounding conditions. Therefore, no uncertainty quantification is required for this parameter.
Containment Pressure	Minimum	N/A	The containment pressure affects degasification, flashing, and NPSH margin. The degasification and flashing calculation conservatively used a minimum amount of accident pressure. The NPSH margin was conservatively calculated without taking credit for accident pressure. Therefore, no uncertainty quantification is required for this parameter.
Pool Temperature	Minimum or Maximum	Minimum or Maximum	The pool temperature affects NPSH margin, degasification, flashing, chemical release, chemical solubility, and head loss. The NPSH margin was conservatively calculated at 212 °F without any credit for accident pressure. Degasification and flashing were conservatively calculated at a range of temperatures to determine the maximum void fraction. A maximum temperature profile was used to conservatively maximize aluminum release and the maximum precipitation temperature. A minimum temperature was used to show that precipitation would not occur prior to 24 hours. Strainer head loss was conservatively calculated for a range of pool temperatures to determine the bounding conditions. Therefore, no uncertainty quantification is required for this parameter.
ECCS Flow Rate	Maximum	Minimum or Maximum	The ECCS flow rate affects head loss, NPSH margin, and in-vessel effects. The maximum ECCS flow rate was conservatively used to calculate strainer head loss and NPSH margin. Both minimum and maximum ECCS flow rates were evaluated in the in-vessel effects calculation to determine the bounding case. Therefore, no uncertainty quantification is required for this parameter.
CS Flow Rate (assuming sprays initiate)	Maximum	Minimum	The CS flow rate affects head loss, washdown transport, and in-vessel effects. The maximum CS flow rate was conservatively used to calculate strainer head loss and NPSH margin. The minimum CS flow rate was conservatively used to calculate fiber debris accumulation in the reactor core. Therefore, no uncertainty quantification is required for this parameter.

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Parameter	Bounding Direction for Strainer Failures	Bounding Direction for Core Failures	Comments
ECCS Switchover Time	N/A	Minimum	The time when the ECCS pumps are switched from refueling water storage tank (RWST) injection to sump recirculation affects in-vessel effects. The fiber debris accumulation in the reactor core was calculated based on the minimum ECCS switchover time. Therefore, no uncertainty quantification is required for this parameter.
CS Switchover Time	N/A	Maximum	The time when the CS pumps are switched from RWST injection to sump recirculation also affects in-vessel effects. The fiber debris accumulation in the reactor core was calculated based on the maximum CS switchover time. Therefore, no uncertainty quantification is required for this parameter.
Hot Leg Switchover Time	N/A	Maximum	The time when the ECCS pumps are switched from cold leg recirculation to hot leg recirculation also affects in-vessel effects. The fiber debris accumulation in the reactor core was calculated based on the maximum switchover time. Therefore, no uncertainty quantification is required for this parameter.
Secure CS Time	Maximum	Minimum	The containment sprays are procedurally secured when the containment pressure is below 3 psig and at least 10 hours of spray operation have elapsed. The containment pressure is expected to be less than 3 psig in approximately 2 hours, so it is likely that both containment spray pumps would be secured right after the 10-hour point. The strainer head loss analysis reasonably assumed that the containment sprays would be secured prior to 24 hours to reduce the strainer flow rate before the onset of chemical effects. The bounding case for fiber debris accumulation in the reactor core is based on the equipment configuration where both CS pumps are conservatively assumed to fail at the start of CS recirculation. Therefore, no uncertainty quantification is required for this parameter.
Thermal Power	N/A	Maximum	The maximum thermal power was used to evaluate in-vessel effects. Therefore, no uncertainty quantification is required for this parameter.
pH	Minimum or Maximum	Minimum or Maximum	The pH affects chemical release and chemical solubility. A maximum pH was conservatively used for chemical release and a minimum pH was conservatively used for solubility in the chemical product generation calculation. This maximizes the precipitate quantity and minimizes the time when precipitation occurs. Therefore, no uncertainty quantification is required for this parameter.
Head Loss	Maximum	N/A	The strainer head loss was determined based on a conservative test protocol and conservative input parameters to identify the maximum head loss for the tested debris quantity. Therefore, no uncertainty quantification is required for this parameter.

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Parameter	Bounding Direction for Strainer Failures	Bounding Direction for Core Failures	Comments
Structural Margin	Minimum	N/A	The strainer structural limit was conservatively determined for both hot and cold temperature conditions and analyzed accordingly. Therefore, no uncertainty quantification is required for this parameter.
NPSH Margin	Minimum	N/A	The minimum NPSH margin was conservatively calculated and used as an acceptance criterion in the strainer head loss analysis. Therefore, no uncertainty quantification is required for this parameter.
Void Fraction Limit at Pump Suction	Minimum	N/A	Based on the guidance in NEI 09-10 (Reference 22), a 2% limit was conservatively used for the degasification and flashing calculation. Therefore, no uncertainty quantification is required for this parameter.
Fiber Penetration	Minimum	Maximum	Fiber debris penetration reduces the quantity on the strainer, which is beneficial for strainer head loss. However, no fiber penetration was credited in the head loss analysis. A conservative test protocol was used to determine maximum fiber penetration for the in-vessel effects calculation. Therefore, no uncertainty quantification is required for this parameter.
Particulate Penetration	Minimum	N/A	Particulate debris penetration reduces the quantity on the strainer, which is beneficial for strainer head loss. However, no particulate penetration was credited in the head loss analysis. The core acceptance criteria are only a function of the fiber debris quantity (Reference 24). Therefore, no uncertainty quantification is required for this parameter.
Core Fiber Limit	N/A	Minimum	Based on the guidance in WCAP-17788 (Reference 24), conservative core fiber limits were used for the in-vessel effects calculation. Therefore, no uncertainty quantification is required for this parameter.

As shown in Table 7, all of the parameters used to analyze strainer failures and core failures can be considered consensus inputs based on the level of conservatism and consideration of competing effects. Therefore, the only parameter that requires uncertainty quantification is the LOCA frequency, which was based on the mean value. To quantify this uncertainty, the 5th and 95th LOCA frequency percentiles (geometric aggregation, 25-year estimate) were calculated for the threshold break size by interpolating the values in NUREG-1829 (Reference 6) Table 7.19. Given a threshold break size of 10 inches (see Section 2.4.1), the 5th percentile LOCA frequency is 3.1E-09 yr^{-1} and the 95th percentile LOCA frequency is 2.2E-06 yr^{-1}. The exceedance frequency for the 10-inch threshold break size is equivalent to ΔCDF. Using a CLERP of 2.83E-05 (see Section 2.4.2), ΔLERF was calculated as shown in Table 8.

Table 8 - Results of Parametric Uncertainty Quantification

Base Case Input	Sensitivity Case Input	ΔCDF (yr^{-1})	ΔLERF (yr^{-1})
25-Year GM LOCA Frequency	25-Year Geometric 5th Percentile	3.1E-09	8.8E-14
	25-Year Geometric 95th Percentile	2.2E-06	6.2E-11

2.6.2 Model Uncertainty

To meet the guidance in NUREG-1855 (Reference 9), model uncertainty must be addressed for any models or approaches for which no consensus exists. WCGS is using standard models that have been widely accepted for deterministic evaluations. However, the following models used for the WCGS risk quantification are included in the model uncertainty evaluation:

- Continuum break model
- Geometric aggregation of LOCA frequencies
- Seismic LOCA frequencies

To address the uncertainty in these models, alternative models were evaluated.

<u>Continuum Break Model</u>

The threshold break size was determined using the continuum break model, which is based on the implicit assumption that breaks of any size up to a DEGB can occur at any weld location. Therefore, the debris generation calculation included a range of break sizes from ½-inch partial breaks up to and including DEGBs at each weld. The threshold break size was defined as the largest break size where none of the breaks of that size or smaller exceed any of the strainer or core acceptance criteria. This was determined to be 10-inches based on the fact that some 12-inch breaks (the next larger partial break size analyzed) generated enough debris to cause strainer and/or core failures.

The continuum break model was included in the uncertainty quantification due in part to a statement in NUREG-1829 that a break of a given size is more likely to result from a complete rupture of a small pipe than a partial rupture of a large pipe (Reference 6). A reasonable alternative model that can be used to quantify the uncertainty is to assume that only DEGBs can occur (i.e., no partial breaks), and then use the smallest DEGB that fails as the threshold break size. Table 9 provides a summary of the DEGB sizes (along with the corresponding nominal pipe sizes) at WCGS.

Table 9: DEGB Sizes (based on Pipe ID)

DEGB Size Based on Pipe ID (inches)	Nominal Pipe Size (inches)
1.338	1.5
1.689	2
2.125	2.5
2.626	3
3.438	4
5.189	6
8.750	10
10.5	12
11.188	14 (Schedule 160)
11.5	14 (Schedule 140)
27.5	27.5
29	29
31	31

The smallest DEGBs that failed were not identified in the strainer head loss analysis and in-vessel effects calculation. However, based on the fact that all breaks smaller than 10 inches passed the acceptance criteria, the smallest DEGB that fails would have to be larger than 10 inches. Therefore, the continuum break model is more conservative than the DEGB-only model and no further evaluation is necessary.

<u>Geometric Aggregation LOCA Frequencies</u>

The WCGS risk quantification was performed using the GM frequency from NUREG-1829 (Reference 6). However, NUREG-1829 also provides an arithmetic aggregation of the LOCA frequencies. To address the guidance in draft RG 1.229 Section C.2.b (Reference 3), the 25-year GM LOCA frequencies were replaced with the 25-year AM LOCA frequencies to calculate the effect on the ΔCDF. The 25-year AM LOCA frequencies are provided in NUREG-1829 (Reference 6) Table 7.13. Given a threshold break size of 10 inches (see Section 2.4.1), the interpolated AM LOCA frequency is 5.2E-06 yr^{-1}. The exceedance frequency for the threshold break size is equivalent to ΔCDF. Using a CLERP of 2.83E-05 (see Section 2.4.2), ΔLERF was calculated as shown in Table 10.

Table 10: Results of LOCA Frequency Model Uncertainty Quantification

Base Case Model	Alternate Model	ΔCDF (yr^{-1})	ΔLERF (yr^{-1})
GM LOCA Frequency	AM LOCA Frequency	5.2E-06	1.5E-10

Seismic LOCA Frequencies

As described in Section 2.3.4, the large break seismic LOCA frequency was calculated to be 6.9E-07 yr^{-1} based on representative fragility parameters from EPRI 3002000709 (Reference 19). However, because this approach for calculating seismic LOCA frequencies was not accepted for STP without the evaluation of an alternative model (Reference 20, pp. 26-27), the large break seismic LOCA frequency was also calculated using site-specific fragility parameters and the guidance in NUREG-1903 (Reference 21). The results of this calculation showed a seismic LOCA frequency of 3.9E-07 yr^{-1}. Because all seismically induced large breaks are assumed to fail due to the effects of debris, the ΔCDF contribution is equal to the seismically induced large break LOCA frequency. Using a CLERP of 2.83E-05 (see Section 2.4.2), the ΔLERF contribution was calculated as shown in Table 11.

Table 11: Results of Seismic LOCA Frequency Model Uncertainty Quantification

Base Case Model	Alternate Model	ΔCDF (yr^{-1})	ΔLERF (yr^{-1})
Representative Fragility Parameters from EPRI 3002000709 (Reference 19)	Site-Specific Fragility Parameters and the Guidance in NUREG-1903 (Reference 21)	3.9E-07	1.1E-11

2.6.3 Completeness Uncertainty

Completeness uncertainty was qualitatively determined to be low. As described below, the WCGS evaluation was rigorous and comprehensive, and the areas that were not explicitly evaluated have a low potential for any significant risk impact:

- The range of hazards, initiating events, and plant operating modes were considered as described in Section 2.1.
- LOCAs were directly evaluated in the risk quantification as described in Section 2.4.
    - The LOCA evaluation included pipe breaks on each ISI weld within the Class 1 pressure boundary inside the first isolation valve.
    - Break sizes ranging from ½-inch to a full DEGB were postulated on each weld.
    - Partial breaks (i.e., breaks smaller than a DEGB) were evaluated in 45-degree increment orientations around the pipe for each break size.
    - Debris quantities were calculated for breaks on ISI welds outside the first isolation valve, and there is no significant difference between the type and quantity of debris generated for these breaks compared to similar size breaks inside the first isolation valve. Even if these breaks result in any GSI-

191 failures, the risk contribution would be negligibly small due to the low likelihood of an isolation valve failing to close, spuriously opening, or developing a large leak. Based on the 2015 update to the NUREG/CR-6928 (Reference 16) component failure rates, the probability of a normally open air operated valve (AOV) or MOV failing to close is less than 4E-04, and the probability of a large leak or spurious operation of an isolation valve (AOV, MOV, or check valve) is on the order of 1E-07 or less (Reference 16, Component Reliability Data Sheets pp. 9, 28). Therefore, the conditional failure probabilities (CFPs) for breaks outside the first isolation valve would be orders of magnitude smaller than the CFPs for equivalent breaks inside the first isolation valve.

    - Non-pipe LOCAs are reasonably represented or bounded by adjacent pipe breaks as described in Section 2.3.1.
    - Both high and low likelihood equipment configurations were addressed with a bounding evaluation as described in Section 2.4.1.
- The risks associated with water hammer induced LOCAs, secondary side breaks, fire induced LOCAs, seismically induced LOCAs, and external hazards were evaluated and shown to be low as described in Sections 2.1 and 2.3.
- All known GSI-191 phenomena and debris failure mechanisms were evaluated in a bounding or reasonably conservative manner.

Although there is also some uncertainty associated with unknown phenomena, this uncertainty is judged to be small. The nuclear industry has been actively addressing GSI-191 concerns for PWRs for well over a decade. In addition, the boiling water reactor (BWR) strainer performance issue dates back to 1992, and unresolved safety issue (USI) A-43 dates back to 1979. Numerous tests have been performed by the U.S. NRC and industry, as well as regulators and utilities around the world over the last 40 years to resolve issues related to debris and strainer performance. This testing has investigated nearly every aspect of GSI-191 including insulation and coatings destruction from break jets; unqualified coatings failure; blowdown and washdown debris transport; containment pool settling, tumbling, and lift-over-curb debris transport; debris erosion; chemical release, solubility, and precipitation; strainer head loss, vortexing, and penetration; ex-vessel component wear; and in-vessel core blockage and boron precipitation. Based on the extensive research that has been performed, it is unlikely that there are unidentified phenomena that would significantly increase the risk of GSI-191 related failures.

2.6.4 Uncertainty Summary

Table 12 provides a summary of the uncertainty quantification results.

Table 12: Uncertainty Quantification Results

Base Case Input or Model	Sensitivity Case Input or Model	ΔCDF (yr^{-1})	ΔLERF (yr^{-1})
Pipe Break Risk Based on 25-year GM LOCA Frequency Input	Pipe Break Risk Based on 25-year Geometric 5th Percentile Input	3.1E-09	8.8E-14
	Pipe Break Risk Based on 25-year Geometric 95th Percentile Input	2.2E-06	6.2E-11
Pipe Break Risk Based on Continuum Break Model	Pipe Break Risk Based on DEGB-Only Model	< 6.6E-07	< 1.9E-11
Pipe Break Risk Based on Geometric LOCA Frequency Model	Pipe Break Risk Based on Arithmetic LOCA Frequency Model	5.2E-06	1.5E-10
Seismic Risk Model Based on Representative Fragility Parameters from EPRI 3002000709 (Reference 19)	Seismic Risk Model Based on Site-Specific Fragility Parameters and the Guidance in NUREG-1903 (Reference 21)	3.9E-07	1.1E-11

Three of the five uncertainty quantification sensitivities show that using alternate inputs or models would decrease ΔCDF and ΔLERF. The two sensitivity cases that show increased risk metrics are the geometric 95th percentile input and the arithmetic aggregation LOCA frequency model. Using the geometric 95th percentile input increases ΔCDF to 2.2E-06 yr^{-1}, which is on the low end of RG 1.174 Region II. Replacing the GM LOCA frequency with the AM LOCA frequency increases ΔCDF by almost an order of magnitude, which is enough to push the results to the middle of RG 1.174 Region II. However, as described in a comparison of the NUREG-1829 geometric and arithmetic means performed for STP, there is a strong basis for the geometric aggregation providing a more accurate GSI-191 risk quantification than the arithmetic aggregation (Reference 17). Also, there are many conservatisms that were included in the WCGS risk quantification that have a significant effect on the results. If these conservatisms were removed, it is expected that the risk results would decrease significantly. The following are some of the most significant conservatisms:

- Competing conservatisms were used in the GSI-191 evaluation (e.g., a conservatively high sump temperature was used for calculations where a maximum temperature is bounding, and a conservatively low sump temperature was used for calculations where a minimum temperature is bounding). Many breaks that are currently predicted to fail would be likely not to fail if all calculations were performed with consistent inputs (e.g., using a consistent temperature for all parts of the GSI-191 evaluation would significantly reduce the level of conservatism and the number of breaks that fail).

- A single, bounding threshold break size was determined for all equipment configurations without taking credit for the fact that the most likely equipment configuration (all pumps running) would have a larger threshold break size. Taking credit for the probability of the various equipment configurations along with the associated threshold break sizes would significantly reduce ΔCDF and ΔLERF.
- All breaks greater than or equal to the threshold break size were assumed to fail even though all of the breaks at the threshold break size and many of the larger breaks would not fail. Taking credit for the successful mitigation of breaks larger than or equal to the threshold break size that do not fail would significantly reduce the CFP and the corresponding ΔCDF and ΔLERF.

Based on the results of the uncertainty quantification and a consideration of the significant conservatisms, it can be concluded with high confidence that the risk associated with the effects of debris at WCGS is very small (i.e., within Region III of RG 1.174, Reference 4).

3.0 Defense-in-Depth and Safety Margin

As described in RG 1.174 (Reference 4), sufficient defense-in-depth and safety margin must be maintained. Both of these aspects were evaluated in detail as described in Attachment IX.

4.0 Monitoring Program

WCNOC has implemented procedures and programs for monitoring, controlling, and assessing changes to the plant that have a potential impact on plant performance related to GSI-191 concerns. Training is provided to personnel accessing containment to raise their awareness of the more stringent containment cleanliness requirements, the potential for sump blockage, and actions being taken to address sump blockage concerns. Procedures have been implemented to ensure the containment building is free of loose debris, to verify the condition of the sump strainers, and to control unattended temporary materials in containment. Strict controls have been imposed on the types and quantities of materials that may be taken into containment. WCNOC has also implemented a coatings condition assessment monitoring program to ensure that the coatings debris limit will not be adversely impacted. Section 2.2.1 of Attachment IX of this submittal provides further details on these procedures and programs.

5.0 Quality Assurance

The WCGS analyses and testing for various GSI-191 phenomena were performed as safety related under the WCGS or vendor quality assurance (QA) programs compliant with 10 CFR 50 Appendix B. The PRA evaluations were not performed as safety related but were developed under the vendor's QA program.

6.0 Periodic Update of Risk-Informed Analysis

The risk-informed GSI-191 analysis will be updated within at least 48 months following initial NRC approval or since the last update. This update will include all parts of the risk-informed evaluation including the systematic risk assessment, consideration of defense-in-depth, and consideration of safety margin. The update will also include any new information on LOCA frequencies that may be developed. The intent of the update is to capture the effects of any plant changes, procedure changes, or new information on the risk-informed analysis and to confirm that the acceptance criteria are still maintained. Reliability data, unavailability data, initiating event frequency data, human reliability data, and other similar PRA inputs are reviewed approximately every two fuel cycles to maintain the base WCGS PRA model consistent with the as-operated plant. In addition, existing guidance is in place for periodic updates of risk-informed applications.

The key elements of the WCGS risk-informed analysis are summarized in Attachment I and in USAR Appendix 6A (as updated in Attachment VI). Changes to these key elements are to be evaluated as a potential departure from a method of evaluation described in the USAR in accordance with 10 CFR 50.59(c)(2)(viii).

7.0 Reporting and Corrective Actions

Nonconformances with existing evaluations, or problem identification, will be entered into the station corrective action program for evaluation and corrective actions, as appropriate. Nonconforming conditions that make the containment sump inoperable for longer than the TS completion time (see markup to the TS in Attachment III) will meet 10 CFR 50.73 reporting criteria for a condition prohibited by TS. WCNOC will also report to the NRC and take corrective actions in the event that the debris-related ΔCDF and ΔLERF exceed the acceptance criteria corresponding to the upper threshold for RG 1.174 Region III (i.e., 1×10^{-6} for ΔCDF and 1×10^{-7} for ΔLERF) (Reference 4) in accordance with 10 CFR 50.72 and 10 CFR 50.73, as applicable.

8.0 License Application

The specific requirements for the license application described in RG 1.174 (Reference 4) are addressed in Attachment I.

9.0 References

1. SRM-SECY-10-0113, “Staff Requirements – SECY-10-0113 – Closure Options for Generic Safety Issue – 191, ‘Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance’,” December 23, 2010
2. SRM-SECY-12-0093, “Staff Requirements – SECY-12-0093 – Closure Options for Generic Safety Issue – 191, ‘Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance’,” December 14, 2012

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3. Draft Regulatory Guide 1.229 (ML16062A016, ML17025A263), Revision 0, “Risk-Informed Approach for Addressing the Effects of Debris on Post-Accident Long-Term Core Cooling”
4. Regulatory Guide 1.174, Revision 3, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” May 2011
5. Regulatory Guide 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” March 2009
6. NUREG-1829, “Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process,” April 2008
7. ASME/ANS RA-Sa-2009, “Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” February 2009
8. Generic Letter 2008-01, “Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems,” January 11, 2008
9. NUREG-1855, Revision 1, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking,” March 2017
10. EPRI Report 1016737, “Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments,” December 2008
11. EPRI Report 1026511, Technical Update, “Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty,” December 2012
12. NEI 04-07, Revision 0, “Pressurized Water Reactor Sump Performance Evaluation Methodology, ‘Volume 1 – Pressurized Water Reactor Sump Performance Evaluation Methodology’,” December 2004
13. NEI 04-07, Revision 0, “Pressurized Water Reactor Sump Performance Evaluation Methodology, ‘Volume 2 – Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02’,” December 2004
14. NRC Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,” September 13, 2004
15. NUREG/CR-6850, “Fire PRA Methodology for Nuclear Power Facilities,” September 2005
16. NUREG/CR-6928, “Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants,” Component Reliability Data Sheets 2015 Update, February 2017
17. STP-RIGSI191-ARAI.01 (ML14149A437), Revision 3, “Means of Aggregation and NUREG-1829: Geometric and Arithmetic Means,” April 18, 2014
18. WCAP-16530-NP-A, “Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191,” March 2008
19. EPRI Report 3002000709, “Seismic Probabilistic Risk Assessment Implementation Guide, Final Report,” December 2013
20. ML17038A223, Enclosure 3, “Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Nos. 212 and 198 to Facility Operating License

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Nos. NPF-76 and NPF-80 STP Nuclear Operating Company, et al. South Texas Project, Units 1 and 2 Docket Nos. 50-498 and 50-499," July 11, 2017

21. NUREG-1903, "Seismic Considerations for the Transition Break Size," February 2008
22. NEI 09-10, Revision 1a-A, "Guidelines for Effective Prevention and Management of System Gas Accumulation," April 2013
23. WCAP-16406-P, Revision 1, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," August 2007
24. WCAP-17788-NP, Volume 1, Revision 1, "Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090)," December 2019
25. WCAP-16882-NP, Revision 1, "PRA Modeling of Debris-Induced Failure of Long Term Core Cooling via Recirculation Sumps," November 2009
26. WCAP-15750, "Risk Informing Hot Leg ECC Switchover Requirements Phase 1: Basis and Recommendations," October 2001
27. ET 08-0045. Docket No. 50-482, "Nine-Month Response to NRC Generic Letter 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems," October 10, 2008
28. 09-00564, "Wolf Creek Generating Station - RE: Closure Letter for Generic Letter 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems" (TAC No. MD7896), October 28, 2009
29. ML17038A223, "South Texas Project, Units 1 and 2 - Issuance of Amendments Re: Changes to Design Basis Accident Analysis Using a Risk-Informed Methodology to Account for Debris in Containment (CAC Nos. MF2400 and MF2401)," July 11, 2017
30. South Texas Project, Units 1 & 2, Docket Nos. STN 50-498, STN 50-499, Response to RAI 4-2 of APLA Round 4 Requests for Additional Information Regarding STP Risk-Informed GSI-191 Licensing Application (TAC Nos. MF2400 and MF2401), July 28, 2016 (ML16221A393)
31. EPRI Report 103959, "Methodology for Developing Seismic Fragilities", June 1994

Wolf Creek Nuclear Operating Corporation
Licensing Submittal for a Risk-Informed Resolution of Generic Letter 2004-02

Attachment VIII

Updated Response to NRC Generic Letter 2004-02

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List of Acronyms

3D	Three-Dimensional
CAD	Computer Aided Design
CFD	Computational Fluid Dynamics
CFR	Code of Federal Regulations
CS	Containment Spray
CSS	Containment Spray System
DEGB	Double-Ended Guillotine Breaks
ECCS	Emergency Core Cooling System
GDC	General Design Criterion
GL	Generic Letter
GSI	Generic Safety Issue
ISI	In-Service Inspection
LAR	License Amendment Request
LOCA	Loss-of-Coolant Accident
NEI	Nuclear Energy Institute
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
RAI	Request for Additional Information
RCS	Reactor Cooling System
RG	Regulatory Guide
RHR	Residual Heat Removal
RWST	Refueling Water Storage Tank
SE	Safety Evaluation
TS	Technical Specification
USAR	Updated Safety Analysis Report
Wolf Creek	Wolf Creek Generating Station

Updated Response to NRC Generic Letter 2004-02

1. Overall Compliance

Provide information requested in GL 2004-02 Requested Information Item 2(a) regarding compliance with regulations.

GL2004-02 Requested Information Item 2(a)
Confirmation that the ECCS and CSS recirculation functions under debris loading conditions are or will be in compliance with regulatory requirements listed in the Applicable Regulatory Requirements section of this GL. This submittal should address the configuration of the plant that will exist once all modifications required for regulatory compliance have been made and this licensing basis has been updated to reflect the results of the analysis described above.

Response to 1:

This submittal by the Wolf Creek Generating Station (referred to as Wolf Creek hereafter) uses a risk-informed approach to address the effects of loss-of-coolant accident (LOCA)-generated debris on emergency core cooling system (ECCS) and containment spray system (CSS) recirculation functions per the requirements of the Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2004-02 (Reference 1). The risk-informed analysis covered a full spectrum of LOCAs, including partial breaks and double-ended guillotine breaks (DEGBs) on all Class 1 in-service inspection (ISI) welds outside of the reactor cavity. Break sizes ranging from 0.375 in up to and including the largest DEGBs on the main loop piping were evaluated to assure that both the most severe LOCAs and those smaller breaks with higher occurrence frequency are considered. With the risk-informed approach, a threshold break size was derived such that all breaks of this size and smaller pass all acceptance criteria (e.g., strainer head loss, air entrainment, structural limit, pump net positive suction head (NPSH), in-vessel and ex-vessel effects). Wolf Creek conservatively relegates to failure the individual breaks greater than the threshold break size. The results of the evaluation in Attachment VII show that the risk from the failures related to LOCA-generated debris is very small and falls in Region III of the risk maps shown in Regulatory Guide (RG) 1.174. The methodology includes conservatisms in the plant-specific testing and analysis, as well as the assumption that all breaks greater the threshold break size are relegated to failure. Conservatisms in the Wolf Creek approach and additional defense in depth measures are discussed in Attachment IX.

The risk-informed approach replaces the existing deterministic approach described in the Wolf Creek licensing basis. Additionally, Wolf Creek plans to adopt the Technical Specification Task Force (TSTF) Traveler TSTF-567 to add a new section to the Technical Specification for the containment sump. Consequently, Wolf Creek proposes to amend the operating license to incorporate these changes per the requirements of Title 10 of the Code of Federal Regulations (CFR) Section 50.59 (10 CFR 50.59). The proposed amendment to the operating license is described in Attachment I of this

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submittal. Attachment II shows the request for exemption from certain requirements of 10 CFR 50.46(a)(1) related to the use of deterministic methodology to evaluate the effects of debris on long-term core cooling.

The following list summarizes the key correspondences issued by the NRC or submitted to the NRC by Wolf Creek regarding the resolution of GL 2004-02 up to 2013. Note that, since then, new testing and analyses have been performed. This new submittal supersedes all of the previous GL 2004-02 submittals and responses entirely.

1. Wolf Creek Letter ET 05-0018 (Reference 2) submitted the initial responses to GL 2004-02 to the NRC.
2. By the letter dated February 9, 2006 (Reference 3), the NRC issued RAIs upon their review of the GL responses.
3. Wolf Creek Letter ET 08-0003 (Reference 4) responded to the requests for additional information (RAIs) issued by the NRC on February 9, 2006.
4. Wolf Creek Letter ET 08-0046 (Reference 5) informed the NRC the completion of new analyses and committed to submit an update to the previous GL 2004-02 responses by December 2008.
5. Wolf Creek Letter ET 08-0053 (Reference 6) submitted a full revision to the Wolf Creek GL 2004-02 responses.
6. By the letter dated July 31,2009 (Reference 7), the NRC issued additional RAIs upon their review of the updated GL responses.
7. Wolf Creek Letter ET 13-0017 (Reference 8) informed the NRC that Wolf Creek will take a risk-informed resolution path for GL 2004-02 and outlined a resolution schedule.

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2. General Description of and Schedule for Corrective Actions

Provide a general description of actions taken or planned, and dates for each. For actions planned beyond December 31, 2007, reference approved extension requests or explain how regulatory requirements will be met as per Requested Information Item 2(b). (Note: All requests for extension should be submitted to the NRC as soon as the need becomes clear, preferably no later than October 1, 2007.)

GL 2004-02 Requested Information Item 2(b)
A general description and implementation schedule for all corrective actions, including any plant modifications, that you identify while responding to this GL. Efforts to implement the identified actions should be initiated no later than the first refueling outage starting after April 1, 2006. All actions should be completed by December 31, 2007. Provide justification for not implementing the identified actions during the first refueling outage starting after April 1, 2006. If corrective actions will not be completed by December 31, 2007, describe how the regulatory requirements discussed in the Applicable Regulatory Requirements section will be met until the corrective actions are completed.

Response to 2:

Wolf Creek has performed physical modification, and comprehensive testing and analyses to determine the impact of LOCA-generated debris on the ECCS and CSS recirculation function during post-LOCA recirculation phase. These testing program and analyses conform, to the greatest extent practicable, to various industrial guidance and associated NRC safety evaluations (SEs) as applicable. Refer to Section 3 of this Attachment for details.

Wolf Creek has completed the following modification, analyses and program updates in the effort of responding to the NRC GL 2004-02 and Generic Safety Issue (GSI) 191.

Physical Modifications

1. Replaced containment emergency sump recirculation screens with new strainers. Each new sump strainer contains stacked perforated plate disks arranged into modules that maximize the strainer surface area. The strainer perforated plates have 0.045 in diameter holes to efficiently capture debris that reaches the strainer. Each of the new strainers has a net surface area of over 3300 ft^2, which is much greater than the surface area of 200 ft^2 of the original sump screen.
2. Installed flow diverters in openings through the secondary shield wall next to Loops A and D that are near the recirculation sumps. The flow diverters are made of perforated plates and prevent the “short path” flow of debris-laden fluid directly to the sumps and force the fluid to take a "long path" through the shield wall openings next to Loops B and C which are farther away from the sumps. This torturous flow path allows more debris to settle out.

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Testing and Analyses

1. Implemented a containment latent debris assessment program and performed periodic latent debris sampling and characterization walkdowns.
2. Performed detailed structural analysis of the new strainers.
3. Developed a three-dimensional (3D) computer aided design (CAD) model of the Wolf Creek containment. The model included pipe welds, insulation, concrete surfaces, large supporting structures and coatings.
4. Performed debris generation analysis to quantify the debris that could be generated by a large spectrum of partial breaks and DEGBs postulated on all Class 1 ISI welds.
5. Performed containment sump pool water level analysis to determine the minimum post-LOCA pool levels.
6. Performed debris transport analysis to determine the fraction of generated debris that could reach the strainer for several representative break locations. As part of the analysis, computational fluid dynamics (CFD) models were used to simulate the recirculation flow patterns in the post-LOCA sump pool.
7. Quantified chemical precipitation debris that could form in the post-LOCA containment environment using the methodology of WCAP-16530.
8. Performed strainer debris head loss and fiber penetration testing following the latest NRC guidance.
9. Performed ex-vessel downstream effects analysis to evaluate potential wear and clogging of various ECCS and CSS components during post-LOCA recirculation phase using the methodology of WCAP-16406.
10. Performed in-vessel downstream effects on long-term core cooling following the latest NRC review guidance.
11. Analyzed head loss testing data to determine bounding strainer head loss values at plant conditions of interest.
12. Updated the NPSH analysis for the residual heat removal (RHR) and containment spray (CS) pumps to incorporate the latest strainer head loss results.
13. Compared the latest head loss results with strainer structural limit to ensure strainer integrity.
14. Performed strainer degasification, flashing and vortexing evaluation to ensure the recirculation functions of the RHR and CS pumps are not adversely affected by air entrainment.
15. Quantified the risk associated with strainer and core failures caused by LOCA-generated debris.

Program Updates

1. Revised design change process procedures and implemented standard design process. Per the new design process, Design Attribute Review forms are used, which require responses and necessary engineering reviews to various screening questions to identify potential impact on long term core cooling by the proposed modification. Selected questions that are related to GSI-191 are listed below.

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- Does the modification affect insulation?
- Does the modification add or remove components in containment?
- Does the modification change the amount of exposed aluminum and/or zinc in containment?
- Does the modification introduce materials that could affect sump performance or lead to equipment degradation?
- Does the modification repair, replace, or install coatings inside containment, including installing coated equipment?
- Does the modification affect installation, replacement, or storage of any structure, system, component or other items in containment that has vendor applied or site applied protective coatings?
- Does the modification affect high/moderate energy line break analysis?
- Does the modification affect the design, performance or operation of pumps?
- Does the modification affect foreign material that would require cleaning to prevent degradation of downstream components?

2. Enhanced the containment entry and material control procedure with requirements of controlling materials during work activities conducted in the containment and radiological postings during plant operational Modes 1 through 4.
3. Revised the clearance order procedure to ensure that GL 2004-02 analyses are considered prior to making future changes to existing requirements that clearance order tags are not installed on components inside the containment being removed from service (tagged out) during plant operational Modes 1 through 4.
4. Revised the work request procedure to ensure that GL 2004-02 analyses are considered prior to making future changes to existing requirements that work request tags are not installed on components inside the containment.
5. Revised the scaffold construction and use procedure to enhance the requirements for controlling scaffold tags and materials used during work activities conducted inside containment during plant operational Modes 1 through 4.
6. Revised procedures to require a survey of latent debris after any invasive or extended maintenance activity.
7. Implemented a containment coatings program for monitoring and assessing the containment building coatings, including administrative controls on coating examinations, deficiency reporting and documentation.
8. Implemented changes to Technical Specifications surveillance procedures to ensure that the sump strainers do not have openings in excess of the strainer's maximum designed opening.
9. Implemented interim compensatory measures in accordance with NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors." These measures will remain in place at a minimum until the full resolution of GL 2004-02 for Wolf Creek.
10. Implemented procedures to ensure that alternative water sources are available to refill the Refueling Water Storage Tank (RWST) or to otherwise provide inventory to inject into the reactor core and spray into the containment atmosphere.

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11. Completed training on sump clogging issues for licensed operators, Operations, Engineering and Emergency Response organization personnel.
12. Implemented procedures to ensure containment drainage paths are unblocked following maintenance.

Wolf Creek has determined that no new corrective actions for modifications or remediation measures are required for the closure of GL 2004-02.

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3. Specific Information Regarding Methodology for Demonstrating Compliance

3.a Break Selection

The objective of the break selection process is to identify the break size and location that present the greatest challenge to post-accident sump performance.

3.a.1 Describe and provide the basis for the break selection criteria used in the evaluation.

Response to 3.a.1

Break selection performed in the Wolf Creek debris generation analysis followed the guidance in NEI 04-07 and the associated NRC SE on NEI 04-07 (Reference 9 pp. 3-5 - 3-26, 4-1 - 4-5; 10 pp. 12-35, 85-91, respectively). The objective of the break selection process is to identify the break conditions that present the greatest challenge to post-accident sump performance. At Wolf Creek, the break selection considered the following:

- Secondary side breaks. Secondary side breaks were shown to be bounded by the breaks on the primary side and therefore were not analyzed in detail (see the Response to 3.a.2).
- Non-piping break LOCAs. Detailed discussion is in Section 2.3.1 of Attachment VII.
- Breaks on in-service inspection (ISI) welds <u>outside</u> the first isolation valve and within the second isolation valves. Debris quantities for these breaks were evaluated in the debris generation analysis and were shown to be bounded by similar sized breaks on ISI welds inside the first isolation valve.
- Breaks at ISI welds <u>inside</u> the first isolation valve and outside of the reactor cavity. These breaks were evaluated in detail in the debris generation analysis and are the focus of the documentation provided in this section.

The remainder of the section will focus on the breaks inside the first isolation valve. Multiple breaks were postulated on every Class 1 ISI pipe weld inside the first isolation valve and outside of the reactor cavity. The use of pipe welds as break locations is acceptable since, as stated in NUREG-1829 (Reference 11 p. xviii), “welds are almost universally recognized as likely failure locations because they can have relatively high residual stress, are preferentially-attacked by many degradation mechanisms, and are most likely to have preexisting fabrication defects.”

Breaks at the reactor nozzle were excluded from the analysis. As the hot and cold legs pass through the primary shield wall, they are held by whip restraints. By design, only lateral movement allowed by the gap in the restraint is possible. It was shown that the maximum allowable lateral movement of the pipes is less than the pipe wall thickness. The reactor coolant pump (RCP) tie rods preclude the movement of the RCP and the possibility of cold leg separating from the reactor vessel for a postulated break at the reactor nozzle. Similarly, the steam generator (SG) lower lateral supports

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preclude the movement of the SG and the possibility of hot leg separating from the reactor vessel for a postulated break at the reactor nozzle. Therefore, no separation between the nozzles and RCS piping would be allowed for a break at the nozzle and there would be no jet flow if the weld at either the cold leg or hot leg were to fail (Reference 12; 13).

The guidance in NEI 04-07 suggests analyzing potential breaks at equal increments along the pipe (Reference 9 pp. 3-9). However, per the NRC's SE on NEI 04-07, evaluating breaks at equal increments is "only a reminder to be systematic and thorough" (Reference 10 p. 17). The Wolf Creek's approach of using Class 1 ISI welds as break locations is both systematic and thorough because there are multiple ISI welds on every pipe in the reactor coolant system (RCS) and the welds cover the full range of possible break locations. In addition, a weld is generally closer to equipment that has a large quantity of insulation, compared to a span of straight pipe (e.g., a break on the hot leg weld at the base of the steam generator will typically generate more debris than a break halfway between the steam generator and reactor vessel). Figure 3.a.1-1 shows locations of example welds which were considered as break locations. Note that this figure may not show all of the weld locations evaluated in the debris generation analysis. The Response to 3.a.3 details the types of breaks analyzed for debris generation.

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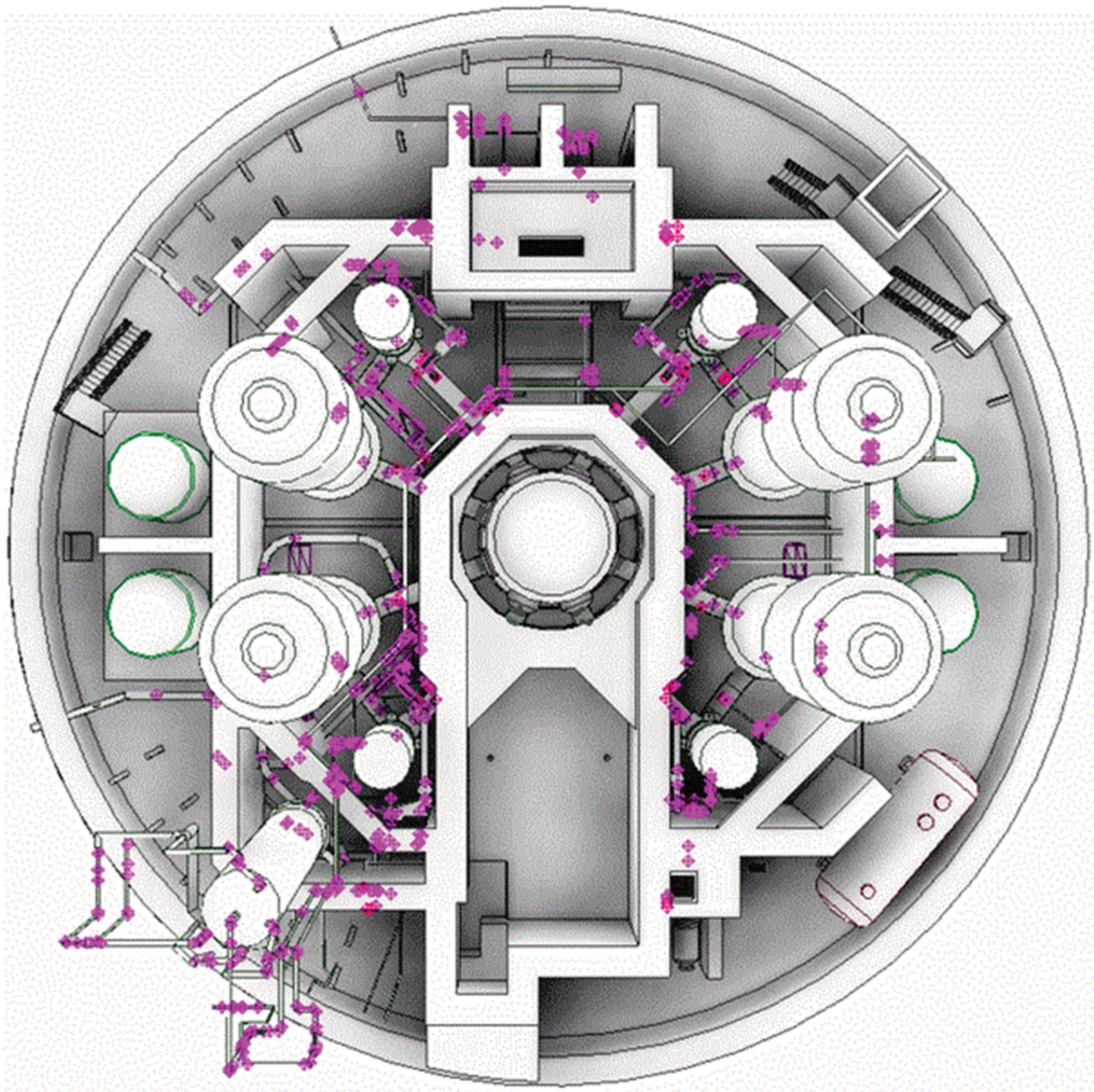

Figure 3.a.1-1: Example of Weld Locations with Postulated LOCAs

3.a.2 State whether secondary line breaks were considered in the evaluation (e.g., main steam and feedwater lines) and briefly explain why or why not.

Response to 3.a.2

Although certain secondary side line breaks may initiate CS, the ECCS would not be required for long term core cooling unless there are subsequent failures, such as a stuck open power operated relief valve (PORV) or loss of auxiliary feedwater, that would require feed and bleed cooling. As discussed in Section 2.3.3 of Attachment VII, a simplified and bounding evaluation was performed to assess the risk contribution from the secondary side line breaks. The evaluation assumed that all secondary side breaks that require ECCS recirculation (e.g., in a feed and bleed scenario) would fail due to the effects of debris. This is a conservative assumption because the overall pressure on the secondary side is lower than the primary side. Therefore, the zone of influence (ZOI) sizes for the secondary side breaks would be smaller than those for the breaks on similarly sized pipes of the primary side. The reduced ZOI sizes result in less debris generated by a secondary side break. Additionally, the flow rate through the strainer required for feed and bleed cooling is significantly lower than the ECCS flow rate for a large break LOCA used in the debris transport analysis, and head loss and fiber penetration testing. The reduction in flow rate and generated debris quantities would result in less debris transport to the strainers, lower strainer head losses and less fiber penetration than the primary side breaks. As a result, the secondary side breaks would not impact the threshold break size determined based on the primary side breaks.

3.a.3 Discuss the basis for reaching the conclusion that the break size(s) and locations chosen present the greatest challenge to post-accident sump performance.

Response to 3.a.3

Debris generation quantities were evaluated for breaks on every Class 1 ISI pipe weld upstream of the first isolation valve and outside of the reactor cavity. The welds are sufficiently close, with sufficient overlap in the ZOIs to provide confidence that the debris load that presents the greatest challenge to post-accident sump performance has been captured (see the Response to 3.a.1).

The debris generation analysis used the BADGER software to place ZOIs on every Class 1 ISI weld inside the first isolation valve and outside of the reactor cavity in a three-dimensional (3D) Computer Aided Design (CAD) model of the Wolf Creek containment. Both DEGBs and partial breaks were evaluated. For a DEGB, its ZOI is represented by a sphere centered at the break location. At each weld, partial breaks were also analyzed for sizes ranging between 0.375 in and up to 26 in (as applicable), and, additionally, for each break size, partial breaks were postulated along the circumference of weld at 8 different angles that are 45° apart. The ZOI of a partial break is a hemisphere center at the edge of the pipe (see Figure 3.a.3-1). BADGER determines the interference between a ZOI and the insulation or coating materials in the CAD model to quantify debris inside the ZOI. For certain debris types (e.g., Nukon), a size distribution was applied based on the distance between the debris and the center of the break. The debris types with non-break specific quantities (e.g., latent debris, unqualified coatings and miscellaneous debris) are quantified separately outside of BADGER.

The Wolf Creek debris generation analysis resulted in nearly 14,000 breaks inside the first isolation valve and outside of the reactor cavity with a wide range of break locations, orientations and sizes. Note that, while DEGBs on the main loop piping are typically bounding with regard to debris loads, smaller partial breaks are more likely

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to occur. Therefore, analyzing the full spectrum of break sizes is necessary to quantify the risks associated with the pipe breaks. With the above approach, the debris generation analysis provides reasonable assurance that the maximum debris loads and the worst-case combination of debris types are captured.

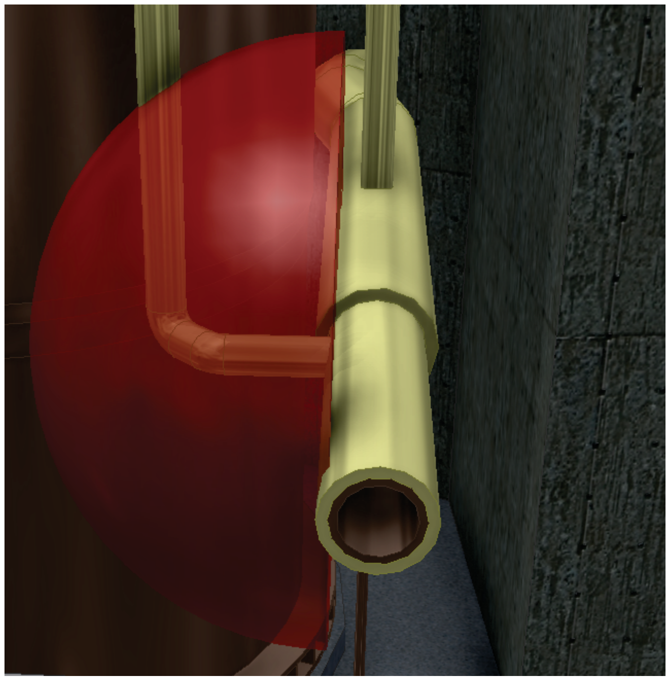

Figure 3.a.3-1: Single Partial Break ZOI

As stated above, at each weld, partial breaks at 8 different angles along the circumference of the weld were analyzed with an angular increment of 45°. This is consistent with the guidance in the NRC SE on NEI 04-07, which states that "licensees will need to simulate various directions around the RCS main-loop piping to determine the limiting break location" (Reference 10 p. 117). Figure 3.a.3-2 shows a graphical representation of the 8 ZOIs at different angles for a given weld. As shown in the figure, there is significant overlap between the ZOIs. Additionally, partial breaks were analyzed for various sizes: 0.375 in, 0.5 in, 2 in, 4 in, 6 in, 8 in, 10 in, 12 in, 14 in, 17 in, 20 in, 23 in and 26 in. The potential for missing significant debris quantity spikes or trends by using these orientation and size increments was evaluated by running a BADGER sensitivity evaluation using the model for a 4-loop Westinghouse plant with low density fiberglass insulation (Reference 14).

The sensitivity evaluation considered 6-in and 8-in breaks at 100 different weld locations and refined the orientation increment from 45° to 15° and the size increment to 0.25 in. The comparison between the refined and original results showed that the maximum quantity of debris generated at the worst case orientation was approximately 2% higher on average with the 15° increment data compared to the 45° increment data. Additionally, reducing the break size increment did not result in any spikes or trends in the maximum debris loads. As will be shown in the Responses to

3.f and 3.n, the plant debris loads for the breaks equal to and less than the threshold breaks size are well bounded by the tested debris loads. Therefore, refining the angle and/or break size increments will not invalidate the head loss or in-vessel threshold break size, and will not adversely impact the risk quantification.

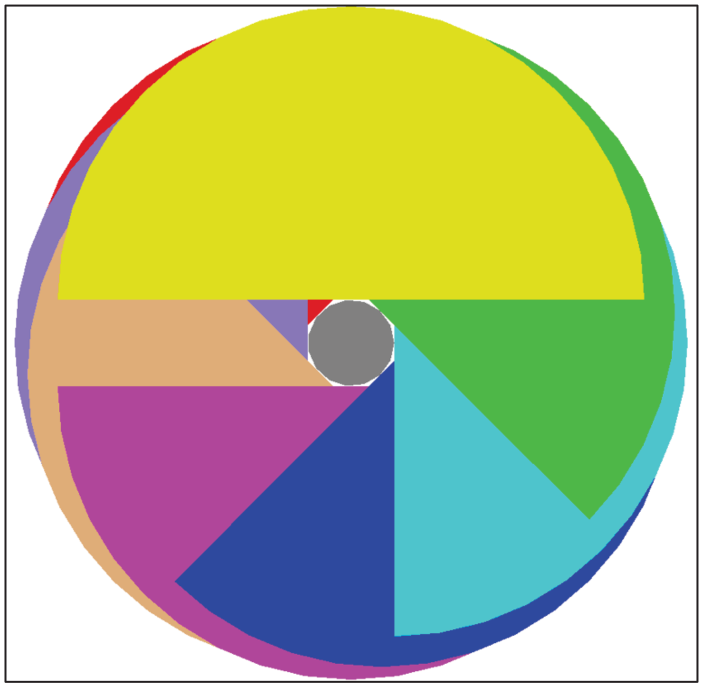

Figure 3.a.3-2: Visualization of Partial Break ZOIs at One Weld with 45° Increment

The debris generation analysis also contains the following conservatisms:

- Materials with undefined debris size distributions were assumed to be destroyed as 100% fines per the NRC SE on NEI 04-07 (Reference 10 Section 3.4.3.3).
- Materials with undefined destruction pressures were evaluated using the maximum ZOI size of 28.6D per the NRC SE on NEI 04-07 (Reference 10 Section 3.4.2.2).
- All qualified coatings on steel and concrete were analyzed as having the worst-case coating system for each surface type (see the Response to 3.h.1).
- Main loop breaks in the steam generator (SG) compartments were grouped by loop and truncated collectively in a way that could result in conservative amounts of debris generated for some breaks.

The tables below show the ranges of debris loads resulting from the Wolf Creek debris generation analysis. Specific generated debris loads of selected bounding breaks are shown in the Response to 3.b.4.

Table 3.a.3-1 and Table 3.a.3-2 show the minimum, average, and maximum amounts of debris generated by DEGBs and partial breaks within the first isolation valve for different break size ranges and debris types. Note that, for a given break size range, the individual quantities for Nukon fines, small pieces, large pieces, and intact blankets

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may come from different breaks and therefore, may not add up to the total Nukon fiber quantity shown in the table. Additionally, the average debris loads for a given break size range represent direct averages over debris loads of individual breaks without considering actual LOCA frequencies of different break sizes. As discussed in the Responses to 3.b.1 and 3.b.4, the quantity of fire barrier (Thermo-lag) debris is identical for all breaks analyzed and is not listed in Table 3.a.3-1 and Table 3.a.3-2.

Table 3.a.3-1 and Table 3.a.3-2 show that no antisweat, Cerablanket, lead blanket or FOAMGLAS materials were destroyed by any analyzed DEGBs or partial breaks inside the first isolation valve and outside of reactor cavity because they are outside of the ZOIs for all analyzed breaks.

Table 3.a.3-1: Debris Generated by DEGBs

Debris Type	Debris Size	Debris Quantity Generated								
		Small Breaks (< 2")			Medium Breaks (2"- 6")			Large Breaks (> 6")		
		Min	Avg	Max	Min	Avg	Max	Min	Avg	Max
Nukon (ft³)	Fines (Individual Fibers)	0.0	0.2	0.8	0.0	2.7	8.9	10.4	91.5	267.8
	Small Pieces (< 6" a side)	0.0	0.8	2.6	0.0	8.8	29.7	32.2	306.3	920.9
	Large Pieces (> 6" a side)	0.0	0.5	1.8	0.0	5.6	19.3	25.3	170.5	461.1
	Intact (Covered) Blankets	0.0	0.6	1.9	0.0	6.1	20.8	27.4	184.2	498.2
	All Debris Within ZOI	0.0	2.1	7.0	0.0	23.3	73.4	96.5	752.7	2084.8
Antisweat (ft³)	Fiber Fines	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Lead Blanket (ft³)		0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
FOAMGLAS (lbm)	Particulate	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
IOZ Qualified Coatings (lbm)		0.0	0.4	2.1	0.0	1.8	14.0	20.5	255.1	722.8
Epoxy Qualified Coatings (lbm)		0.0	<0.1	0.1	0.0	0.2	2.1	0.0	21.0	94.7
Protected Unqualified Coatings (lbm)		0.0	0.0	0.0	0.0	1.2	9.3	0.0	68.1	240.2

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Table 3.a.3-2: Debris Generated by Partial Breaks

Debris Type	Debris Size	Small Breaks (< 2")			Medium Breaks (2"- 6")			Large Breaks (> 6")		
		Min	Avg	Max	Min	Avg	Max	Min	Avg	Max
Nukon (ft^3)	Fines (Individual Fibers)	0.0	<0.1	0.3	0.0	1.1	9.9	0.0	28.0	140.7
	Small Pieces (< 6" a side)	0.0	0.1	1.0	0.0	3.52	32.9	0.0	92.62	467.1
	Large Pieces (> 6" a side)	0.0	<0.1	0.8	0.0	2.55	22.9	0.0	55.84	307.2
	Intact (Covered) Blankets	0.0	<0.1	0.8	0.0	2.75	24.7	0.0	60.33	331.9
	All Debris Within ZOI	0.0	0.2	3.0	0.0	9.9	84.3	0.0	236.8	1231.6
Antisweat (ft^3)	Fiber Fines	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Lead Blanket (ft^3)		0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
FOAMGLAS (lb)	Particulate	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
IOZ Qualified Coatings (lb)		0.0	<0.1	0.1	0.0	2.1	28.4	0.0	83.7	430.2
Epoxy Qualified Coatings (lb)		0.0	0.0	0.0	0.0	<0.1	1.9	0.0	3.3	51.1
Protected Unqualified Coatings (lb)		0.0	0.0	0.0	0.0	0.4	19.5	0.0	18.6	176.1

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3.b Debris Generation/Zone of Influence (excluding coatings)

The objective of the debris generation/ZOI process is to determine, for each postulated break location: (1) the zone within which the break jet forces would be sufficient to damage materials and create debris; and (2) the amount of debris generated by the break jet forces.

3.b.1 Describe the methodology used to determine the ZOIs for generating debris. Identify which debris analyses used approved methodology default values. For debris with ZOIs not defined in the guidance report/SE, or if using other than default values, discuss method(s) used to determine ZOI and the basis for each.

Response to 3.b.1

In the Wolf Creek debris generation analysis, the ZOI for a DEGB is defined as a spherical volume centered at the break in which the jet pressure is higher than the destruction/damage pressure for a certain type of insulation, coatings, or other materials impacted by the break jet. In a pressurized water reactor (PWR) reactor containment building, the worst-case pipe break would typically be a DEGB. In a DEGB, jets of water and steam would blow in opposite directions from the severed pipe. One or both jets could impact obstacles and be reflected in different directions. To take into account the double jets and potential jet reflections, NEI 04-07 (Reference 9, p. 1-3; 10, p. vii) proposes using a spherical ZOI centered at the break location to determine the quantity of debris that could be generated by a given line break.

For any break smaller than a DEGB (i.e., a partial break), the NRC SE on NEI 04-07 accepts the use of a hemispherical ZOI centered at the edge of the pipe (Reference 10, p. 117). Because these types of breaks could occur anywhere along the circumference of the pipe, the partial breaks were analyzed using hemispheres at eight different angles that are 45 degrees apart from each other around the pipe.

Because different insulation types have different destruction pressures, insulation-specific ZOIs were determined. Table 3.b.1-1 shows the primary side break equivalent ZOI radii divided by the break diameter (L/D) for each representative material in the Wolf Creek containment building.

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Table 3.b.1-1: ZOI Radii for Wolf Creek Insulation Types

Insulation Type	Destruction Pressure (psi)	ZOI Radius/Break Diameter (L/D)
Nukon	6[a]	17.0[a]
Antisweat Insulation FOAMGLAS Fire Barrier Material[c]	2.4[b]	28.6[b]
Lead Blankets	24[c]	5.4[c]
Qualified Coatings	40	4.0[d]
Qualified Untopcoated IOZ Coatings	-	10[e]

a NRC SE on NEI 04-07 (Reference 10 p. 30)
b NRC SE on NEI 04-07 (Reference 10 p. Table 3.2)
c See discussion below
d Revised Guidance Regarding Coatings Zone of Influence for Review of Final Licensee Responses to Generic Letter 2004-02 (Reference 15 p. 2). The 40 psi destruction pressure corresponds to a 4D ZOI in Table 3-1 of the SER (Reference 10 p. 27)
e See the Response to 3.h.1

The fire barrier material at Wolf Creek was divided into two categories, covered/ jacketed and uncovered. The covered/jacketed fire barrier material is located in the southwest quadrant of the annulus, outside of the pressurizer compartment and is protected from all break ZOIs by robust barriers. As discussed later in the Response to 3.b.1, robust barriers consist of structures, such as concrete walls that are impervious to jet flow and prevent further expansion of the jet. Insulation in the shadow of large robust barriers can be assumed to remain intact to a certain extent (Reference 10, Section 3.4.3.2). All uncovered fire barrier material was considered to be destroyed for any LOCA. Therefore, the quantity of fire barrier material is identical for any break in containment, see the Response to 3.b.4.

The lead blankets at Wolf Creek are required to withstand 48.1 psig without degradation. Similar lead blankets were tested in the Utility Resolution Guide (URG) for ECCS Suction Strainer Blockage and were shown not to generate debris when subjected to an air jet with a pressure of 40 psig at the blanket surface (Reference 16 p. 173 of Tab 3). Consistent with the NRC SE on NEI 04-07, the air jet pressure was reduced by 40% to account to a PWR two-phase jet (Reference 10 p. 29), reducing the destruction pressure to 24 psig. From Table 3-1 of the NRC SE on NEI 04-07, a destruction pressure of 24 psig corresponds to a ZOI size of 5.4D (Reference 10 p. 27).

The debris generation analysis also evaluated protected unqualified coatings applied to equipment, such as the steam generators and pressurizer. These coatings are protected by the insulation covering each component. Therefore, the unqualified coatings on these components would only become debris if a break destroys the overlaying insulation. As such, the protected unqualified coatings on these components were analyzed within a ZOI corresponding to their protective insulation types.

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In some cases, if the ZOI for a particular material is very large (i.e., the material has a low destruction pressure or is located on a large pipe), it may extend beyond robust barriers located near the break. Robust barriers consist of structures, such as concrete walls that are impervious to jet flow and prevent further expansion of the jet. Insulation in the shadow of large robust barriers can be assumed to remain intact to a certain extent (Reference 10, Section 3.4.3.2). Due to the compartmentalization of containment at Wolf Creek, the insulation on the opposite side of the compartment walls can be assumed to remain intact. However, the steam generator compartments share an opening where a break jet could extend, so this was accounted for by including destruction of some of the insulation in these areas. All ZOIs were truncated to account for robust barriers per the NRC SE on NEI 04-07 (Reference 10 p. vii).

Volumetric debris quantities were determined by measuring the interference between a ZOI and its corresponding debris sources. This was done within the CAD model environment. No jacketed insulation or qualified coatings debris would be generated outside of the ZOIs (Reference 9,pp. 3-19 through 3-20), as accepted in the SE on NEI 04-07 (Reference 10, Section 3.4.3.2).

3.b.2 Provide destruction ZOIs and the basis for the ZOIs for each applicable debris constituent.

Response to 3.b.2

See the Response to 3.b.1.

3.b.3 Identify if destruction testing was conducted to determine ZOIs. If such testing has not been previously submitted to the NRC for review or information, describe the test procedure and results with reference to the test report(s).

Response to 3.b.3

Wolf Creek applied the ZOI refinement discussed in the NRC SE on NEI 04-07 (Reference 10 p. Section 4.2.2.1.1), which allows the use of debris-specific spherical ZOIs. No new destruction testing was used to determine the ZOIs listed above.

3.b.4 Provide the quantity of each debris type generated for each break location evaluated. If more than four break locations were evaluated, provide data only for the four most limiting locations.

Response to 3.b.4

Using the ZOIs listed in the Response to 3.b.1, the breaks selected in the Response to 3.a.1, and the size distributions provided in the Response to 3.c.1, the quantities of generated debris for each break case were calculated for each type of debris. As discussed in Attachment VII, Wolf Creek used a simplified risk-informed approach to

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resolve GSI-191 and the risk quantification was based on a threshold break size of 10 in. Therefore, all DEGBs and partial breaks of 10 in and smaller pass the acceptance criteria (e.g., strainer head loss and structural limit, in-vessel effects, degasification).

The quantities of debris generated for the four most limiting break cases that do not fail any of the acceptance criteria are shown in Table 3.b.4-1. See the Response to 3.h.5 for the quantity of qualified and unqualified coatings for these breaks. See the Response to 3.d.3 for the quantity of latent debris. As discussed in the Response to 3.b.1, the quantity of fire barrier material is identical for all analyzed breaks.

Table 3.b.4-1: Generated Debris Quantities for the Four Worst-Case Breaks that Do Not Fail Any Acceptance Criteria

Break Location		BB01-F406 (SG 1&4)	BB-01-S105-04 (SG 1&4)	BB01-F405 (SG 1&4)	BB-01-S003-2 (PRZR)
Break Size		10"	10"	10"	10"
Break Type		Partial @ 90º	Partial @ 0º	Partial @ 135º	Partial @ 0º
Nukon (lbm)	Fine	62.4	63.34	59.02	26.38
	Small	192.82	212.14	187.75	78.90
	Large	162.84	118.10	138.34	76.18
	Intact	176.0	127.54	149.50	82.32
Thermo-Lag Fiber (lbm)	Fine	9.2	9.2	9.2	9.2
Thermo-Lag Particulate (ft^3)	Particulate	0.522	0.522	0.522	0.522

3.b.5 Provide total surface area of all signs, placards, tags, tape, and similar miscellaneous materials in containment.

Response to 3.b.5

Labels, tags, stickers, placards, and other miscellaneous materials were evaluated via walkdown. The amount of miscellaneous materials found by the walkdown was 7.1 ft^2. This 7.1 ft^2 of miscellaneous debris results in a 5.3 ft^2 (7.1 ft^2 x 75%) (Reference 10 p. 49) of reduction in total strainer surface area (i.e., sacrificial strainer area. For the GSI-191 analyses, a total miscellaneous debris surface area of 20 ft^2 was conservatively used, which results in a reduction in the plant strainer surface area of 15 ft^2 (20 ft^2 x 75%) (Reference 10 p. 49).

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3.c 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 head loss.

3.c.1 Provide the assumed size distribution for each type of debris.

Response to 3.c.1

A summary of the material properties of the debris types found within containment are listed in Table 3.c.1-1. Note that information for coatings debris is shown in the Responses to 3.h.1 and 3.h.6.

Table 3.c.1-1: Wolf Creek Debris Material Properties

Debris	Distribution	Density (lbm/ft³)	Characteristic Size
Nukon	See Following Section	2.4 (bulk) 159 (fiber)	7 μm
Thermal-Lag Fiber	100% Fines	159 (fiber)	7 μm
Thermal-Lag Particulate	100% Particulate	137.3 (particulate)	10 μm

<u>Size Distribution of Nukon Debris</u>

The debris generation analysis uses a four-category size distribution for Nukon based on the guidance in the NRC SE on NEI 04-07 (Reference 10, Appendix II and Appendix VI, p. VI-14). This guidance provides an approach for determining a size distribution for low-density fiberglass using the air jet impact test (AJIT) data, with conservatism added due to the potentially higher level of destruction from a two-phase jet. Within the 17.0D ZOI, the size distribution varies based on the distance of the insulation from the break (i.e., insulation debris generated near the break location consists of more small pieces than insulation debris generated near the edge of the ZOI).

Consequently, the following equations were developed to determine the fraction of fines (individual fibers), small pieces (less than 6 inches), large pieces (greater than 6 inches), and intact blankets as a function of the average distance between the break point and the centroid of the affected debris measured in units of break diameter (C).

$$F_{\text{fines}}(C)\left|\begin{array}{l}(0D \leftrightarrow 4D) = 0.2\\(4D \leftrightarrow 15D) = -0.01364 * C + 0.2546\\(15D \leftrightarrow 17D) = -0.025 * C + 0.425\end{array}\right.$$

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$$F_{smalls}\,(C)\left|\begin{array}{l}(0D \leftrightarrow 4D) = 0.8\\(4D \leftrightarrow 15D) = -0.0682 * C + 1.0724\\(15D \leftrightarrow 17D) = -0.025\ * C\ + 0.425\end{array}\right.$$

$$F_{large}(C)\left|\begin{array}{l}(0D \leftrightarrow 4D) = 0\\(4D \leftrightarrow 15D) = 0.0393 * C - 0.157\\(15D \leftrightarrow 17D) = -0.215 * C + 3.655\end{array}\right.$$

$$F_{intact}(C)\left|\begin{array}{l}(0D \leftrightarrow 4D) = 0\\(4D \leftrightarrow 15D) = 0.0425 * C - 0.170\\(15D \leftrightarrow 17D) = 0.265\ * C\ - 3.505\end{array}\right.$$

3.c.2 Provide bulk densities (i.e., including voids between the fibers/particles) and material densities (i.e., the density of the microscopic fibers/particles themselves) for fibrous and particulate debris.

Response to 3.c.2

See the Response to 3.c.1 for the material and bulk densities of the various types of debris.

3.c.3 Provide assumed specific surface areas for fibrous and particulate debris.

Response to 3.c.3

Specific surface areas could be calculated for each debris type based on the characteristic diameter described in the Response to 3.c.1. However, testing was used to determine strainer head loss and not an analytical method, so specific surface areas were not calculated or used for the Wolf Creek head loss evaluations (see the Response to 3.f).

3.c.4 Provide the technical basis for any debris characterization assumptions that deviate from NRC-approved guidance.

Response to 3.c.4

The Thermo-Lag 330-1 fire barrier material was assumed to be comprised of 90% (by volume) particulate and 10% fiber. To determine the percentage of fiber, it was assumed that the volatiles in the Thermo-Lag 330-1 listed in the material data safety sheet (MSDS) completely evaporate, conservatively increasing the overall percentage

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of fiber. Thermo-Lag 330-1 is 45% volatiles by volume and 5% fiber by volume. Once the volatiles have evaporated, the percentage of fiber was calculated to be 9.1%, as shown below:

$$\%Fiber = \frac{5\%}{(100\% - 45\%)} = 9.1\%$$

This value was then conservatively rounded up to 10%.

The particulate portion of Thermo-Lag 330-1 was assumed to have the density of silica (137.3 lb/ft^3), which is listed as a constituent in the MSDS. The particulate portion was assumed to fail as 10 μm particles, consistent with the NRC SE on NEI 04-07 assumption that coatings and latent particulate debris is composed of 10 μm spheres (Reference 10 p. 22). The fibrous portion of Thermo-Lag 330-1 was assumed to have the same microscopic material properties as Nukon, with a material density of 159 lb/ft^3 and a fiber diameter of 7 μm. This is reasonable as the MSDS lists the fibrous portion as fiberglass.

It was assumed in the debris generation analysis that that RMI did not require evaluation. RMI is installed solely on the reactor pressure vessel (except for the reactor vessel top head) and is shielded from the breaks outside of the reactor cavity by the primary shield wall. Thus, RMI was not analyzed in the debris generation analysis.

It was assumed in the debris generation analysis that Min-K did not require evaluation. Min-K is installed within the primary shield wall penetrations, around the reactor vessel supports, and on the neutron detector wells. The Min-K insulation installed on the reactor vessel top head does not become a debris source, as it would be protected from any LOCA break jet by the reactor vessel head flange. The remainder of the Min-K insulation would not become a debris source because the physical restraints would prevent a jet from damaging it (Reference 12; 13). Thus, Min-K was not analyzed in the debris generation analysis.

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3.d Latent Debris

The objective of the latent debris evaluation process is to provide a reasonable approximation of the amount and types of latent debris existing within the containment and its potential impact on sump screen head loss.

3.d.1 Provide the methodology used to estimate the quantity and composition of latent debris.

Response to 3.d.1

Walkdowns have been completed by Wolf Creek to characterize and quantify miscellaneous and latent debris, following the guidance in Section 3.5.2.2 of NEI 04-07 and the NRC SE on NEI 04-07. These walkdowns were conducted without any preconditioning or pre-inspections and the debris found during the walkdowns is considered representative of normal plant operation under the existing housekeeping programs.

The total latent debris source term from these walkdowns was determined through the collection of debris samples from multiple locations throughout the containment. Samples were collected from six surface types: floors and walls, cable trays, ductwork, major equipment, valve operators, and major piping. For each surface type, a minimum of four samples were collected by wiping with muslin cloth. Afterwards, the sample was bagged and weighed to determine the quantity of debris that was collected. A statistical approach was then used to estimate the mean debris loading on each surface by using conservatively estimated horizontal and vertical surface areas. The total latent debris mass for a surface type is the debris loading multiplied by the conservatively estimated area for that surface type, and the total latent debris is the sum of the latent debris for each surface type.

Two Wolf Creek containment walkdowns were performed to identify and quantify the miscellaneous debris in containment, such as tape, labels, stickers, construction and maintenance debris, and temporary equipment.

The composition of the latent debris is discussed in the Response to 3.d.3.

3.d.2 Provide the basis for assumptions used in the evaluation.

Response to 3.d.2

See the Response to 3.d.3 for assumptions regarding material properties of latent debris.

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3.d.3 Provide results of the latent debris evaluation, including amount of latent debris types and physical data for latent debris as requested for other debris under c. above.

Response to 3.d.3

The quantity of latent debris from the walkdowns are summarized in the debris generation analysis. The five latent debris walkdowns completed at Wolf Creek indicated a maximum of 75 lbm of latent debris. However, a value of 140 lbm is assumed in the debris generation analysis. This conservatively bounds the maximum sampled value of 75 lbm with ample operating margin.

Latent debris is assumed to consist of 15% fiber and 85% particulate by mass, per the NRC SE on NEI 04-07 (Reference 10 p. 50). Based on the NRC SE on NEI 04-07 (Reference 10, pp. 50-52, Appendix vii), the size and density of latent particulate were assumed to be 17.3 μm and 2.7 g/cm^3 (168.6 lbm/ft^3), respectively. Additionally, the bulk density and microscopic density of latent fiber were assumed to be 2.4 lbm/ft^3 and 1.5 g/cm^3 (93.6 lbm/ft^3), respectively. Latent fiber is assumed to have the same characteristic size as Nukon, 7.0 μm. This is reasonable, as Nukon is the predominant fiber type in containment at Wolf Creek (see the Response to 3.b.4). Table 3.d.3-1 summarizes the assumed latent fiber and particulate constituents and their material characteristics.

Table 3.d.3-1: Latent Fiber and Particulate Constituents

	Latent Debris (lbm)	Bulk Density (lbm/ft^3)	Microscopic Density (lbm/ft^3)	Characteristic Size (μm)
Particulate (85%)	119	-	168.6	17.3
Fiber (15%)	21 (8.8 ft^3)	2.4	93.6	7.0
Total	140			

3.d.4 Provide amount of sacrificial strainer surface area allotted to miscellaneous latent debris.

Response to 3.d.4

There was no sacrificial strainer area allotted to miscellaneous latent debris in addition to that documented in the Response to 3.b.5.

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3.e 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.

3.e.1 Describe the methodology used to analyze debris transport during the blowdown, washdown, pool-fill-up, and recirculation phases of an accident.

Response to 3.e.1

The methodology used in the transport analysis is based on the NEI 04-07 Volume 1 guidance (Reference 9 Section 3.6) and the associated NRC SE on NEI 04-07 (Reference 10 Section 3.6) for refined analyses, as well as the refined methodologies suggested by the NRC SE on NEI 04-07 (Reference 10 Appendices III, IV, and VI). The specific effect of each of four modes of transport was analyzed in the debris transport analysis for each type of debris generated. These modes of transport are:

- Blowdown Transport – the transport of debris in all directions to all areas of containment by the break jet
- Washdown Transport – the transport of debris from higher to lower portions of containment, caused by the flow from the break and containment sprays
- Pool Fill-Up Transport – the transport of debris to areas which may be active or inactive during recirculation, caused by break flow and flow from the RWST
- Recirculation Transport – the transport of debris from active regions in the recirculation pool to the sump strainer, caused by flow from the ECCS

The logic tree approach was applied for each type of debris identified in the debris generation analysis. A generic transport logic tree used for the Wolf Creek analysis is shown in Figure 3.e.1-1Figure 3.e.1-1and was developed by refining the baseline logic tree in NEI 04-07 (Reference 9 pp. 3-45 and 3-53). The refinements were added per the additional guidance of the NRC SE on NEI 04-07 (Reference 10 Sections 3.6 and 4.2.4), including considering the potential for transport of large pieces, erosion of small and large pieces, the potential for washdown debris to enter the pool after inactive areas have been filled, and the direct transport of debris to the sump strainers during pool fill-up.

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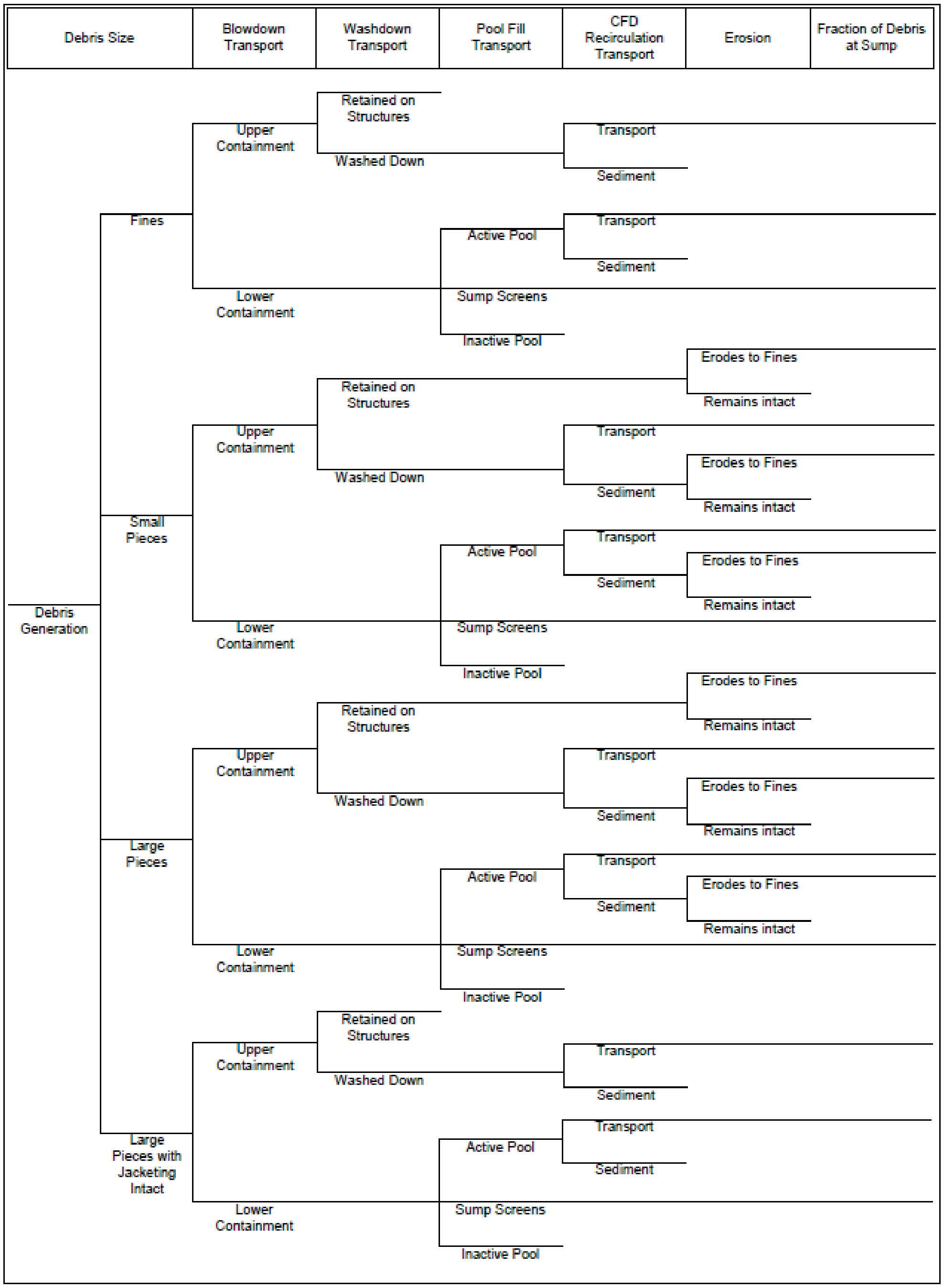

Figure 3.e.1-1: Generic Debris Transport Logic Tree

The basic methodology for the Wolf Creek transport analysis is summarized as follows:

1. The CAD model was provided as input to determine break locations and sizes.
2. The debris generation analysis was used as input for debris types and sizes.
3. Potential upstream blockage points were addressed (see Response to 3.l).

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4. The fraction of debris blown into upper containment and lower containment was determined for different break locations.
5. The fraction of debris washed down by containment spray flow was determined along with the locations where the debris would be washed. As shown in the logic trees, debris that transports to lower containment (e.g., following blowdown and/or washdown) is assumed to be in the pool.
6. During the pool fill-up phase, the quantity of debris transported to the containment sumps and inactive cavities was calculated based on the volumes of the sump or inactive cavities proportional to the pool volume at the time when the sumps or cavities are filled. Per the SE on NEI 04-07 (Reference 10 p. 59), the fraction of debris moving into the inactive cavities was limited to 15%.
7. The location of each type/size of debris at the beginning of recirculation was determined.
8. A computational fluid dynamics (CFD) model was developed to simulate the flow patterns in the pool that would develop during recirculation.
9. A graphical determination of the transport fraction of each type of debris was made using the velocity and turbulent kinetic energy (TKE) profiles from the CFD model output, along with the determined initial distribution of debris.
10. The recirculation transport fractions from the CFD analysis were gathered to input into the logic trees.
11. The quantity of debris that could experience erosion due to the break flow or spray flow was determined.
12. The overall transport fraction for each type of debris was determined by combining each of the previous steps into logic trees.

The debris transport analysis assumed that Nukon debris would not float in the sump pool based on fibrous debris testing in NUREG/CR-6808 (Reference 17). Test data has shown that fiberglass insulation sinks more readily in hotter water (Reference 18 pp. 5-30 and 5-48). This assumption is reasonable given the initial high sump temperature and is also supported by the NRC SE on NEI 04-07 (Reference 10 pp. III-54 and III-55). This approach is consistent with what McGuire used in their GL 2004-02 submittal, which was accepted by the NRC (Reference 19 p. 3 and 36 (Enclosure 1); 20; 21; 22).

<u>Potential Upstream Blockage Points</u>

The upstream effects analysis evaluated all potential upstream blockage points and determined that none of them will prevent flow from reaching the recirculation containment sumps. Upstream effects, including evaluation of potential upstream blockage points, are discussed in the Response to 3.l.

<u>Blowdown Transport</u>

The fraction of debris blown into upper containment and lower containment was determined based on the volumes of upper and lower containment. The evaluation also considered inertial capture. The SE on NEI 04-07 (Reference 10 pp. VI-6) states

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that a portion of the fine and small piece debris would realistically be trapped by inertial capture as the break flow makes sharp changes in direction. If the captured debris is at a location that is not impinged by containment sprays, the debris would remain attached to those surfaces and would not transport to the strainers (Reference 10 pp. VI-7).

The Wolf Creek debris transport analysis did not credit inertial capture for fine debris. For small pieces of fiberglass, inertial capture by miscellaneous structures, grating, and 90° flow turns was considered but was only used to determine the fraction of debris blown to upper containment; it was not used to credit retention of debris on structures that are not impinged by containment sprays. For breaks in each compartment, the number of grating and 90° turns in flow path were analyzed for each break location compartment. See the Response to 3.e.6 for the blowdown transport fractions.

Washdown Transport

When the containment spray is actuated, washdown of debris from upper containment into various areas of lower containment by the spray was assumed to be in proportion to the spray flow split. All fine debris was conservatively assumed to be washed to lower containment.

For small and large debris, holdup by gratings was credited. The debris transport analysis assumed that 40% to 50% of small pieces of fiberglass debris passes through the first layer of grating based on the Drywell Debris Transport Study (DDTS) results. For each additional layer of grating, 0% to 25% of small pieces of fiberglass debris was assumed to be held up. Washdown fractions are calculated for the annulus, inside the bioshield wall through the steam generator compartments and through the refueling canal drain. Because sprays cannot enter the pressurizer compartment, the washdown transport fraction through the pressurizer compartment is 0%.

Large pieces of debris would not pass through grating in the steam generator compartments or the annulus. Therefore, the washdown transport fraction for large debris is 0% through the steam generator compartments and the annulus.

When containment sprays are not activated, the debris transport analysis assumed that 10% of fine debris exposed to condensation flow would be washed down within a time period of 60 minutes. This is a reasonable assumption because condensation tapers off quickly following the accident as the temperatures of the walls and atmosphere reach equilibrium. Additionally, the assumed value is the upper limit determined in NUREG/CR-7172, which showed 1% as the best estimate and 10% as the upper bound estimate (Reference 18 pp. 5-41). This washdown fraction was applied to fiberglass fines, unqualified coatings, degraded qualified coating particulates outside the reactor cavity and latent debris.

The Response to 3.e.6 summarizes the washdown transport fractions.

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Pool Fill-Up Transport

During pool fill-up, debris would transport to the ECCS Sumps A and B, as well as the inactive normal sumps, instrumentation tunnel and reactor cavity. The ECCS sumps are below the containment floor elevation at 2000 ft with a 6" curb around them. The normal sumps are below the floor elevation with no curbs around them. The entrance to the instrumentation tunnel and reactor cavity is at Elevation 2001.8 ft. Therefore, during pool fill-up, debris would transport to the normal sumps first, followed by the ECCS sumps, and finally, to the instrumentation tunnel and reactor cavity.

When both trains are in operation with the sump level higher than the elevation at the entrance of instrumentation tunnel and reactor cavity (2001.8 ft), the ECCS sumps, normal sumps, instrumentation tunnel and reactor cavity would all fill. The quantity of debris transported to the ECCS sumps and inactive cavities was calculated based on their volumes proportional to the pool volume at the time when the sumps or cavities are filled. In accordance with the SE on NEI 04-07 (Reference 10 p. 59), the fraction of debris moving into the inactive cavities during pool fill-up was limited to 15%. Note that for the case of single train failure, the ECCS sump for the failed train was considered an inactive cavity. The total fraction of debris transported to the inactive cavities (including the inactive ECCS sump) was still limited to 15%.

When both trains are in operation with the pool level lower than 2001.8 ft, the transport to the inactive instrumentation tunnel and reactor cavity would be 0%. The pool fill-up transport fractions for the ECCS sumps are the same as the case shown above. For the inactive cavities, the evaluation was based on the volumes of the normal sumps only. Similarly, for the case of single train failure, the pool fill-up transport to the active ECCS sump strainer stays the same. The inactive ECCS sump was considered as an inactive cavity, along with the normal sumps, with their total pool fill-up transport fraction limited to 15%.

The Response to 3.e.6 summarizes the pool fill-up transport fractions.

Transport during pool fill-up was considered for fine debris only since fine debris is more likely to be in suspension. The curbs around the recirculation sumps would stop sunken debris (e.g., small and large pieces, and intact blankets) from washing over during the fill-up phase since only a thin sheet of water would be flowing over the top of the curb as the sumps fill and the fill-up phase has a short duration. Also, it is conservative to neglect transport of debris to inactive cavities during pool fill-up.

CFD Modeling of Containment Recirculation Pool for Recirculation Transport

To assist in the determination of recirculation transport fractions, CFD simulations were performed using Flow-3D, a commercially available software package. A diagram showing the significant features of the CFD model is shown in Figure 3.e.1-2. The ECCS sumps (labeled as "Modeled Alpha Sump Mass Sink" and "Modeled Bravo Sump Mass Sink"), the modeled break locations, and the modeled containment spray regions are highlighted.

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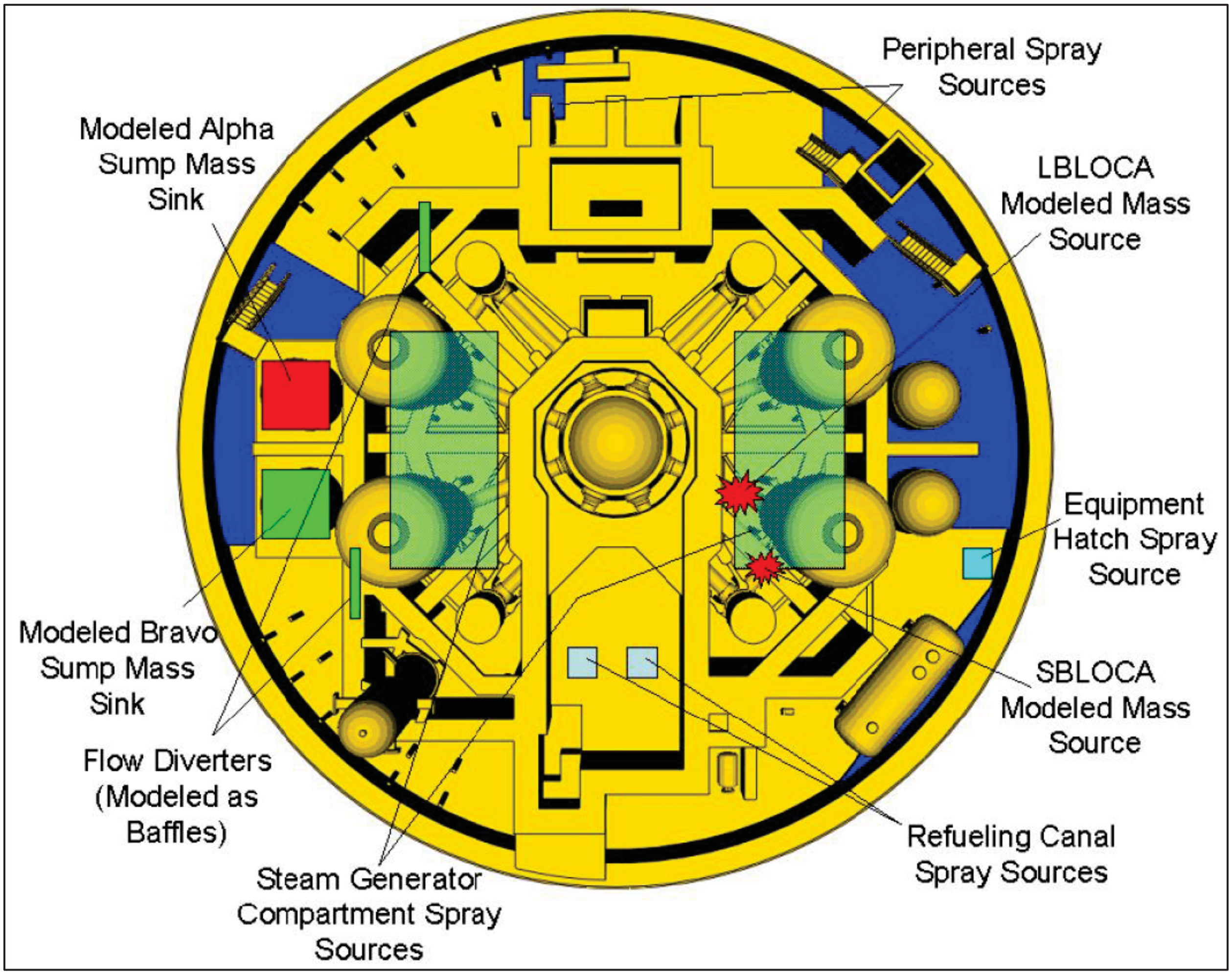

Figure 3.e.1-2: Significant Features in CFD Model

The key CFD modeling attributes/considerations included the following:

Computational Mesh

A rectangular mesh was defined in the CFD models that was fine enough to resolve important features, but not so fine that the simulation would take excessively long to run. A mesh spacing of 6 inches by 6 inches was used in the horizontal directions, and mesh spacing of 3 inches to 6 inches was used in the vertical direction.

Modeling of Containment Spray Flows

Various plan and section drawings, as well as the containment building CAD model, were used to determine the spray flow paths to the pool. Spray water would drain to the pool through many pathways. Some of these pathways include the two steam generator compartments through the open area above the steam generators, the curbed area on the operating deck (above the equipment hatch), the span of grating at the periphery of the containment wall, and the two 10-inch diameter drain lines from the refueling canal. The spray flow rate for these regions was determined from the spray flow rate and applicable floor area, and introduced near the surface of the pool.

Modeling of Break Flow

The water falling from the postulated break would introduce momentum into the containment pool that influences the flow dynamics. This break stream momentum was accounted for by introducing the break flow to the pool at the velocity a freefalling object would have if it fell the vertical distance from the location of the break to the surface of the pool.

Modeling the Emergency Sumps

The emergency sumps strainers at Wolf Creek consist of two cavities with a dividing wall between them. Both sump strainers are enclosed within a 6-inch curb. The mass sinks used to pull flow from the CFD model were defined within the sump strainer curbs. The modeled flow through the two sump strainers was defined as the combined ECCS and CSS flow.

Turbulence Modeling

There are several different turbulence modeling approaches that can be selected for a Flow-3D model. The approaches (ranging from least to most sophisticated) are:

- Prandtl mixing length
- Turbulent energy model
- Two-equation k-ε model
- Renormalized group theory (RNG) model
- Large eddy simulation model

The RNG turbulence model was determined to be the most appropriate for the CFD analysis. The RNG model has a large spectrum of length scales that would likely exist in a containment pool during emergency recirculation. The RNG approach applies statistical methods in a derivation of the averaged equations for turbulence quantities (such as turbulent kinetic energy and its dissipation rate). RNG based turbulence schemes rely less on empirical constants while setting a framework for the derivation of a range of models at different scales.

Steady-State Metrics

The CFD models were started from a stagnant state at a defined pool depth and run long enough for steady-state conditions to develop. Plots of mean kinetic energy were used to determine when steady-state conditions were reached. These steady-state solutions were used to determine the recirculation transport fractions.

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CFD Simulation Cases

For the recirculation transport fractions, CFD simulations were performed for four different LOCA cases in the debris transport analysis. Figure 3.e.1-2 shows the break locations analyzed. Note that the breaks on Loop C were chosen because these breaks are closer to the two open loop door exits and would result in higher turbulence levels in the annulus than a break farther away from these exits. Higher turbulence levels help debris stay in suspension and result in higher transport fractions. The key input parameters for these cases are shown in Table 3.e.1-1.

Table 3.e.1-1: Summary of CFD Simulations

Case	Break Location	Break Size	# of Trains Operating	CSS Sprays On/Off	Flow Rates	Water Level
1	Loop C	LBLOCA	2	On	ECCS: 4,800 gpm/sump CSS: 3,950 gpm/sump	2.09 ft
2	Loop C	LBLOCA	1A	On	ECCS: 4,800 gpm/sump CSS: 3,950 gpm/sump	2.09 ft
3	Loop C	LBLOCA	1B	On	ECCS: 4,800 gpm/sump CSS: 3,950 gpm/sump	2.09 ft
4	Loop C	SBLOCA	2	Off	ECCS: 750 gpm/sump	0.741 ft[(1)]

(1) This water level is slightly less than the minimum SBLOCA water level shown in the Response to 3.g.1. Using a smaller water depth in the CFD model would result in higher flow velocities and turbulence and is therefore conservative for a debris transport analysis.

For the LBLOCA cases in Table 3.e.1-1, operations with one or two strainers were modeled to determine recirculation transport fractions under different flow rates in the pool. Additionally, the containment spray was assumed to be on for the LBLOCA cases to increase the washdown fractions from the upper containment. The ECCS and CSS pump flow rates used for the LBLOCA cases are the maximum pump flow rates, which conservatively increase transport fractions. The minimum sump water level used for the LBLOCA cases result in higher velocities and turbulence levels in the pool and are therefore also conservative for debris transport analysis. Note that Case 4 was done for a SBLOCA but the resulting debris transport fractions were not used in any other analyses.

Debris Transport Metrics

The metrics for predicting debris transport during recirculation are the TKE necessary to keep debris suspended, and the flow velocity necessary to tumble sunken debris along the floor or lift it over a curb. Debris transport metrics have been derived or adopted from data. The metrics utilized in the Wolf Creek transport analysis originate from the sources as follows or calculated using Stokes' Law:

- NUREG/CR-6772 Table 3.1 (Reference 23)
- NUREG/CR-6808 Figure 5.2, Tables 5-1 and 5-3 (Reference 17)

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Distribution of Debris at Start of Recirculation

The distribution of debris at the start of recirculation varies based on debris size and whether the debris was initially blown to the containment floor or washed down by containment sprays.

Fine debris blown into lower containment during blowdown was assumed to be uniformly distributed within the pool. Fine debris blown into upper containment and washed down by spray flow was assumed to remain in the area where it is washed down. However, as fine debris is easily transported, a recirculation transport fraction of 100% was used for all fine debris types (see Table 3.e.6-5) regardless of the initial distribution in the pool.

Small and large pieces of insulation debris blown to the lower containment were assumed to be distributed in the vicinity of the break location at the beginning of recirculation. For breaks inside the bioshield, the debris was assumed to be distributed uniformly inside the bioshield. Because Loop A and D have flow diverters installed, the debris would then be pushed into the annulus through the Loop B and C entrances, and distributed in the annulus around the location of these doorways at the beginning of recirculation, as shown in Figure 3.e.1-3. The small and large piece debris blown to upper containment was washed down by the containment spray into different areas of the lower containment (e.g., inside the bioshield wall, annulus, and through the refueling canal (RFC) drain), as shown in Figure 3.e.1-4.

For breaks inside the annulus, the recirculation transport fraction for debris blown to lower containment was conservatively assumed to be 100%, regardless of initial distribution.

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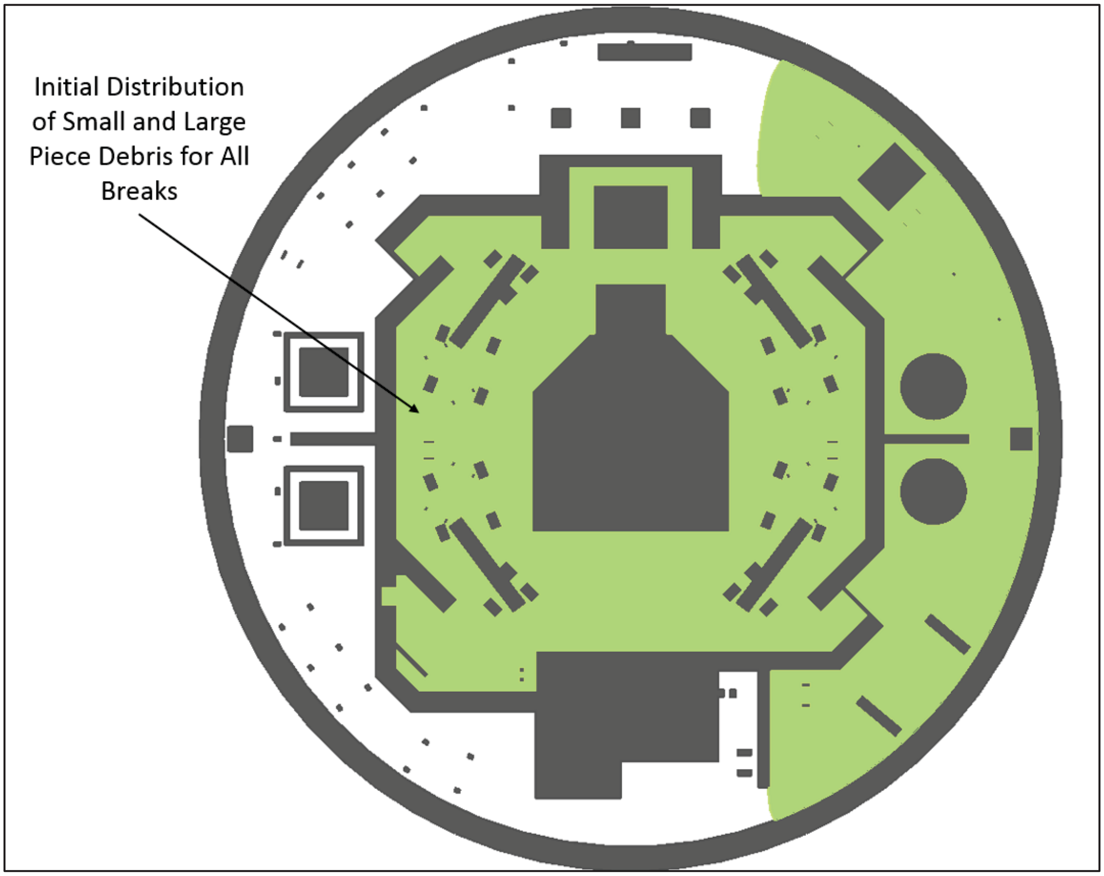

Figure 3.e.1-3: Distribution of Small and Large Debris in Lower Containment for Breaks inside Bioshield

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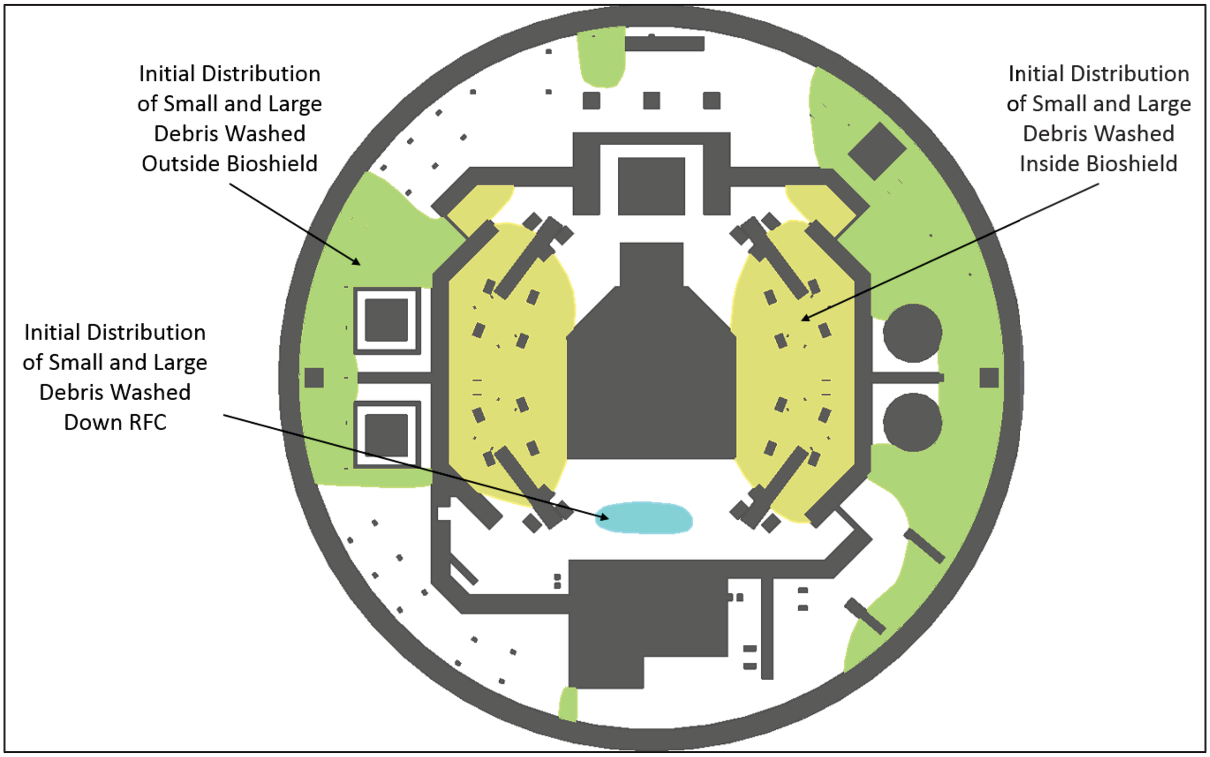

Figure 3.e.1-4: Distribution of Small and Large Debris Washed Down from Upper Containment

Graphical Determination of Debris Transport Fractions for Recirculation

The following steps were taken to determine what percentage of a particular type of debris could be expected to transport through the containment pool to the sump strainers. Detailed explanations of each bullet are provided in the following paragraphs:

- Colored contour velocity and TKE maps were generated from the Flow-3D results in the form of bitmap files indicating regions of the pool through which a particular type of debris could be expected to transport.
- The bitmap images were overlaid on the initial debris distribution plots and imported into AutoCAD with the appropriate scaling factor to convert the length scale of the color maps to feet.
- Closed polylines were drawn around the contiguous areas where velocity and TKE were high enough that debris could be carried in suspension or tumbled along the floor to the sump strainers for uniformly distributed debris.
- The areas within the closed polylines were determined using a CAD querying feature.
- The combined area within the polylines was compared to the initial debris distribution area.
- The percentage of a particular debris type that would transport to the sump strainers was determined based on the above comparison.

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The following figures and discussion are presented as an example of how the transport analysis was performed for small debris generated by a break inside the bioshield. Note that the example shows the small debris blown into the lower containment only. This same approach was also applied for the small debris washed down from the upper containment based on the initial distribution shown in Figure 3.e.1-4. Additionally, the analysis was repeated for other applicable debris types and other break cases analyzed at Wolf Creek.

For a break inside the bioshield, the distribution of small pieces of debris blown into the lower containment is shown in Figure 3.e.1-3 at the start of sump recirculation (depicted by green shading). To determine which sections of the recirculation pool each sump strainer is drawing water from, the CFD results were processed and the sump strainer areas were "seeded" to calculate where water that arrives at these seeds originates. Figure 3.e.1-5 shows the results for the LBLOCA Case 1. The green lines show where the Bravo sump strainer is drawing its water from, and the purple lines show where the Alpha sump strainer is drawing its water from. For the most part, each sump draws water from its respective half of containment. However, the Bravo sump draws a small portion of its water from the opposite side of the containment. This is due to the fact that the containment is not perfectly symmetric and the counter-clockwise flow path along the northern side of containment is less restricted compared to the other side of the containment.

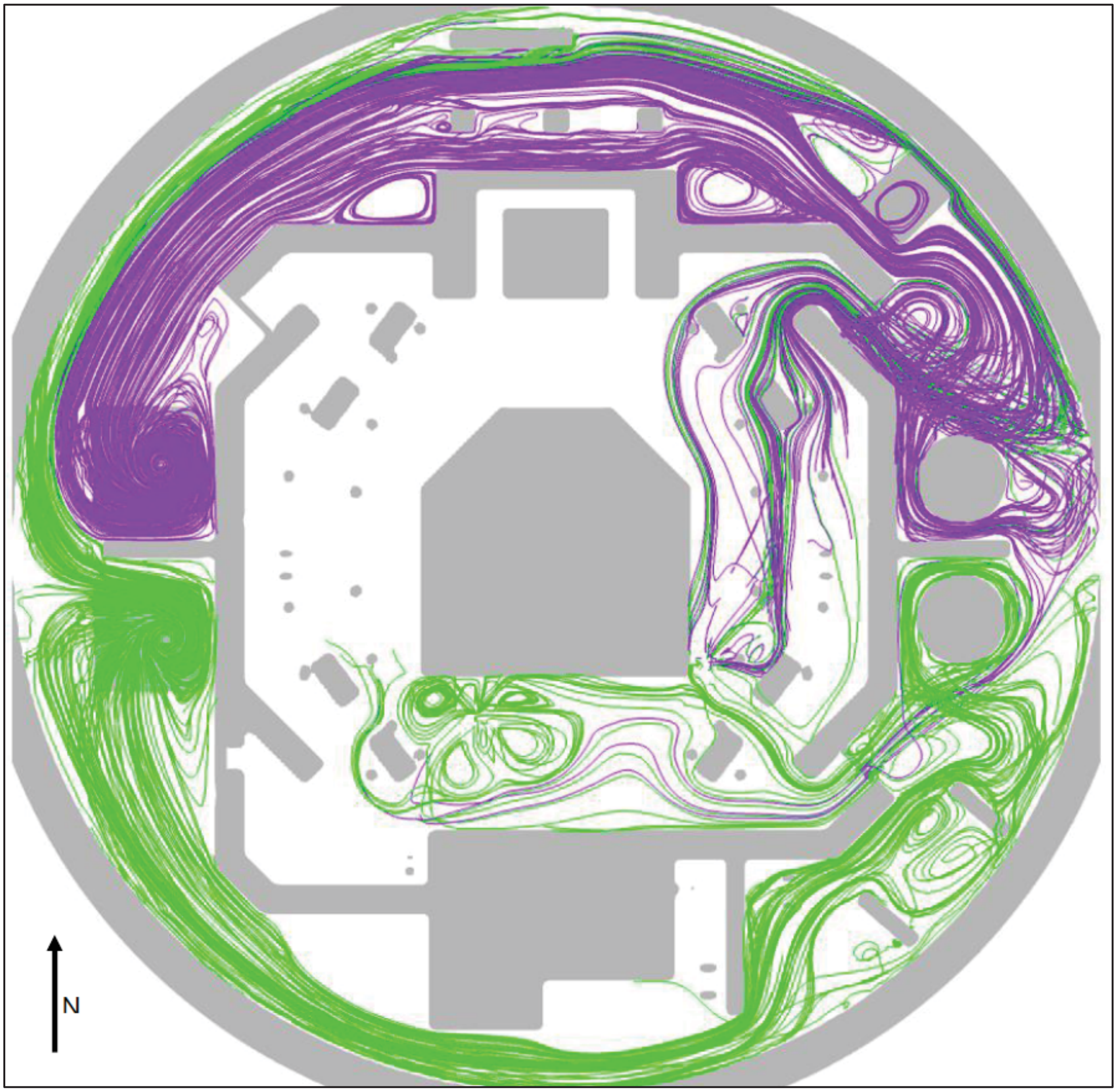

Figure 3.e.1-5: Streamlines Showing Water into Each Sump Strainer during Recirculation (Case 1)

Figure 3.e.1-5 was then overlaid onto the initial debris distribution figure (Figure 3.e.1-3) to determine the areas from which small pieces of debris blown into the lower containment could potentially be transported to each sump during recirculation. Figure 3.e.1-6 shows that small pieces of debris in the purple areas at the start of recirculation could potentially be transported to the Alpha sump and small pieces of debris in the green areas at the start of recirculation could potentially be transport to the Bravo sump.

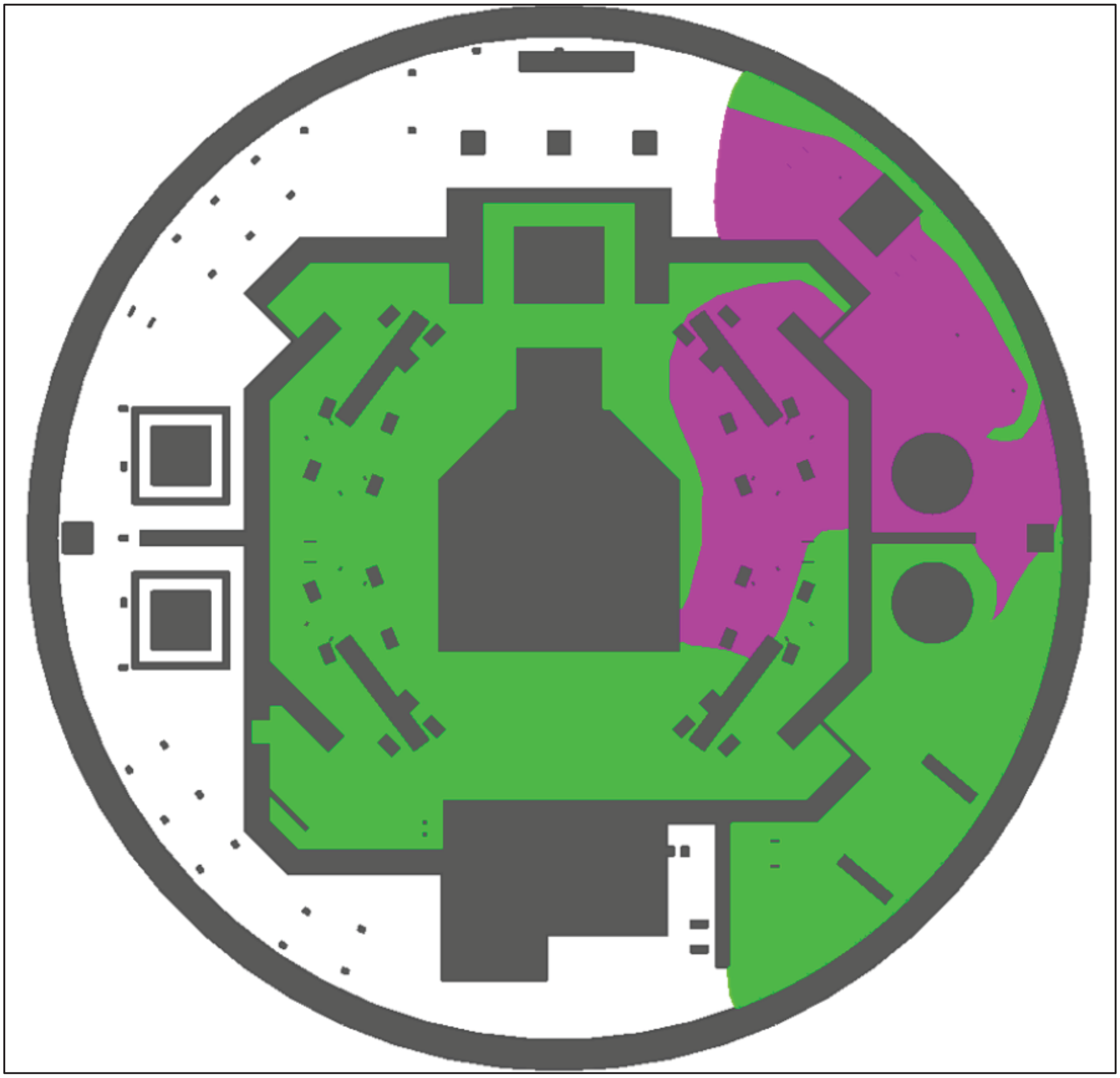

Figure 3.e.1-6: Distribution Area for Small Pieces that could be Transported to Each Sump

Figure 3.e.1-7 shows a plan view of the containment in the CFD model. The regions colored in yellow represent areas where the TKE is high enough to suspend small debris. Note that each yellow area represents a three-dimensional volume and any small debris in the volume with a continuous flow path toward a strainer (determined from the flow vectors) was assumed to transport. The regions colored in red have sufficiently high velocities 1.5 inches above the floor level to drive the sunken small debris to tumble. Small debris in red areas with flow directions towards the strainer (determined from the flow vectors) was assumed to be transportable. Note that the velocity vectors shown in Figure 3.e.1-7 do not indicate magnitude of the velocities. The remainder of the pool with low TKE and velocities is designated by the blue color.

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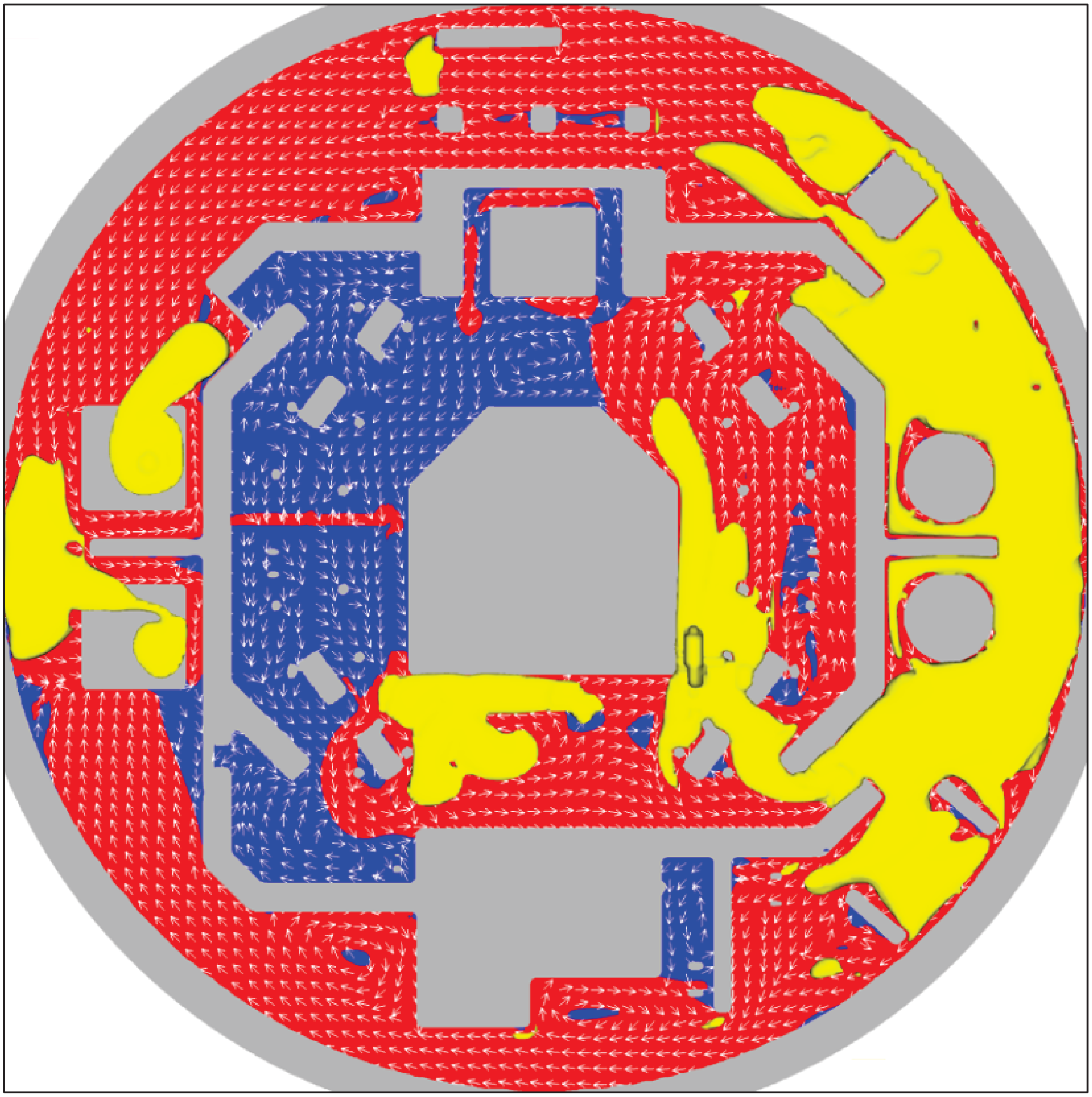

Figure 3.e.1-7: TKE and Velocity with Limits Set at Suspension/Tumbling of Small Generic Debris

The distribution area for the small pieces of debris in Figure 3.e.1-6 was then overlaid on top of the velocity/TKE map shown in Figure 3.e.1-7 to determine the recirculation transport fractions to the two sumps. Small pieces of debris located inside the hatched areas in Figure 3.e.1-8 is transportable to the sumps.

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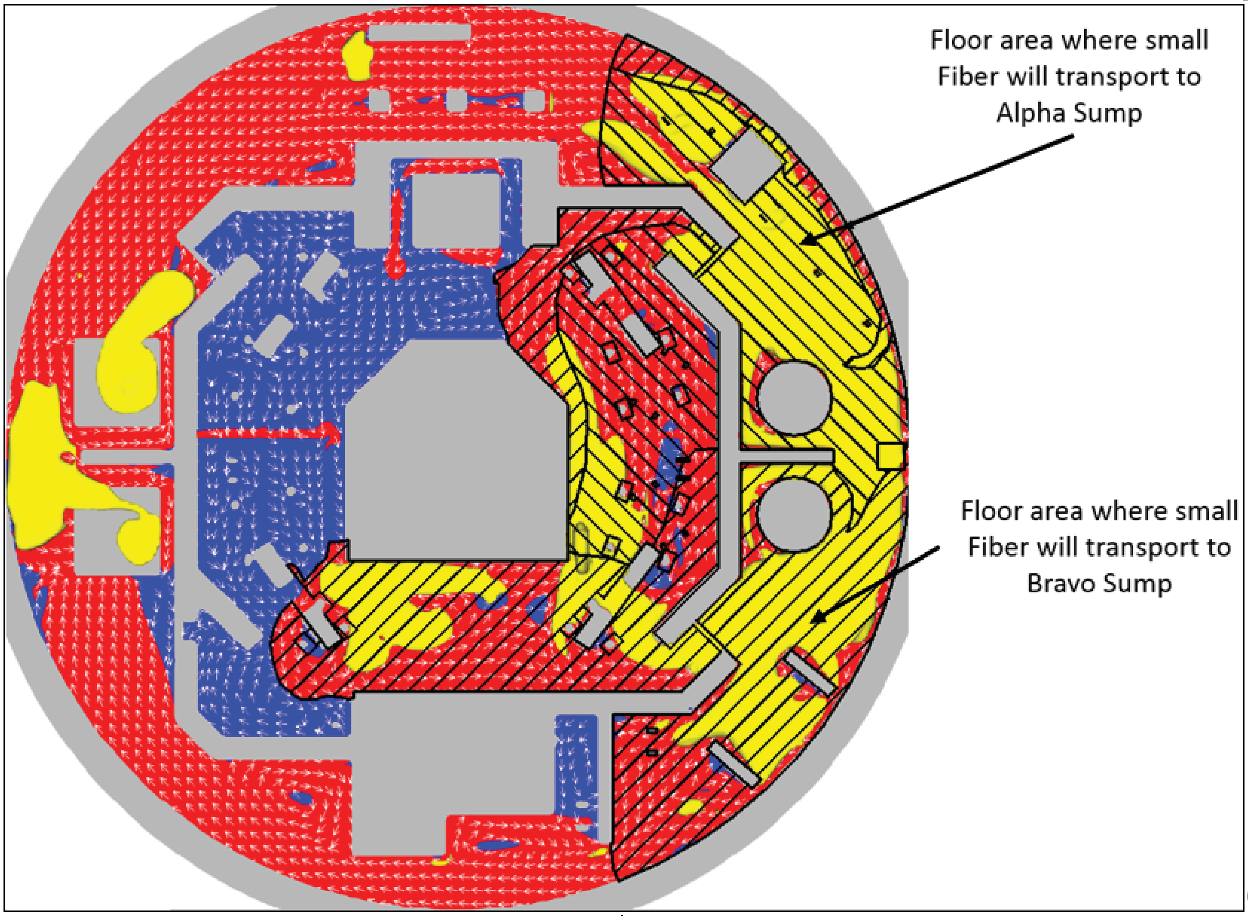

Figure 3.e.1-8: Floor Area where Small Generic Debris Would Transport to the Sump Strainers (Hatched Area)

This same analysis was applied to each type of debris and each CFD case at Wolf Creek to determine the recirculation pool transport fractions, as summarized in the Response to 3.e.6. This includes a recirculation transport fraction for debris blown to lower containment, debris washed down inside the bioshield wall, and debris washed down through the annulus.

Erosion Discussion

Due to the turbulence in the recirculation pool and the force of spray flow, small and large pieces of Nukon debris may erode into fines, which are more transportable to the strainer. Appropriate erosion fractions were applied for small and large pieces of Nukon retained in upper containment and those in the pool (including both the transportable and non-transportable debris), as summarized below.

For the small and large pieces of Nukon debris held up on gratings in the upper containment, an erosion fraction of 1% was used in the debris transport analysis to account for the effects of spray flow (Reference 10 pp. VI-29).

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For the small and large pieces of Nukon that settle in the sump pool, an erosion fraction of 10% was used in the debris transport analysis. This erosion fraction was derived based on 30-day erosion test performed by Alion, and the testing conditions were shown to be applicable for Wolf Creek. The NRC has reviewed the erosion test report and considered this 10% erosion fraction acceptable (Reference 24).

For the transportable small and large pieces of Nukon, a 10% erosion fraction was conservatively applied in the debris transport analysis for the cases with both ECCS/strainer trains in operation. Considering erosion for transported debris is conservative because as debris is carried by the recirculation flow, the relative velocity between the debris and flow stream is small and therefore little erosion is expected.

For the transportable small and large pieces of Nukon, no erosion was considered in the debris transport analysis when only one ECCS/strainer train is in operation. Note that the single train case resulted in no transport of large pieces of Nukon, as shown in Table 3.e.6-8 and Table 3.e.6-9. The debris loads of the single train case were used to determine the amount of Nukon fines and small pieces needed for head loss testing. Rather than assigning an erosion fraction for the transportable small pieces in the debris transport analysis, the transportable small pieces of Nukon were added to the test tank and were allowed to erode into fines when being exposed to the test water flow. Note that the turbulence in the test tank was higher than the containment sump pool due to the mixing introduced during testing to keep the debris in suspension.

3.e.2 Provide the technical basis for assumptions and methods used in the analysis that deviate from the approved guidance.

Response to 3.e.2

The methodology used in the transport analysis is based on the NEI 04-07 (Reference 9 Section 3.6) and the NRC SE on NEI 04-07 (Reference 10 Appendices III, IV, and VI) with one deviation: intact fiberglass blankets were assumed not to transport during blowdown or in the containment pool.

This deviation is acceptable as justified below. The intact blankets refer to large pieces of fiberglass insulation with the blanket cloth intact and are essentially the original insulation blankets that have been blown off piping or equipment. Although a high enough pool velocity could transport these intact pieces along the containment floor per the NRC SE on NEI 04-07 (Reference 10 p. 95), the likelihood is low given the size of these pieces and the potential for the jacketing to get caught on miscellaneous piping or equipment (Reference 10 pp. III-55). At Wolf Creek, Loops A and D have flow diverters installed. Therefore, in order for intact pieces of fiberglass to transport to the sump strainers, they would have to be transported out the Loop B and C entrances into the annulus, around the annulus, and over the sump strainer curbs. It is therefore reasonable to assume that the intact fiberglass blankets will not transport to the strainer, considering the long distance and the number of obstructions along this torturous flow path.

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3.e.3 Identify any computational fluid dynamics codes used to compute debris transport fractions during recirculation and summarize the methodology, modeling assumptions and results.

Response to 3.e.3

To assist in the determination of recirculation transport fractions, four CFD simulation cases were performed using Flow-3D. More detailed discussions on the CFD models and application of the CFD results can be found in the Response to 3.e.1.

3.e.4 Provide a summary of, and supporting basis for, any credit taken for debris interceptors.

Response to 3.e.4

While Wolf Creek does not have debris interceptors, perforated flow diverters are installed in the bioshield wall exits to the annulus for Loops A and D to prevent flow of debris-laden water directly into the sumps. No credit was taken for debris capture by the flow diverters, as discussed in detail below.

Figure 3.e.4-1 shows the locations of the flow diverters with the following design parameters:

1. The Loop A flow diverter is 6 ft wide by 3.67 ft high, for a total surface area of 22.0 ft^2. The Loop D flow diverter is 3 ft wide by 3.67 ft high for a total surface area of 11.0 ft^2.
2. The bottom and top of the flow diverters are at Elevations 2001.3 ft and 2005 ft, respectively. The highest calculated flood level from a LOCA is at Elevation 2004.3 ft.
3. The flow diverters are designed for a different pressure of 12 in between the reactor side and the sump side.
4. The flow diverters are made of perforated plates with 1/8-in diameter openings.

The flow diverters were not credited with capturing or retaining any debris. As shown in Figure 3.e.4-1, the most direct flow paths from the break locations inside the bioshield wall to the sump strainers would be through these diverters. Therefore, these diverters are highly likely to capture *some* debris generated by a break inside the bioshield wall. However, for conservatism, this was not credited in the recirculation transport analysis.

The flow diverters were assumed to be completely blocked in the CFD simulations. This forces the break flow to enter the annulus through the entrances on Loops B and C before reaching the ECCS sumps (see Figure 3.e.4-1). When evaluating recirculation transport fractions, the break locations used in the CFD models are on Loop C (see Figure 3.e.1-2) and are closer to the open annulus entrances on Loops B and C. This setup increases the flow velocities and turbulence levels inside the annulus and therefore, adds conservatism for debris transport analysis.

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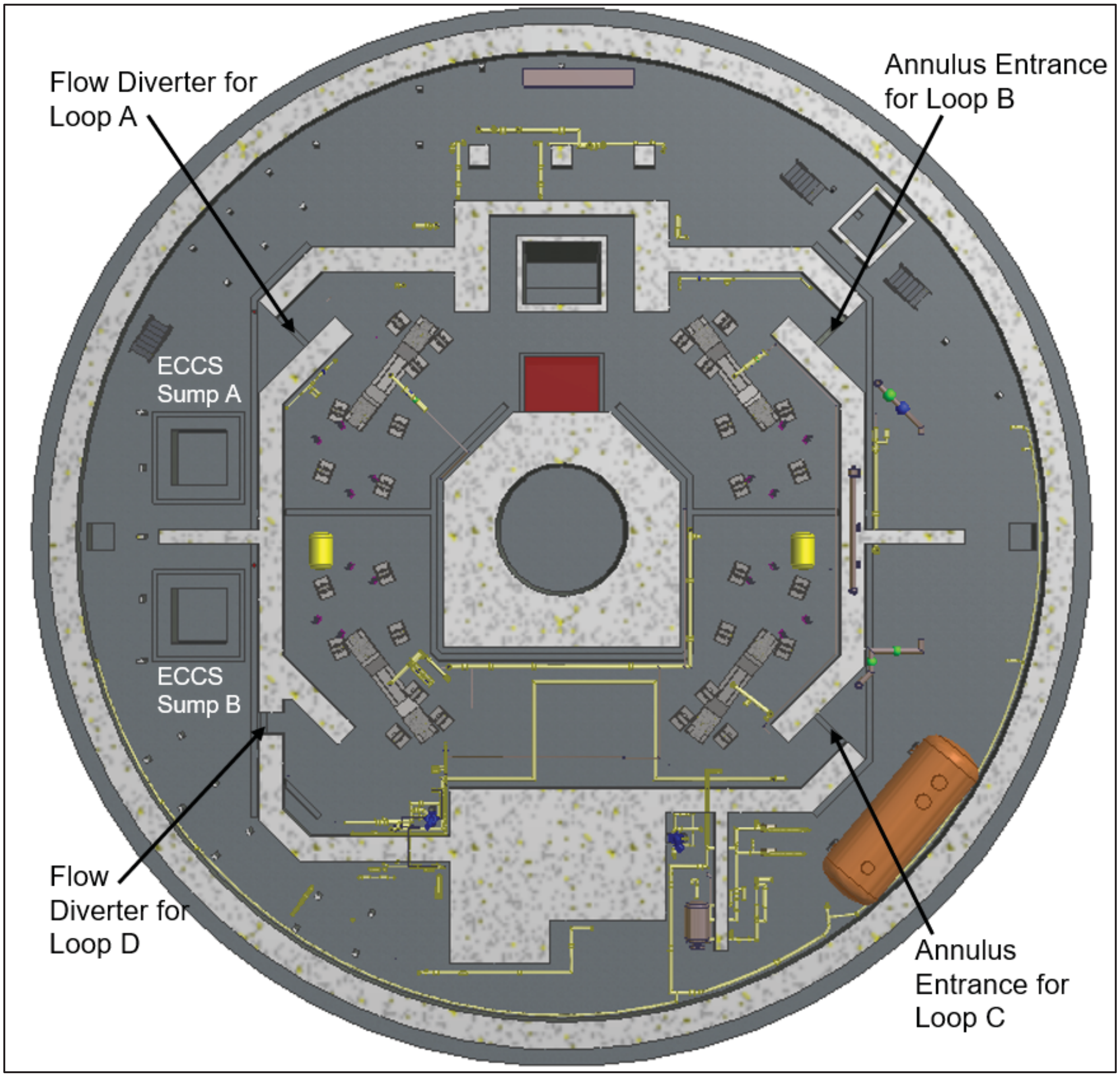

Figure 3.e.4-1: Locations of Flow Diverters and Annulus Entrances

3.e.5 State whether fine debris was assumed to settle and provide basis for any settling credited.

Response to 3.e.5

No credit was taken for the settling of fine debris, as indicated by the 100% recirculation transport fractions for fine debris in Table 3.e.6-5.

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3.e.6 Provide the calculated debris transport fractions and the total quantities of each type of debris transported to the strainers.

Response to 3.e.6

The following debris transport fractions are shown for blowdown, washdown, pool fill-up, and recirculation. Note that these fractions result in the bounding quantity of debris transported to the sump strainer. Cells with a "-" in the tables of this subsection represent values that are not applicable (i.e., debris type not generated for a specific location, debris type not available for washdown/pool fill-up, etc.).

Blowdown Transport

Table 3.e.6-1 shows the bounding (the minimum amount of debris remaining in the compartment) blowdown transport fractions as a function of break location and debris type. Note that only the limiting break locations with respect to the maximum overall debris transport fractions are listed in these tables (annulus breaks are not bounding with respect to debris generated and transported, so they are not listed in these tables).

Table 3.e.6-1: Blowdown Transport Fractions

Break Location	Debris Type	Transport Fraction		
		To Upper Containment (UC)	To Lower Containment (LC)	Remaining in Compartment
Steam Generator Compartments	Fines/Particulate (all)	80%	20%	0%
	Small Nukon	46%	54%	0%
	Large Nukon	25%	75%	0%
	Intact Nukon Blankets	0%	0%	100%
	Unqualified Coatings	0%	0%	0%
	Latent Debris	0%	0%	0%
Pressurizer Compartment	Fines/Particulate (all)	80%	20%	0%
	Small Nukon	64%	17%	19%
	Large Nukon	0%	0%	100%
	Intact Nukon Blankets	0%	0%	100%
	Unqualified Coatings	0%	0%	0%
	Latent Debris	0%	0%	0%

Washdown Transport

Table 3.e.6-2 shows the bounding washdown transport fractions (maximum amount of debris washed to lower containment) for each debris type with containment spray. Note that these transport fractions do not depend on the location of the break.

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Table 3.e.6-2: Washdown Transport Fractions with Containment Spray

Debris Type	Transport Fraction		
	Washed Down in Annulus	Washed Down Inside Bioshield Wall	Washed Down RFC Drains
Fines/Particulate (all)	52%	17%	31%
Small Nukon	36%	17%	31%
Large Nukon	0%	0%	31%
Intact Nukon Blankets	-	-	-
Unqualified Coatings	52%	17%	31%
Latent Debris	-	-	-

Pool Fill-Up Transport

Table 3.e.6-3 and Table 3.e.6-4 show the bounding (minimum) pool fill-up transport fractions as a function of debris type for two train operation and single train operation, respectively. Note that unqualified coatings are assumed to fail at the beginning of recirculation. Therefore, the pool fill-up transport fractions for unqualified coatings are 0%.

Table 3.e.6-3: Pool Fill-Up Transport Fractions – Two Train Operation

Debris Type	Water Level > 1'-10"		Water Level < 1'-10"	
	To Each Sump	To Inactive Cavities	To Each Sump	To Inactive Cavities
Fines/Particulate (all)	13%	15%	13%	4%
Small Nukon	0%	0%	0%	0%
Large Nukon	0%	0%	0%	0%
Intact Nukon Blankets	-	-	-	-
Unqualified Coatings	0%	0%	0%	0%
Latent Debris	13%	15%	13%	4%

Table 3.e.6-4: Pool Fill-Up Transport Fractions – Single Train Operation

Debris Type	Water Level > 1'-10"		Water Level < 1'-10"	
	To Active Sump	To Inactive Cavities	To Active Sump	To Inactive Cavities
Fines/Particulate (all)	13%	15%	13%	15%
Small Nukon	0%	0%	0%	0%
Large Nukon	0%	0%	0%	0%
Intact Nukon Blankets	-	-	-	-
Unqualified Coatings	0%	0%	0%	0%
Latent Debris	13%	15%	13%	15%

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Recirculation Transport

As discussed in the Response to 3.e.1, four different break cases were analyzed for the debris recirculation transport analysis:

- Case 1: LBLOCA in Loop C, 2 trains operating
- Case 2: LBLOCA in Loop C, Train 1A operating
- Case 3: LBLOCA in Loop C, Train 1B operating
- Case 4: SBLOCA in Loop C, 2 trains operating

It was assumed that, for any breaks inside the pressurizer compartment, the recirculation transport fractions for a break in Loop C could be applied. This is reasonable because the flow paths to the sump strainers for these break locations are the same. All debris enters the annulus through the openings in the bioshield wall for Loops B and C, before traveling through the annulus to the sump.

As discussed in the Response to 3.e.1, the recirculation transport fraction for debris blown to lower containment by a break in the annulus were assumed to be 100%. This is a conservative assumption because the breaks in the annulus are scattered around containment, and the velocities and turbulence in the pool for an annulus break would be lower than a break in the steam generator compartment.

The bounding (maximum) recirculation transport fractions for different debris size categories are shown in Table 3.e.6-5 through Table 3.e.6-7.

Table 3.e.6-5: Recirculation Transport Fractions for Fine/Particulate Debris

Case	Sump	Debris in Lower Containment	Washed Inside Bioshield Wall	Washed In Annulus	Washed Down RFC
Case 1	A	50%	50%	50%	50%
	B	50%	50%	50%	50%
Case 2	A	100%	100%	100%	100%
	B	-	-	-	-
Case 3	A	-	-	-	-
	B	100%	100%	100%	100%
Case 4	A	50%	NA	NA	NA
	B	50%	NA	NA	NA

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Table 3.e.6-6: Recirculation Transport Fractions for Small Nukon Debris

Case	Sump	Debris in Lower Containment	Washed Inside Bioshield Wall	Washed In Annulus	Washed Down RFC
Case 1	A	28%	35%	49%	0%
	B	45%	18%	48%	100%
Case 2	A	64%	40%	94%	89%
	B	-	-	-	-
Case 3	A	-	-	-	-
	B	63%	39%	97%	95%
Case 4	A	6%	-	-	-
	B	8%	-	-	-

Table 3.e.6-7: Recirculation Transport Fractions for Large Nukon Debris

Case	Sump	Debris in Lower Containment	Washed Inside Bioshield Wall	Washed In Annulus	Washed Down RFC
Case 1	A	23%	0%	0%	0%
	B	27%	0%	0%	0%
Case 2	A	0%	0%	0%	0%
	B	-	-	-	-
Case 3	A	-	-	-	-
	B	0%	0%	0%	0%
Case 4	A	0%	-	-	-
	B	0%	-	-	-

<u>Overall Debris Transport Fractions</u>

Transport logic trees, which include blowdown, washdown, pool fill-up, recirculation, and erosion, were developed for each size and type of debris generated. These trees were used to determine the total fraction of debris that would reach the sump strainers in each of the postulated cases. The overall transport fractions are provided in Table 3.e.6-8 and Table 3.e.6-9. The transport fractions shown in the tables for the two-train cases are conservative fractions based on transport to sump strainer Bravo, which has higher maximum transport fractions than sump strainer Alpha (both Alpha and Bravo columns under “2 Train” contain Bravo transport fractions).

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Table 3.e.6-8: Overall Transport Fractions for LBLOCA with Sprays On in the Steam Generator Compartment

Debris Type	1 Train		2 Train		
	Alpha	Bravo	Alpha	Bravo	Total
Fine/Particulate (all)[1]	97%	97%	49%	49%	98%
Small Nukon Erosion Fines	3%	3%	7%	7%	14%
Small Nukon	66%	66%	43%	43%	86%
Large Nukon Erosion Fines	8%	8%	4%	4%	8%
Large Nukon	0%	0%	19%	19%	38%
Unqualified Epoxy	100%	100%	50%	50%	100%
Unqualified IOZ	100%	100%	50%	50%	100%
Unqualified Alkyd	100%	100%	50%	50%	100%
Unqualified Carboline 4674	100%	100%	50%	50%	100%
Latent Debris	85%	85%	43%	43%	86%
Miscellaneous Debris	100%	100%	50%	50%	100%

[1] Note that the “Fine/Particulate (all)” category includes the following debris types: Nukon Fines, Antisweat Fines, Lead Blanket Fines, Fire Barrier Fines and Particulate, FOAMGLAS Particulate, Qualified Epoxy Particulate, and Qualified IOZ Particulate.

Table 3.e.6-9: Overall Transport Fractions for LBLOCA with Sprays On in the Pressurizer Compartment

Debris Type	1 Train		2 Train		
	Alpha	Bravo	Alpha	Bravo	Total
Fine/Particulate (all)[1]	97%	97%	49%	49%	98%
Small Nukon Erosion Fines	2%	2%	6%	6%	12%
Small Nukon	54%	56%	36%	36%	72%
Large Nukon Erosion Fines	1%	1%	1%	1%	2%
Large Nukon	0%	0%	0%	0%	0%
Unqualified Epoxy	100%	100%	50%	50%	100%
Unqualified IOZ	100%	100%	50%	50%	100%
Unqualified Alkyd	100%	100%	50%	50%	100%
Unqualified Carboline 4674	100%	100%	50%	50%	100%
Latent Debris	85%	85%	43%	43%	86%
Miscellaneous Debris	100%	100%	50%	50%	100%

[1] Note that Fine/Particulate (all) category includes the following debris types: Nukon Fines, Antisweat Fines, Lead Blanket Fines, Fire Barrier Fines and Particulate, FOAMGLAS Particulate, Qualified Epoxy Particulate, and Qualified IOZ Particulate.

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Transported Debris Quantities

As discussed in Attachment VII, a simplified risk-informed approach was used in this submittal and the risk quantification used a threshold break size of 10 inches, derived based on the strainer head loss and in-vessel acceptance criteria. The transported debris quantities for the four worst breaks that are bounded by the threshold break size are presented in Table 3.e.6-10. These debris loads were calculated using the overall debris transport fractions in Table 3.e.6-8 and Table 3.e.6-9 for single train operation. For conservatism, the higher of the two transport fractions to a sump strainer for each debris type was used. Note that the transported Nukon fine quantity includes fines due to erosion of small and large pieces.

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Table 3.e.6-10: Transported Debris Quantities for the Worst-Case Breaks Bounded by Threshold Break Size (Single Train Operation)

Break Location		BB01-F406 (SG 1&4)	BB-01-S105-04 (SG 1&4)	BB01-F405 (SG 1&4)	BB-01-S003-2 (PRZR)**
Break Size		10"	10"	10”	10”
Break Type		Partial @ 90º	Partial @ 0º	Partial @ 135º	Partial @ 0º
Insulation Debris					
Nukon (lbm)	Fine	79.3	77.2	73.9	28.0
	Small	127.3	140.0	123.9	44.2
	Large	0.0	0.0	0.0	0.0
	Intact	0.0	0.0	0.0	0.0
Thermo-Lag Fiber (lbm)	Fine	8.9	8.9	8.9	8.9
Thermo-Lag Particulate (ft^3)	Particulate	0.506	0.506	0.506	0.506
FOAMGLAS (ft^3)	Particulate	0.0	0.0	0.0	0.0
Coatings Debris					
Qualified Epoxy (ft^3)	Particulate	0.0	0.0	0.0	0.0
Qualified IOZ (ft^3)	Particulate	0.13	0.16	0.11	0.28
Protected Unqualified Coatings (ft^3)	Particulate	0.05	0.00	0.07	0.16
Unqualified Epoxy (ft^3)	Particulate	0.32	0.32	0.32	0.32
Unqualified Alkyd (ft^3)	Particulate	0.57	0.57	0.57	0.57
Unqualified IOZ (ft^3)	Particulate	0.36	0.36	0.36	0.36
Latent Debris					
Latent Fiber (lbm)	Fine	18.0	18.0	18.0	18.0
Latent Particulate (lbm)	Particulate	101.15	101.15	101.15	101.15

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3.f 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.

3.f.1 Provide a schematic diagram of the emergency core cooling system (ECCS) and containment spray systems (CSS).

Response to 3.f.1

A schematic for the ECCS and CSS systems is provided in Figure 3.f.1-1 and Figure 3.f.1-2, respectively.

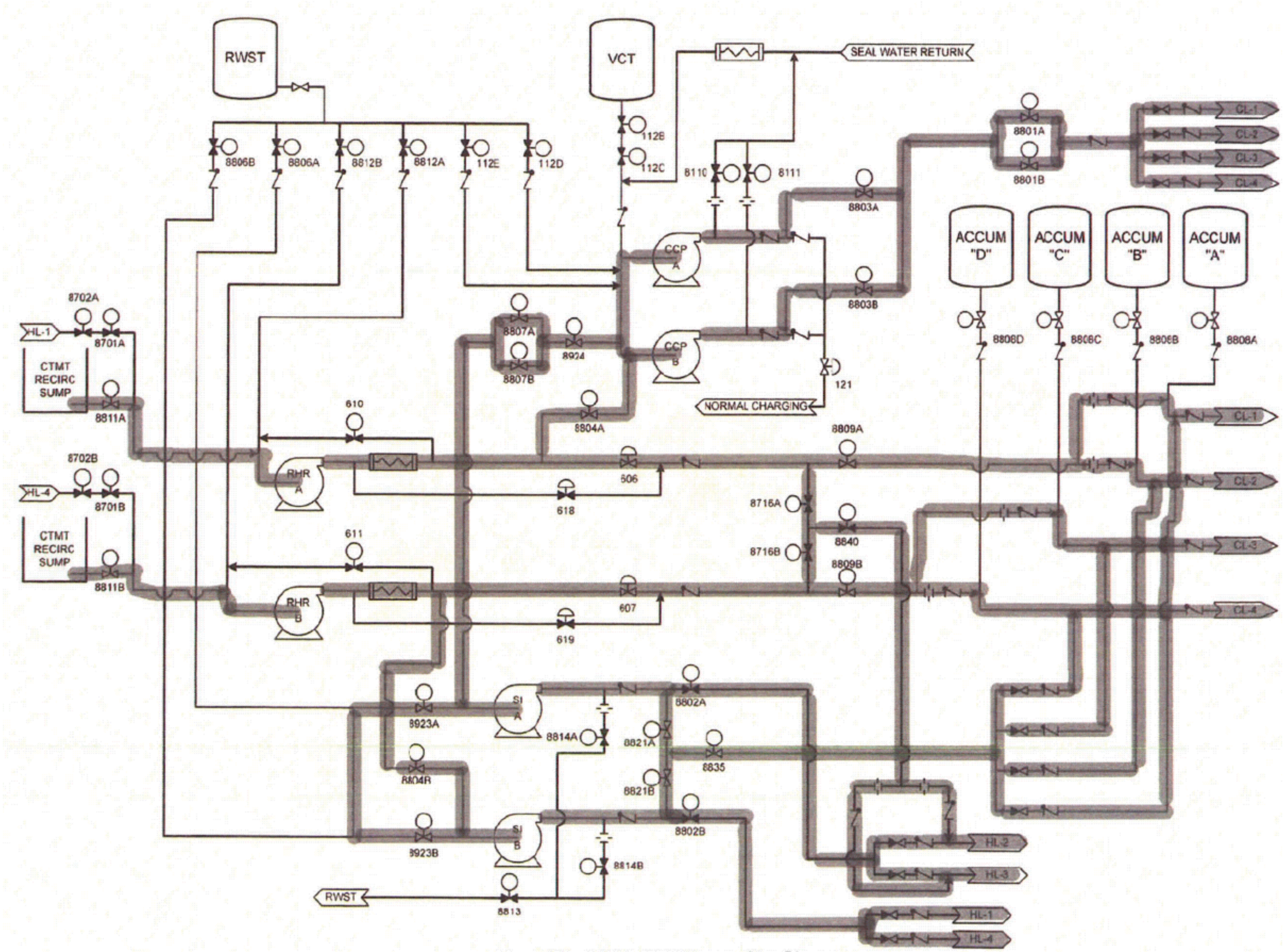

Figure 3.f.1-1: ECCS Schematics

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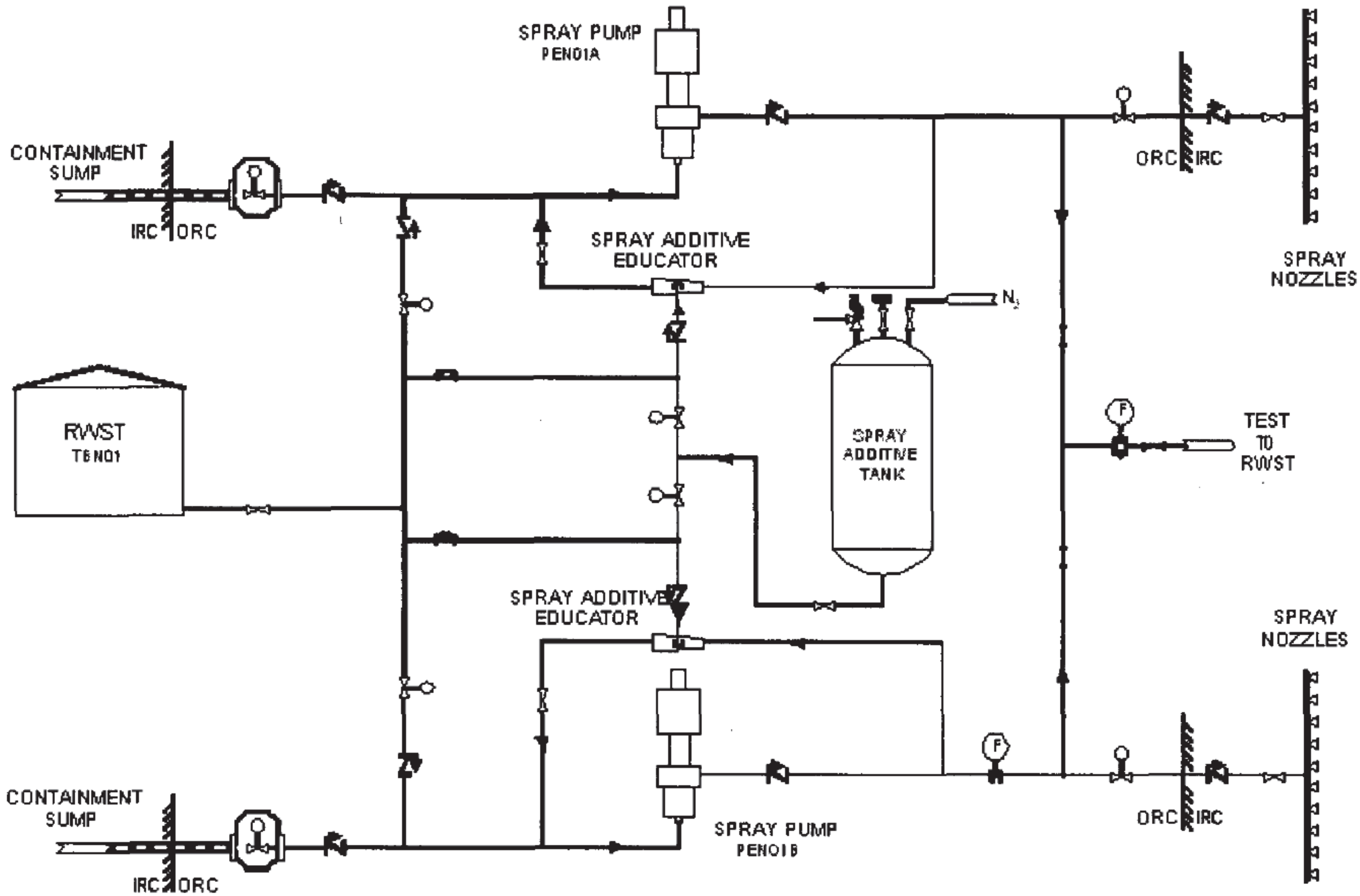

Figure 3.f.1-2: CSS Schematics

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3.f.2 Provide the minimum submergence of the strainer under small-break loss-of-coolant accident (SBLOCA) and large-break loss-of-coolant (LBLOCA) conditions.

Response to 3.f.2

The minimum sump pool water level for a Small Break LOCA (SBLOCA) is at an elevation of 2000.96 ft at ECCS sump switchover. The elevation at the top of the active strainer (i.e. the top of the highest strainer disk) is 2000.92 ft. The minimum strainer submergence is therefore 0.04 ft for SBLOCAs.

For a Large Break LOCA (LBLOCA), the minimum sump pool water level is 2002.09 ft. The minimum strainer submergence is 1.17 ft for LBLOCAs.

3.f.3 Provide a summary of the methodology, assumptions, and results of the vortexing evaluation. Provide bases for key assumptions.

Response to 3.f.3

Wolf Creek conducted prototypical testing and analysis to assess the potential for ingestion of entrained air due to vortexing near the ECCS sump strainer. The testing and analysis are summarized in this section. It was concluded that air-ingesting vortices will not form in the Wolf Creek sump strainer/pumping system.

<u>Analytical Method</u>

To assess the potential for vortexing during post-LOCA recirculation through the sump strainers, the methodology and acceptance criteria in Regulatory Guide (RG) 1.82 (Reference 25 p. Appendix A) were used, as supplemented by the discussion in the book by Jost Knauss, titled “Swirling Flow Problems at Intakes” (Reference 26). The Froude number (Fr) at the entrance of sump suction line was calculated for the plant sump configuration using a conservative combination of operating conditions. A smaller Froude number indicates a lower tendency to develop a vortex (Reference 26). The calculated Fr was then compared with the RG 1.82 acceptance criterion, which states that a Froude number of less than 0.25 is desired to ensure no air ingestion will occur and a Froude number less than 0.5 is desired to limit the air ingestion to less than 2%.

The Fr, using the definition in RG 1.82 (Reference 25), was conservatively calculated to be 0.35 as follows.

$$\mathrm{Fr}_{plant}=\frac{U}{gS}=\frac{5.21\,\frac{\mathrm{ft}}{\mathrm{s}}}{32.2\,\frac{\mathrm{ft}}{\mathrm{s}^2}\,6.96\ \mathrm{ft}}=0.35$$

In the above equation, *U* is the velocity at the suction nozzles of the RHR/CS suction lines that exit the sump. The higher velocity for the RHR suction line, based on a

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bounding RHR pump flow rate of 5,100 gpm, was used for conservatism. *S* is submergence at the centerline of the suction lines. The value used was based on the minimum sump pool water level of a SBLOCA (see the Response to 3.f.2).

Comparing with the RG 1.82 acceptance criterion, the Fr for the plant conditions is greater than the Fr needed to ensure no air ingestion (<0.25) but is smaller than the Fr needed to limit air ingestion to less than 2% (<0.5). Note that this analysis conservatively ignored the installed vortex breaker straightening vanes inside the RHR/CS suction nozzles. With a Froude number of 0.35 and vortex breaking straightening vanes installed to stop swirl in the vicinity of the pipe inlet, it can be concluded that no vortexing will occur. An additional measure of assurance was provided by comparing the tested condition to the plant condition.

<u>Vortexing Testing</u>

Vortexing was monitored for the entire duration of the two latest head loss tests. As described in the Response to 3.f.4, the testing was performed using a test strainer, test tank arrangement, debris loads and flow conditions prototypical to the Wolf Creek plant strainers. The minimum Fr of the testing arrangement was calculated as follows and compared with that of the plant condition to determine if the vortexing test results can be directly applied to the vortexing evaluation of the plant strainer.

$$\mathrm{Fr}_{test} = \frac{U_{test}}{\sqrt{g\, S_{test}}} = \frac{6.86\,\frac{\mathrm{ft}}{\mathrm{s}}}{\sqrt{32.2\,\frac{\mathrm{ft}}{\mathrm{s}^2}\; 88.3125\ \mathrm{in}\frac{ft}{12\ in}}} = 0.45$$

The minimum Fr for testing conditions was calculated with the minimum test flow rate. During the introduction of chemical debris, the test flow rate was maintained within -0/+5% of 544.0 gpm (between 544.0 and 571.2 gpm). Therefore, to calculated Fr for this condition, a velocity (6.86 ft/s) based on a test flow rate of 544 gpm was used at the entrance of the two 4-in suction pipes that exit the test plenum. The test plenum sits below the test strainer stacks (see Figure 3.f.4-1). The submergence, S_{test}, was taken as the elevation difference between the minimum testing water level and the mid-point of the suction plenum.

Since the test condition's minimum Froude number (0.45) is greater than the anticipated limiting plant condition's Froude number (0.35), the conditions in the test were more conducive to developing a vortex. As no air-ingesting vortices occurred in the testing, no air-ingesting vortices are predicted to occur for the plant strainer.

Note that, during the TB head loss test, a surface vortex was observed after adding the 6th batch of fiber debris. The vortex was on the surface only and did not entrain air into the strainers or pumps. The vortex was attributed to the more confined arrangement of the test strainer with respect to the tank wall as compared to the plant sump conditions. The plant strainer has 4-inch gaps of separation between adjacent strainer stacks, and between the exterior strainer stacks and sump pit walls. During head loss testing, the distance between test strainer and surrounding test tank walls was set 2 inches to model the symmetry boundary between adjacent strainer stacks at the plant. The acceleration of flow into these smaller than prototypical gaps (especially at the corners) likely caused the observed flow rotation. The confinement was necessary to ensure bridging between adjacent strainer stacks was correctly represented in the test. As the testing conditions were more conducive to vortexing than plant conditions and no air-entraining vortices were observed during testing, air entrainment due to vortexing will not be an issue at the plant.

3.f.4 Provide a summary of the methodology, assumptions, and results of prototypical head loss testing for the strainer, including chemical effects. Provide bases for key assumptions.

Response to 3.f.4

Wolf Creek performed head loss tests in 2016 to measure the head losses caused by conventional debris (fiber and particulate) and chemical precipitate debris generated and transported to the sump strainers following a LOCA. The test program used a test strainer, test tank arrangement, debris quantities, and flow conditions that were prototypical to Wolf Creek. Two different tests were performed, with one test according to the thin bed (TB) protocol and the other test according to the full debris load (FDL) protocol, following the 2008 NRC Staff Review Guidance (Reference 27). The results of the head loss tests were used to determine the threshold break size for which all breaks of that size or smaller pass the head loss acceptance criteria based on the RHR/CS pump NPSH margin, strainer structural limit and air entrainment due to degasification, flashing and vortexing. The head loss test is described in the following subsections.

<u>Test Loop</u>

The closed test loop included a metal test tank that housed a test strainer submerged in water. During head loss testing, test water was circulated by a pump through the test strainer and various piping components. The test tank consisted of two regions: a rectangular upstream portion for debris introduction and mixing, and a pit region where the test strainer stacks were installed, as shown in Figure 3.f.4-1, where the arrow represents the flow direction.

Debris was introduced at the upstream end of the test tank, away from the test strainer. The upstream debris introduction and transport region of the tank was equipped with hydraulic mixing lines to create adequate turbulence to prevent debris from settling before reaching the strainer. The turbulence level was controlled to keep debris in suspension without disturbing the debris bed on the test strainer.

For the pit region of the test tank, the horizontal gap along the perimeter of the pit, between the tank walls and exterior sides of the strainer stacks, was set at 2 in, which is half the full gap width of 4 in between adjacent strainer stacks and between the strainers and sump pit walls at the plant. The 2 in gap used during head loss testing modeled the symmetry boundaries between adjacent plant strainer stacks and between the strainers and surrounding sump pit walls.

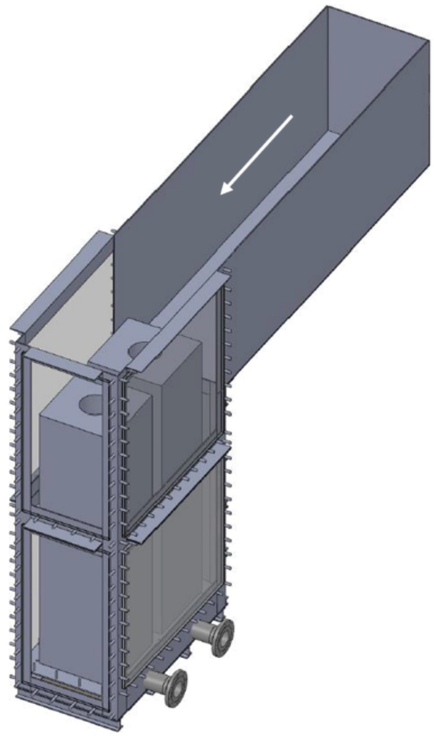

Figure 3.f.4-1: Head Loss Test Tank

The schematics of the test loop are shown in Figure 3.f.4-2. Water traveled through the test strainer, bypassing the filter housings and to the pump, after which a flow meter was in place to measure the total test loop flow rate. A pressure gage was set up to measure the differential pressure across the clean or debris-laden strainer.

Downstream of the pump, valves were configured to control flow split between flow paths to the mixing lines, through a debris hopper and to a transition tank. As discussed above, flow exiting the mixing lines enhanced turbulence in the upstream portion of the test tank to keep debris in suspension. Most types of debris were added to the debris hopper and the metered recirculation flow through the hopper gravity-fed the debris into the mixing region of the test tank. The rest of the flow passed through a three-way valve (TWV), which diverted flow either directly to the mixing region of the test tank or through a transition tank. The transition tank increased the effective volume of the test loop and decreased the amount of draining required. The transition tank was equipped with a separate pump which can recirculate the tank contents back to the mixing region of the test tank.

A heat exchanger loop was used to control test water temperature. A thermocouple was used during each test to continuously measure the test water temperature.

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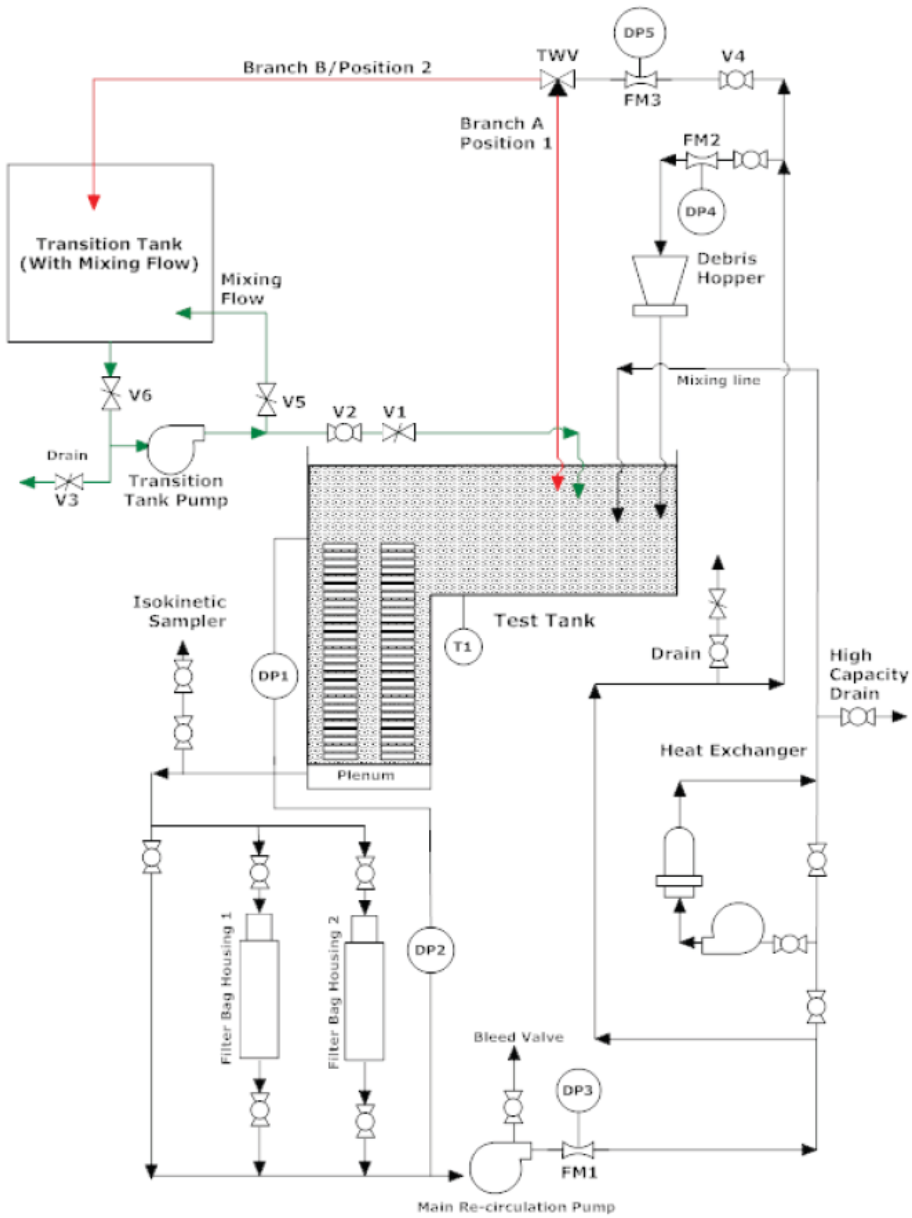

Figure 3.f.4-2: Head loss test piping and instrumentation diagram

<u>Test Strainer</u>

The test strainer for head loss testing consisted of two prototypical strainer stacks that matched the key design parameters of the plant strainer stacks. The gap width between the two stacks was maintained at approximately 4 in, consistent with the plant strainer design. Each test strainer stack was made up of four strainer modules, where the top three modules each had 11 strainer disks and the bottom module had 7 strainer disks. This resulted in a total of 40 disks per test strainer stack and 80 disks total. Within each module, adjacent strainer disks were separated by a partially perforated gap ring located around the central core tube, which runs vertically through the strainer disks. Along the circumference of the core tube, there were four evenly-spaced rectangular openings at the elevation of every strainer disk. The openings on the core tube gradually increased in size from the bottom to the top of the core tube to achieve a uniform flow distribution on the strainer stack. This arrangement is identical to the plant strainer stacks.

The two test strainer stacks were mounted on top of a single plenum, similar to the plant strainer configuration. The height of the test plenum was reduced, compared to the plant strainer plenum, to minimize debris settling inside the plenum. The test plenum had two 4-in exiting suction pipes, which were joined in a downstream manifold. All piping and components between the plenum exits and the manifold were designed to achieve similar losses and to promote equal flow between the two suction pipes.

<u>Debris Types and Preparation</u>

Both conventional (particulate and fibrous) and chemical debris were used in head loss testing.

Nukon was used to represent all types of fiber debris generated at Wolf Creek, which includes LDFG, latent fiber, and Thermo-lag fiber. This is appropriate because all of these fiber types have similar characteristics as Nukon.

Nukon was prepared as fines and small pieces according to the NEI protocol (Reference 28). Preparation of Nukon fines was performed as follows: Nukon insulation sheets, with an overall thickness of 2 in, were baked single-sided by the manufacturer until the binder burnout reached into approximately half the thickness. The heat-treated sheets were then cut into approximately 2 in x 2 in cubes and weighed out according to batch size. Each batch of Nukon was then pressure washed with test water following the NEI protocol to create a debris slurry consisting primarily of Class 2 fine fibers, as defined in NUREG/CR-6224 (Reference 29 pp. B-16). The duration of pressure wash was controlled between batches to ensure consistency. For each batch of prepared debris, a sample was photographed inside a clear column placed on a light table. Figure 3.f.4-3 shows the Nukon fiber pieces before and after the pressure washing.

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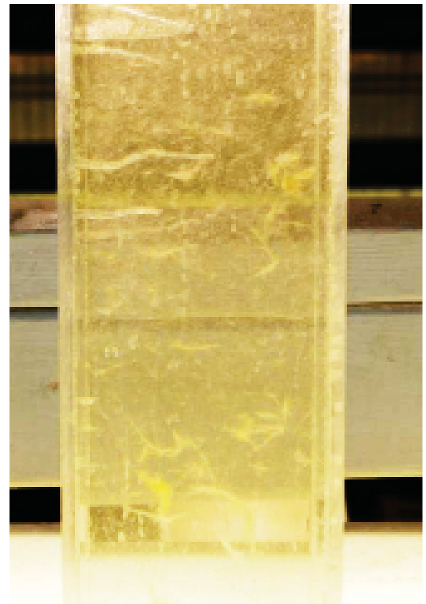

Figure 3.f.4-3: Fine Nukon Fiber Preparation for Head Loss Testing

Preparation of Nukon small pieces was similar to that of Nukon fines except for the size of the pieces that the Nukon sheets were cut into and the duration of pressure washing applied. To prepare small pieces, Nukon sheets were cut into pieces of various sizes: 2 in x 2 in, 2 in x 4 in, 1 in x 4 in, and 1 in x 6 in. Each batch of Nukon small pieces consisted of two equal parts (by weight): one consisting of 2 in x 2 in and 2 in x 4 in pieces and the other consisting of 1 in x 4 in and 1 in x 6 in pieces. The two parts were mixed prior to being wetted and pressure washed. Figure 3.f.4-4 shows a batch of prepared Nukon small pieces photographed on a light table.

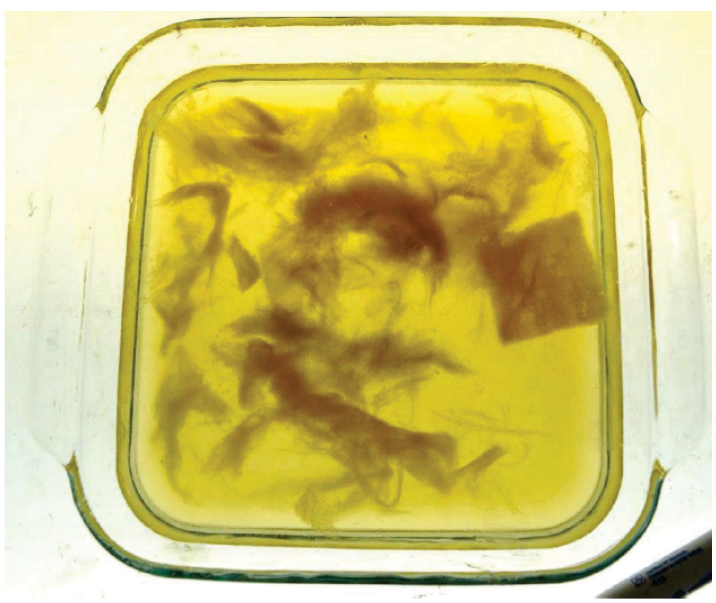

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Figure 3.f.4-4: Prepared Nukon Small Pieces for Head Loss Testing

Particulate debris used in head loss testing included pulverized acrylic, silica flour, paint chips, and PCI Dirt/Dust mix. Pulverized acrylic (with a density of 185 lbm/ft^3) was used to model qualified and unqualified coatings particulate debris on an equal volume basis. Silica flour (with a density of 164.6 lbm/ft^3) was used to model Thermolag and FOAMGLAS particulates, which are both silica-based, on an equal volume basis. Pulverized acrylic and silica flour used in head loss testing had a mean size distribution of 10 microns. Paint chips (with a density of 171 lbm/ft^3) were used to model chips formed by degraded qualified coatings on an equal volume basis, and PCI Dirt/Dust mix was used to model latent particulate.

For the FDL test, the weighed out pulverized acrylic, silica flour, and paint chips of a given batch were mixed with prepared fiber debris prior to introduction. For the TB test, pulverized acrylic and silica flour were wetted separately and diluted with test water before being introduced. No preparation was required for the PCI Dirt/Dust mix which was added to the test tank in its dry form.

Each batch of paint chips were prepared separately using a food processor or blender. The weighed out chips for a given batch were first submerged with room-temperature test water inside the food processor or blender. The food processor/blender was then operated until the desired size distribution was achieved. The prepared paint chips were analyzed after the testing, and the size of the prepared chips ranged between 0.004 to 0.008 in.

Aluminum oxyhydroxide (AlOOH) is the only chemical precipitate debris at Wolf Creek and was used in head loss testing. AlOOH was prepared in accordance with WCAP-16530-NP-A (Reference 30). All prepared AlOOH batches met the acceptance criteria in WCAP-16530-NP. Note that the solution was continuously mixed during preparation and the mixing continued afterwards until the solution was added to the test tank.

<u>Conventional Debris Introduction and Transport</u>

Conventional debris batches containing fine fiber were added to the tank through the debris hopper. Conventional debris batches containing small pieces of Nukon were mixed in a 2-gallon bucket and added directly to the mixing region of the test tank. Introduction of debris batches containing small pieces of Nukon did not start until all batches of fine fiber have been added to the test tank (see Figure 3.f.4-5). Dirt/Dust of each batch was sprinkled directly into the test tank immediately upstream of the test strainer. This was done to prevent the Dirt/Dust from forming large agglomerations with fiber which could clog the debris hopper.

For the TB test, pulverized acrylic and silica flour were first added to the test tank through the debris hopper, followed by the Dirt/Dust sprinkled directly into the test tank. After completion of particulate debris additions, batches of prepared Nukon fines were added through the hopper (see Figure 3.f.4-7). Each batch of fiber debris has a theoretical uniform bed thickness of 1/16-inch.

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No credit was taken for debris settling in head loss testing, and any non-transported debris was captured and either re-introduced to the test tank or quantified post testing and subtracted from the quantity of debris added to the test.

During conventional debris addition, test tank water level was maintained within the acceptable limits by diverting a portion of the returning flow to the transition tank (instead of back to the test tank) via the three-way valve. Before the head loss was allowed to stabilize, the transition tank contents were mixed adequately before being circulated through the main test tank for at least 1 turnover of the transition tank volume. The recirculation continued as necessary until a stable head loss was established.

Chemical Debris Introduction

Chemical debris addition did not start until the full loads of conventional debris had been added and the head loss stabilized. Chemical debris addition was done in a similar manner for the FDL and TB tests. Each batch of prepared AlOOH solution was pumped from the mixing tank into the upstream end of the test tank. Similar to the conventional debris addition, the transition tank was used to provide additional volume for the test loop and the transition tank contents were recirculated back through the main test tank for at least one turnover of the transition tank volume as the head loss stabilized.

Test Parameters

Both head loss tests were conducted under the same test conditions except for variations in debris batching and quantities. The test flow rate was maintained within -0/+5% of 961.2 gpm during conventional debris additions which was scaled from a bounding strainer flow rate of 9100 gpm, which bounds the strainer flow rate with one RHR pump and one CS pump drawing water from the sump (see the Response to 3.g.1). The scaling used the ratio between test strainer surface area (348.2 ft^2) and the net plant strainer surface area for one train (3,296.5 ft^2). This net plant strainer surface area was determined by subtracting 15 ft^2 of strainer sacrificial area (see the Response to 3.b.5) from the strainer surface area of 3311.5 ft^2 to account for blockage by miscellaneous debris (e.g., tags and labels) transported to the strainer.

For chemical debris additions, the test flow rate was maintained within -0/+5% of 544.0 gpm, which was scaled using the same strainer surface areas as above but from a strainer flow rate of 5150 gpm. This strainer flow rate was determined by rounding up the maximum RHR flow rate of 4,760 gpm since chemical precipitate does not occur at Wolf Creek prior to 24 hours following an accident when containment spray has been secured.

The nominal test water temperature used for both head loss tests was 120°F for conventional debris introduction and less than 100°F for chemical debris introduction.

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Test water was prepared by first adding boric acid to deionized water to reach a boron concentration of 2,406 ppm. Afterwards, NaOH was added in small batches to achieve a pH of 8.0.

Prior to each conventional and chemical debris addition, the pH of the water inside the test tank was measured. If the measured pH was higher than 8.2, boric acid was added to the test tank to lower the pH to below 8.2. Such pH control measures were implemented during both the FDL and TB tests.

Strainer submergence was maintained between 14 in and 18.1 in from the top perforated plate of the test strainer. This range of submergence was consistent with the plant strainer submergence for a LBLOCA.

<u>Debris Quantities and Composition</u>

Test debris quantities were determined based on plant debris loads and the ratio of the test strainer area (348.2 ft^2) to the net surface area of one plant strainer (3,296.5 ft^2). For the FDL test, twenty conventional debris batches were added (see Figure 3.f.4-5), of which the first eleven included particulates mixed with fine fiber and the last nine included particulates mixed with small fiber. Table 3.f.4-1 shows the test-scale conventional debris batches used for the FDL test, and Table 3.f.4-2 shows the plant-scale total conventional debris loads used for the FDL test. For pulverized acrylic, silica flour and paint chips, the conversion from mass to volume was done using the densities presented earlier in this section.

Table 3.f.4-1: FDL Test Conventional Debris Batches at Test Scale

Batch	Fiber Fines (g)	Fiber Smalls (g)	Dirt and Dust (g)	Pulverized Acrylic (g)	Silica Flour (g)	Paint Chips (g)
Fiber Fines Batches						
AF1	936.90	0.00	1304.9	5495.4	1971.3	436.4
AF2	936.90	0.00	1304.6	5495.4	1971.2	436.4
CF1	553.22	0.00	240.3	827.3	0.0	0.0
CF2	553.23	0.00	240.6	827.6	0.0	0.0
CF3	553.21	0.00	240.4	827.6	0.0	0.0
CF4	553.21	0.00	240.8	827.8	0.0	0.0
CF5	553.24	0.00	240.7	827.7	0.0	0.0
CF6	553.20	0.00	240.1	827.3	0.0	0.0
CF7	553.24	0.00	240.5	827.8	0.0	0.0
CF8	553.23	0.00	240.0	827.8	0.0	0.0
CF9	553.22	0.00	240.3	827.7	0.0	0.0
Fiber Smalls Batches						
CS1	0.00	1077.31	240.1	827.8	0.00	0.0
CS2	0.00	1077.28	240.2	827.4	0.00	0.0
CS3	0.00	1077.33	240.0	827.5	0.00	0.0

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Batch	Fiber Fines (g)	Fiber Smalls (g)	Dirt and Dust (g)	Pulverized Acrylic (g)	Silica Flour (g)	Paint Chips (g)
CS4	0.00	1077.34	240.4	413.69	0.00	0.0
CS5	0.00	1077.33	240.2	413.63	0.00	0.0
CS6	0.00	1077.32	120.1	0.00	0.00	0.0
CS7	0.00	1077.27	0.00	0.00	0.00	0.0
CS8/ 8A	0.00	1077.31	0.00	0.00	0.00	0.0
CS9A	0.00	37.51	0.00	0.00	0.00	0.0

Table 3.f.4-2: FDL Test Total Conventional Debris Quantities at Plant Scale

Fiber Fines (lbm)	Fiber Smalls (lbm)	Dirt/Dust (lbm)	Pulverized Acrylic (ft^3)	Silica Flour (ft^3)	Paint Chips (ft^3)
141.78	180.67	122.18	2.43	0.50	0.11

During the FDL test, seven batches of prepared AlOOH solution were introduced after the addition of conventional debris was completed and the head loss was stabilized (see Figure 3.f.4-6). The chemical solution batch sizes are shown in Table 3.f.4-3. The total amount of AlOOH added to the test tank was 15.53 kg at test scale or 147.1 kg at the plant scale.

Table 3.f.4-3: FDL Test Chemical Debris Solution Batches

Batch	Volume of AlOOH Solution (gallons)
A1	200
A2	300
A3	60
A4	400
A5	400
A6	300
A7	175

For the TB test, particulate debris was first added to the test tank in the order of pulverized acrylic, silica flour, paint chips, and Dirt/Dust. Afterwards, six batches of Nukon fines, each just under 1/16-in theoretical uniform bed thickness, were added to the test tank (see Figure 3.f.4-7). Table 3.f.4-4 shows the test-scale conventional debris batches used for the TB test, and Table 3.f.4-5 shows the plant-scale total conventional debris loads used for the TB test.

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Table 3.f.4-4: TB Test Conventional Debris Batches at Test Scale

Batch	Fiber Fines (g)	Fiber Smalls (g)	Dirt and Dust (g)	Pulverized Acrylic (g)	Silica Flour (g)	Paint Chips (g)
PA1	0.0	0.0	5859.1	24691.2	6197.1	872.7
F1	1800.1	0.0	0.0	0.0	0.0	0.0
F2	1800.4	0.0	0.0	0.0	0.0	0.0
F3	1800.9	0.0	0.0	0.0	0.0	0.0
F4	1800.6	0.0	0.0	0.0	0.0	0.0
F5	1800.6	0.0	0.0	0.0	0.0	0.0
F6	1800.2	0.0	0.0	0.0	0.0	0.0

Table 3.f.4-5: TB Test Total Conventional Debris Quantities at Plant Scale

Fiber Fines (lbm)	**Fiber Smalls (lbm)**	**Dirt/Dust (lbm)**	**Pulverized Acrylic (ft^3)**	**Silica Flour (ft^3)**	**Paint Chips (ft^3)**
225.57	0	122.35	2.79	0.79	0.11

During the TB test, five batches of prepared AlOOH solution were introduced after the addition of conventional debris was completed and the head loss was stabilized (see Figure 3.f.4-8). The chemical solution batch sizes are shown in Table 3.f.4-6. The total amount of AlOOH added to the test tank was 15.53 kg at test scale or 147.1 kg at the plant scale.

Table 3.f.4-6: TB Test Chemical Debris Batches

Batch	Volume of AlOOH Solution (gallons)
A1	200
A2	400
A3	430
A4	420
A5	380

<u>Head Loss Measurements and Bed Characterization</u>

Pressure differential across the test strainer, flow rate and test water temperature were measured and recorded continuously throughout the entire duration of both tests. Figure 3.f.4-5 and Figure 3.f.4-6 show plots of the raw head loss across the test strainer, the flow rate through the strainer, and the test fluid temperature as a function of time for the conventional debris addition and chemical debris addition phases of the FDL test, respectively. Figure 3.f.4-7 and Figure 3.f.4-8 show plots of the same information for the TB test.

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After adding all conventional debris batches, the head loss was allowed to stabilize. Afterwards, at least one flow sweep was performed by measuring strainer head losses at several different flow rates. Similarly, flow sweeps were also performed at the end of each test, after adding the full load of chemical debris and allowing the head loss to stabilize (see Figure 3.f.4-6 and Figure 3.f.4-8). Such flow sweep data was later used to characterize flow regime through the debris bed and to adjust the measured debris head losses from test conditions to plant conditions (see the Response to 3.f.10).

During each test, flow sweeps were also performed prior to debris addition when the strainer was clean. This flow sweep data was used to characterize flow regime through the strainer and to adjust the clean strainer head loss to different plant conditions (see the Response to 3.f.10).

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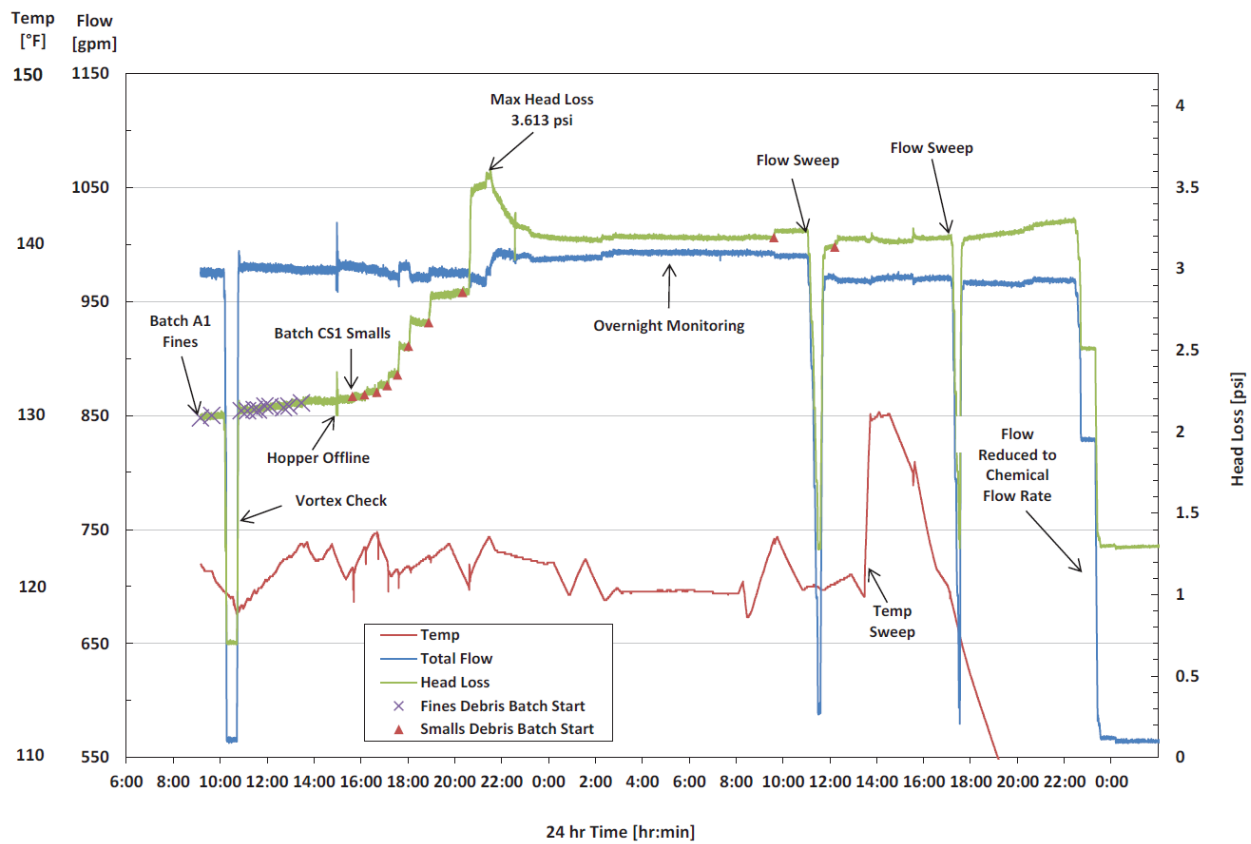

Figure 3.f.4-5: FDL test conventional debris timeline

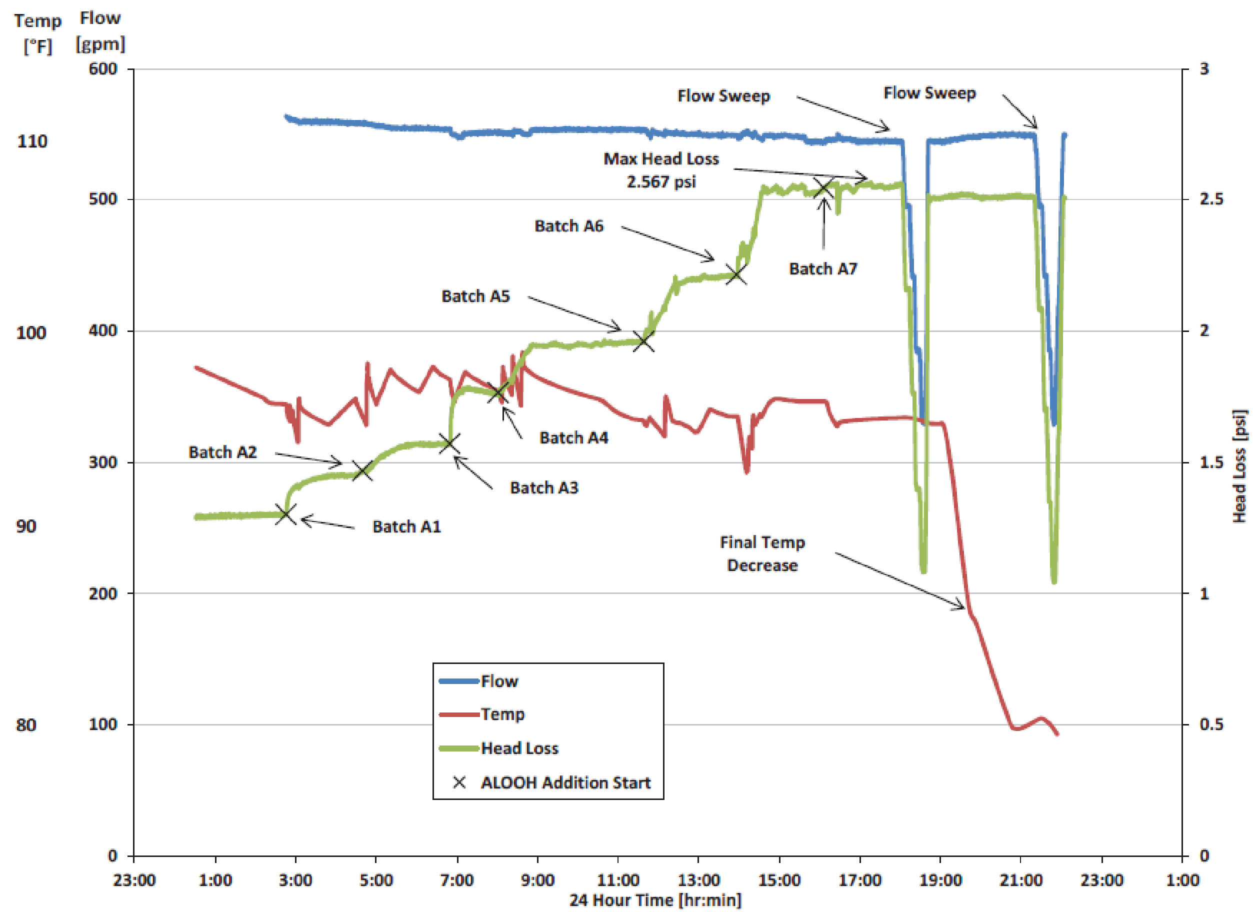

Figure 3.f.4-6: FDL test chemical debris timeline

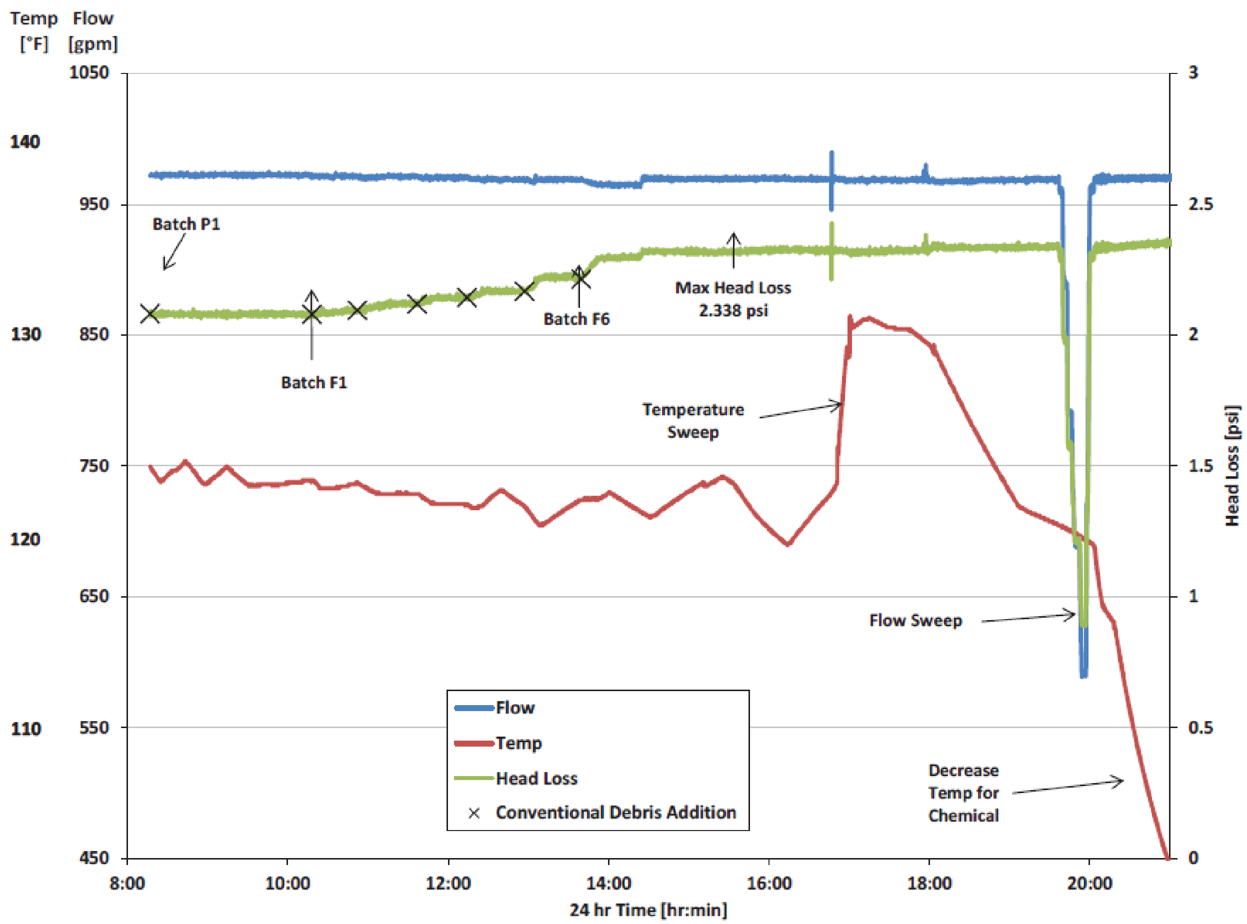

Figure 3.f.4-7: TB test conventional debris timeline

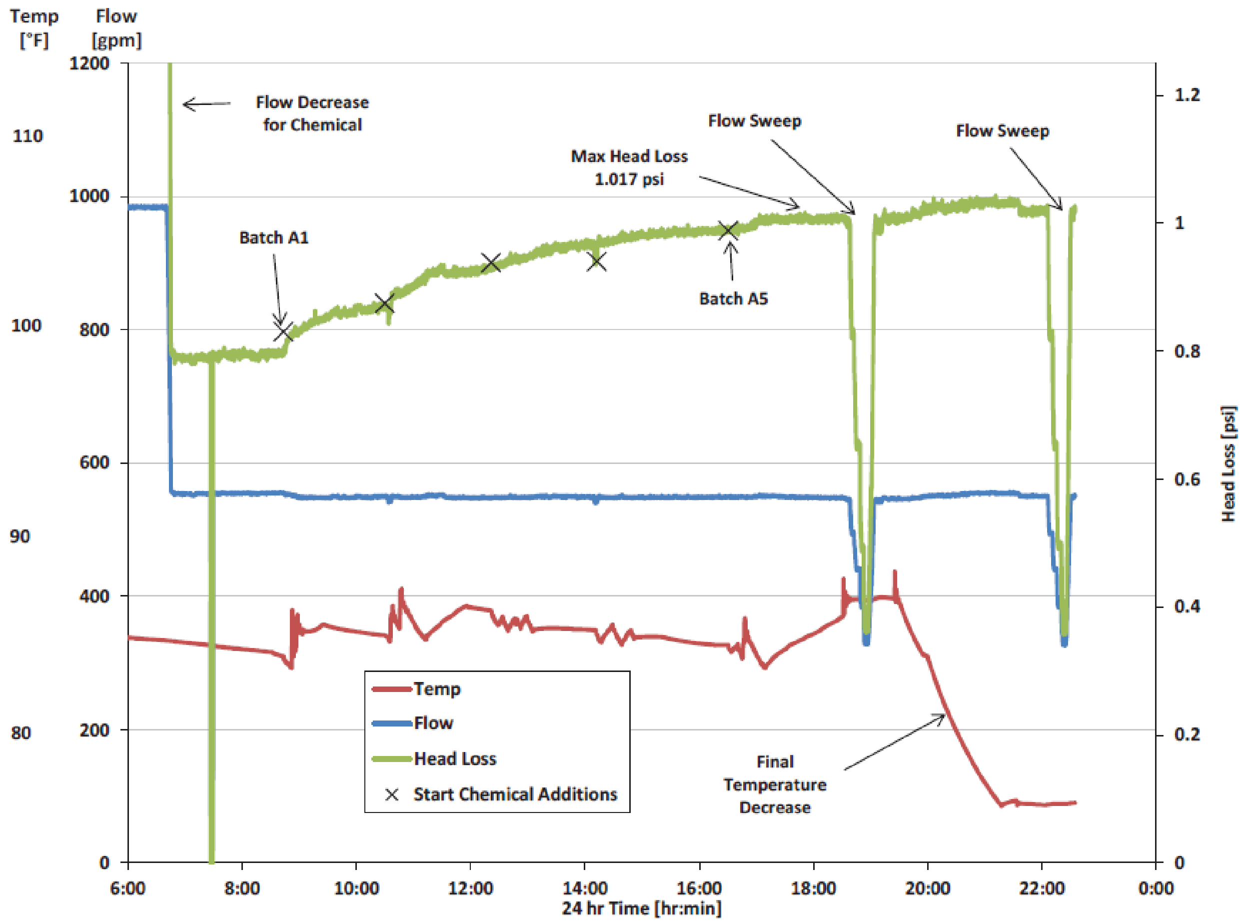

Figure 3.f.4-8: TB test chemical debris timeline

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A summary of the debris head loss results from the FDL and TB tests are provided in Table 3.f.4-7. The head loss values shown in the table already had the test strainer clean strainer head loss subtracted. The maximum debris head losses observed during conventional and chemical debris introduction (bold-faced in the table) were used to derive the head losses for the plant strainer.

Table 3.f.4-7: FDL and TB test head loss results

Test Point	Test Flow Rate (at plant scale) (gpm)	Temperature, (°F)	Debris Head Loss (psi)
FDL Test			
Conventional Debris Maximum Head Loss	**982.3 (9,300)**	**122.7**	**1.487**
Conventional Debris Stabilized Head Loss	967.9 (9,163)	119.4	1.123
Chemical Debris Maximum Head Loss	**544.4 (5,154)**	**95.5**	**1.920**
Chemical Debris Stabilized Head Loss	544.3 (5,153)	95.6	1.914
TB Test			
Conventional Debris Maximum Head Loss	968.6 (9,170)	119.8	0.270
Conventional Debris Stabilized Head Loss	969.0 (9,174)	121.7	0.254
Chemical Debris Maximum Head Loss	548.0 (5,188)	95.9	0.361
Chemical Debris Stabilized Head Loss	545.7 (5,166)	99.7	0.349

3.f.5 Address the ability of the design to accommodate the maximum volume of debris that is predicted to arrive at the screen.

Response to 3.f.5

As discussed in the Response to 3.f.4, the Wolf Creek head loss tests used a test strainer prototypical to the plant strainer designs. Additionally, the test debris loads were scaled based on the ratio of the test strainer surface area and the plant net strainer surface area. The arrangement of the test strainer within the test tank modeled the pit configuration of the plant strainer. As will be shown in the Response to 3.f.7, the head loss tests bounded debris loads of all postulated breaks of 10 in and smaller.

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With these considerations, the impact of the maximum debris volume of the bounded breaks on the plant strainer can be directly determined from the head loss test results.

3.f.6 Address the ability of the screen to resist the formation of a “thin bed” or to accommodate partial thin bed formation.

Response to 3.f.6

The thin bed effect is defined as the relatively high head losses associated with a low-porosity (with a high particulate-to-fiber ratio) debris bed formed by a thin layer of fibrous debris that can effectively filter particulate debris. The Wolf Creek head loss testing included a test for thin-bed effects. In this test, particulate debris was added to the test tank first, followed by fiber fines in batches, each with a theoretical uniform bed thickness of approximately 1/16 in. The batching schedule allowed the formation of debris beds with high particulate to fiber ratios. As a result, any thin-bed effects, should they occur, would be captured by the measured head losses. The TB Test resulted in lower conventional and chemical debris head losses than the FDL Test (see Table 3.f.4-7), and the “thin-bed” effect head loss is therefore bounded by full debris load head loss.

3.f.7 Provide the basis for strainer design maximum head loss.

Response to 3.f.7

A head loss analysis was performed to determine the total strainer head losses by combining the calculated plant strainer CSHL and measured debris head loss from testing. The evaluation used the maximum recorded conventional and chemical debris head losses (as bold-faced in Table 3.f.4-7) and adjusted them from the testing flow rates and temperatures to plant conditions of interest. As summarized in this response, plant conditions (e.g., strainer approach velocity, conventional debris loads, and chemical debris load) for all postulated breaks of 10 inches and smaller are bounded by the head loss testing parameters. Therefore, the derived total strainer head losses represent the maximum head losses for the breaks of 10 inches and smaller at Wolf Creek.

As discussed in the Response to 3.f.4, the strainer approach velocities utilized during head loss testing were based on strainer flow rates of 9,100 gpm and 5,150 gpm for conventional (with both the RHR and CS pumps drawing water from the sump) and chemical debris introductions (with only the RHR pumps drawing water from the sump), respectively. As shown in the Response to 3.f.10, head loss analyses were performed at strainer flow rates of 9,100 gpm and 4,900 gpm for conventional and chemical debris, respectively. These test flow rates are higher than the maximum strainer flow rates of 8,710 gpm and 4,760 gpm for the respective operating conditions (see the Responses to 3.g.1 and 3.g.2). Performing head loss testing at higher flow rates results in conservatively higher head losses because the test debris bed would be compressed more than the plant strainer debris bed.

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Table 3.f.7-1: compares the maximum debris loads for the postulated breaks of 10 inches and smaller with those used in the Wolf Creek head loss tests (see the Response to 3.f.4). As shown in the table, the plant debris loads are bounded by the FDL test for all debris types. Note that the plant debris quantities shown in Table 3.f.7-1: represent the maximum quantity of each debris type for the breaks of 10 inches and smaller. In other words, these plant debris quantities are not from a single break, which is conservative for the purpose of this comparison.

Table 3.f.7-1: Comparison of Plant and Head Loss Test Debris Loads

Debris Type	FDL Test	TB Test	Max Quantity of 10 in Breaks and Smaller
Fiber Fines (lbm)	141.78	225.57	106.2
Fiber Smalls (lbm)	180.67	0	140.0
Total Fiber (lbm)	**322.45**	**225.57**	**244.1**
Latent Particulate (lbm)	122.18	122.35	101.15
ThermoLag/FoamGlas (ft^3)	0.50*	0.79	0.506*
Coatings Particulate (ft^3)	2.43	2.79	1.67
Paint Chips (ft^3)	0.11	0.11	0
AlOOH (kg)	147.1	147.1	147

*Slight difference in values due to rounding between calculation and testing document

As bold-faced in Table 3.f.4-7, the maximum recorded debris head losses for conventional and chemical debris in the FDL test were used to derive the plant strainer head losses. Therefore, the resulting conventional and chemical debris head losses are bounding and applicable for all breaks of 10 inches or less analyzed for Wolf Creek.

3.f.8 Describe significant margins and conservatisms used in head loss and vortexing calculations.

Response to 3.f.8

Conservatisms in Head Loss Analysis

1. Strainer head loss was evaluated assuming a single train failure because operation of one train results in greater debris loading per unit strainer area.
2. The strainer flow rate used for the head loss analysis was 9100 gpm, which bounds the maximum strainer flow rate of 8,710 gpm from combining the maximum RHR and CS pump flow rates of 4,760 and 3,950 gpm.
3. As shown in Table 3.f.4-7, the head loss testing used test flow rates much higher than the maximum strainer flow rates of 8,710 gpm (with RHR and CS pump each drawing water from the sump) and 4,760 gpm (with only the RHR pump drawing water from the sump). Performing head loss testing at higher

flow rates compressed the debris bed more than expected for the plant strainer and would result in conservatively higher debris head losses.

4. As discussed in the Response to 3.f.7, when determining the head loss threshold break size by comparing the tested debris loads with the plant debris loads, the maximum transported plant debris load of each debris type was used. As a result, the plant debris loads for different debris types were not necessarily from the same break. This was a conservative approach to ensure that the tested debris loads bound the plant quantities of all debris types.

<u>Conservatisms in Vortexing Analysis</u>

1. As shown in the Response to 3.f.3, vortexing was evaluated at flow rates of 5,100 gpm for the RHR pump suction line. This flow rate conservatively bounds the maximum RHR pump flow rates of 4,760 gpm.
2. Vortexing was evaluated using a combination of the inputs listed below. Though these conditions do not occur simultaneously at the plant, they were treated as concurrent conditions for conservatism.
3. Vortexing observation was based on strainer head loss testing with contribution from conventional and chemical debris, which bounded all 10-inch breaks. As discussed in the Response to 3.o.2.7.ii, chemical debris does not form until the sump temperature falls to 116.3°F.
4. Minimum water level of a SBLOCA at recirculation switchover.

3.f.9 Provide a summary of the methodology, assumptions, bases for the assumptions, and results for the clean strainer head loss calculation.

Response to 3.f.9

The clean strainer head loss was calculated by the strainer vendor in two parts. Firstly, the head loss through the strainer disk stacks was determined by employing an equation which was experimentally derived from generic PCI Sureflow strainer testing. Secondly, the head loss through the plenum and discharge to the sump was determined analytically using standard hydraulic head loss equations. The calculated head losses for the strainer and plenum were combined to determine the total clean strainer head loss.

<u>Head Loss through Strainer Disk Stacks</u>

Head loss test data from a generic PCI prototype was curve fit to a second-order polynomial function of the strainer's core tube exit velocity. The function was used to calculate the head loss for the Wolf Creek strainer disk stacks based on the Wolf Creek strainer core tube exit velocity. The calculated head loss was increased by 6% to account for the uncertainties in the curve-fit of the test data. Additionally, various differences between the tested PCI prototype and the Wolf Creek strainer design and their potential impact on the clean strainer head loss were analyzed and accounted for accordingly, as summarized below.

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- The core tube diameter and test flow rate for the generic PCI prototype are much larger than the Wolf Creek strainer. However, the core tube exit velocities are similar. Since the measured head loss was shown to depend primarily on the core tube exit velocity, no head loss correction is necessary for the difference in the core tube diameter and flow rate.
- The velocity through the perforated plate openings for the generic PCI prototype is much higher than that of the Wolf Creek strainer. Since higher velocity results in greater head loss, no adjustment was done to the calculated head loss for conservatism.
- Each of the Wolf Creek strainer disk stacks contains multiple modules. A connection sleeve was used to seal the gap between the core tubes of adjacent modules. The head loss due to flow expansion and contraction at the connection sleeve was calculated and added to the overall clean strainer head loss.
- Different from the generic PCI prototype, each of the Wolf Creek strainer disks is internally reinforced by stiffeners made of bent wires. The maximum head loss that could be caused by blockage of these internal stiffeners was calculated to be orders of magnitude less than the overall head loss and therefore was neglected.

<u>Head Loss inside Plenum</u>

The bottom of each strainer disk stack opens to a plenum. The head loss at the exit of the stack was determined by modeling it as flow exiting a pipe. The uniform flow distribution of the Sureflow strainer was considered in the evaluation. Additionally, some of the strainer stacks open to a more confined portion of the plenum before reaching the main part of the sump pit. The head loss associated with this flow expansion was also accounted for.

Combining the head losses above, the resulting total clean strainer head losses are shown in Table 3.f.9-1 for different strainer flow rates and water temperatures.

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Table 3.f.9-1: Clean Strainer Head Loss

CSHL (ft-H_2O)	Temperature (°F)	Strainer Flow Rate (gpm)
0.651	140	8,830
0.205	140	4,880
0.642	212	8,830

3.f.10 Provide a summary of the methodology, assumptions, bases for the assumptions, and results for the debris head loss analysis.

Response to 3.f.10

Total head loss across the strainer was determined as the sum of the calculate CSHL (see the Response to 3.f.9) and the measured debris head loss from testing (see the Response to 3.f.4). The response to 3.f.9 outlines the methodology used for determining CSHL. When determining the total strainer head loss, the calculated CSHL and measured debris head losses were both adjusted to plant conditions of interest. The methodology used for such adjustments is summarized in this section. Note that the FDL test resulted in higher debris head losses than the TB test. Therefore, the test results of the FDL test, as bold-faced in Table 3.f.4-7, were used to determine the total strainer head losses.

Strainer head losses at various sump pool temperatures and plant strainer flow rates are of interest. However, the original CSHL calculation and debris head loss testing were conducted at a few specific temperatures and equivalent plant flow rates. To adjust these head losses to plant conditions, the flow regime (i.e., turbulent vs. laminar) through the strainer or debris bed was first characterized using the flow sweep data recorded during the FDL test.

For conventional debris head loss, the flow sweep data taken after adding the full conventional debris load is plotted in the figure below and was curve fit to a quadratic function of test flow rate.

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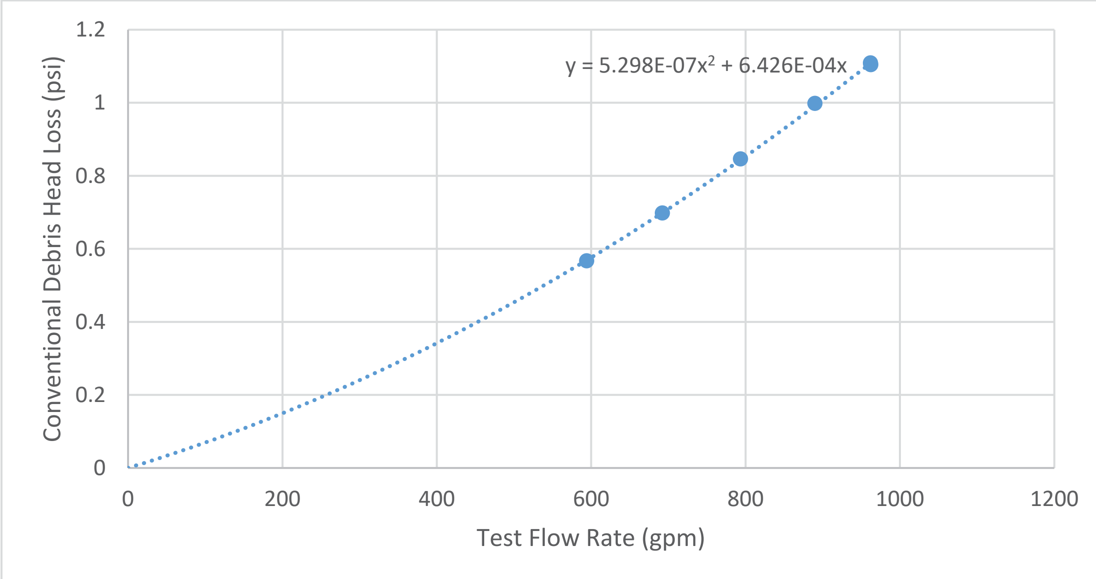

Figure 3.f.10-1: Conventional Debris Head Loss Flow Sweep Curve Fit

$$h_{L,conv,test} = B_1 Q_{Test}^2 + B_2 Q_{Test} = 5.298 \times 10^{-7} Q_{Test}^2 + 6.426 \times 10^{-4} Q_{Test}$$

Using the fitting coefficients in the above equation, the laminar and turbulent fractions for the flow through the conventional debris bed were calculated to be 56% and 44%, respectively, at the target test flow rate of 961.2 gpm (see the Response to 3.f.4).

$$L_{Frac} = \frac{B_2 Q}{B_1 Q^2 + B_2 Q} = \frac{6.426 \times 10^{-4} \times 961.2 \text{ gpm}}{5.298 \times 10^{-7} \times 961.2 \text{ gpm}^2 + 6.426 \times 10^{-4} \times 961.2 \text{ gpm}} = 56\%$$

These laminar and turbulent fractions were then used to adjust the maximum conventional debris head loss from testing (1.487 psi at 9300 gpm and 122.7°F, see Table 3.f.4-7) to plant conditions at 9100 gpm and 212°F. In the example calculation below, the variables with the subscript “1” represent testing conditions while those with subscript “2” are for plant conditions of interest. The water properties at 122.7°F and 212°F were obtained or interpolated as necessary from the ASME steam table. The same approach can be applied for other plant conditions of interest.

$$h_{L,conv,plant} = \left[L_{Frac} \frac{\mu_2}{\mu_1} \frac{Q_{Plant}}{Q_{Reference,conv,test}} + T_{Frac} \frac{\rho_2}{\rho_1} \left(\frac{Q_{Plant}}{Q_{Reference,conv,test}} \right)^2 \right] h_{L,1}$$

$$= \left[\begin{array}{c} 0.56 \times \dfrac{1.894\times10^{-4}\frac{\text{lbm}}{\text{ft}\cdot\text{s}}}{3.656\times10^{-4}\frac{\text{lbm}}{\text{ft}\cdot\text{s}}} \times \dfrac{9{,}100 \text{ gpm}}{9{,}300 \text{ gpm}} + \\ 0.44 \times \dfrac{59.82\frac{\text{lbm}}{\text{ft}^3}}{61.667\frac{\text{lbm}}{\text{ft}^3}} \times \dfrac{9{,}100 \text{ gpm}^2}{9{,}300 \text{ gpm}} \end{array} \right] \times 1.487 \text{psi}$$

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$$= 1.03 \text{ psi or } 2.48\ ft$$

A similar analysis was performed for the chemical debris head loss. The flow sweep data taken after adding the full conventional and chemical debris loads is plotted in the figure below and was curve fit to a quadratic function of test flow rate.

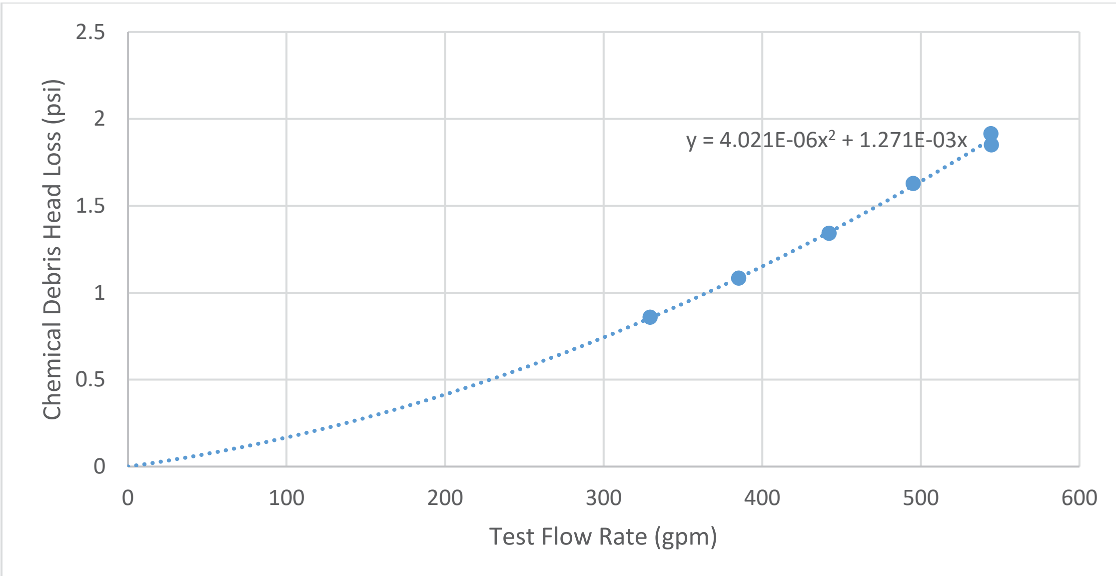

Figure 3.f.10-2: Chemical Debris Head Loss Flow Sweep Curve Fit

$$h_{L,chem,test} = 4.021 \times 10^{-6}\, Q_{Test}^2 + 1.271 \times 10^{-3} Q_{Test}$$

Using the fitting coefficients of the above curve fit and the same formula as that for conventional debris head loss, the laminar and turbulent fractions for the flow through the chemical debris bed were calculated to be 37% and 63%, respectively, at the target test flow rate of 544 gpm. These laminar and turbulent fractions were used to adjust the maximum chemical debris head loss from testing (1.920 psi at 5,154 gpm and 95.5°F, see Table 3.f.4-7) to plant conditions at 4,900 gpm and 55°F using the same formula as above. The resulting chemical debris head loss is 2.24 psi or 5.18 ft. Similarly, water properties at 95.5°F and 55°F were obtained or interpolated (as necessary) from the ASME steam table.

Similar adjustment was also done for the calculated CSHL. Flow sweep data collected during the FDL and TB tests before adding any debris was used to characterize the flow regime through the clean strainer, as shown in Figure 3.f.10-3. Since the resulting curve-fit coefficient of the linear term is negative, the laminar and turbulent fractions are 0% and 100%, respectively.

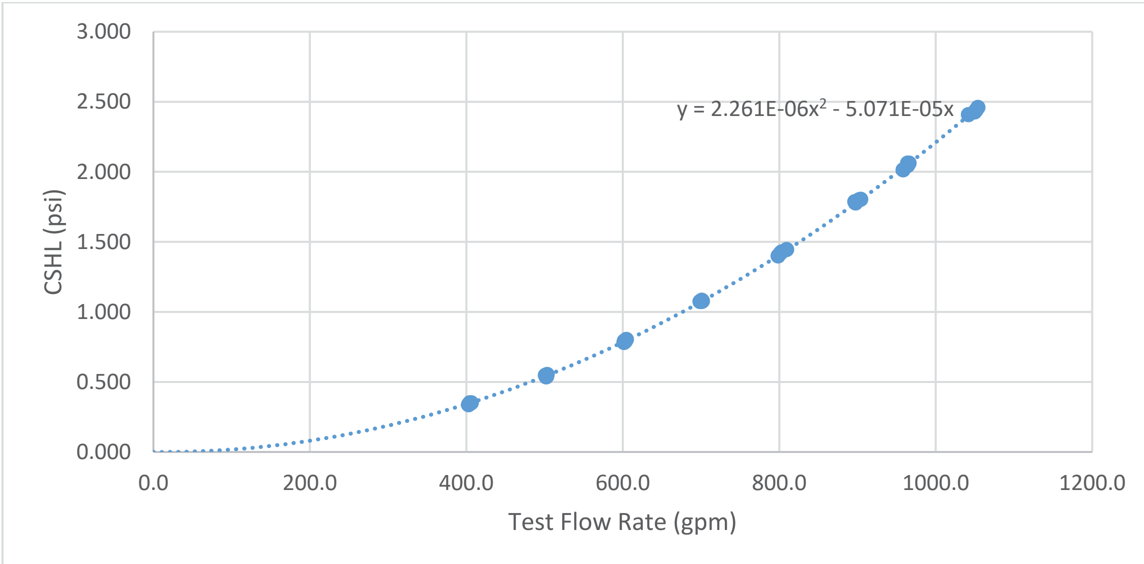

Figure 3.f.10-3: Clean Strainer Flow Sweep Curve Fit

The CSHL originally calculated at 8830 gpm and 212°F (0.642 ft, see Table 3.f.9-1) can then be adjusted to 9100 gpm and 212°F as follows.

$$h_{L,CSHL,plant} = T_{Frac}\left(\frac{Q_{Plant}}{Q_{Reference}}\right)^2 h_{L,1}$$

$$= 1 \times \left(\frac{9{,}100 \text{ gpm}}{8{,}830 \text{ gpm}}\right)^2 \times 0.642 \text{ ft-H}_2\text{O}$$

$$= 0.682 \text{ ft-H}_2\text{O}$$

The total strainer head losses for bounding plant conditions used for pump NPSH margin and strainer structural limit analyses are shown in the table below.

Table 3.f.10-1: Total Strainer Head Loss Results

T (°F)	Strainer Q (gpm)	CSHL (ft)	Debris Head Loss		Total Strainer Head Loss (ft)
			Value (ft)	Debris Head Loss Type	
212	9,100	0.68	2.48	Conventional	3.16
175	9,100	0.68	2.73	Conventional	3.41
116.3	9,100	0.70	3.48	Conventional	4.18
55	4,900	0.21	5.18	Conventional and Chemical	5.39

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As discussed in the Response to 3.g.2, it is the most conservative to evaluate pump NPSH margin at a sump temperature of 212°F. The strainer head loss at this temperature and a strainer flow rate of 9,100 gpm in Table 3.f.10-1 was used to determine the most limiting pump NPSH margin.

The strainer head loss presented in Table 3.f.10-1 at 116.3°F is representative of the conditions immediately before the onset of chemical precipitation and the reduction of strainer flow to 4,900 gpm. Note that, as stated in the Response to 3.o.2.7.ii, chemical precipitation does not occur prior to 24 hours following a LOCA when the CS pump is secured. Since adjusted head losses increase as temperature decreases, the strainer head loss presented in Table 3.f.10-1 at 116.3°F represents the maximum strainer head loss for sump temperatures above 116.3°F.

Similarly, the strainer head loss shown in Table 3.f.10-1 at 55°F represents the maximum strainer head loss for sump temperatures between 55°F and 116.3°F. Note that this total strainer head loss includes the contribution from both conventional and chemical debris.

The structural analysis showed that the sump strainers can sustain a differential pressure of 4 ft at 268°F and 5.5 ft at 175°F (see the Response to 3.k.1). Since the differential pressure limit increases as temperature decreases, it is conservative to apply the 4 ft limit at 268°F to all temperatures between 268°F and 175°F. Similarly, it is conservative to apply the 5.5 ft limit at 175°F to all temperatures below 175°F. As shown in Table 3.f.10-1, the total strainer head losses at 175°F and 212°F are both lower than the differential pressure limit of 4 ft. Additionally, the strainer head losses at 116.3°F and 55°F are both lower than the differential pressure limit of 5.5 ft. It is therefore concluded that the strainer head loss will not challenge structural integrity of the strainer for sump temperatures as low as 55°F.

3.f.11 State whether the sump is partially submerged or vented (i.e., lacks a complete water seal over its entire surface) for any accident scenarios and describe what failure criteria in addition to loss of net positive suction head (NPSH) margin were applied to address potential inability to pass the required flow through the strainer.

Response to 3.f.11

As shown in the Response to 3.f.2, the minimum water levels for both LBLOCA and SBLOCA result in full strainer submergence. Therefore, no evaluation or consideration of alternative failure criteria related to a partially submerged strainer is required for Wolf Creek.

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3.f.12 State whether near-field settling was credited for the head-loss testing, and if so, provide a description of the scaling analysis used to justify near-field credit.

Response to 3.f.12

No near-field settling was credited in Wolf Creek head loss testing. As described in the Response to 3.f.4, the debris introduction region of the test tank was equipped with hydraulic mixing lines that provided a sufficient level of turbulence to keep debris suspended without disturbing the debris bed. Additionally, the pit region of the test tank was designed to represent the prototypical spacing between adjacent strainer stacks and between strainer stacks and the plant sump pit walls. Therefore, no non-prototypical settling occurred around the strainer.

3.f.13 State whether temperature/viscosity was used to scale the results of the head loss test to actual plant conditions. If scaling was used, provide the basis for concluding that boreholes or other differential-pressure induced effects did not affect the morphology of the test debris bed.

Response to 3.f.13

As seen in the Response to 3.f.10, the calculated CSHL and measured debris head losses were adjusted for temperature and flow rate from testing conditions to plant condition. The adjustment was done using flow regime information derived from Wolf Creek-specific flow sweep data collected during the latest head loss testing, instead of assumed laminar and turbulent fractions. Therefore, any boreholes or other differential-pressure induced effects on bed morphology were captured and properly accounted for when scaling the head loss.

3.f.14 State whether containment accident pressure was credited in evaluating whether flashing would occur across the strainer surface, and if so, summarize the methodology used to determine the available containment pressure.

Response to 3.f.14

<u>Flashing Analysis</u>

Flashing would occur if the pressure immediately downstream of the strainer is lower than the water vapor pressure at the sump temperature. The pressure downstream of the strainer can be calculated by adding the strainer submergence (from the top of the strainer) to the containment pressure, before subtracting the strainer head loss. The Wolf Creek analysis showed that, if the containment pressure is assumed to be equal to water vapor pressure at the corresponding sump temperature, a minimum containment accident pressure of 0.97 psi (rounded to 1 psi) needs to be credited to preclude flashing. As shown below, with this containment accident pressure credited,

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the minimum containment pressure required to prevent flashing is at least 1 psi lower than the containment pressure from the safety analysis, and this minimum margin is momentary. Additionally, when using the containment pressure and sump temperature curves from the analysis of a smaller breaks using more realistic inputs, greater margins are shown in the containment pressure. Therefore, crediting 1 psi of accident containment pressure is reasonable and conservative.

The margin in containment pressure for preventing flashing immediately downstream of the strainer was evaluated for time-dependent post-accident containment and sump conditions. The minimum containment pressure that is required to prevent flashing was calculated by adding the strainer head loss to the water vapor pressure and subtracting the minimum strainer submergence calculated from the top of the strainer. Afterwards, this minimum required containment pressure was compared with the expected post-accident containment pressure to determine the margin.

For each given set of conditions, the sump pool temperature and containment pressure are obtained from the analysis of a design basis accident (DBA). The minimum strainer submergence of 1.17 ft (see the Response to 3.g.1) was converted to 0.47 psi conservatively, using the density of water at the highest sump pool temperature after the onset of recirculation (268.8°F). The evaluation used different strainer head loss values for different sump temperature ranges. The head loss value at the lowest temperature of each range is applied conservatively since head loss decreases with increasing temperature. As shown in Table 3.f.14-1, the minimum margin in the containment pressure to prevent flashing is above 1 psi and this minimum margin, which occurs at the sump temperature of 264.8°F, is momentary.

Table 3.f.14-1: Margin in Containment Pressure for Preventing Flashing based on DBA Curves

Time (s)	Sump Pool Temperature (°F)	Vapor Pressure (psia)	Strainer Head Loss (psi)	Min Cont. Pressure Req'd to Prevent Flashing (psia)	Cont. Pressure (psia)	Margin (psi)
1,509	268.4	40.77	1.44	41.73	54.97	13.24
2,010	268.8	41.04	1.44	42.00	48.82	6.82
3,010	264.8	38.40	1.44	39.37	40.40	1.03
4,510	259.4	35.07	1.44	36.03	39.54	3.51
6,510	253.5	31.70	1.44	32.66	35.80	3.14
8,510	246.5	28.04	1.44	29.01	32.90	3.89
10,000	241.3	25.56	1.44	26.52	31.18	4.66
70,000	174.3	6.61	1.79	7.93	19.09	11.16
87,000	171.1	6.15	1.79	7.47	18.66	11.19

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Similar analyses were also performed using containment pressure and sump temperature curves of smaller breaks (e.g., 6 inch and 3 inch breaks) using more realistic inputs. Table 3.f.14-2 shows the results for the analysis of the 3 inch break. The margins between containment accident pressure and minimum pressure required to prevent flashing are greater than 9 psi due mainly to lower sump pool temperatures associated with this smaller break size. It should be noted that the analysis for these smaller breaks conservatively used the maximum head loss (including chemical effects) of 1.95 psi regardless of sump temperature and zero strainer submergence. Based on these analyses, crediting 1 psi of accident containment pressure to prevent flashing immediately downstream of the strainer is reasonable and conservative.

Table 3.f.14-2: Break Margin in Containment Pressure for Preventing Flashing for a 3-inch Break

Time (s)	Sump Pool Temperature (°F)	Vapor Pressure (psia)	Strainer Head Loss (psi)	Min Cont. Pressure Req'd to Prevent Flashing (psia)	Cont. Pressure (psia)	Margin (psi)
6,800.24	218.5	16.70	1.95	18.65	29.56	10.91
7,200.24	213.7	15.20	1.95	17.15	28.86	11.71
9,000.32	200.3	11.59	1.95	13.54	27.32	13.78
11,000.3	199.2	11.33	1.95	13.28	30.18	16.90
16,000.3	201.1	11.79	1.95	13.74	27.61	13.88
18,000.3	207.2	13.35	1.95	15.30	24.62	9.32
20,000.3	199.9	11.51	1.95	13.46	23.19	9.73
22,000.3	194.1	10.18	1.95	12.13	22.10	9.96
24,000.3	189.2	9.18	1.95	11.13	21.27	10.13

<u>Degasification Analysis</u>

The degasification analysis used post-LOCA containment pressure and sump temperature curves as inputs to determine void fractions downstream of the strainer and at the pump suction. Both minimum and maximum safeguards curves were analyzed to ensure the worst case scenario is considered. The quantity of air which comes out of solution to form voids was calculated using Henry's Law. A combination of conservative inputs, which do not occur simultaneously, was used in the degasification analysis to ensure bounding results.

The strainer head loss used included clean strainer head loss and conventional debris head loss for breaks up to the 10 inch threshold break size. Both head losses were adjusted for sump temperature using the approach described in the Response to 3.f.10. The strainer submergence was based on the minimum SBLOCA sump water level.

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The void fraction was first determined downstream of the strainer at the strainer midpoint elevation. The voids formed at the strainer midpoint were assumed to transport intact to the pump suction. The analysis conservatively ignores any re-dissolution of the voids as they are transported to the pump suction and experience an increase in pressure. The void fraction at the pump suction is determined by accounting for compression of the voids due to elevation changes as they transport downward to the pump suction.

The void fraction at the RHR pump suction (α_{RHR}) and CS pump suction (α_{CS}), along with the sump temperature (T_{sump}) and containment pressure (P_{cont}) from the minimum safeguard containment analysis curves, are shown in Table 3.f.14-3 for a range of post-LOCA conditions.

Table 3.f.14-3: Void Fraction Analysis at Pump Suction Using Inputs from Minimum Safeguard Containment Analysis Curves

Parameter	T_{sump}	P_{cont}	α_{RHR}	α_{CS}
Unit	(°F)	(psia)	(%)	(%)
Results	268.4	54.97	0.0002	0.0002
	268.8	48.82	0.0001	0.0001
	264.8	40.40	0.0050	0.0060
	250.1	34.26	0.0100	0.0100
	241.3	31.18	0.0100	0.0110
	213.7	24.82	0.0121	0.0129
	212	24.52	0.0123	0.0131
	210	24.16	0.0125	0.0133
	198.8	22.18	0.0140	0.0150

Similar analysis was also done using the sump temperatures and containment pressures from the maximum safeguards analysis curves. The resulting void fractions <u>at the strainer midpoint</u> (α) for selected conditions are shown in Table 3.f.14-4. This demonstrates the maximum void fraction occurs when using the minimum safeguard containment analysis curves as inputs.

Table 3.f.14-4: Void Fraction Analysis Results at Midpoint of Strainer for Maximum Safeguard Containment Curves

Parameter	T_{sump}	P_{cont}	α
Unit	°F	psia	–
Results	232.7	43.18	0.01%
	212	41.02	0.01%

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The void fractions shown in Table 3.f.14-3 were used to adjust NPSH required (NPSHr) per the guidance provided in RG 1.82 Appendix A (Reference 25) when calculating the NPSH margin in the Response to 3.g.16. As discussed in the Response to 3.g.2, the most limiting NPSH margin occurs at 212°F. For this reason, the degasification analysis is performed for a range of sump temperatures from the maximum sump temperature down to just below 212°F.

The approach to the degasification analysis and correction to the pump NPSHr described above is consistent with the method used by Salem Nuclear Generating Station in their GL 2004-02 response (Reference 31). This method was accepted by the NRC staff in their review of the Salem submittal (Reference 32).

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3.g Net Positive Suction Head (NPSH)

The objective of the NPSH section is to calculate the NPSH margin for the ECCS and CSS pumps that would exist during a loss-of-coolant accident (LOCA) considering a spectrum of break sizes.

3.g.1 Provide applicable pump flow rates, the total recirculation sump flow rates, sump temperature(s), and minimum containment water level.

Response to 3.g.1

<u>Pump/ Sump Flow Rates</u>

Two strainer flow rates were required: at the start of recirculation with one RHR and one CS pump in operation and after chemical precipitation with only one RHR pump in operation. For the first scenario, a flow rate of 9,100 gpm per strainer is used. This flow rate is derived by adding margin (390 gpm) to the total single train flow rate, which is a combination of the maximum RHR pump and CS pump flow rates of 4,760 and 3,950 gpm, respectively (see discussion below). For the second scenario, chemical precipitation was predicted to occur after 24 hours following the accident when CS is secured. Therefore, a flow rate of 4,900 gpm per strainer is used, which is conservatively higher than the actual RHR pump flow rates shown in the table below.

The RHR pump flow rates were determined for several post-LOCA recirculation scenarios, as shown in Table 3.g.1-1. To maximize the pump flow rate, the analysis assumed the failure of one RHR pump and the in-service RHR pump supplies flow to RCS, and two charging and two injection pumps.

Table 3.g.1-1: Post-LOCA Recirculation Flow Rates

Scenario	Flow (gpm)
RHR A Train, Cold Leg Recirculation	4,760
RHR B Train, Cold Leg Recirculation	4,704
RHR A Train, Hot Leg Recirculation	4,748
RHR B Train, Hot Leg Recirculation	4,736

The CS pump recirculation flow rate ranges from 3690 to 3950 gpm as determined from the pump performance curve and CS system curves.

<u>Water Level</u>

As discussed in the Response to 3.f.2, the minimum sump pool water level for a SBLOCA is at an elevation of 2000’-11.51” (or 2000.96 ft) at ECCS sump switchover. For a LBLOCA, the minimum sump pool water level is at an elevation of 2002’-1.02”

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(or 2002.09 ft). The strainer is fully submerged by at least 0.04 ft for SBLOCAs and 1.17 ft for LBLOCAs.

Sump Temperature

The maximum sump water temperature during recirculation is approximately 270°F (see the Response to 3.o.2.3).

Pump NPSH margins were evaluated at 212°F. This resulted in the most limiting NPSH margins, as justified in the Response to 3.g.2.

3.g.2 Describe the assumptions used in the calculations for the above parameters and the sources/bases of the assumptions.

Response to 3.g.2

Pump/ Sump Flow Rates

The following assumptions were used to determine the RHR pump flow rates, which are conservative for the NPSH margin evaluation.

1. During the recirculation phase, the RHR pumps take suction from the containment sump where water level is 2002.09 ft. The sump water temperature is 212°F.
2. RCS pressure and containment pressure are assumed to be at 0 psig.
3. Fluid temperature at all points downstream of the RHR heat exchangers was set at 157.5°F

For determination of the CS pump flow rates, the following assumption was used:

1. The containment sump water level was assumed to be 2003'-10", and the sump water temperature was assumed to be 255°F.

Water Level

The significant assumptions used to calculate the minimum water volume for determining NPSH margin are listed as follows:

1. RWST, RCS, and the SI accumulator inventory were assumed to be the same density as pure water. This assumption is reasonable since the boric acid concentrations are low.
2. The volumes for the portions of Engineering Safety Feature (ESF) piping that are empty prior to emergency operation were evaluated and the result was increased by 5% to maximize the holdup from the sump volume.
3. No credit was given for the spray additive tank (SAT) volume being added to the sump.
4. To conservatively minimize the mass of water within the SI accumulators that could spill into containment, the temperature was assumed to be equal to the maximum initial containment air temperature consistent with the accident

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analysis of 120°F. This approach is conservative because the density of water decreases with increasing temperature.

5. The water density at the sump temperature was used to calculate the post-LOCA RCS volume. This is conservative because the RCS temperatures will be higher than the sump temperatures due to decay heat and residual RCS piping and component heat.
6. All of the injection flow was assumed to flow from a break at the top of the pressurizer surge line. This is the highest potential break elevation in the RCS that could allow full RHR flow through the break.
7. The transit time determined for break flow was based on the free fall of water in a vacuum and no interference with components as the water drains down the containment sump. With a total break flow holdup of about 31 ft^3, this assumption created an insignificant decrease in the holdup volume, since the injection water is falling a relatively short distance and there are not many components between the RCS and the sump pool.
8. No ECCS leakage was assumed to occur outside containment, which is reasonable because emergency response procedures provide guidance to monitor containment sump water level after recirculation initiation and direct operations to make up for any indicated loss of water level.
9. A miscellaneous holdup volume of 250 ft^3 was utilized to account for miscellaneous holdup volumes not specifically quantified in the water level calculation, such as bolt hole depressions, elevator sump, water retained in insulation, holdup on the polar crane, drain piping, holdup in cable trays and floor impressions, and containment cooler condensation drain standpipes.
10. The initial RCS volume was minimized by assuming a minimum pressurizer volume of 38%, based on the nominal pressurizer span at 100% power and an average temperature of 570.7°F.
11. For a break in the RCS loop piping, it was conservatively assumed that the reactor vessel (up to the top elevation of the hot leg piping), RCS loop piping (including reactor coolant pump internals), and pressurizer surge line are refilled with ECCS inventory at the time of ECCS switchover to recirculation. Given the high temperatures within the steam generators and pressurizer and the high temperature in the containment building, it was assumed that the ECCS inventory will not be drawn into these components until the CS switchover time is reached.
12. The containment conditions for the pressurizer surge line break were assumed to be the same as those for the LBLOCA with minimum safeguards. The entire RCS volume was assumed to blow down and then refill to the break elevation at the RHR switchover time, and fill completely (except for the pressurizer) at the CS switchover time.
13. For the purpose of calculating the initial RCS inventory blowdown, 5% steam generator tube plugging was assumed in order to conservatively minimize available inventory. However, when calculating RCS holdup after the initial blowdown, no tube plugging was assumed.

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14. The Wolf Creek containment water level calculation was originally created to bound multiple plants. When calculating the minimum water levels for LBLOCAs, the RWST inventory used in the calculation was conservatively small for Wolf Creek.

<u>Sump Temperature</u>

As stated in the Response to 3.g.1, the maximum sump temperature is approximately 270°F. However, without crediting any containment accident pressure, it is the most conservative to evaluate pump NPSH margin at a sump temperature of 212°F for the following reasons.

Should the NPSH evaluation be performed at a sump temperature higher than 212°F, for example 270°F, the containment pressure would be assumed to be equal to the vapor pressure at 270°F, without crediting containment accident pressure. As a result, the containment pressure and vapor pressure cancel each other out when calculating pump NPSH available (NPSHa). The formula for calculating NPSHa then reduces to elevation difference between the sump water level and RHR suction minus total head loss on the pump suction side. Since head loss increases as temperature decreases due to higher water viscosity, it would be slightly more conservative to calculate the NPSH margin at the lower end of this temperature range, 212°F.

Should the NPSH evaluation be performed at a sump temperature below 212°F, for example 190°F, the containment pressure would be assumed to be equal to the minimum Tech Spec allowable containment pressure of -0.3 psig (14.4 psia), which is higher than the vapor pressure (9.349 psia at 190°F) that would be used when calculating the pump NPSHa. Therefore, the difference between the assumed containment pressure and the vapor pressure (14.4 psia – 9.349 psia) increases the pump NPSHa and NPSH margin. As a result, the pump NPSH margin increases rapidly as sump temperature decreases below 212°F.

3.g.3 Provide the basis for the required NPSH values, e.g., three percent head drop or other criterion.

Response to 3.g.3

The pump vendor curves showed NPSHr values for RHR pump flow rates up to approximately 5500 gpm. For the maximum RHR pump flow rate of 4,760 gpm, an NPSHr of 21.01 ft was read from the curves.

The pump vendor curves showed NPSHr values for CS pump flow rates up to approximately 4000 gpm. For a flow rate of 3950 gpm, an NPSHr of 16.8 ft was read from the curves.

These curves were obtained by the pump manufacturer through testing in accordance with the Hydraulic Institute guidelines in effect at the time.

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Void fraction at the pump suction could affect pump NPSHr and NPSH margin. The methodology in RG 1.82 Appendix A (Reference 25), along with the void fractions shown in Table 3.f.14-3, was used to correct the NPSHr obtained from the vendor curves. As stated in the Response to 3.g.2, the most limiting NPSH margin occurs at a sump temperature of 212°F. Using the void fraction at 212°F, the corrected NPSHr after accounting for the impact of void fraction at pump suction is 21.14 ft for RHR pump and 16.91 ft for CS pump. These values are used to determine the pump NPSH margins in the Response to 3.g.16.

Note that, as shown in Table 3.f.14-3, void fraction continues increasing as temperature drops below 212°F. However, the increase in penalty on NPSHr due to higher void fraction at the lower temperatures was shown to be less than the gain in NPSHa due to reduced vapor pressure at the lower sump temperatures. The sump temperature of 212°F is still the most conservative for determining the pump NPSH margin.

3.g.4 Describe how friction and other flow losses are accounted for.

Response to 3.g.4

The total strainer head loss was calculated by combining the clean strainer and debris bed head losses (see the Response to 3.f.10). The head loss of the ECCS suction piping between the strainer exit and the pump suction was accounted for in a hydraulic model using the Fathom software. For the CS suction piping, the head loss was analyzed in a hand calculation. The piping frictional loss was calculated using the standard Darcy formula with the friction factor determined using the Darcy-Weisbach method. The head losses of the components (e.g., valves, elbows, reducers, and tee junctions) on the pump suction piping were calculated using the loss coefficients from standard industry handbooks.

3.g.5 Describe the system response scenarios for LBLOCA and SBLOCAs.

Response to 3.g.5

The ESF systems include two trains of ECCS pumps and two trains of CS pumps. Each ECCS train consists of one centrifugal charging pump (CCP), one safety injection pump (SIP), and one RHR pump. Each CS train consists of one CS pump.

In response to a LOCA, the RHR pumps, SIPs, and CCPs automatically start upon receipt of an SI signal. These pumps take suction from the RWST and inject to the RCS cold legs. This system line-up is referred to as the ECCS injection phase. The CS pumps start automatically when the containment pressure reaches the Hi-3 pressure setpoint (30 psig including uncertainty) for CS actuation. The CS pumps also take suction from the RWST during the injection phase. The spray cannot be terminated until completion of the injection phase. When the RCS pressure falls below the accumulator pressure, all four accumulators begin to inject borated water into the RCS cold legs.

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Before the RWST inventory is depleted, the suction source of the pumps must be switched to the recirculation sumps. The sump suction valves for the RHR pumps open automatically when the RWST level reaches the Low-Low-1 setpoint. The suction valves from the RWST automatically close after the suction valves from the sump are opened. The switchover for the CS pumps starts manually when the RWST level reaches the low-low-2 setpoint. The length of time that the CSS operates during the recirculation phase is determined by the operator and the procedure requires that the CS pumps be secured when the containment pressure drops to below 3 psig and at least 10 hours of spray operation has elapsed. For the breaks that do not actuate CS, the switchover to sump recirculation for the ECCS pumps follows the same logic.

In cold leg recirculation mode, the ECCS pumps operate in series, with only the RHR pumps capable of taking suction from the containment recirculation sump. The recirculation coolant is then delivered by the RHR pumps directly to the RCS cold legs and to the suctions of the SIPs and CCPs which then deliver coolant via their connections to the cold legs.

Approximately 10 hours following an accident, the ECCS lineup is changed to hot leg recirculation. For this operating mode, the SIPs and CCPs continue taking suction from the RHR pump discharge. The RHR and SI pumps are aligned to supply flow to the RCS hot legs, but the CCPs continue supplying flow to the cold legs.

The differences between the response sequences to an LBLOCA and an SBLOCA are:

- Depending on the size of the break, the RCS pressure may stabilize at a value that does not allow injection from the SI accumulators and/or the RHR pumps.
- For an SBLOCA, the containment pressure will likely remain below the actuation setpoint for the CSS.

For an SBLOCA, the outflow from the RWST may be sufficiently low that the plant may be taken to a safe shutdown condition before the RWST level reaches the Low-Low setpoint. As a result, sump recirculation may not be required.

3.g.6 Describe the operational status for each ECCS and CS pump before and after the initiation of recirculation.

Response to 3.g.6

Operating sequence of the ECCS and CS pumps have been discussed in the Response to 3.g.5. Brief summaries are presented in this section.

<u>Residual Heat Removal Pumps</u>

In the event of a LOCA, the RHR pumps start automatically on receipt of an SI signal. During the injection phase, the RHR pumps take suction from the RWST and supply flow to the RCS cold legs. The transition from injection to sump recirculation for the RHR pumps starts automatically at the RWST low-low-1 level setpoint. Afterwards,

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the RHR pumps take suction from the containment recirculation sump, supplying flow to the CCPs and SIPs and discharge directly to the cold legs. In case of failure of one RHR pump, the operating RHR pump supplies flow to both CCPs and SIPs. Approximately 10 hours after LOCA inception, hot leg recirculation is initiated. The RHR pumps continue taking suction from the containment sump and supply flow to the CCPs and SIPs but discharge to the RCS hot legs.

<u>Centrifugal Charging Pumps</u>

In the event of a LOCA, both CCPs start automatically on receipt of an SI signal and take suction directly from the RWST during the injection phase, supplying flow to the RCS cold legs. After switching to the sump recirculation mode, flow to the CCPs is provided by the RHR pump discharge. The CCPs continue supplying flow to the RCS cold legs during both cold leg and hot leg recirculation.

<u>Safety Injection Pumps</u>

In the event of a LOCA, both SIPs start automatically on receipt of an SI signal. During the injection phase, these pumps take suction from the RWST and deliver water to the RCS cold legs. Similar to the CCPs, flow to the SIPs is supplied from the containment emergency sump via the RHR pumps during the recirculation phase. The SIPs discharge to the RCS cold legs during cold leg recirculation and to the RCS hot legs during hot leg recirculation.

<u>Containment Spray Pumps</u>

Following a LOCA, the CS pumps are automatically actuated by coincidence of two out of four Hi-3 containment pressure signals to take suction from the RWST and supply flow to the spray nozzles during the injection phase. The CS pumps are realigned for the recirculation phase to take suction from the containment recirculation sump when the RWST reaches the low-low-2 level.

3.g.7 Describe the single failure assumptions relevant to pump operation and sump performance.

Response to 3.g.7

Wolf Creek has two separate recirculation sumps and each sump has its own strainer, supplying flow to one train of ECCS pumps and one CS pump. For strainer head loss and pump NPSH evaluation, it was assumed that only one strainer is in service. This scenario is conservative for NPSH evaluation because debris only accumulates on one operating strainer, resulting in higher strainer debris loads and head losses. Additionally, as stated in the Response to 3.g.1, the total flow rate of the operating strainer was also conservatively increased, which also adds conservatism to the NPSH evaluation.

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3.g.8 Describe how the containment sump water level is determined.

Response to 3.g.8

The post-LOCA minimum containment flood water is determined using the following methodology:

1. The quantity of water added to containment from the RWST, SI accumulators, and RCS is calculated.
2. The quantity of water diverted from the containment sump is calculated. Water is diverted from the containment sump by the following effects:

   a. Steam holdup in the containment atmosphere
   b. Water volume required to fill the RHR and CS piping that is empty prior to the LOCA
   c. Additional mass of water that must be added to the RCS due to the increase in the water density at the lower sump water temperature (versus the RCS temperature prior to the LOCA)
   d. Water film on surfaces
   e. Water holdup in the RCS[1]
   f. Water in transit from the containment spray nozzles and the break to the containment sump.
   g. Miscellaneous holdup volumes throughout containment

3. The volume of water available for the sump pool is calculated by subtracting the diverted volumes of Item 2 from the total volume of Item 1. The post-LOCA containment water level is then calculated using a correlation between the containment water level and the sump water volume.

3.g.9 Provide assumptions that are included in the analysis to ensure a minimum (conservative) water level is used in determining NPSH margin.

Response to 3.g.9

The assumptions provided in the Response to 3.g.2 ensure that minimum (conservative) containment water levels are calculated in the Wolf Creek containment water volume calculation.

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[1] The total RCS holdup volume was determined to be 5,066 ft^3 for LBLOCA and 13,988 ft^3 for SBLOCA at the time of ECCS switchover to recirculation.

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3.g.10 Describe whether and how the following volumes have been accounted for in pool level calculations: empty spray pipe, water droplets, condensation, and holdup on horizontal and vertical surfaces. If any are not accounted for, explain why.

Response to 3.g.10

As described in the Response to 3.g.8, the following volumes are treated within the Wolf Creek containment water volume calculation as hold-up volumes that remove water from the containment pool: CS piping that is empty prior to the LOCA, water film on surfaces, and water in transit from the spray headers and break to the containment sump.

3.g.11 Provide assumptions (and their bases) as to what equipment will displace water resulting in higher pool level.

Response to 3.g.11

The water level calculation assumes that some major permanent components will displace water, resulting in a higher pool level. These components include the following:

- Reactor vessel
- Incore tubes, incore tunnel beams, and incore sump curbs
- Concrete walls
- Pressurizer relief tank supports
- Reactor coolant drain tank supports
- Recirculation sump curbs
- Pipes and hangers
- Accumulator, SG and RCP base plates

3.g.12 Provide assumptions (and their bases) as to what water sources provide pool volume and how much volume is from each source.

Response to 3.g.12

The design inputs in Table 3.g.12-1 provide the calculated mass of water sources used to determine the minimum containment water level. Applicable assumptions (and their bases) can be seen in Response to 3.g.2.

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Table 3.g.12-1: Containment Water Sources

Event	Source	Mass (lbm)	Note
LBLOCA	Initial RCS Blowdown	549,054	
	Accumulators	200,006	
	RWST at ECCS switchover	1,889,072	This includes an instrument uncertainty of 1% on the nominal lo-lo-1 alarm setpoint
	RWST at CS switchover	2,690,947	
	SAT	0	
	Initial Containment Vapor	732	Based on 50% relative humidity and a free containment volume of 2,500,000 ft^3
SBLOCA	Initial RCS Blowdown	549,054	
	Accumulators	0	
	RWST at ECCS switchover	1,889,072	This includes an instrument uncertainty of 1% on the nominal lo-lo-1 alarm setpoint
	RWST at CS switchover	N/A	
	SAT	0	
	Initial Containment Vapor	732	Based on 50% relative humidity and a free containment volume of 2,500,000 ft^3

3.g.13 If credit is taken for containment accident pressure in determining available NPSH, provide description of the calculation of containment accident pressure used in determining the available NPSH.

Response to 3.g.13

Containment accident pressure was not credited in the NPSH calculation. When calculating pump NPSH margin at 212°F, the containment pressure was assumed to be the same as water vapor pressure.

3.g.14 Provide assumptions made which minimize the containment accident pressure and maximize the sump water temperature.

Response to 3.g.14

Containment Pressure

As stated in the Response to 3.g.13, the containment pressure was assumed to be equal to the water vapor pressure when calculating pump NPSH margin at 212°F. No containment accident pressure was credited.

Sump Temperature

As discussed in the Response to 3.g.2, the maximum sump pool temperature is approximately 270°F. However, without crediting any accident pressure, it is the most conservative to evaluate pump NPSH margin at a sump temperature of 212°F.

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3.g.15 Specify whether the containment accident pressure is set at the vapor pressure corresponding to the sump liquid temperature.

Response to 3.g.15

See the Response to 3.g.13.

3.g.16 Provide the NPSH margin results for pumps taking suction from the sump in recirculation mode.

Response to 3.g.16

During the recirculation mode, the minimum RHR pump NPSH margin was determined to be 1.21 ft at a sump temperature of 212°F and the maximum RHR pump flow rate of 4,760 gpm. The minimum CS NPSH margin was determined to be 2.01 ft at 212°F and the maximum CS pump flow rate of 3,950 gpm. As shown in the Responses to 3.g.2 and 3.g.3, evaluating the pump NPSH margin at 212°F is the most conservative. The pump NPSH margin was calculated using a strainer head loss at a total strainer flow rate of 9,100 gpm, and the NPSHr was increased to account for the impact of void fraction at the pump suction (see the Response to 3.g.3).

The evaluation conservatively combined the minimum water level at the start of recirculation and strainer head loss associated with a conservatively high strainer flow rate of 9,100 gpm (see the Response to 3.g.1). For added conservatism, the evaluation assumed containment pressure to be equal to the water vapor pressure.

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3.h 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.

3.h.1 Provide a summary of type(s) of coating systems used in containment, e.g., Carboline CZ 11 Inorganic Zinc (IOZ) primer, Ameron 90 epoxy finish coat.

Response to 3.h.1

Design Basis Accident (DBA) Qualified Coatings

Various qualified coating systems are used in containment, as summarized in Table 3.h.1-1.

Table 3.h.1-1: Qualified Coatings Systems

Substrate	System	Coatings	Type	DFT (mil)	Density (lbm/ft^3)
Concrete	100	Carboline 195	Epoxy	Varies	115
	102	Carboline 191 HB	Epoxy	6	171
	103	**Carboline 195 & 191 HB**	**Epoxy**	**40 & 12**	**115 & 171**
	114	Carboline 191 HB	Epoxy	6	171
Steel	**101**	**Carbozinc 11** Optional Topcoat Ameron 90	**IOZ** Epoxy	**5** 6	**208** 94
	105	Carbozinc 11	IOZ	5	208
	108	Carboline 191 HB	Epoxy	6	171
	110	Carboline 193 LF & 191 HB	Epoxy	5 & 6	128 & 171
	113	Carboline 193 LF & 191 HB	Epoxy	5 & 6	128 & 171
	117	Carbozinc 11 & Carboline 191 HB	Epoxy	4 & 6	208 & 171
	119	Carboline 890	Epoxy	6 & 6	110

For the purpose of debris generation analysis, it was assumed that the ‘101’ qualified coatings system was applied to all steel structures, including columns and equipment supports, and the ‘103’ system was applied to all concrete surfaces within containment, as bold-faced in Table 3.h.1-1.

Applying the ‘101’ coating system to all steel structures is reasonable. The ‘101’ coating system has an optional epoxy finish coat. In the debris generation analysis, all steel surfaces inside the steam generator (SG) compartments were assumed to be coated with the IOZ paint Carbozinc 11 only without the optional epoxy finish coat. This is reasonable because the large NSSS equipment supports, which account for a

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large portion of the steel in the SG compartments, were coated with IOZ only. Additionally, without the epoxy finish coat, the ‘101’ coating system was analyzed as untopcoated IOZ using a 10D ZOI. This is a much larger than the 4D ZOI for the qualified epoxy coatings (see the Response to 3.h.5).

Using the ‘103’ coating system for concrete surfaces is conservative because it has the largest number of coats and the largest final dry film thickness (DFT) of all coating systems for concrete.

<u>Unqualified Coatings (Applicable to All Postulated Breaks)</u>

Unqualified coatings are those that fail under design basis accident conditions (regardless if the coatings are inside or outside of the break ZOI) and create debris that could be transported to the containment recirculation sumps. There are several types of unqualified coatings applied within containment. The properties of these unqualified coatings are shown in Table 3.h.1-2.

Table 3.h.1-2: Properties of Unqualified Coatings

Items	Type	DFT (mil)	Density (lbm/ft^3)
Valves & Valve Actuators (primer)	Alkyd	4	98
Valves & Valve Actuators (topcoat)	Epoxy	5	112
Maintenance Truss	IOZ	4	220
Fans & Fan Housings	IOZ Epoxy Alkyd	4 5 4	220 112 98
Containment Tool Room Cabinets	Alkyd	2	112
MCC Panels	Epoxy	3	112
Fire Extinguishers	Epoxy	6	112
Azimuth Markers	Epoxy	1	112
Cable Rack Assemblies	Epoxy	1	112

The debris generation analysis also evaluated protected unqualified coatings applied to the SGs and pressurizer. The coatings are protected by the insulation covering each component. However, if a break were to destroy the overlying insulation inside the ZOI, it would also destroy the unqualified coatings underneath. As such, the protected unqualified coatings on these components were analyzed within a ZOI corresponding to their respective protective insulation types.

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3.h.2 Describe and provide bases for assumptions made in post-LOCA paint debris transport analysis.

Response to 3.h.2

The following assumptions related to coatings were made in the debris transport analysis:

1. It was conservatively assumed that the coatings particulate debris has a recirculation transport fraction of 100%. This is reasonable as particulate debris types are easily suspended in the recirculation pool.
2. It was assumed that all unqualified coatings outside the reactor cavity fail at the beginning of recirculation. This is conservative because it results in 100% of unqualified coatings outside the reactor cavity being present in the pool at the start of recirculation.
3. It was assumed that the pool fill-up transport fractions to inactive cavities and to the sumps for unqualified coatings is 0%. This is conservative because it results in 100% of this debris being present in the pool at the beginning of recirculation, resulting in higher overall transport fractions to the sump strainers.
4. Qualified coatings debris generated by a LOCA were assumed to transport to upper containment in proportion to the ratio of upper containment volume to the total containment volume. This is a reasonable assumption because fine debris generated by a LOCA would easily travel with the blowdown flow. Note that the break jet would not necessarily be directed toward upper containment. However, as the lower containment pressurizes, a significant portion of the blowdown flow would move toward upper containment.
5. All coatings debris blown to upper containment were conservatively assumed to be washed back down by the containment spray flow.
6. It was assumed that the protected Carboline 4674 unqualified coatings that are destroyed when the overlying insulation is destroyed would be in lower containment. This is a conservative assumption because this places the debris in the pool at the beginning of recirculation.
7. It was assumed that the unqualified coatings are uniformly distributed in the recirculation pool. This is a reasonable assumption because the unqualified coatings are scattered around containment.
8. It was assumed that qualified coatings debris is uniformly distributed in the pool, and that the percentage of particulate debris transported is equal between the two sumps for two train operation. Particulate debris is homogenously distributed throughout the pool, transports by remaining in suspension, and is not impacted by tumbling debris that can be swept along the floor. As particulate debris transports uniformly with the flow, and the flow rate of the two sump strainers are equal (for two train operation), this is a reasonable assumption. Note that for single train operation, 100% of particulate debris would transport to the active sump.

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3.h.3 Discuss suction strainer head loss testing performed as it relates to both qualified and unqualified coatings and what surrogate material was used to simulate coatings debris.

Response to 3.h.3

See the Response to 3.f.4 for detailed information on coating surrogates and the amount added to the test tank during head loss testing. Pulverized acrylic was used as surrogate for qualified and unqualified coatings particulate debris. Paint chips, which were processed using a food processor or blender, was used to simulate the coating chips debris. As stated in the Response to 3.h.5, all qualified and unqualified coatings fail as 10 μm particles.

3.h.4 Provide bases for the choice of surrogates.

Response to 3.h.4

See the Response to 3.f.4.

3.h.5 Describe and provide bases for coatings debris generation assumptions. For example, describe how the quantity of paint debris was determined based on ZOI size for qualified and unqualified coatings.

Response to 3.h.5

The following assumption related to coatings were made in the debris generation analysis:

1. Qualified epoxy coatings were analyzed with a 4.0D ZOI. This ZOI size has been previously accepted by the NRC (Reference 33). Qualified un-topcoated IOZ coatings were analyzed within a 10.0D ZOI (Reference 10 p. viii).
2. Qualified coatings within the ZOIs were assumed to fail completely as 10 μm diameter spheres; qualified coatings outside the ZOIs were assumed to remain intact. This is based on direction from NEI 04-07 (Reference 9, Table 3-3) and was found to be acceptable in the NRC SE on NEI 04-07 (Reference 10 p. 22)
3. Various qualified coating systems are used in the Wolf Creek containment, as summarized in Table 3.h.1-1. It was assumed that the '101' qualified coatings system was applied to steel structures, including columns, and equipment supports, and the '103' system was applied to all concrete surfaces within containment. Refer to the Response to 3.h.1 for the justification of this assumption.
4. The unqualified coatings on the SGs, reactor vessel, and pressurizer were assumed to be applied per the manufacturer's recommendations with two coats at 2 mils each.
5. The unqualified coatings are assumed to fail as 100% particulate with a characteristic diameter of 10 μm.

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6. All unqualified coatings for which documentation indicated an equal distribution in upper and lower containment were assumed to be located in lower containment. This is the most conservative approach because coatings in lower containment would be immediately available for transport.
7. Coatings data are often listed with a tolerance – for example solids by weight could be listed as 97% ± 2%. All coatings densities are calculated using their nominal values. This is reasonable because while some batches of coatings may be slightly more or slightly less dense than others, they should average out to the nominal density.
8. When characteristics (e.g., liquid coating density and percent solids by weight) for certain coatings are unavailable, the liquid coating density was estimated based on the shipping weight of a given volume of coating product. The percent solids by weight was conservatively assumed to be 100%.
9. The unqualified coatings on the reactor vessel head above the permanent cavity seal ring were assumed not to become debris for the postulated breaks. This is reasonable because the coatings are held in place by the reactor head insulation which is protected by the permanent cavity seal ring.

The generated amounts of qualified and protected unqualified coating debris vary with breaks. Table 3.h.5-1 shows the coatings debris loads for the four worst breaks that do not fail any of the strainer head loss or in-vessel acceptance criteria.

Table 3.h.5-1: Generated Qualified and Protected Unqualified Coatings Debris for Four Worst-Case Breaks that Do Not Fail Any Acceptance Criteria

Break Location	BB01-F406 (SG 1&4)	BB-01-S105-04 (SG 1&4)	BB01-F405 (SG 1&4)	BB-01-S003-2 (PRZR)
Break Size	10"	10"	10”	10”
Break Type	Partial @ 90º	Partial @ 0º	Partial @ 135º	Partial @ 0º
Qualified Epoxy (ft^3)	0.0	0.0	0.0	0.0
Qualified IOZ (ft^3)	0.14	0.17	0.12	0.28
Protected Unqualified Coatings (ft^3)	0.05	0.0	0.07	0.16

The generated quantities of the unqualified coatings are shown in Table 3.h.5-2. Note that these debris quantities are applicable for all postulated breaks inside the containment.

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Table 3.h.5-2: Generated Quantities of Unqualified Coatings

Coating Type	Quantity (ft^3)
Unqualified Epoxy	0.315
Unqualified Alkyd	0.566
Unqualified IOZ	0.360

3.h.6 Describe what debris characteristics were assumed, i.e., chips, particulate, size distribution and provide bases for the assumptions.

Response to 3.h.6

In accordance with the guidance provided in NEI 04-07 (Reference 9) and the associated NRC SE on NEI 04-07 (Reference 10), all qualified and unqualified coatings debris was treated as particulate and therefore transported entirely to the sump strainer (with the exception of qualified coatings transported to inactive cavities). See the Responses to 3.h.1, 3.h.2, and 3.h.5 for additional description of coatings debris characteristics.

3.h.7 Describe any ongoing containment coating condition assessment program.

Response to 3.h.7

Wolf Creek conducts condition assessments of coatings inside containment every refueling outage to assure that the coatings are still performing their intended design function and are still considered qualified. When the condition assessment or coating inspection identifies problem areas, a more comprehensive investigation is performed as necessary. Inspection reports are transmitted in a work order for engineering review, assessment, and acceptance. The periodic condition assessments and resulting repair and replacement activities assure that the amount of coatings that may be susceptible to detachment from the substrate during a LOCA event is minimized.

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3.i 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.

Provide the information requested in GL 2004-02 Requested Information Item 2(f) regarding programmatic controls taken to limit debris sources in containment.

> *GL 2004-02 Requested Information Item 2(f)*
>
> *A description of the existing or planned programmatic controls that will ensure that potential sources of debris introduced into containment (e.g., insulations, signs, coatings, and foreign materials) will be assessed for potential adverse effects on the ECCS and CSS recirculation functions. Addressees may reference their responses to GL 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," to the extent that their responses address these specific foreign material control issues.*

In responding to GL2004-02 Requested Information Item 2(f), provide the following:

3.i.1 A summary of the containment housekeeping programmatic controls in place to control or reduce the latent debris burden. Specifically for RMI/low-fiber plants, provide a description of programmatic controls to maintain the latent debris fiber source term into the future to ensure assumptions and conclusions regarding inability to form a thin bed of fibrous debris remain valid.

Response to 3.i.1

Wolf Creek procedure "Containment Entry and Material Control" establishes requirements for control of transient materials brought into containment. It also provides requirements for the control of trash and debris that can be generated as a result of maintenance activities. The procedure applies to all material brought into containment in all modes of operation. "Housekeeping Control" procedure addresses walkdowns to ensure that all debris is removed prior to STARTUP, and that dust, dirt, oil, etc. are removed from all accessible surfaces.

Wolf Creek procedure "Containment Inspection" documents the inspection requirements of the containment for any debris which could impair the recirculation sumps from performing their design functions. Additionally, Wolf Creek procedure "Containment Sump Inspection" documents the inspection requirements of the containment recirculation sumps to ensure that each sump inlet is not restricted by debris.

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3.i.2 A summary of the foreign material exclusion programmatic controls in place to control the introduction of foreign material into the containment.

Response to 3.i.2

The foreign materials exclusion program provides the guidance that the containment building is considered a system during plant modes 1 through 4 and refers to the containment entry and materials control procedure. The containment entry and materials control procedure provides guidance following containment entries during plant modes 1 through 4 to thoroughly clean the immediate work area and other areas where debris may have migrated during the work activity. A containment inspection surveillance is then conducted following containment entries during plant modes 1 through 4 to ensure cleanliness. During plant modes 5, 6, and defueled, the containment entry and materials control procedure also provides guidance for general containment cleaning.

3.i.3 A description of how permanent plant changes inside containment are programmatically controlled so as to not change the analytical assumptions and numerical inputs of the licensee analyses supporting the conclusion that the reactor plant remains in compliance with 10 CFR 50.46 and related regulatory requirements.

Response to 3.i.3

Wolf Creek has implemented the standard design process. Design Attribute Review is required during the development of an engineering modification to determine potential impacts on engineering disciplines, engineering programs, and stakeholders. The Design Attribute Review requires responses to various screening questions, including questions related to GSI-191 compliance. Selected questions pertaining to ECCS and CSS recirculation functions are listed below:

- Does the modification affect insulation?
- Does the modification add or remove components in containment?
- Does the modification change the amount of exposed aluminum and/or zinc in containment?
- Does the modification introduce materials that could affect sump performance or lead to equipment degradation?
- Does the modification repair, replace, or install coatings inside containment, including installing coated equipment?
- Does the modification affect installation, replacement, or storage of any structure, system, component or other items in containment that has vendor applied or site applied protective coatings?
- Does the modification affect high/moderate energy line break analysis?
- Does the modification affect the design, performance or operation of pumps?
- Does the modification affect foreign material that would require cleaning to prevent degradation of downstream components?

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Any “yes” answer to the above questions requires involvement of appropriate engineering disciplines for additional assessment.

In addition to the design change controls, Wolf Creek procedurally tracks all transient materials taken inside Containment. During normal operations, all items taken into Containment are logged. At the completion of the Containment entry the items are accounted for. At the completion of work activities inside Containment during normal operations, the work area is thoroughly cleaned and inspected (including the area below the work activity, if the work was performed on grating), prior to leaving Containment. Items left in Containment during normal operations are either stored in a container with a latchable door or cover, or secured to a structural member to prevent possible transport to the sumps.

3.i.4 A description of how maintenance activities including associated temporary changes are assessed and managed in accordance with the Maintenance Rule, 10 CFR 50.65.

Response to 3.i.4

Procedures are in place to control maintenance activities and evaluate temporary changes that have the potential to affect the debris source term.

The containment entry and material control procedures contain requirements for control of materials during work activities conducted in the containment building. Following maintenance activities in the containment building, procedures that control the containment cleanliness verification process specifically require both general area and target area cleaning.

Changes implemented as temporary alterations in support of maintenance that impact plant design are required to be developed in accordance with the same change procedures that are used for all plant modifications. As described in Section 3.i.3, the plant modification procedures contain administrative controls that specifically address potential impacts of debris on the ECCS performance.

3.i.5 If any of the following suggested design and operational refinements given in the guidance report (guidance report, Section 5) and SE (SE, Section 5.1) were used, summarize the application of the refinements.

a. *Recent or planned insulation change-outs in the containment which will reduce the debris burden at the sump strainers.*

   Response to 3.i.5.a

   There are no planned insulation change-outs that would reduce the debris burden at the sump strainers.

b. *Any actions taken to modify existing insulation (e.g., jacketing or banding) to reduce the debris burden at the sump strainer.*

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Response to 3.i.5.b

There are no planned actions to modify existing insulation (e.g., jacketing or banding) to reduce the debris burden at the sump strainers.

c. *Modifications to equipment or systems conducted to reduce the debris burden at the sump strainers.*

Response to 3.i.5.c

There are no planned modifications to equipment or systems to reduce the debris burden at the sump strainers.

d. Actions taken to modify or improve the containment coatings program.

Response to 3.i.5.d

There are no planned actions to modify the existing containment coatings to reduce the debris burden at the sump strainers.

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3.j Screen Modification Package

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

3.j.1 Provide a description of the major features of the sump screen design modification.

Response to 3.j.1

The currently installed sumps strainer consist of two PCI Sure-FlowTM strainers with one strainer in each sump pit. Each sump strainer is made up of 72 modules. The modules are arranged in a square matrix of 16 modules on each level, except for the bottom level that has only eight modules. Eight modules are seven plate/disks high and 64 modules are eleven plates/disks high. The interior of the disks contain wire stiffeners for support, made up of wires - 7 gauge and 8 gauge. The disks are completely covered with perforated plate having 0.045 inch diameter holes. The bottom disk of a module is separated 3 inches from the top disk of the adjacent module. The space between adjacent modules is encased in a sleeve connecting the central core tube of each module. Each module has cross-bracing on all four exterior vertical surfaces. The strainers are installed on a strainer substructure assembly, which is installed at the bottom of the containment recirculation sump pit. The strainers superstructure consists of four vertical supports on the 2000' elevation concrete pad. These supports are inside the sumps 6" concrete curb. A series of horizontal channels connect to the four vertical supports and provide lateral restraint for the module stacks. The strainers are robust so as to also serve as the trash racks, as described in a license amendment application and approved in the associated NRC safety evaluation. The materials for the strainer supports, both the lower support platform and the superstructure, are also stainless steel.

The sump strainers have 3311.5 ft^2 of effective surface area per sump that can handle the amount of debris generated and carried to the sumps. A significant design feature of the PCI Sure-Flow^{TM} strainers ensures uniform flow rate through all sections of the modules. This ensures that during post-accident operation, debris is not preferentially distributed to certain areas of the strainer. The approach velocity of the recirculation coolant flow at the sump strainer face will be less than 0.01 feet per second.

3.j.2 Provide a list of any modifications, such as reroute of piping and other components, relocation of supports, addition of whip restraints and missile shields, etc., necessitated by the sump strainer modifications.

Response to 3.j.2

Debris barriers have been installed in all openings through the secondary shield wall near the emergency recirculation sumps, specifically these were installed at the Loop A and Loop D passageway entrances. Furthermore, debris barriers were installed in

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drain trenches and other openings in the secondary shield wall near the sumps. The barriers prevent the flow of debris-laden fluid directly to the sumps and force the fluid to take a "long path" through shield wall openings farther away from the sumps. Using perforated plates, the debris barriers are designed to restrict passage of debris while allowing water to pass through the barrier. Blockage of intact pieces of fiberglass debris through the Loop A and Loop D passageways and other openings was included in the transport modeling. No other debris was assumed to be caught on these debris barriers.

As a result of the new strainer design, sump level indication was also replaced. The new instrumentation provides an indication of strainer differential pressure. The differential pressure measurement provides the control room operators a qualitative indication of how well the strainer is performing during the recirculation functions of the Emergency Core Cooling System and Containment Spray System following all postulated accidents for which the operation of these systems is required.

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3.k 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.

Provide the information requested in GL2004-02 Requested Information Item 2(d)(vii).

> *GL 2004-02 Requested Information Item 2(d)(vii)*
> *Verification that the strength of the trash racks is adequate to protect the debris screens from missiles and other large debris. The submittal should also provide verification that the trash racks and sump screens are capable of withstanding the loads imposed by expanding jets, missiles, the accumulation of debris, and pressure differentials caused by post-LOCA blockage under flow conditions.*

3.k.1 Summarize the design inputs, design codes, loads, and load combinations utilized for the sump strainer structural analysis.

Response to 3.k.1

Sump Strainer Structural Analysis

The Wolf Creek sump strainer structural qualification analysis evaluated the strainer modules as well as the supporting structures. There are two (Train A and Train B) ECCS sumps inside the Wolf Creek containment and each sump is equipped with its own strainer. Each strainer is composed of three parts. The strainer modules themselves are composed of individual modules and are bolted together in vertical stacks. These module stacks are supported by a steel framed substructure which acts as a flow plenum which is sealed against the sides of the sump pit walls. At the top the module stacks are supported in the lateral direction by a superstructure which is bolted to the concrete floor. This superstructure only supports the modules laterally and its weight is carried by the modules down to the substructure. The strainer was qualified using a combination of manual calculations generated in Mathcad, as well as finite element analyses using the GTSTRUDL software and the ANSYS software.

Applicable Strainer Codes

Some parts of the strainers (radial stiffeners, connecting rods, edge channels, seismic stiffeners, etc.) are classified as part of the support structure. The governing code for the qualification of the strainer is the Wolf Creek code of record, the American Institute of Steel Construction (AISC) “Manual of Steel Construction,” 7th Edition (Reference 34). Additionally, ANSI/AISC N690-1994,"Specification for the Design, Fabrication, and Erection of Steel Safety Related Structures for Nuclear Facilities" (Reference 35) is used to supplement the AISC in any areas related specifically to the structural qualification of stainless steel. The strainer also has several components made from

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thin gage sheet steel, and cold formed stainless sheet steel. SEI/ASCE 8-02, "Specification for the Design of Cold-Formed Stainless Steel Structural Members" (Reference 36) is used for certain components where rules specific to thin gage and cold form stainless steel should be applicable. This was further supplemented by the AISI Code (Reference 37) where the ASCE Code does not provide specific guidance. Finally, guidance was also taken from AWS D1.6, "Structural Welding Code - Stainless Steel" (Reference 38) as it relates to the qualification of stainless steel welds.

<u>Loads and Load Combinations for the Strainer</u>

The strainers are designed for the following load combinations:

- Seismic loads - The strainers are designed to meet Category I Seismic Criteria. Both the Operating Basis Earthquake (OBE) and the Safe Shutdown Earthquake (SSE) loads are developed from response spectra curves that envelope the response spectra curves for Wolf Creek. The structures are considered "Bolted steel structures" and the damping values for seismic loads are taken from Regulatory Guide 1.61 (Reference 39) as 4% for the OBE and 7% for the SSE.
- Live Loads - Live loads include the weight of the debris accumulated on the strainer and the differential pressure that the strainer and plenum members can withstand in the operating condition.
- Thermal Loads - Thermal expansion is considered in the design and layout of the structures. The strainers themselves are free to expand in the vertical direction as the structure is designed with a sliding connection allowing the strainer modules to expand upward without constraint. In the lateral direction, the seismic supports are gapped, leaving enough room to accommodate the thermal growth of the strainers and their supports without restraint. The design temperature for the strainers is 268°F, which is the maximum calculated containment sump water temperature during a large break LOCA.
- Hydrodynamic loads - Hydrodynamic loads on the strainers from the motion of the water surrounding the strainer during a seismic event were also considered.

The evaluation of the strainer was performed for the following load combinations:

Table 3.k.1-1: Load Combinations for the Strainer

Load Combination	Combination	Allowable
Load Combination 1	D + DP + DEB	1.0 S
Load Combination 2	D + E	1.0 S
Load Combination 2a	D + DP + DEB + Ew	1.5 S < Sy
Load Combination 4	D + DP + DEB + E'w	1.6 S < Sy

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where,

D	= Dead Weight Load
DP	= Differential Pressure Live Load (across a debris covered strainer)
DEB	= Debris Weight Live Load
E	= Operating Basis Earthquake
Ew	= Operating Basis Earthquake (including underwater earthquake effects)
E'w	= Safe Shutdown Earthquake (including underwater earthquake effects)
S	= The required section strength based on elastic design methods and the allowable stresses defined in the AISC "Manual of Steel Construction," 7th Edition. Other codes are also used for acceptance criteria for areas not covered under the AISC, such as special considerations associated with stainless steel, thin gage materials, and perforated plate. In addition, the structure is designed to maintain elastic behavior under all design loading conditions such that maximum stress is limited to the yield strength of the material.
Sy	= Yield Strength

DP is the pressure load during accident conditions when the strainers are covered with debris. This is the maximum allowable differential pressure that will not result in interaction ratios (IR) greater than 1.0 for the Cold Case. Conservatively, the density of water at temperature of 68°F is considered to represent equivalent water height of differential pressure:

DP = 5.5 ft or 2.38 psi   Cold Case at 175°F
DP = 4 ft or 1.73 psi   Hot Case at 268°F

Note that Load Combinations #1, #2, and #4 in Table 3.k.1-1 are considered the critical load combinations and are specifically evaluated. Load Combination #2a is bounded by Load Combination #4. Conservatively, the allowable stresses associated with Load Combination #2a are used for Load Combination #4 such that both load combinations can be enveloped by one analysis.

Stainless Steel Members in Compression

The allowable compression stress of the stainless steel members is based on the lower allowables from ANSI/AISC N690-1994, "Specification for the Design, Fabrication, and Erection of Steel Safety Related Structures for Nuclear Facilities" (Reference 35) as opposed to those provided in the AISC Code. Per Q1.5.9.2 of ANSI/AISC N690-1994, the allowable stresses for tension, shear, bending and bearing for stainless steel can be taken as the same allowables provided for carbon steel. The AISC 7th Edition was used for allowables for these types of stresses.

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Perforated Plates Combination

For the perforated plates, the equations from Appendix A, Article A-8000 of the ASME B&PV Code, Section III, 1974 Edition through Winter 1974 addenda (Reference 40) was used to calculate the perforated plate stresses. Note that Article A-8000 refers to Subsection NB for allowable stresses, which are defined in terms of stress intensity limits, Sm. However, since the strainers are Class II components, the allowable stresses are based on the lower ASME principal stress allowable of S. Therefore, the allowable stresses are taken from Table NC-3821.5-1 of the ASME B&PV Code (Reference 40) which is for steel tanks. This same table was adopted for vessels in later editions of the code.

Table 3.k.1-2: Load Combinations for the Perforated Plates and Allowables

Load Condition	Stress Type	Allowable Stress
Normal/Upset (Load Comb 1&2)	**Primary Membrane Stress** **Primary Local Membrane + Bending Stress**	**1.0 S** **1.5 S**
Emergency (Load Comb 4)	**Primary Membrane Stress** **Primary Local Membrane + Bending Stress**	**1.5 S** **1.8 S**

A structural and seismic evaluation was also performed on the instrument support elements associated with the new strainers.

3.k.2 Summarize the structural qualification results and design margins for the various components of the sump strainer structural assembly.

Response to 3.k.2

The Wolf Creek strainer structural analysis determined the structural design margins for the various components of the sump strainer assembly, as summarized in Table 3.k.2-1. A few conservatisms in the analysis are listed below:

- Results provided in the table below are based on the maximum debris weight rather than the debris weight associated with the threshold break size.
- The thermal stress at 268°F was used for all temperatures above 175°F, and the thermal stress at 175°F was used for all temperatures below 175°F. This combines the maximum thermal stress at high temperatures with the maximum head loss at lower temperatures.
- When analyzing the OBE load case, the allowable stress is 1.5 S. When analyzing the SSE case, the allowable stress is 1.6 S. For some of the components, the analysis combined those requirements and used the allowable stress of 1.5 S for the SSE load case.

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The interaction ratios for the components in the models are provided in Table 3.k.2-1. The Interaction Ratio is nominally "Actual" divided by "Allowable." Since seismic loads are present, most components are subject to loads in multiple directions acting simultaneously. The results of the calculation show that the interaction ratios for the strainer assembly components are below 1.0 for both Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) loads, and the strainers meet the acceptance criteria for all applicable loadings.

Table 3.k.2-1: Summary of Structural Analysis Results for Strainer Components

Strainer Component	Seismic Case	IR(1)	IR(2)	IR (max)
External Radial Stiffener (Including Collar and Plates)	OBE	0.15	0.17	0.17
	SSE	0.79	0.88	0.88
Tension Rods	OBE	0.43	0.48	0.48
	SSE	0.51	0.57	0.57
Spacers	OBE	0.71	0.79	0.79
	SSE	0.79	0.88	0.88
Edge Channels	OBE	0.1	0.109	0.109
	SSE	0.96	0.997	0.997
Cross Bracing Cables	OBE	0.08	0.09	0.09
	SSE	0.41	0.46	0.46
Hex Couplings	OBE	0.17	0.19	0.19
	SSE	0.69	0.76	0.76
Core Tube	OBE	0.03	0.03	0.03
	SSE	0.11	0.12	0.12
Substructure Angle Iron Support Legs	OBE	0.57	0.63	0.63
	SSE	0.71	0.79	0.79
Substructure Angle Iron Framing (including coped sections and angle braces)	OBE	0.98	0.79	0.98
	SSE	0.91	0.93	0.93
Substructure Channels (including coped sections)	OBE	0.92	0.82	0.92
	SSE	0.94	0.84	0.94
Cover Plates	OBE	0.33	0.36	0.36
	SSE	0.46	0.52	0.52
Superstructure Square Tubing Support Legs	OBE	0.19	0.21	0.21
	SSE	0.74	0.83	0.83
Superstructure Channels	OBE	0.13	0.14	0.14
	SSE	0.6	0.66	0.66
Perforated Plate (DP Case)	OBE	0.8	0.89	0.89
	SSE	0.66	0.73	0.73
Perforated Plate (Seismic Case)	OBE	0.31	0.34	0.34
	SSE	0.29	0.32	0.32
Perforated Plate (Inner Gap)	OBE	0.972	0.73	0.972

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Strainer Component	Seismic Case	IR(1)	IR(2)	IR (max)
	SSE	0.9996	0.76	0.9996
Wire Stiffener (4)	---	0.7	0.78	0.78
Perforated Plate (Core Tube End Cover DP Case)	OBE	0.7	0.78	0.78
	SSE	0.6	0.67	0.67
Perforated Plate (Core Tube End Cover Seismic Case)	OBE	0.07	0.08	0.08
	SSE	0.08	0.09	0.09
Radial Stiffening Spokes of the End Cover Stiffener	OBE	0.12	0.14	0.14
	SSE	0.11	0.12	0.12
Core Tube End Cover Sleeve	OBE	0.08	0.09	0.09
	SSE	0.05	0.06	0.06
Weld of Radial Stiffener to Core Tube	OBE	0.07	0.08	0.08
	SSE	0.31	0.35	0.35
Weld of mounting tabs to End Cover Stiffener	OBE	0.02	0.02	0.02
	SSE	0.01	0.02	0.02
Weld of End Cover Stiffener to End Cover Sleeve	OBE	0.07	0.07	0.07
	SSE	0.05	0.05	0.05
Edge Channel Rivets	OBE	0.05	0.06	0.06
	SSE	0.67	0.75	0.75
Inner Gap Hoop Rivets	OBE	0.09	0.1	0.1
	SSE	0.08	0.09	0.09
End Cover Rivets	OBE	0.03	0.03	0.03
	SSE	0.02	0.02	0.02
Connecting Bolts and Pins	OBE	0.5	0.55	0.55
	SSE	0.63	0.7	0.7
Mounting Pin Weld	OBE	0.43	0.48	0.48
	SSE	0.8	0.9	0.9
Substructure Sealing Plates (4)	---	0.99	0.8	0.99
Substructure Bolted Connections	OBE	0.76	0.49	0.76
	SSE	0.93	0.84	0.93
Substructure Welded Connections	OBE	0.46	0.51	0.51
	SSE	0.77	0.86	0.86
Substructure Post Jack Bolt and Baseplate	OBE	0.63	0.71	0.71
	SSE	0.71	0.79	0.79
Substructure Wall Jack Bolts	OBE	0.39	0.44	0.44
	SSE	0.39	0.43	0.43
Substructure Bolted Connections	OBE	0.08	0.09	0.09
	SSE	0.79	0.87	0.87

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Strainer Component	Seismic Case	IR(1)	IR(2)	IR (max)
Substructure Welded Connections	OBE	0.22	0.25	0.25
	SSE	0.82	0.92	0.92
Superstructure Expansion Anchors (3)	ShearX	0.15	0.16	0.16
	ShearY	0.88	0.98	0.98
	Tension	0.64	0.71	0.71
Superstructure Anchor Base Plate (3)	ShearX	0.16	0.18	0.18
	ShearY	0.64	0.72	0.72
	Tension	0.76	0.85	0.85
Superstructure Anchor Base Plate Stiffener Welds (3)	ShearX	0.18	0.2	0.2
	ShearY	0.76	0.85	0.85
	Tension	0.4	0.44	0.44

1. Results for the Cold Case at a temperature of 175°F
2. Results for the Hot Case at a temperature of 268°F
3. Worst case OBE /SSE for ShearX, ShearY, and Tension
4. The wire stiffeners and substructure sealing plates are not incorporated in the model except for mass. The seismically insensitive items are evaluated for differential pressure only. The wire stiffeners and substructure sealing plates are analyzed for loads endured immediately following the LOCA when only conventional debris is assumed and 24 hours after the LOCA when chemical precipitation is assumed to occur.

3.k.3 Summarize the evaluations performed for dynamic effects such as pipe whip, jet impingement, and missile impacts associated with high-energy line breaks (as applicable).

Response to 3.k.3

The recirculation sump strainers are installed inside the sump pits with approximately one foot extending above the containment floor. The sumps are located outside the secondary shield walls and are also protected by a concrete slab above (see Figure 3.e.1-2). The arrangement of the sump strainers protects them from missiles, pipe whip or jet impingement from a LOCA. Therefore, the loads associated with these dynamic effects do not need to be considered for the qualification of the strainers.

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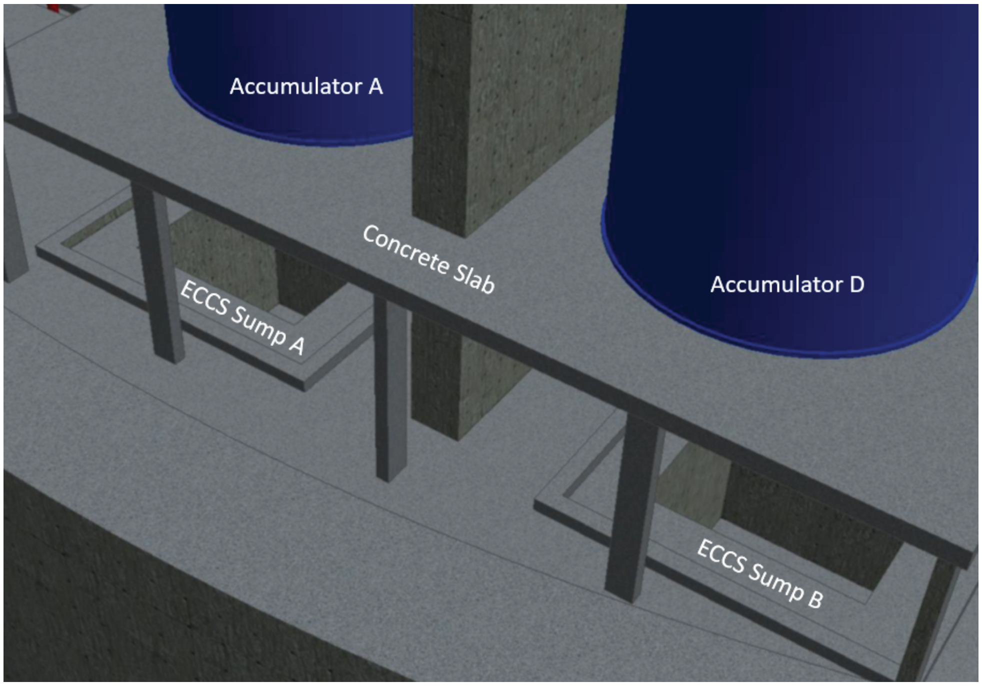

Figure 3.k.3-1: Significant Features in CFD Model

3.k.4 If a backflushing strategy is credited, provide a summary statement regarding the sump strainer structural analysis considering reverse flow.

Response to 3.k.4

Wolf Creek is not crediting a backflushing strategy for mitigating an excessive strainer head loss condition.

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3.I Upstream Effects

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

Provide a summary of the upstream effects evaluation including the information requested in GL 2004-02 Requested Information Item 2(d)(iv).

> *GL 2004-02 Requested Information Item 2(d)(iv)*
> *The basis for concluding that the water inventory required to ensure adequate ECCS or CSS recirculation would not be held up or diverted by debris blockage at choke-points in containment recirculation sump return flowpaths.*

3.I.1 Summarize the evaluation of the flowpaths from the postulated break locations and containment spray washdown to identify potential choke points in the flow field upstream of the sump.

Response to 3.I.1

The WCGS upstream effects evaluation includes an assessment of the WCGS containment geometry and transport pathways that containment spray flow and ECCS flow from the break will follow to the lower elevations of the containment building. The evaluation is based upon a review of WCGS design drawings and photographs of inside the containment building. Each elevation of the containment building was reviewed to identify the physical and structural features that affect the flow of debris and water to the lower elevations of the containment building. The containment building was divided into seven general compartments for individual evaluation, separated by grating, concrete walls, and concrete floors.

Upper Containment (including lay down area) (elevation 2068'-8" to dome elevation 2205'-0")

The overall area at this elevation is open with numerous areas of floor grating, which would allow water to pass through to the lower elevations unencumbered. There is one small area of concrete flooring near the pressurizer valve rooms, but water in this location will flow to the grated or open areas surrounding it. This area is open inside and outside the secondary shield walls down to the operating floor at elevation 2047'-6". Standpipes downstream of the containment cooler condensation overflow lines are accounted for as a miscellaneous holdup volume in the containment water level analysis. No potential choke points or hold-up points were identified in this area.

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Operating Floor (elevation 2047'-6")

A hold-up point in this area of the containment building is the reactor head storage and decontamination area. Water collected on the head stand will drain to the surrounding floor but water will be retained by a curb surrounding the head stand. The area inside the curb has a 4" floor drain that directs water to a common drain header and then to the drain trenches at the ground floor elevation. However, the drain could become plugged with debris and is not considered functional for this evaluation.

Annulus and Inside Secondary Shield (elevation 2026'-0")

Major equipment and features in this area include the main steam and feedwater lines, the tops of the A and D safety injection accumulators, several HVAC openings, and a compartment for the letdown orifices, the top of which is located at elevation 2036'-0". The northern, northwestern, and southwestern sides of the annulus have mostly concrete floors while the rest of the elevation outside the secondary shield is grated. There are no curbs associated with the concrete floors, so water inventory will flow to the lower elevations without holdup. No potential choke points or hold-up points were identified at this elevation.

Refueling pool (elevation 2009'-9" and elevation 2007'-2")

The refueling pool floor (elevation 2009'-9") contains two 10" drains that are sealed with flanges during refueling operations and are completely open during power operations. There are debris exclusion devices (trash rack cages) installed during power operations to prevent the drain from becoming a choke point. The containment water level analysis accounted for the holdup by the drain flange and due to the hydraulic head required for draining the pool. There is an upending pit below the refueling pool floor elevation at elevation 2007'-2". The drain in the upending pit is a 4" line which is normally isolated with a normally closed valve. This area is a hold-up point that would retain water inventory following a postulated design basis accident (post-DBA). The holdup volume by the upending pit was accounted for in the containment water level analysis.

Ground Floor Inside Secondary Shield (elevation 2001'-4")

There are only four significant openings through which post-DBA recirculation water may pass through the secondary shield wall. These passageways provide personnel and equipment access through the secondary shield wall in an area near each of the four reactor coolant pumps (RCPs), and include steps to transition from the 2001'-4" floor elevation inside the secondary shield wall to the 2000'-0" floor elevation outside the wall. Three of the four openings are approximately six feet wide. The fourth opening, entering under the pressurizer near the "D" loop RCP, is approximately three feet wide. As shown on Figure 3.I.1-1, the opening near the "A" loop RCP is shown to the left of the sump pits and the opening near the "D" loop RCP is shown to the right of the sump pits.

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Additionally, there is a system of small drain trenches, approximately one foot wide by one foot deep, that surround the primary shield wall and transfer drain water to outside the secondary shield wall. Trenches and drain piping outside the secondary shield walls direct drainage to the normal containment sumps (which are not part of the ECCS system) located in the containment ground floor annulus at elevation 2000'-0". Since the containment flood level will exceed the floor elevation inside the secondary shield wall, the trench system is expected to transport water to the containment annulus. Debris barriers have been installed in the loop "A" and loop "D" passageway entrances through the secondary shield wall, which are near the containment recirculation sumps. Debris barriers have also been installed in the portions of the drain trenches and other openings in the secondary shield wall that are near the recirculation sumps. A portion of the drain trench in the containment annulus region can be seen in Figure 3.I.1-1, with a trench opening through the secondary shield wall just to the right of the loop "D" passageway. Using perforated plates with the same hole size as the sump strainers, the debris barriers are designed to restrict passage of debris while allowing water to pass through the barrier. The barriers prevent the flow of debris laden fluid directly to the sumps and force the fluid to take a "long path" through shield wall openings farther away from the sumps. The remaining two open six foot wide passageways through the secondary shield wall will transport the ECCS break flow and CSS flow from inside the secondary shield to the containment annulus without restriction. In addition, the remaining trenches penetrating the secondary shield wall will also pass a significant quantity of water from inside the secondary shield wall to the containment annulus. Given these large passageways and large total trench length, large debris or mounds of debris would not create a choke point or hold-up point preventing the recirculation fluid from transporting to the sump.

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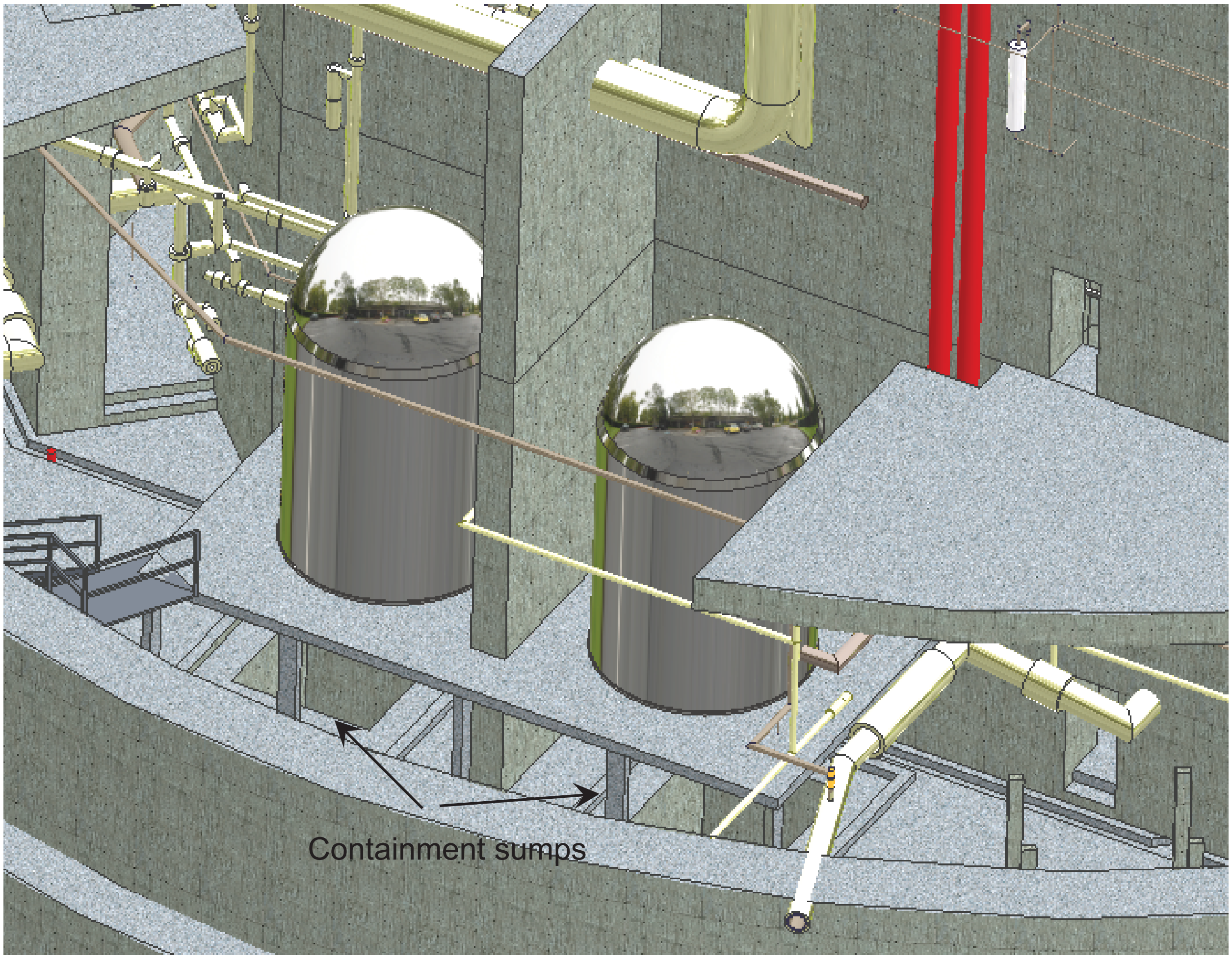

Figure 3.I.1-1: Isometric View of Sump Pit Area

<u>Ground Floor, Annulus (elevation 2000'-0")</u>

The containment building emergency recirculation sumps are located in this annular region between the secondary shield wall and the containment wall, as shown in Figure 3.I.1-1. A six inch curb surrounds each sump pit, creating a six inch deep hold-up volume above the 2000'-0" floor elevation. As discussed above, the normal containment sumps receive water flow from the drain trenches and piping in this area. This represents an additional hold-up volume below the 2000'-0" floor elevation. Given the large flow passages in the annulus region, significant mounds of debris would not create a choke point preventing the recirculation fluid from transporting to the sump.

<u>Reactor Cavity and Instrumentation Tunnel and Sump (elevation 1970'-6")</u>

This area of evaluation encompasses the area under the reactor vessel in the reactor cavity as well as the incore instrumentation tunnel. Post-DBA water inventory flow to this area will come from the elevation 2001'-4" hatch north of the primary shield wall when the flood height exceeds 2001'-10" due to the protective 6" curb. In addition,

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flow to this area will also come from the permanent cavity seal ring access covers. This tunnel and area under the reactor cavity will retain water inventory during post-DBA recirculation mode operations. No potential choke points were identified in this area.

3.I.2 Summarize measures taken to mitigate potential choke points.

Response to 3.I.2

Administrative controls ensure the drains from the refueling cavity to lower containment are not obstructed during power operations.

3.I.3 Summarize the evaluation of water holdup at installed curbs and/or debris interceptors.

Response to 3.I.3

As discussed above, a six inch curb surrounds each containment recirculation sump pit, creating a six inch deep hold-up volume above the 2000'-0" floor elevation. Debris barriers installed in the loop "A" and loop "D" passageway entrances through the secondary shield wall do not impact water hold-up since loop "B" and loop "C" passageways allow debris laden fluid to flow into the containment building annulus area and to the recirculation sumps.

3.I.4 Describe how potential blockage of reactor cavity and refueling cavity drains has been evaluated, including likelihood of blockage and amount of expected holdup.

Response to 3.I.4

The refueling pool floor (elevation 2009'-9") contains two 10 inch diameter drains that are open during power operations. There are debris exclusion devices (trash rack cages) installed during power operations over each of the 10 inch drains to prevent large pieces of debris from plugging the drains. The trash rack cages measure 33" x 33" x 15" with 5" openings. Figure 3.I.4-1 shows a trash rack cage sitting on the refueling pool floor near the 10" drain, which has its blind flange installed for non-power operations (refueling preparations). The 10 inch drains go straight through the refueling cavity floor slab and discharge into the open area below; thus, the drain pipes themselves would not become plugged with debris. Administrative controls ensure the drains from the refueling cavity to lower containment are not obstructed during power operations.

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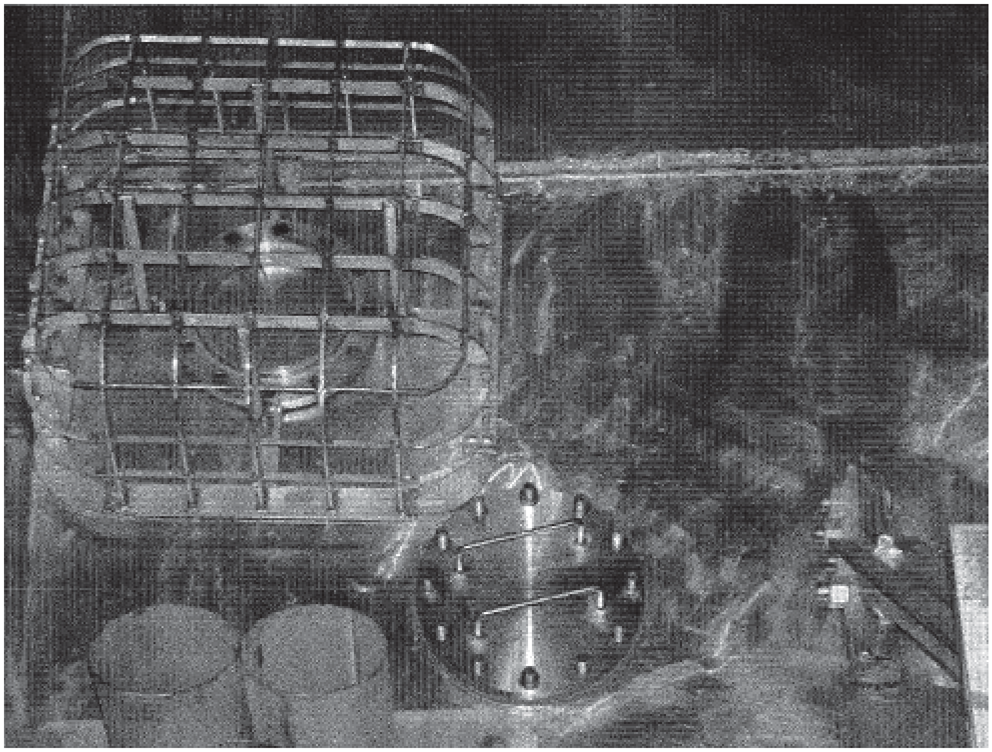

Figure 3.I.4-1: Refueling Pool Trash Rack (Callaway pictured but similar at Wolf Creek)

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3.m 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. Provide the information requested in GL 04-02 Requested Information Item 2(d)(v) and 2(d)(vi) regarding blockage, plugging, and wear at restrictions and close tolerance locations in the ECCS and CSS downstream of the sump.

GL 2004-02 Requested Information Item 2(d)(v)
The basis for concluding that inadequate core or containment cooling would not result due to debris blockage at flow restrictions in the ECCS and CSS flowpaths downstream of the sump screen (e.g., a HPSI throttle valve, pump bearings and seals, fuel assembly inlet debris screen, or containment spray nozzles). The discussion should consider the adequacy of the sump screen's mesh spacing and state the basis for concluding that adverse gaps or breaches are not present on the screen surface.

GL 2004-02 Requested Information Item 2(d)(vi)
Verification that the close-tolerance subcomponents in pumps, valves and other ECCS and CSS components are not susceptible to plugging or excessive wear due to extended post-accident operation with debris-laden fluids.

3.m.1 If NRC-approved methods were used (e.g., WCAP-16406-P with accompanying NRC SE), briefly summarize the application of the methods. Indicate where the approved methods were not used or where exceptions were taken, and summarize the evaluation of those areas.

Response to 3.m.1

In order to evaluate the wear on the equipment within the ECCS and CSS recirculation flow paths, the wear models developed in WCAP-16406-P-A, Rev 1 (Reference 41), were used without exceptions. Unapproved methods were not used in the evaluations.

The erosive wear rate developed for annealed steel in the WCAP was applied to the equipment, which included pumps, heat exchangers, orifices, and spray nozzles. This wear rate was dependent upon the mass concentration of the debris that passes through the sump screen and enters the recirculation flow path, the mass fraction multiplier, the hardness of the material being eroded as compared to the hardness of carbon steel, and contained a restriction on the flow velocity. If the velocity was greater than 15 feet/sec, the erosive wear rate was accelerated.

For pumps, abrasive wear was also considered. The rotor dynamics for multistage ECCS pumps is affected by the wear ring clearances on the suction and discharge side of the pump. These clearances experience abrasive wear due to the debris in the

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pumped fluid. WCAP-16406-P-A, Rev 1 (Reference 41), provided two wear models to calculate the amount of wear and associated clearance increase of the two running clearances. The first was the "free flowing abrasive wear model", and the second model is the Archard abrasive wear model, which addressed packing type wear.

The hydraulic performance of the ECCS and CS pumps were evaluated by determining the impact of wear on the pump internals and on head and flow from the pump performance curve. Per WCAP-16406-P-A, Rev 1 (Reference 41), the increased internal to external leakage of the pump fluid due to wear does not impact the required NPSH, so only the impact on the flow must be evaluated. Per this WCAP, all pumps must undergo this hydraulic evaluation, which is based on the minimum pump performance curve. If a pump met the following criteria, then no further hydraulic evaluation was required:

1. Hydraulic flow margin positive at beginning of containment recirculation
2. Wear ring material ≥ 400 BHN
3. Impeller hub material ≥ 400 BHN

If any of the above criteria were not satisfied, the change in the pump wear ring gap due to abrasive wear had to be calculated and the resulting reduction in the pump discharge flow evaluated. Then as long as positive flow margin existed, no further evaluation was required. Table 3.m.1-1 shows the results of this evaluation.

Table 3.m.1-1: Hydraulic Performance Evaluation Results

Pump	Hydraulic Flow Margin	Wear Ring Material	Impeller Hub Material	Evaluation Required
RHR	Positive	311 BHN	470 BHN	Yes
SI	Positive	496 BHN	470 BHN	No
CCP	Positive	496 BHN	470 BHN	No
CS	Positive	264 BHN	300 BHN	Yes

Per WCAP-16406-P-A, Rev 1 (Reference 41), as long as the resultant wear gap clearance, including the effects of normal, abrasive and erosive wear was within two times the initial design clearance, no further evaluation was required. From this WCAP, the change in the wear ring gap due to normal wear was assumed to not exceed 3 mils.

The RHR and CS pumps above did not meet the criteria for wear and impeller hub materials hardness greater than 400 BHN, so a wear evaluation was completed for these pumps. For these pumps, the increased clearance due to the erosive and abrasive wear is less than two times the design clearance. Therefore, the hydraulic performance of the pumps, will not be affected by the sump debris. Table 3.m.1-2 provides the evaluation data.

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Table 3.m.1-2: Hydraulic Performance Evaluation for RHR and CS Pumps

Pump	Normal Wear (mils)	Erosive Wear (mils)	Abrasive Wear (mils)	Design Clearance (mils)	Increased Clearance (mils)	2X Design Clearance (mils)
RHR	3.0	0.006	0.027	28	31.033	56
CS	0.0	0.006	0.024	23	23.03	46

3.m.2 Provide a summary and conclusions of downstream evaluations.

Response to 3.m.2

Debris Blockage

The new containment recirculation sump strainers are covered with perforated plate with nominal 0.045 inch (+/- 0.002 inch) openings, or a maximum opening of 0.047 inches. An evaluation pertaining to the potential blockage in components downstream of the strainers was performed. This evaluation utilized the following assumptions, consistent with WCAP-16406-PA, Rev 1 (Reference 41), pertaining to debris size:

- The width of deformable particulates that may pass through the sump strainer is limited to the size of the flow passage hole in the sump strainer, plus 10%.
- The thickness of deformable particulates that may pass through the sump strainer is limited to one-half the size of the flow passage hole.
- The maximum length of deformable particulates that may pass through the flow passage hole in the sump strainer is equal to two times the diameter of the passage hole.
- The thickness and/or width and maximum length of non-deformable particulates that may pass through the sump strainer is limited to the size of the flow passage hole in the sump strainer.

The short-term and long-term (cold leg and hot leg) alignments for the ECCS and CSS were reviewed to ensure that all of the flow paths and components impacted by the debris passing through the sump strainers were considered. The methodology developed evaluated whether the system valves, piping, instrument tubing and heat exchangers could be susceptible to blockage from the debris that passes through the sump strainers. The conclusion from this evaluation was that all of the ECCS and CSS components evaluated can accommodate sump bypass particles without blockage.

Debris Ingestion

The concentration of debris in the recirculating fluid that passes through the sump is characterized in terms of lbm per lbm (lbm/lbm). For downstream effects, the debris concentration of the individual debris is defined as the ratio of the solid mass of the debris in the pumped fluid to the total mass of water that is being recirculated by the ECCS and CSS. The debris amounts are based on the highest transported debris

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amounts. All particulate/coatings that transport to the strainer are assumed to penetrate the strainer. The individual concentrations for WCGS are:

- Particulate concentration at ECCS switchover = 1.118×10^{-3} lbm/lbm
- Particulate concentration at CSS switchover = 7.973×10^{-4} lbm/lbm
- Chemical debris concentration = 1.976×10^{-4} lbm/lbm
- Maximum fiber concentration = 4.387×10^{-4} lbm/lbm

This data was used for the wear evaluations.

Erosive Wear

WCGS heat exchangers, orifices and spray nozzles were evaluated for the effects of erosive wear for a constant debris concentration over the mission time of 30 days. The erosive wear on these components was determined to be insufficient to affect the system performance. Debris depletion was not utilized for components experiencing only erosive wear in this analysis note because the evaluation passed with constant debris concentration.

Pumps

For pumps, the effect of debris ingestion through the sump strainers on three aspects of operability (including hydraulic performance, mechanical shaft seal assembly performance and mechanical performance (vibration) were evaluated per WCAP-16406-P-A, Rev 1 (Reference 41). The hydraulic and mechanical performances of the pumps were determined to be unaffected by the recirculating sump debris. The mechanical shaft seal assembly performance evaluation resulted in two action items:

- The evaluation of cyclone separators
- The evaluation of the pumps' carbon/graphite backup seal bushings

The CS pumps at WCGS have cyclone separators. A wear evaluation of the cyclone separator was performed. This evaluation concluded that, considering both the small dimensions of the fibrous debris and the low concentration of fibrous debris, the ports of the cyclone separators would not clog or block due to fibrous debris. Additionally, the primary seals were not expected to fail as a function of debris.

Valve Wear

WCGS has 12 throttle valves in the ECCS System, which were evaluated against established wear criteria, per WCAP-16406-P-A, Rev 1, Section 8.2.2 (Reference 41). This evaluation determined that all 12 valves analyzed passed the acceptable criteria under their current positions. Therefore, they are unaffected by wear erosion. Other valves within the system were evaluated per WCAP-16406-P-A and were found to not require a detailed evaluation for wear, as it was determined that they were not affected by the debris loaded fluid during the recirculation mode of operation.

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3.m.3 Provide a summary of design or operational changes made as a result of downstream evaluations.

Response to 3.m.3

It was determined that no design or operational changes were needed as a result of the downstream evaluations.

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3.n 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 core cooling.

3.n.1 Show that the in-vessel effects evaluation is consistent with, or bounded by, the industry generic guidance (WCAP-16793), as modified by NRC staff comments on that document. Briefly summarize the application of the methods. Indicate where the WCAP methods were not used or where exceptions were taken, and summarize the evaluation of those areas.

Response to 3.n.1

Wolf Creek performed fiber penetration testing and used the test results in the in-vessel downstream effects analysis. The analysis followed the latest NRC staff review guidance (Reference 42) and pressurized water reactor owners group (PWROG) guidance (Reference 43), and used the methodology and acceptance criteria in WCAP-17788-P, Revision 1 (Reference 44; 45). The analysis concluded that post-accident long-term core cooling (LTCC) will not be challenged by accumulation of debris within the reactor core for all postulated LOCAs up to 10 inches. Therefore, the in-vessel threshold break size is 10 in. A summary of the fiber penetration testing and in-vessel analyses is provided below. Note that LOCADM analysis is omitted from the submittal since the NRC review guidance concluded that "review of plant-specific LOCADM analyses is not necessary to reasonably assure compliance with LTCC requirements" (Reference 42 p. 2).

<u>Fiber Penetration Testing</u>

Wolf Creek conducted fiber penetration testing in 2016 at Alden Research Laboratory (Alden). The purpose of the testing was to collect time-dependent fiber penetration data of the prototypical strainer. One large-scale fiber-only penetration test was conducted with test parameters selected to be representative of the most conservative plant strainer configuration and post-accident conditions (e.g., debris characteristics and composition, flow rate, and water chemistry), as detailed later in this section. The test results were used to derive a curve-fit to quantify fiber penetration at plant conditions. The penetration test is described in the sections below.

Test Loop Design

The test loop used for fiber penetration testing is similar to that for head loss testing (see the Response to 3.f.4). The closed test loop included a metal test tank with acrylic windows that housed a test strainer submerged in water. For fiber penetration testing, test water was circulated by a pump through the test strainer, a fiber filtering system, and various piping components. The test tank consisted of two parts: the rectangular upstream portion for debris introduction and mixing, and the pit region where the test strainer was installed, as shown in Figure 3.f.4-1. Debris was introduced at the upstream end of the test tank, away from the test strainer. The upstream portion of the test tank was equipped with hydraulic mixing lines to create adequate turbulence in order to prevent the debris from settling before reaching the test strainer. The turbulence level was controlled to keep fiber in suspension without disturbing the fiber bed on the strainer. The pit region of the test tank was designed such that the spacing between the test strainer and the surrounding acrylic walls conservatively models the gaps between adjacent strainer modules and between the strainers and the sump pit walls at the plant.

The test loop used for penetration testing was similar to that used for head loss testing (described in the Response to 3.f.4) with the following exceptions:

1. At least one of the two in-line filters was always online during the fiber penetration test. This ensured that fiber debris that passed through the test strainer was collected inside filter bags.
2. For the pit region of the test tank, the gap between the tank walls and exterior sides of the strainer stacks was increased from 2 in used during head loss testing to the full gap width of 4 in. The 2 in gap used during head loss testing modeled the symmetry boundary between adjacent strainer modules. The 4 in gaps used during penetration testing minimized the potential for fiber bridging between the test strainer and tank walls to allow fiber to travel through the gaps and reach the strainer perforated plates. This is conservative for the purpose of fiber penetration testing.

Test Strainer

The test strainer for penetration testing was comprised of two prototypical strainer stacks that matched the key design parameters of the plant strainer stacks. The gap width between the two stacks was maintained at approximately 4 in, consistent with the plant strainer design. Overall, the test strainer was similar to that used in head loss testing (as described in the Response to 3.f.4) with a few modifications to promote fiber penetration. Every other disk of the strainer modules was removed for penetration testing, resulting in 22 disks per test strainer stack, rather than the 40 disks per stack used for head loss testing. This modification more than doubled the gaps between adjacent disks to prevent a fiber bridge from developing across adjacent disks. This conservatively promoted fiber penetration by allowing more fiber to enter the gaps and reach the strainer perforated plates. The core tube for the test strainer was also modified to cap the slots corresponding to the removed disks and to maintain the ratio of penetrable disk area to penetrable core tube area. Additionally, the seismic cables, which are part of the plant strainer design, were removed from the test strainer modules to avoid fiber debris being captured by the cables.

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Debris Types and Preparation

Nukon was the only fiber debris type used in penetration testing. This is appropriate because the only types of fibrous debris generated in containment for the bounding breaks of each break size are LDFG, Thermo-lag fiber, and latent fiber, which all have similar characteristics to Nukon.

Note that particulate debris was not used for penetration testing. This is conservative for penetration testing because as a fiber bed forms on the strainer, introduction of particulate would serve to hasten bed formation and inhibit further fiber penetration. This could also increase pressure drop across the debris bed which serves to further compress the fiber bed and reduce penetration.

All Nukon fiber was prepared as fines according to the NEI protocol (Reference 47) following the same procedures used for the head loss tests, described in the Response to 3.f.4. Preparation of Nukon debris was performed as follows: Nukon sheets, with an overall thickness of 2 in, were baked single-sided until the binder burnout reached into approximately half the thickness. The heat-treated sheets were then cut into approximately 2 in x 2 in cubes and weighed out according to batch size. Nukon was then pressure-washed with test water following the NEI protocol to create a debris slurry consisting predominantly of Class 2 fine fibers, as defined in NUREG/CR-6224 (Reference 29 pp. B-16). Figure 3.n.1-1 shows the prepared Nukon fines after pressure washing.

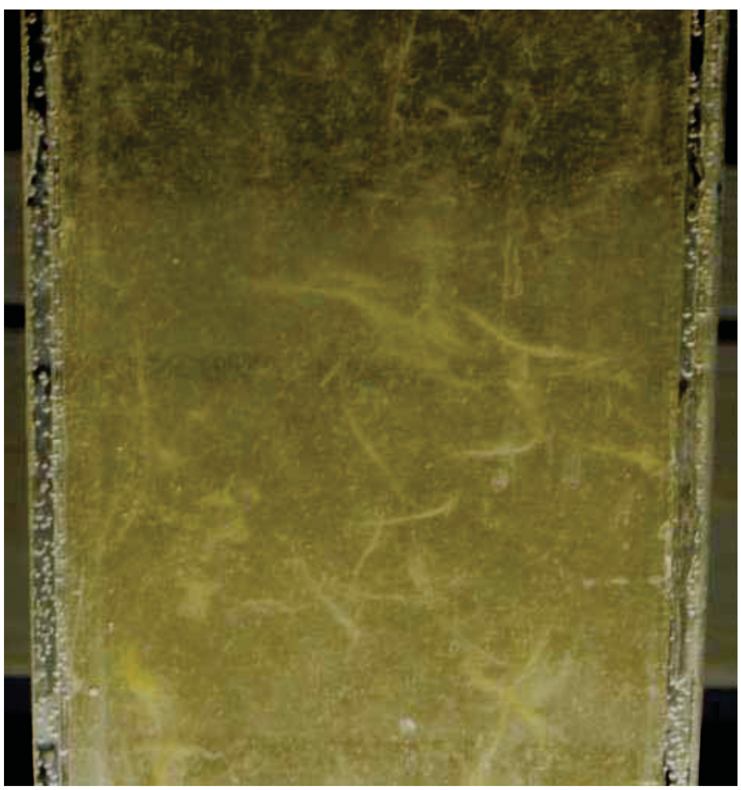

Figure 3.n.1-1: Nukon Fines Prepared for Wolf Creek Penetration Testing

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Debris Introduction and Transport

Prepared Nukon debris was introduced in seven separate batches. The first two batches each had a theoretical uniform bed thickness of 1/16 in. The third through sixth batch each had a theoretical uniform bed thickness of just over 1/8 in. The last batch had a theoretical uniform bed thickness of just over 1/16 in. Although the batching size increased after the first two batches, this had little effect on the test results because the fiber concentration was maintained lower than that expected at the plant by controlling the addition time of each batch. This approach promoted penetration by slowing down bed development during testing, compared with plant conditions. The total tested fiber load bounded the largest fiber load for all of the postulated breaks.

Debris was added to the test tank through a hopper. The prepared debris slurry was transferred from the preparation barrel to the hopper using 5-gallon buckets. As discussed above, for each batch, the introduction was timed to achieve a prototypical debris concentration in the test tank. During this process, the debris slurry inside the barrel was stirred to promote a homogeneous mixture. Additionally, the debris was stirred, as necessary, to break up any fiber agglomeration in the hopper and in the test tank.

After the introduction of each batch, transportation of fiber from the debris hopper into the tank and to the strainer was verified. Any non-transported fiber was collected, and later dried and quantified. The total non-transported fiber for the entire test amounted to less than 1 gram (vs. total tested fiber load of >13000 grams). Therefore, no fiber settling was credited.

Collection of Debris Penetration

Fiber can penetrate through the strainer by two different mechanisms: prompt penetration and shedding. Prompt penetration occurs when fiber reaching the strainer travels through the strainer immediately. Shedding occurs when fiber that already accumulated on the strainer migrates through the bed and ultimately travels through the strainer. Both mechanisms were considered during the Wolf Creek fiber penetration testing.

Fibers that passed through the strainer were collected by the in-line 5-micron filters downstream of the test strainer and upstream of the pump. All of the flow downstream of the strainer travelled through 5-micron filter bags installed inside filter housings before returning to the test tank. The filtering system allowed the installation of two sets of filter bags in parallel lines such that one set of filter bags could be left online at all times, even during periods in which filter bags were swapped.

Before each test, all of the filter bags required for the test were uniquely marked and dried, and their weights were recorded. After testing, the debris-laden filter bags were rinsed with deionized (DI) water to remove residual chemicals before being dried and weighed. The weight gain of the filter bags during testing was used to quantify fiber penetration. When processing the filter bags, in either a clean or debris laden state, the bags were placed in an oven for at least an hour before being cooled and weighed inside a humidity-controlled chamber. This process was repeated for each bag until two consecutive bag weights were within 0.10 g of each other.

The capture efficiency of the 5-micron filter bags used for fiber penetration testing was verified to be above 99.4%. The verification process involved adding a set quantity of dry Nukon fine fiber to processed clean filter bags. The debris-laden filter bags were then soaked in DI water, rinsed and dried, before being weighed. The capture efficiency was then quantified by comparing the weight gain of the filter bag with the known fiber weight. At least one filter bag with Nukon fiber was soaked in borated water, dried, weighed, soaked in DI water, then dried and re-weighed. This process helped determine a minimum soaking/rinsing time to remove the dissolved boric acid from the debris-laden fiber bag post-testing. The high capture efficiency of the filter bags indicates that <0.6% of fiber that passed through the strainer may not be captured by the filter bags during testing. This amount is insignificant and is bounded by the conservatisms in the analyses.

Before introducing a new debris batch into the test tank, a clean set of filter bags were placed online and were left online for a minimum of three pool turnovers (PTOs) to capture the prompt fiber penetration. Afterwards, at least one additional filter bag set was used for each batch to capture the fiber penetration due to shedding. For Batches 2 and 7, an additional set of filter bags was used to capture long-term shedding data. The bag sets used for shedding penetration were each online for at least 30 minutes. This approach allowed the testing to capture time-dependent fiber penetration data, which was used to develop a model for the rate of fiber penetration as a function of time and fiber quantity on the strainer.

Test Parameters

The test water used for fiber penetration testing had a chemical composition prototypical to Wolf Creek. The plant condition selected for testing was that of the minimum boron concentration of 2117 ppm and buffer (NaOH) concentration of 4.5797 g/L. This water chemistry corresponds to the maximum sump pH condition at Wolf Creek. Test water was prepared by adding pre-weighed chemicals to DI water per the prescribed concentrations.

The fiber penetration test flow rate was determined based on an approach velocity of 0.0061 ft/s, which was determined from the maximum plant recirculation flow rate of 9100 gpm per strainer and a surface area of 3311.5 ft^2 per strainer (see the Response to 3.f.4). This strainer flow rate was derived by adding margin to the total flow rate of 8,710 gpm, which is a combination of the maximum RHR and CS pump flow rates of 4,760 and 3,950 gpm, respectively (see the Response to 3.g.1). It should be noted that the Wolf Creek strainer is the PCI “Sure-Flow®” strainer with a uniform flow distribution. During testing, the pump flow rate was maintained within –0/+5% of the target flow rate.

The strainer submergence was allowed to range between 14.0 in and 18.1 in, which corresponds to the strainer submergence range for a LBLOCA between ECCS and CS switchover.

The test water was maintained at approximately 120°F for the duration of the test.

Strainer Penetration Curve-Fit Development

Data gathered from the fiber penetration test was used to develop two curve-fits for quantifying the strainer fiber penetration under plant conditions: a low-fiber curve and a high-fiber curve. The low-fiber curve was derived using test data from the first 4 batches and is applicable for breaks with transportable fine fiber loads up to 257.9 kg (or 568.6 lbm) per strainer. The high-fiber curve was derived using test data of all batches and is applicable for breaks with transportable fine fiber loads ranging between 387.7 kg (or 854.7 lbm) and 484.6 kg (or 1068.4 lbm) per strainer. The models were developed per the following steps:

- General governing equations were developed to describe both the prompt fiber penetration and shedding through the strainer as a function of time and fiber quantity on the strainer. The equations contain coefficients whose values were determined based on the test results.
- The test results of interest were curve fit to the governing equations developed in the previous step using various optimization techniques to refine the coefficients. This produced a unique set of equations as the penetration curve.

As shown later in this section, only the low-fiber penetration curve was used for the in-vessel effects analysis. Therefore, Figure 3.n.1-2 compares the fiber penetration results of the test (shown as circles) with the fiber penetration quantities determined by applying the low-fiber curve to the test conditions (shown as blue solid line). As stated above, the low-fiber curve was based on test data of the first 4 batches only. Therefore, the comparison between the test results and fitted curve should only be done for time up to approximately 22,000 seconds in Figure 3.n.1-2. As shown in the figure, the model results adequately represent the test data.

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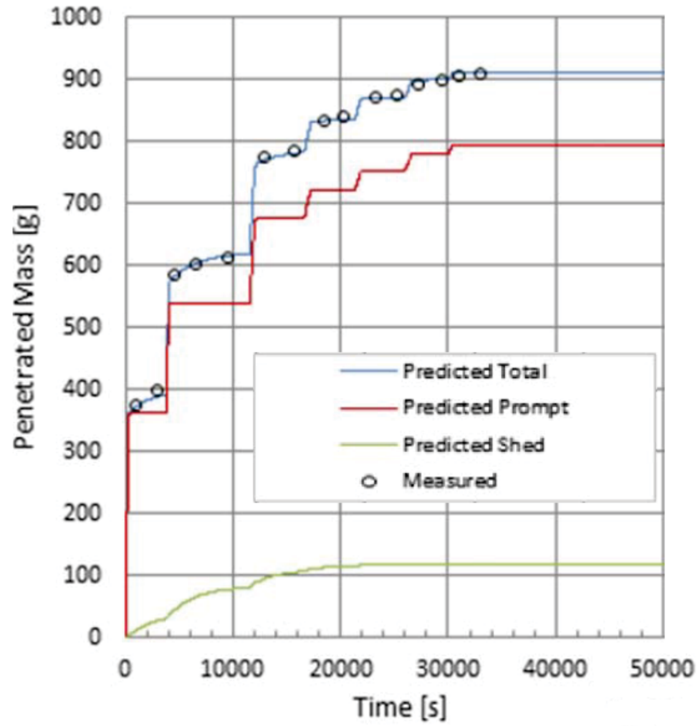

Figure 3.n.1-2: Wolf Creek Penetration Model Fit

The penetration curve was then used to determine the prompt fiber penetration fraction and shedding fraction for a given time and amount of fiber accumulated on the strainer. Coupled with a fiber transport model, an example time-dependent evaluation was performed to quantify the total amount of fiber that could pass through the strainer under plant conditions, as shown below. For the time-dependent analysis, the recirculation duration was divided into time steps. For each time step, the fiber penetration rates and quantities were calculated. Figure 3.n.1-3 shows the resulting cumulative fiber penetration through the strainer over time.

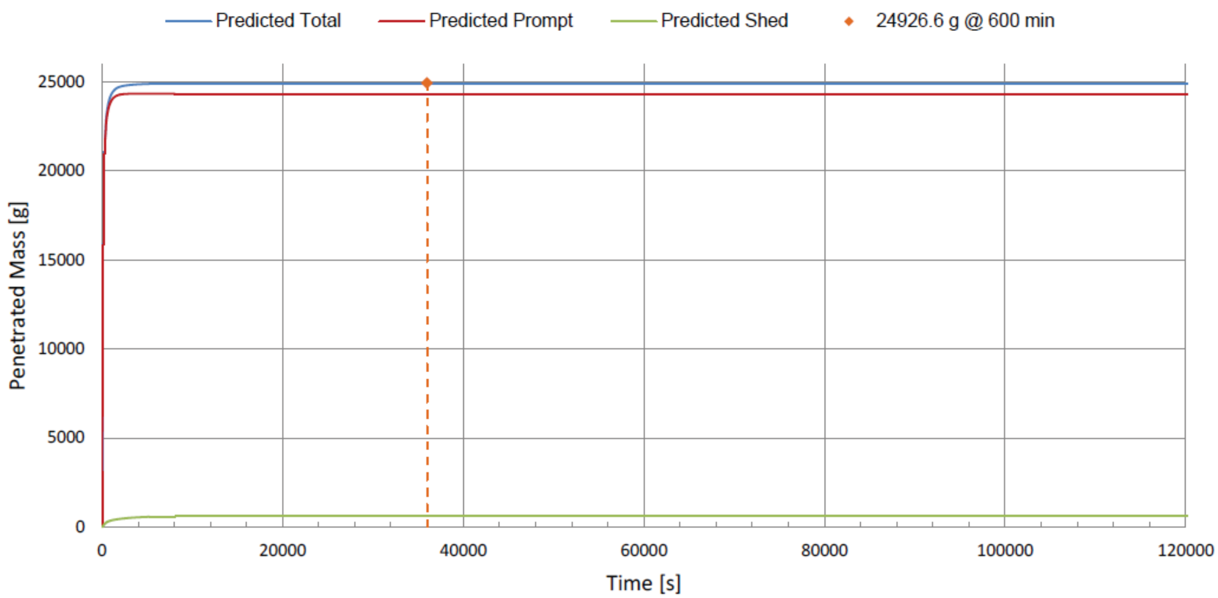

Figure 3.n.1-3: Plant-Scale Fiber Penetration vs. Time from Example Application

Figure 3.n.1-4 shows the prompt fiber penetration fraction as a function of fiber quantity on the strainer. As expected, the prompt penetration fraction decreases as a fiber debris bed forms on the strainer.

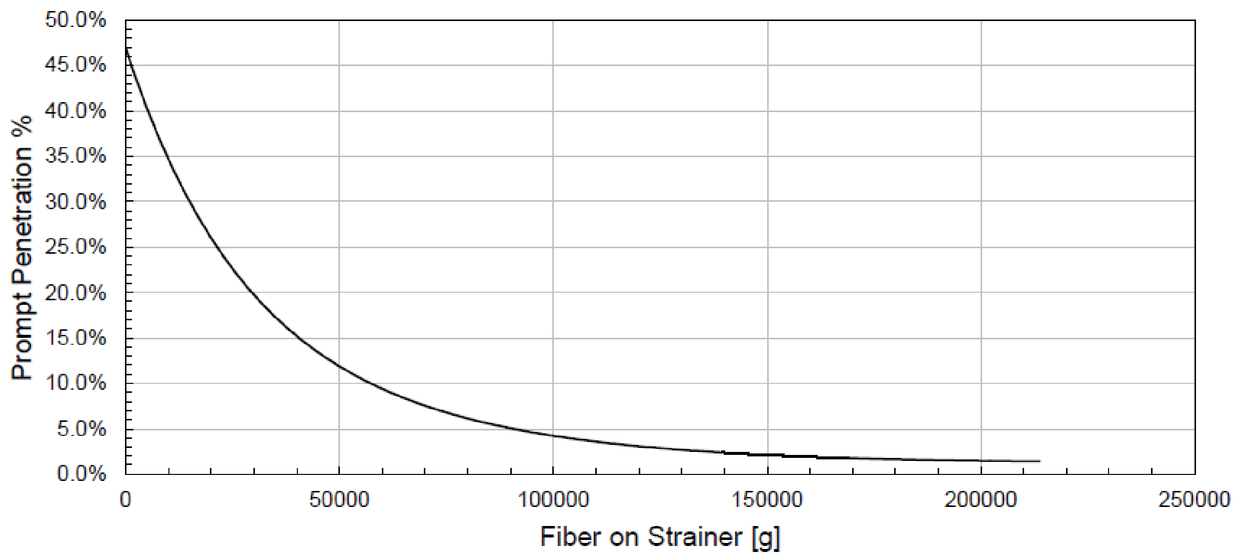

Figure 3.n.1-4: Prompt Fiber Penetration Fraction vs Fiber on Strainer

Figure 3.n.1-5 shows the shedding rate as a function of time. Note that shedding penetration depends on the fiber quantity on the strainer and time. As shown in the figure, the shedding rate decreases over time for a given amount of fiber on the strainer.

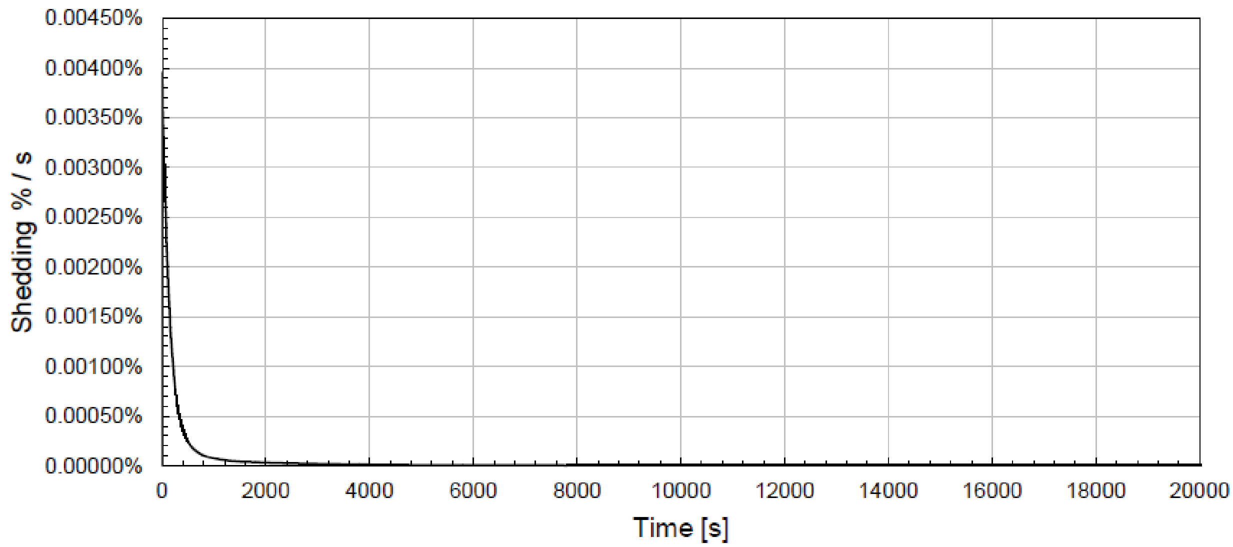

Figure 3.n.1-5: Shedding Rate vs Time

<u>Evaluation of Fiber Accumulation inside Reactor Vessel</u>

During the post-LOCA sump recirculation phase, debris that passes through the strainer could accumulate at the reactor core inlet or inside the reactor vessel, thereby potentially challenging LTCC. This was evaluated following the NRC review guidance on the resolution of in-vessel effects. Per this guidance, analysis of in-vessel fiber load for cold leg breaks (CLBs) is unnecessary (Reference 42 p. 3) and is therefore omitted. For the hot leg breaks (HLBs), the evaluation uses the methodology and acceptance criteria from WCAP-17788-P. This evaluation used time-dependent fiber penetration fractions obtained from Wolf Creek testing based on plant-specific inputs, as described earlier in this response. The penetration fraction varies with the amount of fine fiber (including erosion fines generated from small and large pieces) collected on the strainer and the amount of time passed since the onset of recirculation.

The time dependent evaluation divided the recirculation phase into small time steps. For each time step, the following computation was performed to quantify the fiber that passes through the strainer:

1. The fractions of prompt and shedding penetration were calculated using the low-fiber penetration curve based on the quantity of fine fiber collected on the strainer and the time since the onset of recirculation at the beginning of each time step. For the analysis presented in this submittal, only the low-fiber penetration curve was used because the transportable fine fiber loads used for the evaluation fall in the applicable range of this curve.
2. The amount of fine fiber that arrived at the strainer during the current time step was calculated by multiplying the fine fiber concentration in the pool by the strainer flow rate and time step. The fiber debris was assumed to be uniformly distributed in the sump pool.

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3. The amount of prompt penetration was calculated by multiplying the prompt penetration fraction from Step 1 by the amount of fine fiber arriving at the strainer during the current time step from Step 2.
4. The amount of shedding penetration was calculated by multiplying the shedding penetration fraction from Step 1 by the amount of fiber collected on the strainer at the beginning of the time step.
5. The fiber that passes through the strainer is split based on the ratio in flow rate between the emergency core cooling system (ECCS) pumps and CS pump.
6. The fiber transported by the ECCS pumps reaches the reactor and is assumed to accumulate at the core inlet only, without crediting the alternate flow paths (AFPs). This is consistent with the NRC review guidance (Reference 42). The fiber carried by the containment spray pump is returned to the sump pool. The pool fiber concentration is updated as an initial condition for the next time step.

The steps shown above were implemented in an Excel spreadsheet and the total in-vessel fiber load for HLBs was calculated by summing up the amount of fiber that reaches the reactor during each time step. The total fiber quantity was then increased by 2.5% to account for the uncertainties in the curve fit of the fiber penetration test data. Note that this percentage increase was determined by comparing the measured total fiber penetration with model results based on testing conditions.

To ensure the worst conditions were captured, the evaluation analyzed various equipment lineups and different combinations of inputs (e.g., pool volume, transport fiber load, number of RHR and CS trains in operation, RHR and CS pump flow rates, sump recirculation and hot leg switchover times, and CS duration). The worst design basis case results from the scenario with both RHR pumps in operation and failure of one CS pump. The worst beyond design basis case had both RHR pumps in operation and failure of both CS pumps. The resulting total fiber loads at the core inlet are 92.81 g/FA for the worst design basis case and 94.29 g/FA for the worst beyond design basis case for the breaks of 10 inches and smaller.

<u>Resolution of In-Vessel Downstream Effects per NRC Review Guidance</u>

The NRC Review Guidance for the resolution of in-vessel downstream effects (Reference 42) provided four different paths (identified as Box 1 through Box 4 paths) that PWR licensees can use to resolve the issue based on the AFP analysis in WCAP-17788-P, Revision 1. Wolf Creek used the Box 4 path to demonstrate that in-vessel effects will not challenge LTCC for breaks up to the threshold break size of 10 inches. Table 3.n.1-1 summarizes the parameters that must be compared between Wolf Creek and those used in the WCAP analysis. More detailed discussions of these parameters are presented after the table.

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Table 3.n.1-1: Summary of In-Vessel Effects Parameters

<table>
<tr><th>Parameters</th><th>WCAP-17788 Revision 1 Values</th><th colspan="2">Wolf Creek Values</th></tr>
<tr><td>Nuclear Steam Supply System (NSSS) Design</td><td>Various</td><td colspan="2">Westinghouse</td></tr>
<tr><td>Fuel Type</td><td>Various</td><td colspan="2">Westinghouse 17 x 17</td></tr>
<tr><td>Barrel/Baffle Configuration</td><td>Various</td><td colspan="2">Upflow</td></tr>
<tr><td>Minimum Chemical Precipitation Time</td><td>143 minutes (t_{block} from WCAP-17788, Vol 1, Table 6-1)</td><td colspan="2">24 hours</td></tr>
<tr><td>Maximum HLSO Time</td><td>N/A</td><td colspan="2">10 hours</td></tr>
<tr><td rowspan="2">Maximum Core Inlet Fiber Load for HLB</td><td rowspan="2">WCAP-17788, Vol 1, Table 6-3</td><td>DB</td><td>BDB</td></tr>
<tr><td>92.81 g/FA</td><td>94.29 g/FA</td></tr>
<tr><td rowspan="2">Total In-Vessel Fiber Limit for HLB</td><td rowspan="2">WCAP-17788, Vol 1, Section 6.4</td><td>DB</td><td>BDB</td></tr>
<tr><td>92.81 g/FA</td><td>94.29 g/FA</td></tr>
<tr><td>SSO Time</td><td>20 minutes</td><td colspan="2">13 minutes</td></tr>
<tr><td>Maximum Rated Thermal Power</td><td>3658 MWt</td><td colspan="2">3565 Mwt</td></tr>
<tr><td>AFP Resistance</td><td>WCAP-17788, Vol 4, Table 6-1</td><td colspan="2">WCAP-17788, Vol 4, Table RAI-4.2-24</td></tr>
<tr><td rowspan="2">ECCS Flow per FA</td><td rowspan="2">8 – 40 gpm/FA</td><td>DB</td><td>BDB</td></tr>
<tr><td>52.9 gpm/FA</td><td>37.8 gpm/FA</td></tr>
</table>

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Comparison of Wolf Creek Chemical Precipitation Time with HLSO Time and t_{block}

For Wolf Creek, chemical precipitation was shown to occur after the latest HLSO time and after the time that complete core inlet blockage can be tolerated, which is defined in WCAP-17788 as t_{block}.

1. Wolf Creek chemical precipitation time (t_{chem}) – Chemical precipitation was shown not to occur within 24 hours for containment sump temperatures above 87°F following the accident based on the autoclave testing in WCAP-17788, Volume 5 (Reference 48). This was determined using a precipitation map to compare the sump aluminum concentration estimated from the WCAP-16530 methodology with all NaOH group autoclave test results and the WCAP-17788 precipitation boundary equation (see Figure 3.n.1-6). Autoclave tests performed at a pH greater than 10 were omitted as non-representative of Wolf Creek, which has a maximum final containment sump pool pH of 9.39. The minimum final containment sump pool pH is 8.78. Using the maximum sump aluminum concentration at 24 hours and a sump pH of 8.78, the minimum pH + p[Al] was calculated to be 11.6, which crosses the precipitation boundary at 87°F. Containment sump temperatures below 87°F by 24 hours would be indicative of a significantly less severe accident than simulated using the WCAP-16530 methodology. Therefore, aluminum precipitation will not occur within 24 hours.

   The WCAP-17788 precipitation map is shown below in Figure 3.n.1-6. The 24-hour precipitation temperature of 87°F is determined by the intersection of the Wolf Creek minimum pH+p[Al] line (red dotted line) and the WCAP precipitation boundary at a pH of 8.78 (blue line). The WCAP precipitation boundary at a pH of 9.39 is shown for comparison (green dot-dashed line). Because the pH+p[Al] difference between the two plots is less than the difference between the minimum and maximum final pH values, both plots are conservative when using the maximum calculated Wolf Creek aluminum concentration.

   Figure 3.n.1-6 also shows the Argonne National Laboratory (ANL) aluminum solubility equation (black dashed line). This is included in the figure to demonstrate the relative conservatism of the various solubility equations, but they are not used to determine the short-term aluminum precipitation timing.

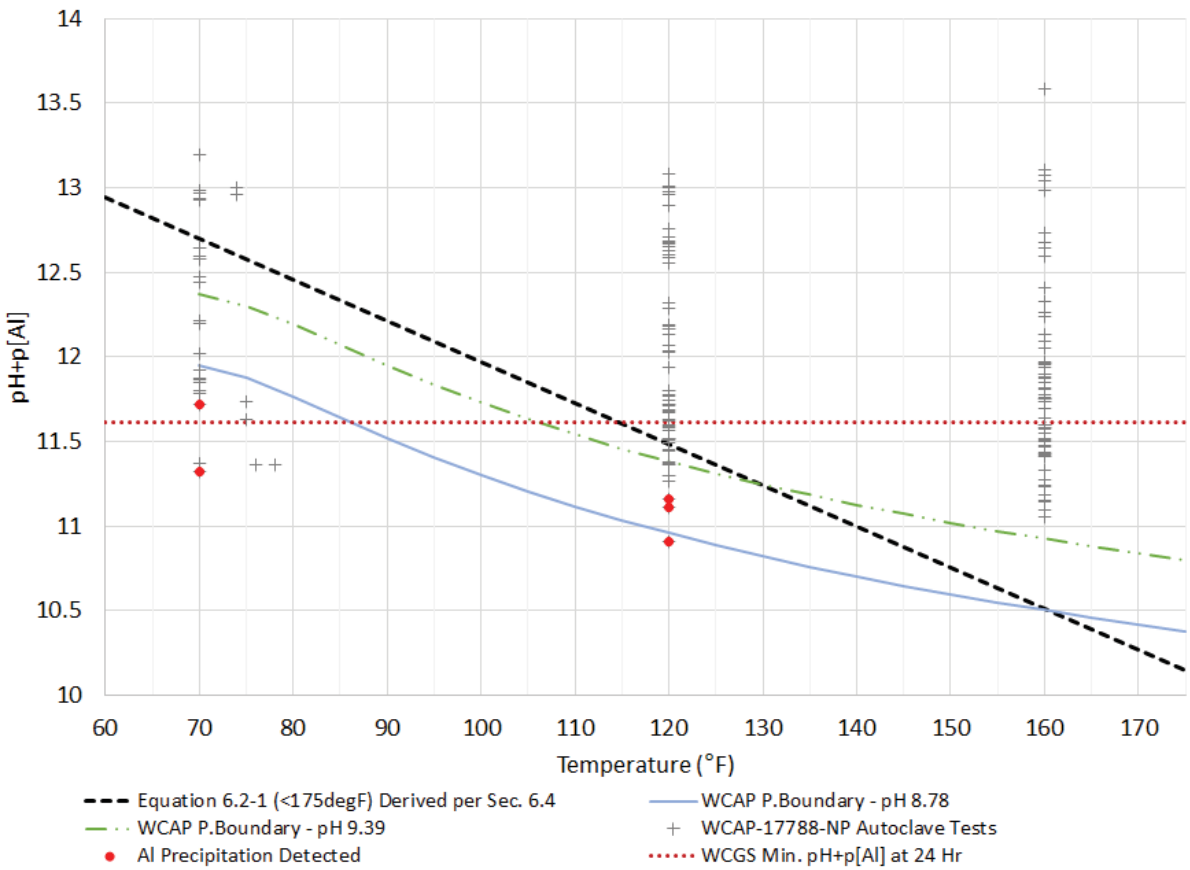

Figure 3.n.1-6: 24 Hour Aluminum Solubility

To demonstrate the conservatism of the above approach for demonstrating no chemical precipitation within the first 24 hours, the bounding Wolf Creek post-LOCA conditions were also compared with specific autoclave test groups in WCAP-17788-P, Volume 5. It was shown that Group 9 and Group 15 are representative of the high and low pH conditions, respectively.

Table 3.n.1-2 shows the key Wolf Creek post-LOCA conditions and debris loads related to chemical precipitation at plant and test scales.

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Table 3.n.1-2: Key Wolf Creek Parameter Values for Chemical Precipitation

Parameter	Plant Scale	Test Scale
Buffer	Sodium Hydroxide	Sodium Hydroxide
Sump pH (Long-term)	8.5 – 9.39	8.5 – 9.39
Minimum Sump Volume	53,526 ft^3 (3,340,000 lb_m/62.4 lb_m/ft^3)	1.76 ft^3 (50 L)
Maximum Sump Pool Temperature	268.4°F	268.4°F
Maximum Calcium Silicate	0 g	0 g*
Maximum E-Glass	2,283,190 g	75.1 g*
Maximum Silica	0 g	0 g*
Mineral Wool	0 g	0 g*
Maximum Aluminum Silicate	0 g	0 g*
Maximum Concrete	Not Determined**	Not Determined**
Maximum Interam™	0 g	0 g*
Aluminum	882 ft^2 (total combined)	0.0290 ft^2*
Galvanized Steel	Not Determined***	Not Determined***

* Test Scale = Plant Scale × (Test Volume/Wolf Creek Minimum Sump Volume)
** Concrete is not a significant contributor to chemical precipitation
*** The galvanized steel surface area was not determined for Wolf Creek. Both test Groups 9 and 15 included tests with no galvanized steel present to inhibit aluminum release.

The Group 9 and Group 15 tests used sodium hydroxide as buffer. The test pH values of Group 9 and Group 15 reflect the maximum and minimum sump pH at Wolf Creek.

The material quantities for Test Group 9 are comparable with those of Wolf Creek with the Group 9 tests having slightly less E-Glass but slightly more aluminum metal than Wolf Creek. The Group 15 tests contained less E-Glass than Wolf Creek but had approximately 50% more aluminum. The Wolf Creek chemical effects analysis showed that aluminum released from aluminum metal is significantly greater than from E-Glass. Additionally, mineral wool, which is not present as a Wolf Creek debris type, was used in the Group 15 tests and would contribute additional aluminum release.

Precipitation was not detected by filtration tests for Group 9 down to a temperature of 120°F and Group 15 down to a temperature of 160°F over the 24-hour test duration. Filtration tests were not performed below these temperatures for these groups. This supports the conclusions reached above using the precipitation map. Aluminum precipitation will not occur within 24 hours.

2. Wolf Creek HLSO time – Wolf Creek maximum HLSO time is 10 hours after the event.

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3. Time of t_{block} used in WCAP-17788 – Wolf Creek is a Westinghouse NSSS plant with a upflow barrel/baffle design. WCAP-17788 used a t_{block} of 143 minutes for this reactor and fuel configuration, as shown in Table 6-1 of WCAP-17788, Volume 1 (Reference 44).

Comparison of Wolf Creek Maximum Thermal Power with that Assumed in WCAP-17788

Wolf Creek maximum rated thermal power is less than the analyzed power level in WCAP-17788 for a Westinghouse NSSS with an upflow barrel/baffle design.

1. Wolf Creek rated thermal power – Wolf Creek maximum rated thermal power is 3565 MWt.
2. Thermal power assumed in WCAP-17788 – The WCAP analysis used a thermal power of 3658 MWt for a Westinghouse upflow plants, as shown in Table 6-1 of WCAP-17788, Volume 4 (Reference 45).

Comparison of Wolf Creek Reactor AFP Resistance with that Assumed in WCAP-17788

The Wolf Creek reactor AFP resistance is less than that analyzed in WCAP-17788.

3. Wolf Creek reactor AFP resistance – The Wolf Creek AFP resistance is presented in Table RAI-4.2-24 of WCAP-17788-P, Volume 4 (Reference 45) as "Total Unadjusted K/A^2 (ft^{-4}) in the table.
4. Maximum AFP resistance assumed in WCAP-17788 – The maximum AFP resistance used in the WCAP analysis is presented in Table 6-1 of WCAP-17788, Volume 4 (Reference 45) as "Barrel/Baffle Total K/A^2 (ft^{-4})".

Comparison of Wolf Creek Flow Rate with that Analyzed in WCAP-17788

The Wolf Creek ECCS flow per fuel assembly (FA) for design basis cases exceeds the flow rates analyzed in WCAP-17788, while for the beyond design basis cases it is within the range of flow rates analyzed in WCAP-17788.

1. Wolf Creek ECCS flow rate – The Wolf Creek ECCS flow rate per fuel assembly is 52.9 gpm/FA based on the ECCS flow rate used in the in-vessel analysis for the most limiting design basis pump configuration with two ECCS trains in operation at maximum flow rate. For the beyond design basis cases, the analysis used an ECCS flow rate of 37.8 gpm/FA with two ECCS trains operating at minimum flow rates to maximize in-vessel fiber load.
2. ECCS flow rates analyzed in WCAP-17788 – For a Westinghouse upflow plants, the analyzed ECCS flow rate is 8 gpm/FA to 40 gpm/FA, provided in Table 6-1 of WCAP-17788, Volume 4 (Reference 45).

The NRC review guidance (Reference 42) requires consistency between the minimum ECCS recirculation flow rate used in the WCAP AFP analyses and that at the plant. As stated in the NRC Technical Evaluation Report (Reference 49), the debris bed at the reactor core inlet has the highest resistance when the bed is formed at the lowest flow rate. It was also observed during the fuel assembly testing that debris bed became unstable at higher flow rates, resulting in bed breakthrough (Reference 49). As stated above, the ECCS flow rates that resulted in the worst in-vessel fiber loads for Wolf Creek are either higher than or slightly below the maximum flow rate of 40 gpm/FA analyzed in the WCAP. It is therefore expected that the debris bed formed at the Wolf Creek reactor core inlet would have lower resistance than that analyzed in the WCAP. As a result, the Wolf Creek plant conditions are bounded by the WCAP analysis.

Comparison of Wolf Creek In-Vessel Fiber Load with WCAP-17788 Limit

The maximum amount of fiber that may arrive at the core inlet for the breaks up to threshold size for Wolf Creek exceeds the core inlet fiber limit but is less than the total in-core fiber limit presented in WCAP-17788.

1. WCAP-17788 core inlet fiber limit – The core inlet fiber limit that is applicable for Wolf Creek (i.e., Westinghouse upflow plant with Westinghouse fuel) is in Table 6-3 of WCAP-17788 Volume 1 (Reference 44). Since the Wolf Creek fuel assembly has the same pitch as was used in WCAP-17788 Volume 1, no adjustment to this fiber limit is necessary
2. WCAP-17788 total in-core fiber limit – The total in-core fiber limit is in Section 6.4 of WCAP-17788, Volume 1 (Reference 44).
3. Wolf Creek in-vessel fiber load – The maximum Wolf Creek core inlet fiber load for the threshold break size of 10 inches is 92.81 g/FA for the most limiting design basis pump configuration and 94.29 g/FA for the most limiting beyond design basis case. Per the latest NRC review guidance (Reference 42), the flow split between the core inlet and AFPs was not credited. All the fiber that reaches the RV is assumed to accumulate at the reactor core inlet.

The WCAP-17788 core inlet fiber limit is based on the assumption that debris accumulates uniformly at the core inlet. In reality, the debris bed at the core inlet will not be uniform due to non-uniform flow distribution. As a result, it would take more debris than determined by WCAP-17788 to completely block the core inlet and activate the AFPs as discussed in the Appendix B of the NRC review guidance (Reference 42). Because of the expected non-uniform debris loading, the debris head loss at the core inlet would be lower than predicted in WCAP-17788. Lower head loss would allow additional fiber accumulation beyond the core inlet fiber limit where complete core blockage is predicted to occur in the WCAP. By definition, if the head loss at the core inlet is not high enough to activate flow through the AFPs, the core is continuing to receive sufficient flow for LTCC through the core inlet. As described in WCAP-17788, LTCC is assured as long as the total amount of fiber to the RCS remains below the total in-core fiber limit. Therefore, it is reasonable to use the total in-core fiber limit as the acceptance criterion for HLBs. For Wolf Creek, the maximum quantity of fiber predicted to reach the reactor core (92.81 g/FA for design basis case and 94.29 g/FA for beyond design basis) for the threshold break size of 10 inches, is

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lower than the WCAP-17788 total in-core fiber limit and therefore will not challenge LTCC.

Comparison of Wolf Creek SSO Time with that Assumed in WCAP-17788

The earliest SSO time for Wolf Creek is shorter than that assumed in the WCAP-17788 analysis.

1. Wolf Creek SSO time – The SSO time marks the beginning of sump recirculation and fiber accumulation inside the reactor. For Wolf Creek, the shortest duration for injection from the RWST is 13 minutes.
2. The SSO time assumed in the WCAP-17788 analysis is 20 minutes as shown in Table 6-1 of WCAP-17788, Volume 4 (Reference 45).

Although the earliest SSO time for Wolf Creek is not bounded by that used in the WCAP analysis, other plant-specific attributes are shown to be adequate to ensure LTCC will not be compromised for breaks of 10 inches and smaller, as summarized below.

1. Conservatisms in Sump Switchover Time

The 13 minutes SSO time is calculated in a conservative manner. Key assumptions used in the analysis are summarized below:

- The analysis assumed that two trains of ECCS are in operation with the maximum flow rates from the RHR pumps, SIPs, CCPs and CS pumps. No delay was credited for pump startup.
- The pump flow rates used were based on 0 psig containment pressure. The RCS depressurization and the containment pressurization are not considered.
- The analysis used the minimum Technical Specification RWST volume with the instrument uncertainties accounted for. Typically, significant margin exists between this minimum RWST volume and the actual RWST volume maintained during plant operation.

2. Margin in Thermal Power

The applicable AFP analysis for Westinghouse upflow plants in WCAP-17788 assumed a thermal power of 3658 MWt, which leads to a decay heat of 87.4 MWt at the assumed SSO time of 20 minutes using the Appendix K decay heat curve (i.e., 1971 ANS-5.1 decay heat curve plus 20% uncertainty). Wolf Creek has a rated thermal power of 3565 MWt. At the earliest 13 minutes SSO time, the Wolf Creek decay heat is calculated to be 78.8 MWt using a more realistic decay heat curve of 1979 ANS-5.1 standard with 2σ uncertainty. This decay heat is bounded by that analyzed in the WCAP.

3. Debris Transport Behavior

The in-vessel debris analysis in WCAP-17788 Volume 4 for Westinghouse plants assumed that all debris arrives over 60 seconds starting at the time of SSO. The assumption that complete core inlet blockage occurs at the time of SSO is substantially conservative. It is not possible for the entire amount of debris to arrive at the core inlet and form a uniform bed coincident with the initiation of SSO. Additionally, the core inlet fiber limit derived in WCAP-17788 is conservatively low due to the assumption that fiber debris would accumulate uniformly at the reactor core inlet. Additionally, the Wolf Creek in-vessel analysis for breaks up to 10 inches shows that it takes 428 seconds (~7.1 minutes) after the start of SSO for the core-inlet fiber load to reach the applicable WCAP core-inlet fiber limit.

The wolf Creek in-vessel analysis assumed that the RHR pumps, SIPs, and CCPs begin taking suction from the sump at the earliest SSO time of 13 minutes after the accident, and conservatively delayed the time when the CS pumps start taking suction from the sump. This approach is conservative because, although the ECCS switchover from injection to sump recirculation is initiated automatically and there is no interruption in the RHR flow to RCS during the transfer to sump recirculation, additional time is required to manually switch the CCPs and SIPs to recirculation. The minimum validated time critical action to align the CCPs and SIPs is ~4.5 minutes. This operator action time for aligning the CCPs and SIPs to take suction from the sump will further delay the time to reach the core-inlet fiber limit to beyond the 7.1 minutes discussed above.

Furthermore, the Wolf Creek maximum in-vessel fiber load for the breaks up to the threshold break size (94.29 g/FA) is on the order of 2 times the applicable WCAP core inlet fiber limit. The Wolf Creek in-vessel analysis showed that it takes over an hour after the start of SSO to reach this fiber load.

Sensitivity studies were performed using a more gradual build-up of core inlet resistance to demonstrate the conservatisms in the WCAP-17788 AFP analysis, which assumed that the core inlet resistance ramps up in 1 minute after the SSO time. As discussed in Section 8.0 and the response to NRC RAI-4.7 contained in Volume 4 of WCAP-17788-P Revision 1, two sensitivity studies (Case 3A and Case 3B) were conducted using the Westinghouse upflow plant model. In these sensitivity runs, the core inlet resistance was increased linearly over a 1-hour or 2-hour period after start of SSO, which simulates a more realistic build-up of debris. As shown in Table 8-3 of Volume 4 of WCAP-17788-P Rev. 1, Case 3A and Case 3B result in a peak cladding temperature less than 525°F, which is below the acceptance criteria of 800°F. Additionally, the sensitivity cases showed no core-wide uncovery because the downcomer was filled as the core inlet resistance increased gradually and core cooling was not interrupted. With the gradual increase in core inlet resistance, the sensitivity cases predicted a core inlet fiber limit that is more than 3 times higher than that determined in the WCAP-17788 base case.

Considering the conservatisms in the comparison between plant parameters and those assumed in WCAP-17788, the WCAP AFP analysis for the Westinghouse upflow Plants is applicable for and bounds Wolf Creek plant conditions. Therefore, for the breaks up to the threshold break size, accumulation of debris within the reactor vessel will not challenge LTCC.

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3.o Chemical Effects

The objective of the chemical effects section is to evaluate the effect that chemical precipitates have on head loss and core cooling.

3.o.1 Provide a summary of evaluation results that show that chemical precipitates formed in the post-LOCA containment environment, either by themselves or combined with debris, do not deposit at the sump screen to the extent that an unacceptable head loss results, or deposit downstream of the sump screen to the extent that long-term core cooling is unacceptably impeded.

Response to 3.o.1

The chemical effects strategy for Wolf Creek includes:

- Quantification of chemical precipitates using the WCAP-16530-NP-A (Reference 30) methodology.
- Introduction of those pre-prepared precipitates in prototypical array testing.
- Application of an aluminum solubility correlation to determine the maximum precipitation temperature.
- Use of autoclave test results to determine the minimum precipitation timing.
- Time-based determination of acceptable head losses.

As discussed in the Response to 3.a.1, Wolf Creek has evaluated debris generation quantities for breaks postulated on every Class 1 ISI pipe weld inside the first isolation valve and outside of the reactor cavity. The amount of chemical precipitates was quantified for bounding quantities of LOCA generated debris. Other plant-specific inputs such as pH, temperature, aluminum quantity, and spray times were selected to maximize the generated amount of precipitates. In the Response to 3.f.7, the generated amounts of chemical precipitates are compared with those used in the strainer head loss tests to determine the resulting head loss across the strainers.

Before the tests were conducted, aluminum oxyhydroxide (AlOOH) was prepared according to the WCAP-16530-NP-A recipes and was verified to meet the WCAP-16530-NP-A settling criteria within 24 hours of the test. During the test, a fiber and particulate debris bed was established on the strainer surfaces, the stabilization criteria was satisfied, and the pre-prepared precipitates were added to the test tank in batches. Further details on the head loss measured after introduction of chemical precipitates are described in the Response to 3.f.4.

The effect of chemical precipitate deposition on fuel rod surfaces is evaluated separately from the strainer head loss described in this section. See the Response to 3.n.1 for details of the in-vessel effects evaluations.

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3.o.2 Content guidance for chemical effects is provided in Enclosure 3 dated March 2008 to a letter from the NRC to NEI (ADAMS Accession No. ML080380214).

Response to 3.o.2

The NRC identified evaluation steps in "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Plant-Specific Chemical Effect Evaluations" in March of 2008 (Reference 50). Wolf Creek's responses to the GL Supplement Content evaluation steps are summarized below. The numbering of the following subsections to the Response to 3.o.2 follow the numbering scheme provided in Section 3 and Figure 1 of the March 2008 guidance (Reference 50). Figure 3.o.2-1 highlights the Wolf Creek chemical effects evaluation process using the flow chart in Figure 1 of the March 2008 guidance (Reference 50). The applicable items in Figure 3.o.2-1 are blocks 2 through 5, 7 through 10, 12, 13, and 15 through 17.

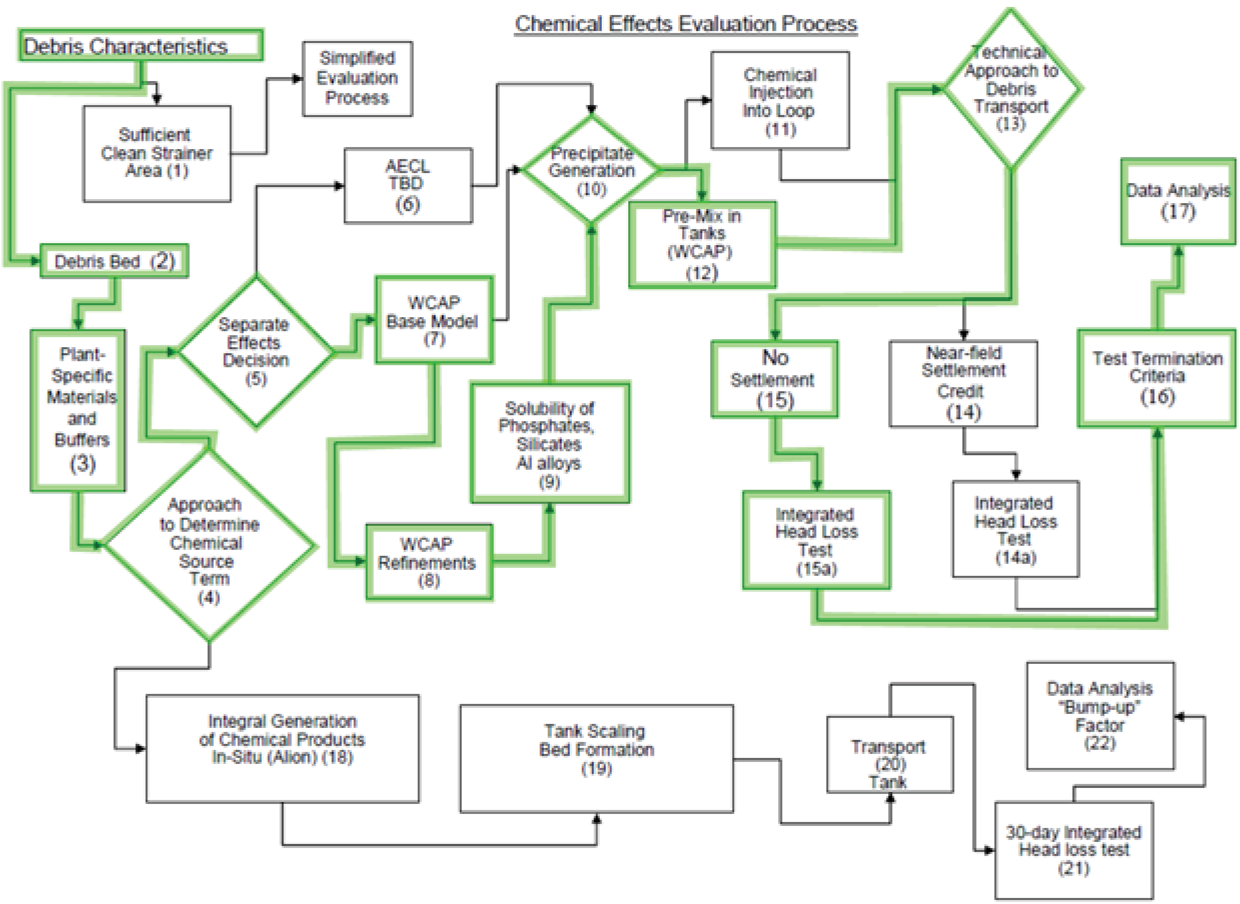

Figure 3.o.2-1: Chemical Effects Evaluation Process for Wolf Creek

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1. <u>Sufficient 'Clean' Strainer Area</u>:

   i. *Those licensees performing a simplified chemical effects analysis should justify the use of this simplified approach by providing the amount of debris determined to reach the strainer, the amount of bare strainer area and how it was determined, and any additional information that is needed to show why a more detailed chemical effects analysis is not needed.*

   Response to 3.o.2.1.i

   Wolf Creek is not crediting clean strainer area to perform a simplified chemical effects analysis. See Figure 3.o.2-1.

2. <u>Debris Bed Formation</u>:

   i. *Licensees should discuss why the debris from the break location selected for plant-specific head loss testing with chemical precipitate yields the maximum head loss. For example, plant X has break location 1 that would produce maximum head loss without consideration of chemical effects. However, break location 2, with chemical effects considered, produces greater head loss than break location 1. Therefore, the debris for head loss testing with chemical effects should be based on break location 2.*

   Response to 3.o.2.2.i

   A single bounding chemical debris load, determined using the inputs and assumptions described in the Response to 3.o.2.3.i, was used for all break locations. Assuming the bounding quantity of chemical debris at all locations conservatively maximizes the head loss determined by plant specific head loss testing. A FDL test and a TB test were performed for Wolf Creek by Alden Research Laboratory (Alden). The FDL test introduced incremental quantities of fibrous debris mixed with corresponding particulates to target certain plant break locations and/or debris loads of interest, followed by batches of chemical debris up to the bounding quantity. During the TB test, the quantity of particulate debris determined as acceptable in the FDL test was first introduced to the test tank, followed by small batches of fiber debris. The bounding quantity of chemical debris was added to the test tank after all the fiber and particulate debris was introduced. The main purpose of the TB test was to determine if a low-porosity debris bed (high particulate and low fiber) could form on the strainer, resulting in head losses higher than those observed during the FDL test. The head loss results of the FDL test, for both the conventional and chemical debris beds, bounded the results of the TB test. The results of the FDL test were used to develop the head loss contributions from conventional debris and chemical debris because this test resulted in higher head loss values than the TB test. See the Responses to 3.f.4 and 3.f.10 for a detailed

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summary of the methodology, assumptions, bases, and results of the prototypical head loss testing and the debris head loss analyses.

3. *Plant-Specific Materials and Buffers:*

   i. *Licensees should provide their assumptions (and basis for the assumptions) used to determine chemical effects loading: pH range, temperature profile, duration of containment spray, and materials expected to contribute to chemical effects.*

Response to 3.o.2.3.i

The chemical model requires a number of plant-specific inputs. Each input is chosen to maximize the calculated quantity and minimize the solubility (aluminum only) of the chemical precipitates.

Wolf Creek uses sodium hydroxide (NaOH) to buffer the post-LOCA containment sump pool. The maximum final containment sump pool pH of 9.39 (at a reference temperature of 25°C) was conservatively used for the entire 30-day post-LOCA event to determine the maximum rate of aluminum release. The containment sprays deliver the NaOH to the containment sump pool. The bounding high containment spray pH profile conservatively assumed to determine the maximum rate of aluminum release is shown in Table 3.o.2.3.i-1. The maximum containment spray duration was bounded by 3200 minutes, which was used to conservatively maximize the calculated aluminum release.

Table 3.o.2.3.i-1: Maximum Containment Spray pH Profile

Time (min)	pH at 25°C
0	9.73
20.99	9.73
21	10.12
31.3	10.42
41.7	11.10
52.3	11.98
63.1	12.30
73.9	12.49
88.1	12.64
96.2	12.70
96.21	9.39
3200	9.39

Acids generated through radiolysis may decrease the containment sump pool pH over the 30-day post-LOCA event. Because aluminum solubility decreases with lower pH, the pH used to determine the aluminum solubility limit was the minimum final containment sump pool pH of 8.78 (at a reference temperature of 25°C). This assumption conservatively maximized the potential for the precipitation of aluminum products. Different pH values for release and solubility are combined in a non-physical way, bounding the effects of all potential pH profile variations.

Limiting containment sump pool and containment temperature profiles were used to maximize chemical release rates. The temperature profiles are shown in Figure 3.o.2.3.i-1 and Figure 3.o.2.3.i-2.

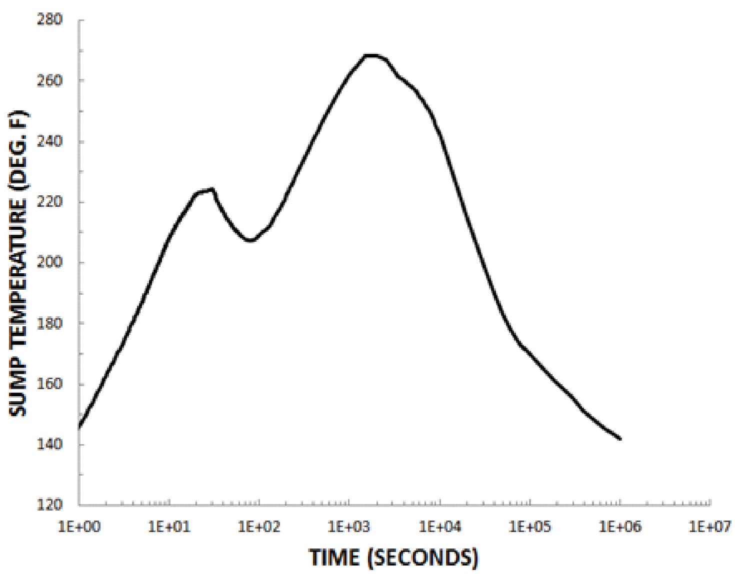

Figure 3.o.2.3.i-1: Maximum Containment Sump Pool Temperature Profile

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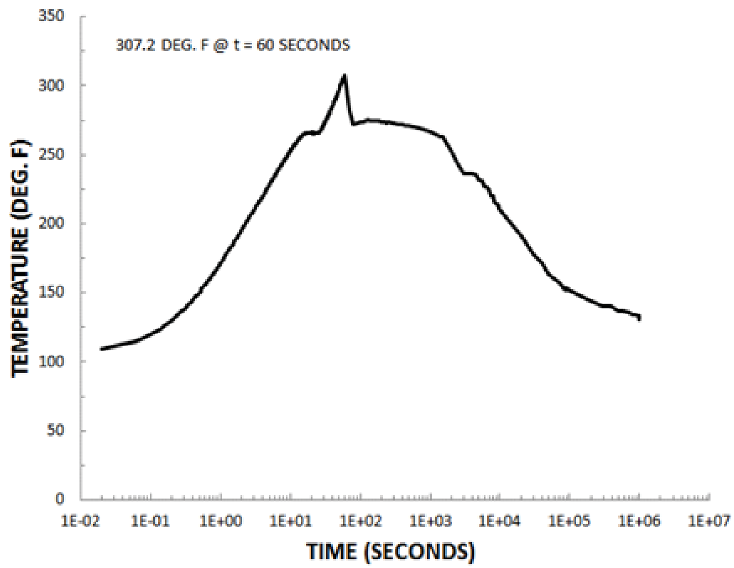

Figure 3.o.2.3.i-2: Maximum Containment Spray Temperature Profile

The total amount of analyzed unsubmerged aluminum exposed to containment sprays was 839 ft^2. The total amount of analyzed submerged aluminum exposed to the containment sump fluid at Wolf Creek was 43 ft^2. No mass limit was set on the quantity of aluminum released from these sources.

The maximum amount of Nukon destroyed by the LOCA and assumed to be submerged in the containment sump pool was 2,084.8 ft^3 with an as-fabricated bulk density of 2.4 lbm/ft^3. This quantity includes all Nukon debris within the ZOI, including fines, small pieces, large pieces, and intact blankets. The amount of latent fiber analyzed in containment was 30 lbm.

Cerablanket insulation is used inside the reactor cavity around the perimeter of the reactor vessel and near the neutron detector wells. As stated in the Response to 3.a.1, reactor cavity breaks are excluded, and Cerablanket is shielded from the breaks outside of the reactor cavity by the primary shield wall. Therefore, Cerablanket is not considered a debris source. It should be noted, however, the chemical effects analysis included 9.1 ft^3 of Cerablanket (with an as-fabricated bulk density of 8 lbm/ft^3) for operating margin.

The total amount of concrete assumed to be exposed and submerged in the containment sump pool was 10,000 ft^2. The quantity of chemical precipitates was negligibly impacted by this large assumed surface area of exposed concrete. Therefore, exposed concrete is not a significant impact to chemical product generation in the Wolf Creek post-LOCA containment sump pool and is not tracked for this purpose.

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The containment sump pool was assumed to be well mixed. This assumption conservatively maximized aluminum release by not considering the concentration gradient that will form around submerged source materials at low pool velocity conditions.

Minimum and maximum water mass cases were run to determine both maximum generation of precipitates and maximum precipitation temperatures because aluminum release rates from some materials are concentration dependent. At Wolf Creek, the maximum analyzed containment sump pool mass that is available for chemical dissolution was 4,107,546 lbm. The minimum analyzed containment sump pool mass that is available for chemical dissolution was 3,340,000 lbm.

Thermo-Lag and Darmatt KM1 were excluded as sources of aluminum in containment. WCAP-16530-NP-A excludes Thermo-Lag as a contributor to chemical effects as an organic mastic. Darmatt KM1 is not classified under any material type in WCAP-16530-NP-A. All Darmatt KM1 in the Wolf Creek containment is covered or jacketed, it is located in the southwest quadrant of the annulus (outside of the pressurizer compartment), and it is protected from being destroyed by a LOCA by robust barriers. Only 153 in^3 (0.09 ft^3) of the Darmatt KM1 in containment is not protected from sprays by an overhead concrete slab. This is a negligible quantity as compared with the assumed 2,084.8 ft^3 of Nukon and 9.1 ft^3 of Cerablanket. Darmatt KM1 was excluded from further chemical effects analysis due to its protection from destruction during a LOCA and its low exposure to containment spray.

As justified in the Response to 3.o.2.7.i, the WCAP-16530-NP-A methodology was simplified to consider only AlOOH precipitate. Therefore, silicon and calcium releases were not tracked by the chemical analysis. Min-K debris is not required to be considered for chemical precipitation by Wolf Creek. This debris type does not contribute to aluminum release per WCAP-16530-NP-A.

4. *<u>Approach to Determine Chemical Source Term (Decision Point)</u>:* [separate effects vs. integrated chemical effects]

   i. *Licensees should identify the vendor who performed plant-specific chemical effects testing.*

Response to 3.o.2.4.i

Wolf Creek is using the separate chemical effects approach to determine the chemical source term. Alden Research Laboratory, Inc. performed the testing in their test lab in Holden, MA.

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5. *Separate Effects Decision (Decision Point): Within this part of the process flow chart, two different methods of assessing the plant-specific chemical effects have been proposed. The WCAP-16530-NP-A study (Box 7 WCAP Base Model) uses predominantly single-variable test measurements. This provides baseline information for one material acting independently with one pH-adjusting chemical at an elevated temperature. Thus, one type of insulation is tested at each individual pH, or one metal alloy is tested at one pH. These separate effects are used to formulate a calculational model, which linearly sums all of the individual effects. A second method for determining plant-specific chemical effects that may rely on single-effects bench testing is currently being developed by one of the strainer vendors (Box 6, AECL).*

   Response to 3.o.2.5

   Wolf Creek is using the WCAP-16530-NP-A chemical effects base model to determine the chemical source term. The application of an aluminum solubility correlation to determine a maximum precipitate formation temperature is discussed in the Response to 3.o.2.7.i and Response to 3.o.2.9.i. Additionally, the use of autoclave test data to determine the minimum precipitation timing is discussed in the Response to 3.o.2.7.i and Response to 3.o.2.9.i.

6. *AECL Model:*

   i. *Since the NRC is not currently aware of the complete details of the testing approach, the NRC staff expects licensees using it to provide a detailed discussion of the chemical effects evaluation process along with head loss test results.*

      Response 3.o.2.6.i

      This question is not applicable because Wolf Creek is not using the AECL model. See Figure 3.o.2-1.

   ii. *Licensees should provide the chemical identities and amounts of predicted plant-specific precipitates.*

      Response 3.o.2.6.ii

      This question is not applicable because Wolf Creek is not using the AECL model. See Figure 3.o.2-1.

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7. *WCAP Base Model:*

i. *Licensees proceeding from block 7 to diamond 10 in the Figure 1* [*Figure 3.o.2-1*] *flow chart should justify any deviations from the WCAP base model spreadsheet (i.e., any plant specific refinements) and describe how any exceptions to the base model spreadsheet affected the amount of chemical precipitate predicted.*

Response 3.o.2.7.i

The Wolf Creek chemical model quantifies chemical precipitates using the WCAP-16530-NP-A (Reference 30) methodology with the following deviations from the WCAP base model spreadsheet:

1. The application of an aluminum solubility correlation to determine a maximum precipitate formation temperature (see the Response to 3.o.2.9.i).
2. The use of autoclave test data to determine the minimum precipitation timing (see the Response to 3.o.2.9.i).
3. The use of a new base model spreadsheet that follows the WCAP-16530-NP-A methodology.

An aluminum solubility correlation was used to determine a maximum precipitate formation temperature, which effectively delays the onset of aluminum precipitation. Additionally, WCAP-17788, Volume 5 (Reference 48) autoclave test results were used to determine the minimum precipitation timing (see the Response to 3.o.2.9.i). Per Section 4.0, Item 3, of the WCAP-16530-NP-A Safety Evaluation, if a licensee performs strainer head loss tests with surrogate precipitate and applies a time-based pump NPSH margin acceptance criteria, they must use an aluminum release rate that does not under-predict the initial 15 day aluminum concentrations in Integrated Chemical Effects Test 1 (ICET 1). Therefore, to allow for time-based head loss acceptance criteria, a new spreadsheet was developed which doubles the aluminum release rate from aluminum metal over the initial 15 days. Additionally, the aluminum solubility was used to conservatively decrease the aluminum concentration after precipitation occurs, which increases the rate of release from insulation materials and concrete post-precipitation.

Comparison of the simulated ICET 1 test results (Figure 3.o.2.7-1) with the measured aluminum concentrations (Figure 3.o.2.7-2) verifies that the new spreadsheet accurately predicts ICET 1 aluminum release over the 30-day duration.

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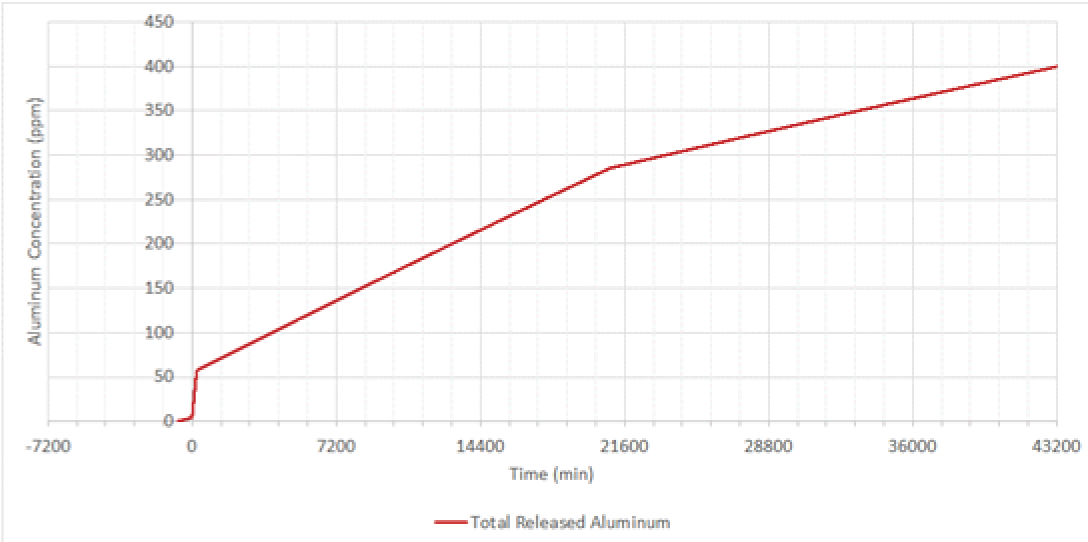

Figure 3.o.2.7-1: Simulation of ICET 1 Al Concentration

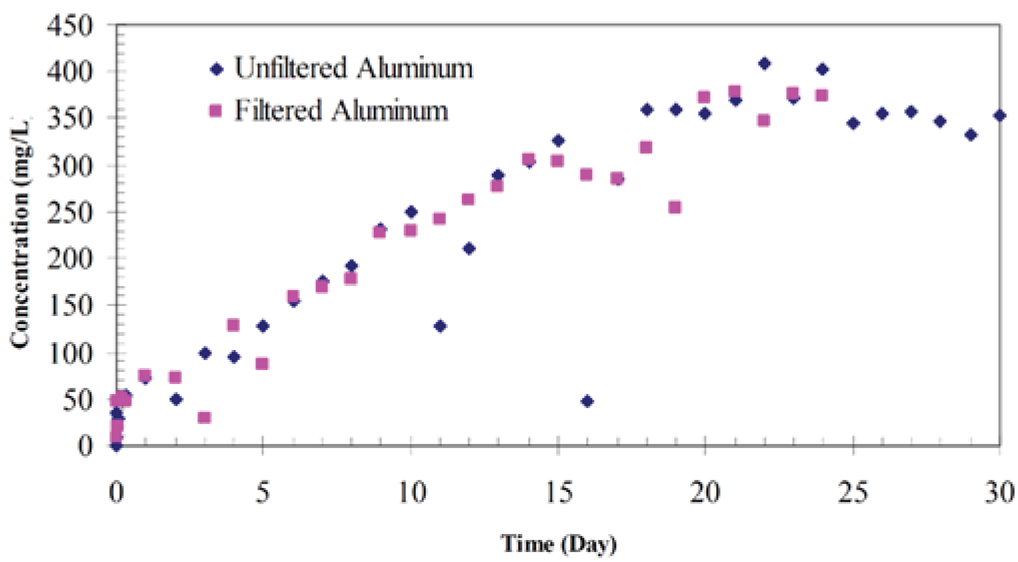

Figure 3.o.2.7-2: Measured Aluminum Concentrations in ICET 1

The chemical precipitates assumed as surrogates for all chemical debris that could form in the containment sump by the WCAP-16530-NP-A methodology for plants that use NaOH buffer are AlOOH and sodium aluminum silicate ($NaAlSi_3O_8$). The WCAP-16530-NP-A precipitation model assumes that $NaAlSi_3O_8$ would precipitate first if there is dissolved silicon, and the remaining aluminum would precipitate as AlOOH. However, per the WCAP-16530-NP-A Safety Evaluation, either aluminum precipitate is an acceptable surrogate for aluminum precipitate in head loss testing. Therefore, to simplify head loss testing, only AlOOH is predicted to form by the new spreadsheet.

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ii. *Licensees should list the type (e.g., AlOOH) and amount of predicted plant-specific precipitates.*

Response 3.o.2.7.ii

Using the WCAP-16530-NP-A (Reference 30) methodology as described in the Response to 3.o.2.7.i, a bounding AlOOH precipitate mass of 147 kg was calculated for Wolf Creek. Using the aluminum solubility correlation described in the Response to 3.o.2.9.i, the maximum temperature where aluminum precipitation could occur in the containment sump pool was calculated to be 116.3°F. As described in the Response to 3.o.2.9.i, comparison of the maximum aluminum concentration (total released at 24 hours) with autoclave test results demonstrates that aluminum precipitation will not occur prior to 24 hours for containment sump pool temperatures as low as 87°F.

8. *WCAP Refinements:*

Response to 3.o.2.8

Refinement to the model for aluminum solubility is discussed in the Response to 3.o.2.9.i. No other refinements to the WCAP-16530-NP-A methodology were used.

9. *Solubility of Phosphates, Silicates and Al Alloys:*

i. *Licensees should clearly identify any refinements (plant-specific inputs) to the base WCAP-16530 model and justify why the plant-specific refinement is valid.*

Response to 3.o.2.9.i

The base WCAP-16530-NP-A model assumes that aluminum precipitates form immediately upon the release of aluminum into solution. However, as discussed in the Response to 3.o.2.7.i, the Wolf Creek chemical model includes the following application of an aluminum solubility correlation and autoclave testing to determine formation temperature and timing. This allows for the application of strainer chemical head loss to be delayed to lower accident temperatures when a greater head loss margin is available.

The aluminum solubility limit was determined using Equation 3.o.2.9.i-1.

$$C_{Al,sol} = \begin{cases} 26980 \cdot 10^{(pH+\Delta pH)-14.4+0.0243T}, & \text{if } T \leq 175\ ^\circ F \\ 26980 \cdot 10^{(pH+\Delta pH)-10.41+0.00148T}, & \text{if } T > 175\ ^\circ F \end{cases} \quad \text{(Equation 3.o.2.9.i-1)}$$

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Nomenclature:

ΔpH = pH change due to radiolysis acids
T = solution temperature, °F

Chemical precipitation is shown not to occur within 24 hours for Containment Sump Pool temperatures above 87°F following the accident based on the autoclave testing in WCAP-17788, Volume 5 (Reference 48). See the Response to 3.n.1 for additional details on the determination of the minimum precipitation timing.

The aluminum solubility limit equation (Equation 3.o.2.9.i-1), developed by Argonne National Laboratory (ANL), was used to determine the temperature and timing of aluminum precipitation after 24 hours and to determine the aluminum concentration in solution for use in the aluminum release equations for concrete and insulation. When precipitation was predicted by this equation after 24 hours, the full amount of aluminum released was assumed to precipitate. The aluminum solubility limit equation was not used to reduce the predicted quantity of precipitate by crediting the amount remaining in solution. Additionally, as discussed in the Response to 3.o.2.2.i, bounding chemical quantities were introduced in head loss testing. Lastly, as discussed in the Response to 3.o.2.7.i, the aluminum release rate from aluminum metal was doubled over the initial 15 days to allow for time-based head loss acceptance criteria. Therefore, the refinement is valid because the overall chemical effects evaluation remains conservative.

ii. *For crediting inhibition of aluminum that is not submerged, licensees should provide the substantiation for the following: (1) the threshold concentration of silica or phosphate needed to passivate aluminum, (2) the time needed to reach a phosphate or silicate level in the pool that would result in aluminum passivation, and (3) the amount of containment spray time (following the achieved threshold of chemicals) before aluminum that is sprayed is assumed to be passivated.*

Response to 3.o.2.9.ii

Silicon and phosphate inhibition of aluminum release were not credited.

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iii. *For any attempts to credit solubility (including performing integrated testing), licensees should provide the technical basis that supports extrapolating solubility test data to plant-specific conditions. In addition, licensees should indicate why the overall chemical effects evaluation remains conservative when crediting solubility given that small amount of chemical precipitate can produce significant increases in head loss.*

Response to 3.o.2.9.iii

The overall chemical effects evaluation remains conservative because a reduction in precipitate quantity due to residual solubility of aluminum after precipitation occurs was not credited. Additionally, bounding chemical quantities were introduced in head loss testing. Lastly, the aluminum release rate from aluminum metal was doubled over the initial 15 days to allow for time-based head loss acceptance criteria. See the Response to 3.o.2.9.i for the technical discussion regarding the credit of aluminum solubility for temperature and timing.

iv. *Licensees should list the type (e.g., AlOOH) and amount of predicted plant-specific precipitates.*

Response to 3.o.2.9.iv

Using the WCAP-16530-NP-A (Reference 30) methodology as described in the Response to 3.o.2.7.ii, a bounding AlOOH precipitate mass of 147 kg was calculated for Wolf Creek. Using the aluminum solubility correlation described in the Response to 3.o.2.9.i, the maximum temperature where aluminum precipitation could occur in the containment sump pool was calculated to be 116.3°F. As described in the Response to 3.o.2.9.i, comparison of the maximum aluminum concentration (total released at 24 hours) with autoclave test results demonstrates that aluminum precipitation will not occur prior to 24 hours for containment sump pool temperatures as low as 87°F.

10. *Precipitate Generation (Decision Point):*

Response to 3.o.2.10

As discussed in the Response to 3.o.2.12.i, Wolf Creek pre-mixed surrogate chemical precipitates in a separate mixing tank for chemical head loss testing. The direct chemical injection method was not used in head loss testing.

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11. *Chemical Injection into the Loop:*

   i. *Licensees should provide the one-hour settled volume (e.g., 80 ml of 100 ml solution remained cloudy) for precipitate prepared with the same sequence as with the plant-specific, in-situ chemical injection.*

   Response to 3.o.2.11.i

   The direct chemical injection method was not used in head loss testing for Wolf Creek. See Figure 3.o.2-1.

   ii. *For plant-specific testing, the licensee should provide the amount of injected chemicals (e.g., aluminum), the percentage that precipitates, and the percentage that remains dissolved during testing.*

   Response to 3.o.2.11.ii

   The direct chemical injection method was not used in head loss testing for Wolf Creek. See Figure 3.o.2-1.

   iii. *Licensees should indicate the amount of precipitate that was added to the test for the head loss of record (i.e., 100 percent, 140 percent of the amount calculated for the plant).*

   Response to 3.o.2.11.iii

   The direct chemical injection method was not used in head loss testing for Wolf Creek. See Figure 3.o.2-1.

12. *Pre-Mix in Tank:*

   i. *Licensees should discuss any exceptions taken to the procedure recommended for surrogate precipitate formation in WCAP-16530.*

   Response to 3.o.2.12.i

   The WCAP-16530-NP-A precipitate formation methodology for AlOOH was followed with no exceptions.

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13. *Technical Approach to Debris Transport (Decision Point):*

Response to 3.o.2.13

Wolf Creek chemical effects testing used hydraulic and manual agitation and turbulence in the test tank to ensure that essentially all debris added to the test tank reached the strainer in head loss testing. The debris quantities were corrected for the mass of debris trapped in the debris introduction hopper. The debris removed from the test tank through the draining of test fluid from the transition tank was considered to have a negligible effect on head loss given the debris had more than one opportunity to filter at the debris bed. Therefore, Wolf Creek did not credit any near field settlement in head loss testing. Refer also to the Response to 3.f.12.

14. *Integrated Head Loss Test with Near-Field Settlement Credit:*

i. *Licensees should provide the one-hour or two-hour precipitate settlement values measured within 24 hours of head loss testing.*

Response to 3.o.2.14.i

Wolf Creek is not crediting near field settlement of chemical precipitate in chemical head loss testing. See Figure 3.o.2-1.

ii. *Integrated Head Loss Test with Near-Field Settlement Credit: Licensees should provide a best estimate of the amount of surrogate chemical debris that settles away from the strainer during the test.*

Response to 3.o.2.14.i

Wolf Creek is not crediting near field settlement of chemical precipitate in chemical head loss testing. See Figure 3.o.2-1.

15. *Head Loss Testing Without Near Field Settlement Credit:*

i. *Licensees should provide an estimate of the amount of debris and precipitate that remains on the tank/flume floor at the conclusion of the test and justify why the settlement is acceptable.*

Response to 3.o.2.15.i

Measures taken during the test, as described in the Response to 3.f.12, to keep debris suspended and transportable to the test strainer, prevented notable

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settling of debris or precipitate in the test tank, upstream of the test strainers, during the FDL test. After drain down of the TB test, a negligible quantity of debris was found to have settled upstream of the strainers.

A description of the test apparatus can be found in the Response to 3.f.4. There were two ways for debris to be removed from the test during normal test operations. First, each time the transition tank was drained, its contents exited the test loop. The debris removed from the test tank through the transition tank was considered to have a negligible effect on head loss given the debris had more than one opportunity to filter at the debris bed. Second, conventional debris trapped in the debris introduction hopper could also leave the test loop when the hopper was drained after its isolation following fines additions. Of the conventional debris, 296.46 grams (0.62%) was drained from the test loop during the FDL test, 262.50 grams of which were drained from the hopper. Debris trapped in the hopper did not have an opportunity to be filtered by the debris bed and was subtracted out from the quantity of debris added to the strainer during the test. Only 24.13 grams (0.05%) of conventional debris was drained from the test loop during the TB test. Therefore, the amount of conventional debris that remained on the tank/flume floor that had not transported to the strainers is negligible for both tests.

Of the chemical debris, 18.94 grams (0.12%) was removed from the FDL test and 209.72 grams (1.35%) was removed from the TB test through the transition tank. The chemical debris drained from the transition tank had to first travel through the debris bed prior to being directed into the transition tank. Any chemical debris drained from the transition tank therefore did not have a meaningful effect on head loss. Lastly, due to the measures to keep debris suspended, chemical debris that remained on the tank/flume floor after drain down had passed through the strainer during the test, and was therefore shown not to impact strainer head loss.

ii. Licensees should provide the one-hour or two-hour precipitate settlement values measured and the timing of the measurement relative to the start of head loss testing (e.g., within 24 hours).

Response to 3.o.2.15.ii

All precipitates met the acceptance criteria provided in WCAP-16530-NP-A (Reference 30). AlOOH precipitate settling was measured within 24 hours of the time the surrogate was used and the 1-hour settled volume (for an initial 10 ml solution volume) was 6 ml or greater and within 1.5 ml of the freshly prepared surrogate.

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16. *Test Termination Criteria:*

i. *Licensees should provide the test termination criteria.*

Response to 3.o.2.16.i

For both the FDL and TB tests, head loss was stabilized after all of the chemical debris was introduced. A period of time of at least 5 pool turn-overs and during which the increase in strainer head loss was less than 1% in two consecutive 30 minute intervals was observed before continuing to debris bed characterization. The debris beds in this state were characterized using both a temperature and a flow sweep.

17. *Data Analysis:*

i. *Licensees should provide a copy of the pressure drop curve(s) as a function of time for the testing of record.*

Response to 3.o.2.17.i

The pressure drop curves as a function of time are shown for the FDL test and the thin bed test in Figure 3.f.4-5, Figure 3.f.4-6, Figure 3.f.4-7 and Figure 3.f.4-8.

ii. *Licensees should explain any extrapolation methods used for data analysis.*

Response to 3.o.2.17.ii

Head loss tests were conducted in a manner where the final test conditions allowed for the maximum head loss 30 days after the start of an accident to be evaluated without extrapolation of the test data. Testing was never moved to the next phase or terminated when head loss was increasing. The maximum stabilized head loss, which was not increasing or decreasing by more than 1% in an hour, at the completion of chemical debris additions was 1.914 psi at 95.6°F and 0.364 at 79.2°F for the FDL test and TB test respectively. This value was observed prior to the start of the final temperature and flow sweeps for the FDL test and after the temperature sweep before the final flow sweep for the TB test. As the head loss was stable at the time the previously mentioned head loss measurement was taken, no extrapolation was required when estimating 30-day head loss values.

Head loss adjustments for temperature and flow rate are described in the Not applicable for Wolf Creek Response to 3.f.10.

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18. <u>Integral Generation (Alion)</u>:

    i. *Licensees should discuss why the test parameters (e.g., temperature, pH) provide for a conservative chemical effects test.*

    Response to 3.o.2.18.i

    Wolf Creek is using the separate chemical effects approach to determine the chemical source term. This section is not applicable to the Wolf Creek chemical effects analysis. See Figure 3.o.2-1.

19. <u>Tank Scaling / Bed Formation</u>:

    i. *Scaling factors for the test facilities should be representative or conservative relative to plant-specific values.*

    Response to 3.o.2.19.i

    Wolf Creek is using the separate chemical effects approach to determine the chemical source term. This section is not applicable to the Wolf Creek chemical effects analysis. See Figure 3.o.2-1.

    ii. *Bed formation should be representative of that expected for the size of materials and debris that is formed in the plant specific evaluation.*

    Response to 3.o.2.19.ii

    Wolf Creek is using the separate chemical effects approach to determine the chemical source term. This section is not applicable to the Wolf Creek chemical effects analysis. See Figure 3.o.2-1.

20. <u>Tank Transport</u>:

    i. *Transport of chemicals and debris in testing facility tanks should be representative or conservative with regard to the expected flow and transport in the plant-specific conditions.*

    Response to 3.o.2.20.i

    Wolf Creek is using the separate chemical effects approach to determine the chemical source term. This section is not applicable to the Wolf Creek chemical effects analysis. See Figure 3.o.2-1.

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21. *<u>30-Day Integrated Head Loss Test</u>:*

i. *Licensees should provide the plant-specific test conditions and the basis for why these test conditions and test results provide for a conservative chemical effects evaluation.*

Response to 3.o.2.21.i

Wolf Creek is using the separate chemical effects approach to determine the chemical source term. This section is not applicable to the Wolf Creek chemical effects analysis. See Figure 3.o.2-1.

ii. *Licensees should provide a copy of the pressure drop curve(s) as a function of time for the testing of record.*

Response to 3.o.2.21.ii

Wolf Creek is using the separate chemical effects approach to determine the chemical source term. This section is not applicable to the Wolf Creek chemical effects analysis. See Figure 3.o.2-1.

22. *<u>Data Analysis Bump Up Factor</u>:*

i. *Licensees should provide the details and the technical basis that show why the bump-up factor from the particular debris bed in the test is appropriate for application to other debris beds.*

Response to 3.o.2.22.i

Wolf Creek is using the separate chemical effects approach to determine the chemical source term. This section is not applicable to the Wolf Creek chemical effects analysis. See Figure 3.o.2-1.

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3.p Licensing Basis

The objective of the licensing basis section is to provide information regarding any changes to the plant licensing basis due to the sump evaluation or plant modifications.

3.p.1 Provide the information requested in GL 2004-02 Requested Information Item 2(e) regarding changes to the plant licensing basis. The effective date for changes to the licensing basis should be specified. This date should correspond to that specified in the 10 CFR 50.59 evaluation for the change to the licensing basis.

GL 2004-02 Requested Information Item 2(e)
A general description of and planned schedule for any changes to the plant licensing bases resulting from any analysis or plant modifications made to ensure compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this GL. Any licensing actions or exemption requests needed to support changes to the plant licensing basis should be included.

Response to 3.p.1:

As shown in this submittal, Wolf Creek is using a simplified risk-informed approach to respond to GL 2004-02. The overall approach is similar to what Vogtle used in their GL 2004-02 submittals (Reference 52; 53) with the exception that Wolf Creek used a simplified threshold break size approach to quantify the risk increase due to strainer and reactor core failures associated with LOCA-generated debris. The proposed change, replacing the current deterministic methodology with a risk-informed methodology, requires changes to the descriptions of how Wolf Creek meets the 10 CFR 50.46(a)(1) requirements. An exemption to certain requirements of 10 CFR 50.46(a)(1) is provided in Attachment II of this submittal.

Wolf Creek's risk-informed approach to assess the effects of LOCA debris replaces the existing deterministic approach described in the Wolf Creek licensing basis. This, in turn, requires an amendment to the Wolf Creek operating license to incorporate the revised methodology per the requirements of 10 CFR 50.59. This proposed amendment to the operating license is included in Attachment I of this submittal.

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4. NRC Request for Additional Information

Table 4-1 summarizes the outstanding requests for additional information (RAIs) issued by the NRC on the previous Wolf Creek GL 2004-02 submittals (Reference 54). A brief summary of the response is provided for each RAI, along with a reference to relevant sections of this submittal.

Table 4-1 Responses to Outstanding RAIs from Past Wolf Creek GL 2004-02 Submittals

RAI No.	RAI	Response
1	Please verify that the insulation types and amounts are distributed relatively symmetrically between all loops to validate that the focus on loop D provided a conservative break selection evaluation, or otherwise justify the assumption.	This RAI is no longer applicable because the assumption is no longer used. The new Wolf Creek debris generation analysis evaluated breaks on all Class 1 in-service ISI welds inside the containment (see the Response to 3.a.1).
2	Please justify that the 3-inch charging line break provides the greatest debris generation for the partially submerged conditions. Please state whether there other breaks, potentially on larger lines, that could result in a larger debris term, yet still result in partial submergence. Please provide results of evaluations and testing that verify that the debris generated by the limiting break that results in partial submergence will not result in unacceptable head loss (strainer failure). Please either state that this evaluation is based on a U.S. Nuclear Regulatory Commission (NRC) staff-accepted test methodology or justify use of a different methodology. Note that the NRC staff considers testing conducted at Alden Labs prior to 2008 likely to be non-conservative. Alternately, please verify that the strainer will be fully submerged for all small break loss-of-coolant accident (SBLOCA) conditions as described on page 47 of the December 22, 2008, supplemental response (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090060877), such that the large break loss-of-coolant accident (LBLOCA) testing bounds the SBLOCA.	The strainer is shown to be fully submerged for all breaks (see the Response to 3.g.1). Wolf Creek has performed new head loss testing in 2016 following the NRC March 2008 guidance (see the Response to 3.f.4).

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RAI No.	RAI	Response
3	Although American National Standards Institute (ANSI)/American Nuclear Society (ANS) standard 58-2-1988, "Design Basis for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Rupture," predicts higher jet centerline stagnation pressures associated with higher levels of subcooling, it is not intuitive that this would necessarily correspond to a generally conservative debris generation result. Please justify the initial debris generation test temperature and pressure with respect to the plant-specific reactor coolant system (RCS) conditions, specifically the plant hot and cold leg operating conditions. If ZOI reductions are also being applied to lines connecting to the pressurizer, then please also discuss the temperature and pressure conditions in these lines. Please describe results of any tests conducted at alternate temperatures and pressures to assess the variance in the destructiveness of the test jet to the initial test condition specifications.	This RAI is no longer applicable. All of the ZOI sizes used in the new Wolf Creek debris generation analysis were approved by the NRC (see the Response to 3.b.1).
4	Please describe the jacketing/insulation systems used at Wolf Creek Generating Station (WCGS) for which the ZOI reduction is sought and compare those systems to the jacketing/insulation systems tested, demonstrating that the tested jacketing/insulation system adequately represent the plant jacketing/insulation system. The description should include differences in the jacketing and banding systems used for piping and other components for which the test results are applied. At a minimum, the following areas should be addressed:	This RAI is no longer applicable because the new Wolf Creek debris generation analysis did not credit ZOI reduction based on jacketing of the insulation (see the Response to 3.b.1).
4.a	Please describe how the characteristic failure dimensions of the tested jacketing/insulation compared with the effective diameter of the jet at the axial placement of the target. The characteristic failure dimensions are based on the primary failure mechanisms of the jacketing system (e.g., for a stainless steel jacket held in place by three latches where all three latches must fail for the jacket to fail, then all three latches must be effectively impacted by the pressure for which the ZOI is calculated). Applying test results to a ZOI based on a centerline pressure for relatively low LID nozzle to target spacing would be non-conservative with respect to impacting the entire target with the calculated pressure.	

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RAI No.	RAI	Response
4.b	Please explain whether the insulation and jacketing system used in the testing was of the same general manufacture and manufacturing process as the insulation used in the plant. If not, please explain what steps were taken to ensure that the general strength of the insulation system tested was conservative with respect to the plant insulation. For example, it is known that there were generally two very different processes used to manufacture calcium silicate whereby one type readily dissolved in water but the other type dissolves much more slowly. Such manufacturing differences could also become apparent in debris generation testing, as well.	
4.c	Please provide results of an evaluation of scaling the strength of the jacketing or encapsulation systems to the tests. For example, a latching system on a 30-inch pipe within a ZOI could be stressed much more than a latching system on a 10-inch pipe in a scaled ZOI test. If the latches used in the testing and the plants are the same, the latches in the testing could be significantly under-stressed. If a prototypically sized target were impacted by an undersized jet it would similarly be under-stressed. Evaluations of banding, jacketing, rivets, screws, etc., should be made. For example, scaling the strength of the jacketing was discussed in the Ontario Power Generation report, "Jet Impact Tests -Preliminary Results and Their Application, N-REP-34320-10000," dated April 18, 2001 (ADAMS Accession No. ML020290085), on calcium silicate debris generation testing.	
5	There are relatively large uncertainties associated with calculating jet stagnation pressures and ZOIs for both the test and the plant conditions based on the models used in the WCAP reports. Please describe steps were taken to ensure that the calculations resulted in conservative estimates of these values. Please provide the inputs for these calculations and describe the sources of the inputs.	This RAI is no longer applicable. See the response to RAI #3.
6	Please describe the procedure and assumptions for using the ANSI/ANS-58-2-1988 standard to calculate the test jet stagnation pressures at specific locations downrange from the test nozzle. As part of this description, please address the following points.	This RAI is no longer applicable. See the response to RAI #3.
6.a	In WCAP-16710-P, "Jet Impingement Testing to Determine the Zone of Influence (ZOI) of Min-K and NUKON Insulation, for Wolf Creek and Callaway Nuclear Operating Plants," please explain why the analysis was based on the initial condition of 530°F whereas the initial test temperature was specified as 550°F.	

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RAI No.	RAI	Response
6.b	Please explain whether the water subcooling used in the analysis was that of the initial tank temperature or the temperature of the water in the pipe next to the rupture disk. Test data indicated that the water in the piping had cooled below that of the test tank.	
6.c	The break mass flow rate is a key input to the ANSI/ANS-58-2-1988 standard. Please explain how the associated debris generation test mass flow rate was determined. If the experimental volumetric flow was used, then explain how the mass flow was calculated from the volumetric flow given the considerations of potential two-phase flow and temperature-dependent water and vapor densities. If the mass flow was analytically determined, then describe the analytical method used to calculate the mass flow rate.	
6.d	Noting the extremely rapid decrease in nozzle pressure and flow rate illustrated in the test plots in the first tenths of a second, please explain how the transient behavior was considered in the application of the ANSI/ANS-58-2-1988 standard. Specifically, please explain whether the inputs to the standard represented the initial conditions or the conditions after the first extremely rapid transient (e.g., say at one tenth of a second).	
6.e	Given the extreme initial transient behavior of the jet, please justify the use of the steady-state ANSI/ANS-58-2-1988 standard jet expansion model to determine the jet centerline stagnation pressures rather than experimentally measuring the pressures.	
7	Please describe the procedure used to calculate the isobar volumes used in determining the equivalent spherical ZOI radii using the ANSI/ANS-58-2-1988 standard. Please include discussions of the following points.	This RAI is no longer applicable. See the response to RAI #3.
7.a	Please provide the assumed plant-specific RCS temperatures and pressures and break sizes used in the calculation. Note that the isobar volumes would be different for a hot-leg break than for a cold-leg break since the degrees of subcooling is a direct input to the ANSI/ANS-58-2-1988 standard and which affects the diameter of the jet. Note that an under-calculated isobar volume would result in an under-calculated ZOI radius.	
7.b	Please describe the calculational method used to estimate the plant-specific and break-specific mass flow rate for the postulated plant loss-of-coolant accident (LOCA), which was used as input to the standard for calculating isobar volumes.	

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RAI No.	RAI	Response
7.c	Given that the degree of subcooling is an input parameter to the ANSI/ANS-58-2-1988 standard and that this parameter affects the pressure isobar volumes, please describe steps taken to ensure that the isobar volumes conservatively match the plant-specific postulated LOCA degree of subcooling for the plant debris generation break selections. Please explain whether multiple break conditions were calculated to ensure a conservative specification of the ZOI radii.	
8	Please provide a detailed description of the test apparatus, specifically including the piping from the pressurized test tank to the exit nozzle including the rupture disk system. Please also address the following related points:	This RAI is no longer applicable. See the response to RAI #3.
8.a	Based on the temperature traces in the test reports it is apparent that the fluid near the nozzle was colder than the bulk test temperature. Please explain how the fact that the fluid near the nozzle was colder than the bulk fluid was accounted for in the evaluations.	
8.b	Please explain how the hydraulic resistance of the test piping which affected the test flow characteristics was evaluated with respect to a postulated plant-specific LOCA break flow, where such piping flow resistance would not be present.	
8.c	Please provide the specified rupture differential pressure of the rupture disks.	
9	WCAP-16710-P discusses the shock wave resulting from the instantaneous rupture of piping. Please address the following points regarding the shock wave.	This RAI is no longer applicable. See the response to RAI #3.
9.a	Please describe results of analysis or parametric testing conducted to get an idea of the sensitivity of the potential to form a shock wave to different thermal-hydraulic conditions. Please state and justify whether temperatures and pressures prototypical of PWR hot legs were considered.	
9.b	Please explain whether the initial lower temperature of the fluid near the test nozzle was taken into consideration in the evaluation, and if not, why not. Specifically, please explain and justify whether the damage potential was assessed as a function of the degree of subcooling in the test initial conditions.	
9.c	Please provide the basis for scaling a shock wave from the reduced-scale nozzle opening area tested to the break opening area for a limiting rupture in the actual plant piping.	
9.d	Please compare how the effect of a shock wave was scaled with distance for the test nozzle, and compare that with the expected plant condition.	

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RAI No.	RAI	Response
10	Please provide the basis for concluding that a jet impact on piping insulation with a 45 degree seam orientation is a limiting condition for the destruction of insulation installed on steam generators, pressurizers, reactor coolant pumps, and other non-piping components in the containment. For instance, considering a break near the steam generator nozzle, once insulation panels on the steam generator directly adjacent to the break are destroyed, the LOCA jet could impact additional insulation panels on the generator from an exposed end, potentially causing damage at significantly larger distances than for the insulation configuration on piping that was tested. Furthermore, it is not clear that the banding and latching mechanisms of the insulation panels on a steam generator or other RCS components provide the same measure of protection against a LOCA jet as those of the piping insulation that was tested. Although WCAP-16710-P asserts that a jet at WCGS or Callaway cannot directly impact the steam generator, but will flow parallel to it, it seems that some damage to the SG insulation could occur near the break, with the parallel flow then jetting under the surviving insulation, perhaps to a much greater extent than predicted by the testing. Similar damage could occur to other component insulation. Please provide a technical basis to demonstrate that the test results for piping insulation are prototypical or conservative of the degree of damage that would occur to insulation on steam generators and other non-piping components in the containment.	This RAI is no longer applicable. See the response to RAI #3.
11	Some piping oriented axially with respect to the break location (including the ruptured pipe itself) could have insulation stripped off near the break. Once this insulation is stripped away, succeeding segments of insulation would have one open end exposed directly to the LOCA jet, which appears to be a more vulnerable configuration than the configuration tested by Westinghouse. As a result, damage would seemingly be capable of propagating along an axially oriented pipe significantly beyond the distances calculated by Westinghouse. Please provide a technical basis to demonstrate that the reduced ZOIs calculated for the piping configuration tested are prototypical or conservative with respect to the degree of damage that would occur to insulation on piping lines oriented axially with respect to the break location.	This RAI is no longer applicable. See the response to RAI #3.

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RAI No.	RAI	Response
12	WCAP-16710-P noted damage to the cloth blankets that cover the fiberglass insulation, in some cases resulting in the release of fiberglass. The tears in the cloth covering were attributed to the steel jacket of the test fixture and not the steam jet. Please justify the assumption that damage that occurs to the target during the test would not be likely to occur in the plant. Please explain whether the potential for damage to plant insulation from similar conditions was considered. For example, the test fixture could represent a piping component or support, or other nearby structural member. Please provide the basis for the statement in the WCAP that damage similar to that which occurred to the end pieces would not be expected to occur in the plant. It is likely that a break in the plant will result in a much more chaotic condition than that which occurred in testing. Therefore, it would be more likely for the insulation to be damaged by either the jacketing or other objects nearby.	This RAI is no longer applicable. See the response to RAI #3.
13	For the Min-K panel testing, one specimen was ejected from the test fixture and impacted a tree some 150 feet away. This impact resulted in minor damage to the encapsulation. Please provide the results of evaluations of the potential for a similar occurrence in the plant, including at distances much closer than 150 feet as applicable to the plant. Please provide the result if the panel lodged within the jet ZOI, as well as whether the encapsulating material could fatigue, fail, and allow the insulating material to be released.	See the Responses to 3.a.1 for the exclusion of the reactor nozzle breaks and the Response to 3.c.4 for Min-K insulation not being impacted.

Updated Response to NRC Generic Letter 2004-02

RAI No.	RAI	Response
14	Please provide a comparison of the Thermal-Wrap and Nukon insulation systems that justifies that the Thermal-Wrap system is at least as structurally robust as the Nukon. The licensee's response only describes the similarity of the base material fibers and claims similarity was asserted in the NRC staff's safety evaluation (SE) dated December 6,2004 (ADAMS Accession No. IVIL043280641), on NEI 04-07. This conclusion in the SE was reached on the basis of a large 17D ZOI being assumed, one that was likely conservative for both materials, whether the jacketing is similar or perhaps even present. However, when conservatisms are removed to arrive at a smaller ZOI (i.e., effectively 5D), it becomes necessary to demonstrate similarity of the jacketing, banding, scrim, and cloth covers to show sufficient similarity. Based on testing done by Westinghouse/Wyle for Arkansas Nuclear One (ANO) (Entergy Operations, Inc. letter dated February 28, 2008, ADAMS Accession No. ML08071 0544), some damage was seen for Thermal Wrap at 12D and at 7D, which suggests a potential for increased vulnerability. It was not clear to the NRC staff why this Thermal Wrap test is not more applicable to the Thermal Wrap at WCGS than a test performed for a different material (i.e., Nukon). Therefore, please justify treating Thermal Wrap with a reduced ZOI based on Nukon testing, in terms of the jacketing, banding and/or latching, scrim, and cloth cover for the Thermal Wrap insulation to provide confidence that it is comparable to the jacketing system for the Nukon insulation system that was tested.	This RAI is not applicable because Wolf Creek does not have this debris type.

Updated Response to NRC Generic Letter 2004-02

RAI No.	RAI	Response
15	The supplemental response dated February 29, 2008 (pg 14 of 82), stated that the Min-K at WCGS is located near the reactor vessel. Please state whether spherical resizing was performed for the Min-K ZOI and, if so, justify that it is appropriate for this location considering that substantial physical obstructions could result in a significantly non-spherical ZOI. The NRC staff's May 16, 2007, audit report for San Onofre Nuclear Generating Station discusses a potentially similar issue (Open Item 1 in Section 3.2, ADAMS Accession No. ML071240024) regarding Microtherm insulation that was located on the reactor vessel, for which spherical resizing was considered inappropriate by the NRC staff due to the constraints imposed by the biological shield wall and reactor vessel. The WCAP report states that a 1/8-inch offset was assumed rather than full separation for the Min-K break thus spherical scaling would not have been specified per the ANSI/ANS-58-2-1988 model. The supplemental response and debris generation test report do not discuss how the scaling for the 1/8-inch offset case is handled. It is not clear that the ANSIIANS-58-2-1988 (non-spherical) model for the limited separation case was done for the Min-K. Please justify that an offset circumferential break is the only type of break necessary to consider, or provide results of evaluations of other configurations. For example, should a longitudinal break be postulated? If the Min-K is damaged by a restrained break, please justify that a 30-second duration jet impingement is adequate to model the blowdown from the break. Please explain and justify whether the blowdown should be longer for a restrained break with a smaller effective break area.	See the Responses to 3.a.1 for the exclusion of the reactor nozzle breaks and the Response to 3.c.4 for Min-K insulation not being impacted.

Updated Response to NRC Generic Letter 2004-02

RAI No.	RAI	Response
16	The assumed debris size distribution of 60 percent small fines and 40 percent large pieces for low-density fiberglass within a 5D ZOI is inconsistent with Figure 11-2 of the NRC staffs SE dated December 6, 2004 (ADAMS Accession No. ML043280641), on NEI 04-07, which considers past air jet testing and indicates that the fraction of small fines should be assumed to reach 100 percent at jet pressures in the vicinity of 18-19 pounds per square inch (psi). At 5D, the jet pressure is close to 30 psi, which significantly exceeds this threshold. Furthermore, the licensee's assumption that the size distribution for debris in a range of 5D to 7D is 100 percent intact blankets also appears to be inconsistent with existing destruction testing data. These assumptions for low-density fiberglass debris size distributions appear to be based on the recent Westinghouse/Wyle ZOI testing discussed in WCAP-16710-P. However, that testing was not designed to provide size distribution information, and much of the target material was exposed to jet pressures much lower than would be expected for a prototypically sized break. Furthermore, given the assumption that insulation between 5D and 7D is 100 percent intact pieces that do not transport or erode, the licensee has effectively assumed a 5D ZOI rather than a 7D ZOI for low-density fiberglass. In light of the discussion above concerning previous testing experience, please provide a basis for considering the assumed debris size distribution of 60 percent small fines and 40 percent large pieces within a 5D ZOI to be conservative or prototypical.	This RAI is no longer applicable. The new Wolf Creek debris generation analysis used a 17D ZOI for Nukon, along with the four-category size distribution, as provided in the NRC SE on NEI 04-07 (see the Responses to 3.b.1 and 3.c.1).

Updated Response to NRC Generic Letter 2004-02

RAI No.	RAI	Response
17	The NEI 04-07, along with the NRC staff SE on that document, provides information regarding the treatment of the characteristics of fibrous debris generated from a break. The guidance report states that small fines are individual fibers. However, the staff SE notes that this is likely to result in problems in the treatment of fibrous debris. The amount of fines and small pieces of debris should be defined separately so that inputs for transport analyses and head loss testing are well defined. The estimation of fine debris amounts is especially important for testing that allows near-field settling. The guidance documents, such as Appendix II to the SE on NEI 04-07, indicate that reduced ZOIs generally result in increased percentages of small and fine debris. The supplemental response dated December 22, 2008, stated that 30 percent of the small fibrous debris added to the head loss test was estimated to be in the form of fines, but the response did not provide a basis for this assumption, such as an analytical evaluation of expected quantities of the plant fibrous debris determined to be fines. The ZOI reduction taken for Nukon should reflect the phenomenon demonstrated in SE Appendix II of increased debris fragmentation near the break location when the debris sizing is estimated, or the licensee should justify otherwise. Please identify the amounts of fine fibrous debris predicted to be generated from the analyzed limiting breaks.	This RAI is no longer applicable. See the Response to RAI #16.
18	To the extent Foamglass and Cerablanket are included in the limiting debris loading case, please provide an evaluation of their characteristics, such as characteristic size distribution (small pieces and fines, etc.) and density, and please provide justification for deviation from staff-approved evaluation methodologies.	See the Response to 3.a.3. In the updated Debris Generation analysis, no antisweat, Cerablanket, lead blanket or FOAMGLAS materials were destroyed by any analyzed DEGBs or partial breaks inside the first isolation valve and outside of reactor cavity because they are outside of the ZOIs for all analyzed breaks.

Updated Response to NRC Generic Letter 2004-02

RAI No.	RAI	Response
19	The NRC staff's December 6, 2004, SE for NEI 04-07 states that a maximum of 15 percent holdup of debris should be assumed in inactive holdup regions during pool fill up. For the case of single-train sump operation for WCGS, a two-sump plant, the sump that is not operating essentially becomes an inactive holdup region. From this point of view, the staff observed that WCGS appeared to credit a 15 percent inactive holdup volume in the containment pool, plus 14 percent holdup in the inactive recirculation sump for single-train cases, for a total of 29 percent of debris held up in inactive volumes for these single-train cases (e.g., the Loop D cross-over break, the case considered by the licensee to be bounding). The staff considers this credit a deviation from the approved guidance in the SE, which stated that the limit for inactive hold up should be 15 percent unless a computational fluid dynamics (CFD) analysis was performed that considered the time-dependent containment pool flows during pool fill up. Please provide additional basis for the assumed total inactive holdup fraction of 29 percent or revise this value to within the accepted SE range.	The debris transport analysis credits transport to the inactive cavities in accordance with the NRC staff's SE for NEI 04-07. The transport of debris to inactive cavities during pool fill-up is limited to 15%. See the Response to 3.e.1.
20	The licensee's supplemental responses, including the one dated December 22, 2008, discuss crediting Stokes' Law, but do not specifically quantify the credit taken for application of this methodology. Please state the quantities of fine debris assumed to settle onto the containment floor by applying the Stokes' Law methodology. If credit is taken for such settling, please provide technical justification regarding the following points: (1) lack of experimental benchmarking of analytically derived turbulent kinetic energy (TKE) metrics; (2) uncertainties in the predictive capabilities of TKE models in CFD codes, particularly at the low TKE levels necessary to suspend individual fibers and 10-micron particulate; (3) the basis for analytical prediction of settling velocities in quiescent and non-quiescent water due to the specification of shape factors and drag coefficients for irregularly shaped debris; and (4) the basis for the theoretical correlation of the terminal settling velocity to turbulent kinetic energy that underlies the Alion Science &Technology methodology for fine debris settling. Please address these points to demonstrate that any credit taken for fine debris settling is technically justified.	The debris transport analysis utilizes Stokes' Law to calculate the settling velocity of particulate debris. This analysis is reasonable because the particulate debris analyzed is generally spherical, small in size, and would settle slowly. This approach has little impact on the debris transport analysis because fine debris and particulate have a 100% recirculation transport fraction. See the Responses to 3.e.6.
21	Please provide a description of the testing performed to support the assumption of 10 percent erosion of fibrous debris pieces in the containment pool. Please specifically include the following information:	See the responses below to the sub-bullets of this RAI.

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RAI No.	RAI	Response
21.a	Please describe the test facility used and demonstrate the similarity of the flow conditions (velocity and turbulence), chemical conditions, and fibrous material present in the erosion tests to the analogous conditions applicable to the plant condition.	The generic erosion testing done by Alion was justified to be applicable in a Wolf Creek-specific report. This is consistent with the NRC guidance "*Proprietary Erosion Testing of Submerged Nukon Low-Density Fiberglass Insulation in Support of Generic Safety Issue 191 Strainer Performance Analysis.* June 2010." See the Response to 3.e.1.
21.b	Please provide specific justification for any erosion tests conducted at a minimum tumbling velocity if debris settling was credited in the test flume for velocities in excess of this value (e.g., in front of the curb around the sump pit).	See the Response to RAI 21.a.
21.c	Please identify the duration of the erosion tests and how the results were extrapolated to the sump mission time.	The duration of the erosion test is 30 days, see the Response to 3.e.1.
22	The supplemental response dated December 22, 2008, indicates that a significant percentage of small and large pieces of Nukon were assumed to transport to the strainers (Le., nearly 100 percent of small pieces and approximately 75 percent of large pieces in several cases). This analytical result minimized the quantity of settled small and large pieces of fiberglass that were analytically assumed to erode in the containment pool. However, for the strainer head loss testing conducted by Performance Contracting, Inc. (PCI), the NRC staff considers it likely that a significant fraction of small pieces and most or all of the large pieces of debris that were analytically considered transportable actually settled in the test flume rather than transporting to the test strainer. The head loss testing did not model the erosion of this debris. The licensee's consideration of debris erosion, therefore, appears to be non-conservative, because neither the analysis nor the head loss testing accounted for the erosion of debris that settled during the head loss testing. Please estimate the quantity of eroded fines from large and small pieces of fiberglass debris that would result had erosion of the settled debris in the head loss test flume been accounted for and justify the neglect of this material in the head loss testing program.	A new debris transport analysis has been performed and the resulting transport fractions of small and large pieces are shown in the Response to 3.e.6. Refer to the Response to 3.e.1 for the discussion of erosion of small and large pieces.

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RAI No.	RAI	Response
23	Based upon discussions with the licensee and PCI in January and February 2008, the head loss testing conducted by PCI modeled flow conditions during the recirculation phase of a LOCA and modeled all debris (other than a small quantity of latent debris added with the recirculation pump stopped) as entering the containment pool one flume-length (nominally 33 feet) away from the containment sump strainers. Flow conditions during the pool-fill phase of the LOCA were not considered by the testing, nor was the potential for debris to enter the containment pool closer than one flume-length from the strainer due to the effects of blowdown, washdown, and pool fill transport. The lack of modeling of these two transport aspects of the head loss testing appeared to result in a non-prototypical reduction in the quantity of debris reaching the test strainer and, ultimately, non-conservative measured head loss values. Please provide the technical basis for not explicitly modeling transport modes other than recirculation transport, considering the following points:	The new Wolf Creek debris transport analysis evaluated debris transport during blowdown, washdown, pool fill, and pool recirculation. Transport fractions for all of these phases were used to calculate the overall transport fractions and determine the quantity of debris at the strainer (see the Response to 3.e.6). The transported debris loads were then used to inform the new head loss testing performed in a test tank. No credit was taken for debris settling in the latest head loss testing. See the Response to 3.f.12.
23.a	As shown in Appendix III of the NRC staff's SE on NEI 04-07, containment pool velocity and turbulence values during fill up may exceed those during recirculation, due to the shallowness of the pool. Some debris that would not transport during recirculation may transport during the pool-fill phase. In addition, latent debris on the containment pool floor could be stirred into suspension by these high-velocity, turbulent flows, unlike the latent debris added to the quiescent PCI flume.	
23.b	The pool fill phase will tend to move debris from inside the shield wall into the outer annulus away from the break location and nearer to the recirculation sump strainers.	
23.c	For plants that have strainers located below the floor grade level, the transport of large and small pieces of debris during pool fill can result in this debris accumulating directly on the strainer surfaces, potentially resulting in the formation of a limiting debris layer at the entrance to the sump pit.	
23.d	Representatively modeling the washdown of some fraction of the debris nearer the strainer than one flume-length away would be expected to increase the quantity of debris transported to the strainer and the measured head loss. This statement applies both to debris that tends to settle in the head loss test flume, as well as debris considered to settle analytically, such as various types of paint chips.	

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RAI No.	RAI	Response
24	Please provide the technical basis for the conclusion that large pieces of fibrous debris and, as applicable, reflective metal insulation (RMI) debris, coating chips, and other debris types, have a transport fraction of zero during the pool-fill phase of a LOCA.	Refer to the Response to 3.e.1 for the justification that pool fill-up transport was only considered for fine debris.
25	The supplemental response dated December 22, 2008, states that, based on testing documented in NUREG/CR-2791, "Methodology for Evaluation of Insulation Debris Effects, Containment Emergency Sump Performance Unresolved Safety Issue A-43," dated September 1982, Nukon was assumed not to float on the surface of the containment pool. However, page 51 of NUREG/CR-2791 indicates that large floating fragments of fiberglass under hot (60°C) sprays will sink in 2 to 5 days. As a result, NUREG/CR-2791 stated that it is reasonable to assume that all large floating fragments of fiberglass sink in the vicinity of the strainers. In light of these test results from NUREG/CR-2791, please provide additional basis for the assumption that large or intact pieces of fiberglass debris cannot float for a sufficient period of time to reach the strainers prior to sinking in the containment pool. If floating debris sinks on strainers located in a sump pit, there is the potential for forming a limiting layer of debris at the entrance to the strainer pit.	The new debris transport analysis assumed that Nukon debris would not float in the sump pool based on fibrous debris testing in NUREG/CR-6808. See the Response to 3.e.1.
26	The supplemental response dated December 22,2008, indicates that most miscellaneous debris settled prior to reaching the test strainer. Based on previous observations of other testing performed at PCI, the NRC staff observed that tags, labels, and miscellaneous materials were added to the test flume by submerging them beneath the surface of the test fluid. Submerging miscellaneous debris would not allow the potential for transport to the strainers by floatation to be evaluated. Due to the pit strainer configuration at WCGS, if miscellaneous debris can float toward the strainers and subsequently sink over the strainers, then part of the opening to the strainer pit could be blocked off. Please describe the addition process for miscellaneous debris (e.g., tags, labels, and stickers), and discuss how the potential for transport via floatation was considered in the head loss testing program or by analysis.	This RAI is no longer applicable since it is related to the previous flume testing. In the new head loss tank testing, no miscellaneous debris was added to the test tank. Instead, the impact of miscellaneous debris was accounted for by reducing the effective strainer surface area when scaling the debris loads and flow rate for testing. See the Response to 3.f.4.

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RAI No.	RAI	Response
27	Based on discussions with the licensee and PCI in January and February 2008, the NRC staff considered the modeling of the boundary conditions for the localized CFD model for the vicinity of the sump strainers to be non-conservative. Although the total flow rate from each side of the annulus was taken from the full-containment CFD model by using a constant, averaged velocity boundary condition, the localized CFD model did not simulate the significant channeling of the flow predicted by the full-containment model. As a result, velocities in the vicinity of the strainer were significantly underestimated. Since the localized CFD model was used as the basis for the head loss test flume velocities, the staff considered the test flume velocities as underestimating the velocities at which much of the debris would actually transport to the plant sump strainers. Although the staff recognized that some improvements had been made to the localized CFD model (e.g., a finer mesh resolution), these improvements did not compensate for the inaccuracy in the specification of the inlet boundary conditions. Please provide any information in addition to the information already discussed with the staff that could demonstrate the adequacy of the flow conditions used for the head loss test.	
28	Based on discussions with the licensee and PCI in January and February 2008, the NRC staff understood that the test debris was not categorized into specific subgroups of fines and small pieces. Based on a rough visual inspection, the licensee estimated that 30 percent of the small pieces of fibrous debris added to the test were fines. However, the staff does not consider the licensee's estimate to be sufficient because (1) visual estimation of the relative quantities of fine and small piece debris in a given sample is inherently inaccurate and subjective, and would be expected to vary significantly from sample to sample and (2) the high concentration of debris in the prepared debris slurries resulted in significant debris agglomeration, which likely prevented the fines from transporting prototypically in the test flume in any case. As a result, the NRC staff questions whether the licensee's head loss testing resulted in debris settling under debris preparation conditions that are not prototypical of the limiting plant condition. Please provide any information in addition to the information already discussed with the staff that could demonstrate the adequacy of the transport behavior of the fine debris added to the head loss test.	This RAI is no longer applicable since it is related to the previous flume testing, and new head loss tank testing has been performed since the previous submittal. See the Response to 3.f.4.

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RAI No.	RAI	Response
29	Please discuss any sources of drainage that enter the containment pool near the containment sump strainers (Le., within the range of distances modeled in the head loss test flume). Please identify whether the drainage would occur in a dispersed form (e.g., droplets) or a concentrated form (e.g., streams of water running off of surfaces). Please discuss how these sources of drainage are modeled in the test flume. Please provide contour plots of the calculated turbulence (which include a numerical scale with units) for the CFD calculation for the test flume with that for the full-containment plant CFD calculation. Please address whether the test flume turbulence values are prototypical of the plant condition.	The latest head loss testing was performed as tank tests and did not credit settling of debris. The turbulence inside the test tank was maintained high enough to transport the debris. See the responses to 3.f.4 and 3.f.12.
30	The supplemental response indicates that a correlation for determining the tumbling velocity for paint chips was developed based on NUREG/CR-6916, "Hydraulic Transport of Coating Debris," dated October 2006, data. Please describe the correlation and its application to the WCGS strainer analysis. Please further identify whether paint chips were included in the head loss tests conducted for WCGS. If paint chips were included, then please describe the size distribution of the chips used for head loss testing.	This RAI is no longer applicable because all coatings were assumed to be particulate. See the Response to 3.h.5.
31	Please provide a description of the debris transport barriers installed in the secondary shield wall exits for Compartments A and D, including the following information:	Refer to the Response to 3.e.4 for the design parameters of the flow diverters.
31.a	the total surface area of these barriers	
31.b	their height compared to the maximum containment pool water level	
31.c	their design differential pressure	
31.d	their perforation size (on page 36 of the December 22, 2008, supplemental response, the perforated openings are stated to be 1/8 inch; however, page 72 appears to imply they are 0.045 inch)	
32	Please provide the basis for considering the single-train test for Loop D to be bounding. The amount of debris settling in the head loss flume is an unknown variable. Depending on the extent of debris settling in the test flume, a more limiting condition could potentially have resulted from doubling the velocity in the test flume and dividing the debris between two strainers.	The RAI is no longer applicable. As discussed in the Response to RAI #1, the assumption that a Loop D break is the most conservative is no longer used. Additionally, new head loss tank testing did not credit near-field settling. See the response to 3.f.12.

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RAI No.	RAI	Response
33	The thin bed test described in the supplemental response dated December 22, 2008, was actually a fiber-only test. This test was used to observe the transport of fibrous debris without particulate debris clouding the water. The supplemental response stated that the fibrous debris did not clump and moved gently downstream from the introduction point where most of it settled on the flume floor. The licensee concluded that the observations verified that fibrous fines contained as part of the smalls were free to transport and were not captured by the small fibrous pieces. The observations appear to have been qualitative. If the fines added with the smalls are to be credited as fines, the please provide quantitative evidence that fine fibers credited as fines were not entangled in the larger debris pieces. Please state and justify how much of the small debris transported separately and actually behaved as fines. This issue is important in the determination of the amount of fine fibrous debris added to the head loss test. The addition of less transportable debris prior to or at the same time as more easily transportable debris is not consistent with the understanding that the staff reached with PCI/AREVA NP (NRC February 20,2008, memorandum, "Summary of Phone Calls with Performance Contracting, Inc. (PCI)/AREVAIAlden to Discuss Head Loss Test Protocol," ADAMS Accession No. ML080310263) on head loss testing procedures.	The new head loss tank testing included one full debris load test and one thin-bed test. Refer to the Response to 3.f.4.
34	Please provide the amount of fine fibrous debris predicted to be generated and transported to the strainer, including erosion and considering the reduced ZOI credited for debris generation. Please also provide information that verifies that the properly scaled amount of fine fibers was added to the test in a manner that did not inhibit their transport.	The bounding amount of fine fibrous debris predicted to be generated and transported to the strainer for the breaks up to the threshold break size are shown in the Response to 3.e.6. During the new head loss testing, no near-field settling was credited. See the Response to 3.f.12.
35	Please provide information that justifies that the agglomeration of the fine fibrous debris, observed during head loss testing, did not adversely affect the transport of the debris to the strainer.	Refer to the Response to 3.f.4 for the new head loss tests.
36	Please provide information that justifies that the addition of debris to the test flume without the recirculation pump running is realistic or conservative or prototypical with respect to the plant condition.	New head loss tank tests were performed. Refer to the Response to 3.f.4.

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RAI No.	RAI	Response
37	Please provide information that justifies that the debris addition sequence was conservative or prototypical and that it resulted in a valid thin bed test being conducted. A review of the debris addition sequence described in the supplemental response dated December 22, 2008, indicates that some less transportable debris may have been added prior to more transportable debris. Also, the design basis test appears to use a stratified addition sequence (page 60). First, part of the latent fiber is added (with the pump stopped), then some of the coating particulate, then fines (from erosion and latent fines), then coating chips, then latent particulate and Thermolag, then coating chips, and then small Nukon fibers (including 30 percent fines), followed by miscellaneous debris and other fibers. This is contrary to the guidance in "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Strainer Head Loss and Vortexing," dated March 2008 (ADAMS Accession No. ML080230038), that states that all particulate should be added prior to the addition of fibrous debris for thin bed testing.	The new head loss tank testing included one full debris load test and one thin-bed test. For the thin-bed test, particulate debris was added to the test tank before the fiber debris was batched in. Refer to the Response to 3.f.4.
38	Please provide the clean strainer head loss (CSHL), including a breakdown of the portion attributable to the correlation (core tube) and the portion attributable to the standard calculation (plenum). The CSHL should be provided for the lowest temperature case for conservatism.	Refer to the Response to 3.f.9 for the discussion of the CSHL.
39	Please provide information that justifies that the extrapolation of head loss results provided a realistic or conservative head loss prediction for the end of the mission time. Alternately, re-perform the extrapolation in a conservative manner consistent with NRC staff guidance referenced in RAI# 27 and provide an explanation that is consistent with the methodology. Please provide adequate data to demonstrate that a suitable time frame was considered during the extrapolation. Please address the following points in your response:	New head loss testing required the strainer head loss to be stabilized before a test was terminated. As a result, extrapolation of recorded head loss data was no longer needed. See the Responses to 3.f.4 and 3.f.10.
39.a	The supplemental response describes an exponential function for extrapolation of the final test head loss value to the strainer mission time. The submittal states that the debris head loss is proportional to the average debris bed thickness at a given moment in time. Apparently this assumption is part of the basis for the extrapolation of the test data. Please justify this assumption, which the staff does not believe to be correct. The head loss may have a relationship to debris bed thickness, but there are other variables that may have a larger affect on head loss; for example, debris bed morphology and compaction.	

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RAI No.	RAI	Response
39.b	The supplemental response provides a relationship for extrapolation of the head loss to the mission time. The constant C1 is stated to be the clean strainer head loss. However, in the examples provided, the constant C1 represents the maximum head loss attained during the test.	
39.c	The examples and extrapolation curves provided do not appear to correspond to the description of the relationship. The curve fit drawn on the data plot does not appear to actually fit the data or to be conservative. It appears that an exponential function was assumed and made to fit the data as well as possible. However, multiple data points taken during the first day of the test exceed the final, 30-day extrapolated head loss. This is clearly non-conservative.	
40	Please provide a plot of the head loss test data from the initiation of the test to the end of the test with significant test evolutions annotated on the plot.	Refer to the Response to 3.f.4 for the head loss vs. time figures.
41	Please verify that the core tube is fully flooded under all conditions for which recirculation is required, or re-evaluate the potential for vortex formation. If testing is credited, details of the test conditions should be provided. It should be noted that, if the core tube has air in it, a vortex may not be observed on the surface of the test tank and that some other measurement of air entrainment would have to be employed. On page 48, of the supplemental response dated December 22, 2008, stated that the SBLOCA case will result in 2.5 inches of the strainer stack top module being exposed. On page 47, the licensee stated that the SBLOCA may result in a water level less than approximately 6 inches below the top of the strainer at switchover. Please verify which statement is accurate. In addition, please verify that the accumulators will discharge to add to the sump liquid inventory under all LOCA conditions. If the accumulators do not discharge for all LOCAs, please evaluate strainer submergence, and potential for vortex formation and air entrainment, at alternate sump level conditions that do not assume accumulator discharge.	The SBLOCA sump water level analysis has been refined to use Wolf Creek-specific inputs (instead of the inputs that bounded both Callaway and Wolf Creek). With this refinement, the strainer is fully submerged for SBLOCA. Note that when analyzing the minimum sump water level for SBLOCA, injection from the accumulators was not credited. See the Response to 3.g.12. Refer to the Response to 3.f.3 for the updated discussion of vortexing based on the latest strainer testing.

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RAI No.	RAI	Response
42	Because no containment accident pressure was applied during the evaluation and the sump temperature can reach 212 of, the supplemental response dated December 22, 2008, stated that boiling and flashing could occur across the strainer debris bed. The supplemental response concluded that any voids would re-condense in the interior of the strainer modules before leaving the strainer assembly and entering the suction piping of the containment spray/emergency core cooling system pumps. Because the strainers are relatively tall vertical stacks in a sump pit this is likely true. However, the supplemental response did not discuss the potential effect of voiding on the head loss across the debris bed and how changing the head loss could lead to additional voiding. Please provide information regarding the amount of accident pressure required to prevent voiding within the debris bed and strainer and verify that the required containment pressure is available at the required times during the postulated event to prevent flashing. Please provide the minimum margin to flashing. In addition, please provide an evaluation of gas evolution downstream of the strainer that could reach the pump suction. Please provide the percentage of evolved gas estimated at the pump inlet.	Refer to the Response to 3.f.14 for the updated flashing and degasification analysis.
43	Please provide the head loss value that was used as the basis for the value extrapolated to alternate fluid temperatures including any extrapolation to the mission time and the temperature at which the head loss was measured. Please provide the temperature corrected head loss including the conditions to which it is corrected. Please provide the methodology for extrapolation and temperature scaling of the head loss. For example, please state whether the test clean strainer head loss was subtracted from the measured value prior to extrapolation and/or temperature correction. Please explain how the calculated clean strainer head loss was combined with the final debris head loss to determine the final overall head loss. If the net positive suction head (NPSH) analysis is time-or temperature-dependent, please provide details as to how the debris and strainer head loss was calculated for the evaluated conditions.	A new head loss testing program conducted with Alden Laboratories was used in a new strainer head loss analysis. The adjustment of head loss from testing conditions to plant conditions is discussed in the Response to 3.f.10.

Updated Response to NRC Generic Letter 2004-02

RAI No.	RAI	Response
44	Page 81 of the December 22,2008, supplemental response indicates that a water volume required to fill the RCS steam space is accounted for as a hold-up volume not contributing to the containment building water level. However, page 83 of the same document indicates that an approximate volume of 8900 cubic feet (ft^3) from the 12,135 ft^3 inventory of the RCS is credited in the containment water level calculation at switchover. Please clarify this apparent discrepancy, as it pertains to both the calculated large-break and small-break post-LOCA containment water levels.	There is not a discrepancy in the stated values for containment water level. The terminology of "RCS steam space" is not clear. The Response to 3.g.8 has been clarified that the total holdup by the RCS is accounted for when determining the minimum sump water level.
45	For degraded qualified coatings, the Keeler and Long report, "Design Basis Accident Testing of Coating Samples from Unit 1 Containment, TXU Comanche Peak SES," dated April 13, 2006 (ADAMS Accession No. ML070230390), and industry testing are cited by the licensee as justification of epoxy chip sizes. While the NRC review guidance has accepted use of the Keeler and Long report as justification for degraded qualified epoxy coatings failing as chips, the resulting chip sizes from the Keeler and Long report are smaller than those described in table 3h-2 of the submittal dated December 22, 2008. Please provide justification for using chips larger than those determined in the Keeler and Long report. In addition, please supply the industry testing reference used on page 87 of 128 of the December 22, 2008, supplemental letter, to determine the size distribution of degraded qualified coatings.	The new debris generation analysis assumed that all coatings fail as particulate. See the Response to 3.h.5.
46	Please describe how the quantity of curled chips was determined. In addition, please justify the simplification of the size distribution of the curled chips to a 1.5-inch chip size.	The new debris generation analysis assumed that all coatings fail as particulate. See the Response to 3.h.5.

Updated Response to NRC Generic Letter 2004-02

RAI No.	RAI	Response
47	The NRC staff does not consider in-vessel downstream effects to be fully addressed at WCGS, as well as at other pressurized-water reactors (PWRs). WCGS's submittal refers to draft WCAP-16793-NP, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous, and Chemical Debris in the Recirculating Fluid." The NRC staff has not issued a final SE for WCAP-16793-NP. The licensee may demonstrate that in-vessel downstream effects issues are resolved for WCGS by showing that the licensee's plant conditions are bounded by the final WCAP-16793-NP and the corresponding final NRC staff SE, and by addressing the conditions and limitations in the final SE. The licensee may also resolve this item by demonstrating without reference to WCAP-16793 or the NRC staff SE that in-vessel downstream effects have been addressed at WCGS. Please report how the in-vessel downstream effects issue has been addressed for WCGS within 90 days of issuance of the final NRC staff SE on WCAP-16793.	In-vessel downstream effects was reanalyzed following the latest NRC review guidance (ML19228A011) published in 2019.
48	The caption for Figure 30-1 on page 120 in the December 22, 2008, supplemental response appears to imply that this figure provides the results of settling tests for the aluminum oxyhydroxide (AIOOH) used in head loss testing for WCGS. This figure is identical to Figure 7.6-1 of WCAP-16530-NP-A, which is not plant-specific. Please provide the 1-hour settlement values for all batches of AIOOH used in head loss testing forWCGS.	A new head loss testing program conducted with Alden Laboratories was used in the resolution of GSI-191. All the chemical debris used during testing met the acceptance criteria in WCAP-16530. See the Response to 3.o.2.15.ii.
49	The licensee did not consider Min-K and Darmat KM1 as part of the debris generated based on where they are located in the containment and that they are outside the ZOI for destruction. Please state and justify whether either of these materials is subject to wetting by containment spray. If so, please state and justify whether the leached ionic material from these insulations been included in the inventory of chemicals found in the containment sump liquid following the spray phase of the LOCA for input to head loss testing.	Min-K and Darmatt KM1 are dispositioned in the Response to 3.o.2.3.i.

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RAI No.	RAI	Response
50	Table 30-1 on page 120 in the December 22,2008, supplemental response, identifies the amounts of precipitate formed from various components in containment. Although the NUKON fabric coating prevented loss of the insulation fibers, please state and justify whether the material leached from the NUKON during the spray and recirculation phase was accounted for in the concentration of ionic materials in the containment sump. Please state whether the mass of aluminum in the sodium aluminum silicate calculated for the Reactor Cavity column includes dissolved aluminum from all the Cerablanket that would be wetted in containment. If not, please discuss why the aluminum in wetted Cerablanket outside the reactor cavity does not contribute to chemical effects.	The Response to 3.o.2.3.i states that the Nukon quantity includes all Nukon debris within the ZOI, including fines, small pieces, large pieces, and intact blankets. The Response to 3.o.2.3.i also summarizes how Cerablanket is accounted for in the chemical effects analysis.

Updated Response to NRC Generic Letter 2004-02

5. References

1. **NRC Generic Letter 2004-02 (ML042360586).** Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors. September 13, 2004.
2. **Wolf Creek Letter ET 05-0018.** Docket 50-482: Wolf Creek Nuclear Operating Corporation Response to Generic Letter 2004-02: Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors. August 31,2005.
3. **NRC Letter.** Wolf Creek Generating Station, Unit 1, Request for Additional Information RE: Response to Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design-Basis Accidents at Pressurized-Water Reactors” (TAC NO. MC4731). February 9, 2006.
4. **Wolf Creek Letter ET 08-0003.** Docket No. 50-482: Wolf Creek Response to Request for Additional Information RE: Response to Generic Letter 2004-02: “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors". February 29, 2008.
5. **Wolf Creek Letter ET 08-0046.** Docket No. 50-482: Completion of Corrective Actions Associated with NRC Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors”. September 30, 2008.
6. **Wolf Creek Letter ET 08-0053.** Docket No. 50-482: Revision 1 to Wolf Creek Response to Request for Additional Information RE: Response to Generic Letter 2004-02: “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors". December 22, 2008.
7. **NRC Letter.** Wolf Creek Generating Station - Request for Additional Information Regarding Response to Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors" (Tac No. MC4731). July 31,2009.
8. **Wolf Creek Letter ET 13-0017.** Docket No. 50-482: Wolf Creek Nuclear Operating Corporation Proposed Path to Closure of Generic Safety Issue -191, “Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance”. May 16, 2013.
9. **NEI Guidance Report NEI 04-07 Volume 1.** Pressurized Water Reactor Sump Performance Evaluation Methodology 'Volume 1 - Pressurized Water Reactor Sump Performance Evaluation Methodology'. December 2004. Revision 0.
10. **NEI Guidance Report NEI 04-07 Volume 2.** Pressurized Water Reactor Sump Performance Evaluation Methodology 'Volume 2 - Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02'. December 2004. Revision 0.
11. **NUREG/CR-1829, Volume 1.** Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process. April, 2008.

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12. **PWROG Letter ML100710710.** Transmittal of Responses to NRC Request for Additional Information Re: Pressurized Water Reactor Owners Group Bases For Licensee Debris Generation Assumptions for GSI-191. March 5, 2010.
13. **NRC Letter ML100570364.** NRC Conclusions Regarding Pressurized Water Reactor Owners Group Response to Request for Additional Information Dated January 25, 2010 Regarding Licensee Debris Generation Assumptions for GSI-191. March 31, 2010.
14. **ENERCON.** BADGER Orientation and Size Increments used for Debris Generation Calculation (ADAMS Accession No. ML16217A084). Revision 1, August 3, 2016.
15. **NRC ML100960495.** NRC Revised Guidance Regarding Coatings Zone of Influence for Review of Final Licensee Responses to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors". April 6, 2010.
16. **C.D.I. Report No. 96-06.** Air Jet Testing of Fibrous and Reflective Metallic Insulation, included in Volume 3 of GE Nuclear Energy Document No. NEDO-32686-A, DRF A74-00004, "Utility Resolution Guide for ECCS Suction Strainer Blockage". Revision A, October 1998.
17. **NUREG/CR-6808.** Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance. February 2003.
18. **NUREG/CR-7172.** Knowledge Base Report on Emergency Core Cooling Sump Performance. January 2014.
19. **Duke Letter ML080730131.** Duke Energy Corporation McGuire Nuclear Station, Units 1 and 2 Docket Nos. 50-369 and 50-370 NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidnets at Pressurized-Water Reactors". February 28. 2008.
20. **NRC Letter ML083080350.** McGuire Nuclear Station, Units 1 and 2, Request for Additional Information (RAI) Regarding Supplemental Responses to Generic Letter 2004-02 (TAC Nos. MC4692 and MC4693). November 18, 2008.
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24. **NRC Letter.** Proprietary Erosion testing of Submerged Nukon Low-Density Fiberglass Insulation in Support of Generic Safety Issue 191 Strainer Performance Analysis. s.l. : June 2010.
25. **NRC Regulatory Guide 1.82.** Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident. Revision 4.
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28. **NEI.** ZOI Fibrous Debris Preparation: Processing, Storage, and Handling (ADAMS Accession No. ML120481057). Revision 1, January 24, 2012.
29. **NUREG/CR-6224.** Parametric Study of the Potential for BWR ECCS Strainer Blockage due to LOCA Generated Debris. October 1995 : s.n.
30. **WCAP-16530-NP-A.** Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191. March 2008.
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36. **ASCE Standard SEI/ASCE 8-02.** Specification for the Design of Cold-Formed Stainless Steel Structural Members.
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51. **WCAP-17788-NP, Volume 5.** Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090) – Autoclave Chemical Effects Testing for GSI-191 Long-Term Cooling. July 2015. Revision 0.
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Licensing Submittal for a Risk-Informed Resolution of Generic Letter 2004-02

Attachment IX

Defense-in-Depth and Safety Margin

Table of Contents

Defense-in-Depth and Safety Margin

List of Acronyms

BDBEE	Beyond-Design-Basis External Event
CCP	Centrifugal Charging Pump
CDF	Core Damage Frequency
CET	Core Exit Thermocouple
CS	Containment Spray
CST	Condensate Storage Tank
DID	Defense in Depth
ECCS	Emergency Core Cooling System
EMG	Emergency Management Guidelines
FLEX	Diverse and Flexible Coping Strategies
GSI-191	Generic Safety Issue 191
ISI	In-Service Inspection
LERF	Large Early Release Frequency
LOCA	Loss-of-Coolant Accident
MSLB	Main Steam Line Break
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PRA	Probabilistic Risk Assessment
PWSCC	Primary Water Stress Corrosion Cracking
RCP	Reactor Coolant Pumps
RCS	Reactor Cooling System
RG	Regulatory Guide
RHR	Residual Heat Removal
RVLIS	Reactor Vessel Level Indications
RWST	Refueling Water Storage Tank
SAMG	Severe Accident Management Guidelines
SI	Safety Injection
Wolf Creek	Wolf Creek Generating Station

Defense-in-Depth and Safety Margin

1.0 Introduction

For the purpose of this Wolf Creek Generating Station (Wolf Creek) risk-informed Generic Safety Issue 191 (GSI-191) submittal, defense-in-depth (DID) is defined as the response to the question of what happens if the analysis is wrong about a successful end state and it actually turns out to be a failure. DID measures include mitigative design features and actions that address protection of the public from radiation in the event that a loss-of-coolant accident (LOCA) results in strainer blockage or loss of long-term core cooling due to effects of LOCA-generated debris. It identifies operator actions that can be taken to mitigate the event and describes the robustness of the radiation barriers at Wolf Creek.

Similarly, safety margin is defined as elements of the analysis that increase the confidence that a declared success is a success. Therefore, safety margin is a combination of built-in conservatisms that increase confidence that scenarios that go to success remain in success and why some scenarios that are assumed to fail might actually succeed.

The conclusion of the evaluation is that there is substantial DID and safety margin.

2.0 Defense-in-Depth

The evaluation of DID first addresses whether the impact of the proposed licensing basis change (individually and cumulatively) is consistent with the DID philosophy, as outlined in Regulatory Guide (RG) 1.174 (Reference 1). This section also presents the measures available to Wolf Creek for preventing, detecting, and mitigating conditions that could challenge long-term core cooling due to strainer blockage and inadequate cooling flow to the reactor core.

2.1 Evaluation for RG 1.174 DID Philosophy

Wolf Creek is proposing a licensing basis change to use a risk-informed approach to address the concerns of GSI-191 with respect to maintaining long-term core cooling following a LOCA. An evaluation was performed to determine whether the change meets the DID principles defined in RG 1.174 (Reference 1). As stated in the RG, consistency with the DID philosophy is achieved if the following occurs:

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. As summarized in the Response to Section 2 (see Attachment VIII), Wolf Creek has performed various physical and procedural changes, for example, installation of new strainers with increased surface areas and a reduced opening size to reduce strainer head loss and debris penetration, installation of flow diverters to prevent debris-laden fluid to directly reach the sumps, implementation of the standard design change process that identifies potential impact to GSI-191 compliance by planned modifications, and comprehensive program controls to ensure the debris load limits are not

Defense-in-Depth and Safety Margin

exceeded. The new risk-informed elements of the analysis showed a very small increase in risk of containment or reactor failures related to GSI-191, as demonstrated by the very small changes in core damage frequency (CDF) and large early release frequency (LERF) per the RG 1.174 criteria (Reference 1). Therefore, the existing balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved.

- Over-reliance on programmatic activities as compensatory measures associated with the change in the licensing basis is avoided. The proposed licensing basis change does not adversely impact any of the programmatic activities, such as the in-service inspection (ISI) program, plant personnel training, reactor coolant system (RCS) leakage detection program, or containment cleanliness inspection activities. Therefore, the licensing change will not cause any over-reliance on these activities.
- System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers). The proposed licensing basis change for the use of a risk-informed methodology does not change the redundancy, independence, and diversity of the emergency core cooling system (ECCS) or containment spray (CS) system. These systems have been fully analyzed relative to their contribution to nuclear safety through plant-specific probabilistic risk assessment (PRA). The risk contribution related to GSI-191 due to the proposed licensing basis change has also been evaluated for the full spectrum of LOCA events. As described in Attachment VII, Section 2.6, the uncertainties in the risk-informed approach were examined. Based on the results of the uncertainty quantification and a consideration of the significant conservatisms, it was concluded with high confidence that the risk associated with the effects of debris at WCGS is very small or within Region III of RG 1.174 (Reference 1).
- Defenses against potential common-cause failures are preserved, and the potential for the introduction of new common-cause failure mechanisms is assessed. The potential for new common-cause failure mechanisms has been assessed for the GSI-191 issues. The primary failure mechanism includes clogging of the sump strainers and/or reactor core, which is not a new failure mechanism. The defenses against these clogging mechanisms are strengthened by the physical and procedural changes made by Wolf Creek. Additionally, the new risk-informed approach does not introduce any new common-cause failures or reduce the current plant defenses against common-cause failures.
- Independence of barriers is not degraded. The three barriers to a radioactive release are the fuel cladding, RCS pressure boundary, and reactor containment building. For the evaluation of a LOCA, the RCS barrier is postulated to be breached. The proposed licensing basis change for the use of a risk-informed approach to evaluate the effects of LOCA-generated debris does not affect the design and analysis requirements for the fuel. Therefore, the fuel barrier independence is not degraded.
- The post-LOCA recirculation function is provided by the ECCS located inside the auxiliary building. During the recirculation phase, the residual heat removal (RHR) pumps take suction from the containment recirculation sumps and supply flow back

to the reactor directly and/or through the centrifugal charging pumps (CCPs) and safety injection (SI) pumps. The pumps, system piping and other components on the recirculation flow path serve as the barrier to release. The auxiliary building has a dedicated ventilation system to control airborne radioactivity during emergency conditions and the building is capable of handling recirculating water leakage. The proposed licensing basis change does not alter the design and operating requirements for ECCS or auxiliary building. Analyses have been performed to show that, assuming a single failure that results in the loss of one air cooling train and one CS train, the containment fan coolers and the CS system can remove sufficient thermal energy from the containment atmosphere following a LOCA or main steam line break (MSLB) to maintain the peak containment pressure below design values. The licensing basis change does not alter the design or operating requirements of these systems. It is therefore reasonable to conclude that the independence of the barriers is maintained and not degraded by the licensing basis change.

- Defenses against human errors are preserved. The use of the risk-informed methodology in the GSI-191 analysis does not impose any additional operator actions or increase the complexity of existing operator actions. Thus, the defenses that are already in place with respect to human errors are not impacted by the proposed licensing basis change.
- The intent of the plant's design criteria is maintained. The proposed licensing basis change does not alter any of the ECCS acceptance criteria specified in 10 CFR 50.46. Additionally, the proposed change does not affect the design or design requirements of the plant equipment associated with GSI-191. As discussed above, the risk-informed analysis shows that the risk increase due to GSI-191 related failures is very small and meets the RG 1.174 acceptance criteria (Reference 1). Therefore, the intent of the plant's design criteria is maintained.

2.2 Detecting and Mitigating Adverse Conditions

For the purposes of GSI-191 resolution, the primary regulatory objective is specified in 10 CFR 50.46(b)(5) as maintaining long-term core cooling. Adequate DID is maintained by ensuring the capability exists for operators to detect and mitigate adverse conditions due to potential impacts of debris blockage, such as inadequate flow through the strainers and/or through the reactor core. This section evaluates the Wolf Creek DID measures for detecting and mitigating adverse conditions in order to support the Wolf Creek application for a risk-informed approach to resolve GSI-191.

Inadequate strainer flow refers to the condition where significant pump cavitation occurs due to inadequate RHR and/or CS pump net positive suction head (NPSH) margin associated with the high head losses across the sump strainers and debris bed. Additionally, accumulation of debris on the strainer and high head loss through the debris bed would increase the structural loading on the strainer assembly and challenge the structural integrity of the strainer. For Wolf Creek, testing was performed to measure the debris bed head losses using a prototypical strainer configuration and post-LOCA conditions. The effect of debris head loss was conservatively accounted for in the risk-informed analysis.

Inadequate reactor core flow refers to the condition where the normal core cooling flow path has become impeded (blocked) and is not allowing sufficient cooling water to reach the core. This condition could result from the formation of a debris bed at the reactor core inlet or at the fuel grid inside the core due to debris that passes through the sump strainers. The effect of debris accumulation in the reactor core was conservatively accounted for in the risk-informed analysis following the latest NRC guidance (Reference 2).

2.2.1 Prevention of Strainer Blockage

The primary means to delay or prevent strainer blockage is to monitor and reduce the flow through the sump strainers as necessary, and control debris sources inside containment. Specific measures are laid out as follows.

- Various Wolf Creek Emergency Management Guidelines (EMGs) provide the operators with guidance on monitoring sump strainer blockage. If sump blockage is detected, one of the EMGs provides actions that operators should take to respond to the condition.
- For small to medium break LOCAs, depletion of the refueling water storage tank (RWST) can be delayed by following an EMG, which provides actions to cool down and depressurize the RCS to reduce the break flow, thereby lowering the injection flow necessary to maintain RCS subcooling and inventory. It is possible to bring the plant to cold shutdown conditions before the RWST is drained to the sump recirculation switchover level. Therefore, sump recirculation may not be required and, in that case, sump blockage would not be an issue.
- The Technical Specification minimum required RWST volume is 394,000 gallons, which corresponds to a water level of 506.2 inches measured from the tank bottom. The RWST low-level alarm was set at a water level of 521 inches. The maximum RWST volume measured up to its overflow nozzle is 419,000 gallons, corresponding to a water level of 537 inches. It is expected that the RWST water level is normally maintained above the low-level alarm setpoint. As a result, additional inventory is available from the RWST, compared with the Technical Specification minimum volume credited in the analyses.

Several measures are in place to control the debris sources inside the Wolf Creek containment building.

- Training is provided to personnel accessing containment to raise their awareness of the more stringent containment cleanliness requirements, the potential for sump blockage, and actions being taken to address sump blockage concerns.
- To meet the Technical Specification 3.5.2 (ECCS-Operating), Wolf Creek has implemented procedures which require that, prior to entering Mode 4 (Hot Shutdown) from Mode 5, several walk-downs be performed by all personnel to

ensure the containment building is free of loose debris. For subsequent entries, inspections of the travel path and work locations are required to ensure the areas are free of loose debris.

- For the Technical Specification Surveillance Requirement 3.5.2.8, Wolf Creek implemented a procedure to verify by visual inspection that the containment sump inlets are not restricted by debris and that the suction inlet strainers show no evidence of gaps in the perforated plate, structural distress, or abnormal corrosion. The procedure also ensures that the Loop A and D debris barriers and the Loop A and D floor drain trench debris baskets are inspected, as well as secondary shield wall penetrations which have debris barriers installed.
- Wolf Creek implemented a procedure to control unattended temporary materials in containment. The program includes periodic surveillance and assessment of containment material conditions during Modes 1-4. It imposes strict controls on the types and quantities of materials that may be taken into containment.
- Wolf Creek implemented a coatings condition assessment monitoring program in accordance with RG 1.54 (Reference 2), as supplemented by the 10CFR50.65, ASTM D5163-05A (Reference 3) and EPRI 1003102 (Reference 4) guidelines. The program also requires that coatings surveillance personnel shall meet qualification requirements per ANSI N45.2.6 (Reference 5). The program ensures that the coatings debris limit will not be adversely impacted.

2.2.2 Detection of Strainer Blockage

During sump recirculation following a LOCA, accumulation of fiber, particulate, and chemical debris on the strainer could cause high flow head losses which may challenge the operation of the RHR and CS pumps. This, in turn, could result in a condition where insufficient cooling is provided for reactor core cooling and/or containment pressure control. When such a condition exists, it is important for the plant operators to be able to detect this condition in a timely manner. Wolf Creek maintains a post-accident monitoring instrumentation program, which ensures the capability to monitor plant variables and system status during and following an accident. This program includes those instruments that indicate system status and furnish information regarding the release of radioactive materials, in accordance with RG 1.97 (Reference 6). Wolf Creek has the following methods for detection of sump strainer blockage conditions.

- Wolf Creek procedure monitors flow rate, discharge pressure and motor current of the RHR pump for any signs of pump cavitation, as an indication for sump strainer blockage.
- Wolf Creek has core exit thermocouple (CET) and reactor vessel level indications (RVLIS) in the control room to allow monitoring for any potential reduction in core cooling flow due to sump blockage. This indication is also displayed on the computer systems as part of the critical safety system status tree indicators, monitored by the reactor operators and shift technical advisor. The status tree indicators provide changes based on status tree logic to enhance operator recognition of a distress condition developing.

Defense-in-Depth and Safety Margin

2.2.3 Mitigation of Strainer Blockage

Multiple methods are available to mitigate an inadequate recirculation flow condition caused by the accumulation of debris on the sump strainer.

- The Wolf Creek EMGs contain steps to reduce flow through the system up to and including stopping all pumps taking suction from a clogged sump strainer. It has been observed, during strainer head loss testing, that stopping all flow through a debris-laden strainer could dislodge portions of the debris bed from the strainer because the force that holds the debris bed in place was the flow head loss through the debris. This is also an important measure to avoid permanent pump damage that could be caused by the loss of suction condition.
- Wolf Creek implemented EMGs to minimize the number of pumps required depending on plant conditions and directs shutting down all pumps as applicable. Additionally, the EMGs contain steps to shut down SI pumps and CCPs that piggyback off a potentially cavitating RHR pump during recirculation.
- The Wolf Creek EMG contains steps to initiate makeup to the RWST from, for example, the spent fuel pool. This would allow realignment of SI and CS pumps to the direct injection flow path from the RWST and provide necessary cooling for an extended period. The operators would establish the minimum flow required for core decay heat removal depending on sub-cooling conditions.
- In response to the Nuclear Regulatory Commission (NRC) Order EA-12-049 (Reference 7), “Mitigation Strategies for Beyond-Design-Basis External Event (BDBEE)”, Wolf Creek developed FLEX to maintain fuel cooling (spent fuel pool and core) and containment integrity. Various modifications have been implemented such that non-emergency equipment can be credited during a BDBEE. For example, the Auxiliary Feedwater System can be used to deliver cooling water from the condensate storage tank (CST) to the steam generators for reactor core cooling. Makeup capabilities were added to refill the CST and Reactor Make-up Water Storage Tank, which would serve as suction sources for core cooling.

2.2.4 Prevention of Inadequate Reactor Core Flow

The set of actions identified in Section 2.2.1 for reducing or controlling flow through the emergency sump strainers during the recirculation phase can have a similar positive impact on reducing the potential for fuel blockage. Controlling flow to the reactor vessel to maintain fuel coverage and match decay heat has benefits through reduced head loss and delayed onset of any chemical precipitates.

The Wolf Creek plant design has simultaneous hot leg and cold leg injection capability once the RWST is depleted and the RHR and SI pumps have been realigned during the recirculation phase. Initially all of the ECCS pumps would be aligned for cold leg injection. At 10 hours after the initiating event, the switchover to simultaneous hot/cold leg injection would be made. For this configuration, the RHR and SI pumps provide cooling water through the hot leg while the CCP continues injecting coolant through the cold leg. It is expected that, with most of the flow traveling through the hot leg, the motive force that holds the debris at the core inlet would be removed and the flow from the hot legs would travel down the heated core to the inlet, which could dislodge the debris bed at the core inlet.

2.2.5 Detection of Inadequate Reactor Core Flow

Multiple methods exist for detection of a core blockage condition as manifested by an inadequate RCS inventory or inadequate RCS and core heat removal conditions. The primary methods for detection include CET temperature indication and reactor water level, as monitored by the RVLIS. An additional method for detection of a core blockage condition includes monitoring of containment radiation levels.

- Core exit temperature behavior is the primary indicator of adequate core cooling. If cold leg recirculation has been established with flow maintained into the RCS, core exit temperature should be stable or slowly lowering during accident recovery. Increasing core exit temperatures while injection flow is maintained, regardless of reactor vessel water level behavior, could be an indication of insufficient core flow. In this regard, Wolf Creek's functional restoration procedure would attempt to establish injection flow of clean water from the RWST. CETs are monitored during EMG usage as well as for status tree functional restoration entries and the safety parameter display system.
- Reactor vessel water level is also monitored and a decreasing water level could indicate a lower core region flow blockage. Wolf Creek employs the RVLIS to provide instrumentation for the detection of inadequate core cooling. The RVLIS utilizes two sets of differential pressure cells to measure reactor vessel level continuously. The measurement provides an approximate indication of the relative void content of the circulating fluid.
- Increasing radiation levels are indicated by alarms in the control room with specific procedural steps in both alarm response procedures and EMGs for addressing the condition. Radiation monitor indication in the auxiliary building may be indication of a LOCA outside containment or provide initial entry conditions due to increasing radiation levels. Abnormal containment radiation could be an indication of fission product barrier degradation, which is monitored by the control room. Due to the sensitivity of the monitors and the low alarm set points, identification of degrading core conditions is expected well before a significant release of radioactivity to containment occurs.

2.2.6 Mitigation of Inadequate Reactor Core Flow

Multiple methods are available for Wolf Creek to mitigate an inadequate reactor core flow condition. Upon identification of an inadequate RCS inventory or an inadequate core heat removal condition, the EMGs direct the operators to take actions to restore cooling flow to the RCS including:

- Reestablish SI flow to the RCS
- Reduce RCS pressure by performing rapid secondary depressurization
- Restart reactor coolant pumps (RCPs) and open pressurizer power operated relief valves

These actions are to be performed sequentially. Success, as indicated by improved core cooling and increasing vessel inventory, is evaluated prior to performing the next action in the sequence. Re-initiation of high pressure SI may be, depending on the cause of inadequate core cooling, the most effective method to recover the core and restore adequate core cooling. If some form of high-pressure injection cannot be established or is ineffective in restoring adequate core cooling, the operator takes actions to reduce the RCS pressure in order for the SI accumulators and low-head pumps to inject. Analyses have shown that a rapid secondary depressurization is the most effective means for achieving this objective. If secondary depressurization is not possible, or primary to secondary heat transfer is significantly degraded, and at least one idle SG is available, the operator can start the RCPs associated with the available idle steam generators. The RCPs will provide forced two-phase flow through the core and temporarily improve core cooling until some form of makeup flow to the RCS can be established.

Wolf Creek has also implemented procedures per the severe accident management guidelines (SAMG) which provide the operator with actions to protect fission product boundaries and return the plant to a controlled stable condition when the emergency operating procedures are no longer effective in controlling the casualty. Entry into the SAMG procedures is directed by the emergency operating procedures when certain conditions are met. Some of the operator actions outlined in the SAMG procedures can help maintain reactor core flow, for example, injection into steam generators and RCS, depressurization of RCS, makeup to RWST, realignment to injection from RWST, and flooding the containment.

Cooling can also be provided to the reactor core using the flow paths established by the FLEX strategy or by reinitiating injection through a refilled RWST, as discussed in Section 2.2.3. If it is determined that the inadequate core cooling condition is caused by clogged sump strainers, the actions discussed in Section 2.2.3 can also be taken to reestablish cooling flow through the strainers.

2.3 Barriers for Release of Radioactivity

The purpose of this section is to demonstrate that there are additional defense in depth measures to protect the current barriers for release of radioactivity. The three barriers are the fuel cladding, the RCS boundary, and the reactor containment building. Each of these barriers is addressed in the subsections below.

2.3.1 Fuel Cladding

Following a LOCA, the ECCS provides both the initial phase of accident mitigation and long-term cooling to the fuel cladding barrier. For the initial phase of accident mitigation, the proposed licensing basis change for the use of a risk-informed approach to evaluate the effects of debris does not alter the fuel cladding limits, or previous analysis and testing programs that demonstrate the acceptability of ECCS.

The primary goal of the Wolf Creek SAMG procedures is to protect fission product boundaries and mitigate any ongoing fission product releases in the event that conditions warrant entry into the SAMGs. Some of the operator actions outlined in the SAMG procedures can help maintain reactor core flow and integrity of the fuel cladding, for example, injection into SGs and RCS, depressurization of RCS, makeup to RWST, realignment to injection from RWST and flooding the containment.

2.3.2 RCS Pressure Boundary

The integrity of the RCS pressure boundary is assumed to be compromised for the GSI-191 sump performance evaluation. However, the proposed licensing basis change does not modify the previous analyses or testing programs that demonstrate the integrity of the RCS. Additional measures are in place to prevent and detect pipe breaks, as discussed below.

- The Wolf Creek ISI Program provides rules for the examination and testing of ASME Class 1, 2 and 3 components and component supports. The ISI Program Plan addresses those examinations and tests required by ASME Section XI and Wolf Creek augmented ISI commitments. The integrity of the Class 1 welds, piping, and components are maintained at a high level of reliability through the inspection program. The Wolf Creek ISI Program also ensure that inspections are performed in accordance with the schedule requirements of the ASME code.
- Wolf Creek developed a program plan to manage the risk of Primary Water Stress Corrosion Cracking (PWSCC) degradation in Alloy 600 components and Alloy 82/182 welds. The plan is in accordance with 10 CFR 50.55a, ASME Code Cases N-722-1 (Reference 8) and N-770-2 (Reference 9), and NEI 03-08 (Reference 10). The plan identifies all Alloy 600/82/182 locations, ranks the locations based on their risks of developing PWSCC, provides inspection requirements, and presents mitigation/replacement options. Wolf Creek has either implemented or planned mitigation measures for the welds of concern. Periodic inspections of the Alloy 600 components and Alloy 82/182 welds are covered in the ISI Program.
- RCS overpressure protection is provided by the pressurizer safety valves, steam generator safety valves, and the reactor protection system and associated equipment. Combinations of these systems ensure compliance with the overpressure requirements of the ASME Boiler and Pressure Vessel Code, Section III, Paragraphs NB-7300 and NC-7300, for pressurized water reactor systems.
- The leak detection program at Wolf Creek is capable of early identification of RCS leakage in accordance with RG 1.45 (Reference 11) to provide time for appropriate operator action to identify and address RCS leakage. The effectiveness of this program is not reduced by the proposed licensing basis change to the risk-informed approach for GSI-191.

- Some of the operator actions outlined in the Wolf Creek SAMG procedures can help maintain integrity of the RCS when directed by the emergency operating procedures. Such actions include injection into steam generators and RCS, depressurization of RCS, makeup to RWST, realignment to injection from RWST and flooding containment.

2.3.3 Reactor Containment Integrity

The Wolf Creek containment building is designed such that for all break sizes, up to and including a double-ended guillotine break of an RCS pipe or secondary system pipe, the containment peak pressure is below the design pressure with adequate margin. This has been demonstrated by previous analyses based on conservative assumptions (e.g., minimum heat removal and maximum containment pressure). The analyses also considered the worst single active failure affecting the operation of the ECCS, CS system, and containment fan coolers during the injection phase, and the worst active or passive single failure during the recirculation phase. For primary system breaks, loss of offsite power is also assumed. The analyses showed that the containment fan coolers, in conjunction with the CS system, can remove sufficient thermal energy and decay heat from the containment atmosphere following a LOCA or MSLB to maintain the containment pressure below design values. Therefore, the containment building remains a low leakage barrier against the release of fission products for the duration of the postulated LOCAs.

The evaluation of post-LOCA debris effects using a risk-informed approach is not part of the analyses that demonstrate containment integrity. The proposed licensing basis change does not affect the methodology, acceptance criteria, or conclusion of the existing analysis. Therefore, the reactor containment integrity is not affected.

Additionally, some of the operator actions outlined in the Wolf Creek SAMG procedures can help maintain integrity of the containment when directed by the emergency operating procedures. Such actions include control of containment pressure and hydrogen concentration.

2.4 Emergency Plan Actions

The proposed change to the licensing basis to use the methodology of a risk-informed approach does not involve any changes to the emergency plan. There is no change to the strategies for preventing core damage and containment failure, or for consequence mitigation. The use of the risk-informed approach does not impose any additional operator actions or complexity. Implementation of the proposed change would not result in any changes to the response requirements for emergency response personnel during an accident.

Defense-in-Depth and Safety Margin

3.0 Safety Margin

The GSI-191 testing and analyses have various built-in conservatisms, as summarized in the table below.

Table 9-1 – Description of Safety Margin

#	Topic	Conservatism Credited as Safety Margin	Realistic Conditions	Impact on Evaluation
1.	Scenario Frequency	All secondary side break scenarios that require ECCS strainer recirculation are assumed to fail due to the effects of debris	Most (if not all) secondary side breaks would be successfully mitigated due to the relatively low strainer flow rates and debris loads for these scenarios	Overall likelihood of failure is over-predicted for secondary side breaks
2.	Thermal-Hydraulics	No credit was taken for containment accident pressure in NPSH calculations and minimal credit taken for flashing evaluation	The post-LOCA containment pressure would be significantly higher than the saturation pressure	NPSH margin is under-predicted, and flashing is over-predicted
3.	Thermal-Hydraulics	Design basis sump temperature profile used for all break sizes	Sump temperature profiles would be significantly lower for smaller break sizes	Chemical release (precipitate quantities), degasification, and flashing, are over-predicted
4.	Debris Generation	100% failure of unqualified coatings for all breaks	Some types of unqualified coatings may have a relatively low failure fraction	Particulate debris quantity on strainers is over-predicted
5.	Debris Generation	Unqualified epoxy fails as 100% particulate	Epoxy coatings are likely to fail in a range of sizes (including both particulate and chips)	Unqualified coatings debris transport and particulate debris quantity on strainers are over-predicted

Defense-in-Depth and Safety Margin

#	Topic	Conservatism Credited as Safety Margin	Realistic Conditions	Impact on Evaluation
6.	Debris Generation	The ZOIs for the main loop piping breaks in the steam generator compartments were grouped by loop and truncated collectively without crediting shadowing by the pressurizer wall. Additionally, equipment shadowing was not credited	Shadowing by the pressurizer wall and equipment reduces fiber debris loads.	The fiber debris loads for some of the breaks were over-predicted.
7.	Debris Generation	Unqualified coatings outside the reactor cavity fail at the start of recirculation	Unqualified coatings would fail gradually and may not fail until much later in the event	Delay in coating failure would cause coatings to arrive at the strainers later in the event when strainer flow rate, and therefore head loss, is lower Unqualified coatings that fail in upper containment after sprays are secured would not transport to the lower containment or strainer
8.	Chemical Effects	Maximum pH for chemical release and minimum pH for solubility	Consistent time-dependent pH profile resulting in lower release and/or increased solubility	Aluminum precipitate quantity is over-predicted and precipitates would form later than predicted
9.	Chemical Effects	No aluminum remains in solution after the solubility limit has been reached or 24 hours (whichever comes first)	Some breaks would never exceed the solubility limit, and breaks that do exceed the solubility limit would still have some aluminum in solution	Aluminum precipitate quantity and strainer head loss are over-predicted

Defense-in-Depth and Safety Margin

#	Topic	Conservatism Credited as Safety Margin	Realistic Conditions	Impact on Evaluation
10.	Chemical Effects	All insulation debris is assumed to be in the sump for the chemical release calculation	In reality, a large fraction of the debris would be captured in upper containment, and the release of chemicals would be significantly reduced for breaks where containment sprays are not initiated	Aluminum release from insulation is over-predicted, resulting in an over-prediction of the aluminum precipitate quantity
11.	Debris Transport	Fine debris has a high condensate washdown fraction (10%) when sprays are not initiated	A condensate washdown of 1% is a realistic estimate and 10% an upper bound estimate per NUREG/CR-7172.	The quantity of fine debris washed down to lower containment (and subsequently transported to the strainers and core) is over-predicted for breaks that do not initiate containment sprays
12.	Debris Transport	Fine debris has a high spray washdown fraction (100%) when sprays are initiated	Some fine debris would be blown to locations shielded from containment sprays and would be retained in these locations for the duration of the event	The quantity of fine debris washed down to lower containment (and subsequently transported to the strainers and core) is over-predicted for breaks that initiate containment sprays
13.	Debris Transport	Fine debris has a high recirculation transport fraction (100%) for all breaks	Some fine debris would settle and be retained in stagnant regions of the recirculation pool (especially for cases where fewer pumps are operating)	The quantity of fine debris transported to the strainers and core is over-predicted

Defense-in-Depth and Safety Margin

#	Topic	Conservatism Credited as Safety Margin	Realistic Conditions	Impact on Evaluation
14.	Debris Transport	Small and large pieces of Nukon transport at the incipient tumbling velocity for the respective debris sizes (note that the incipient tumbling velocity is defined as the minimum fluid velocity at which an individual piece would begin to move). All small pieces of Nukon (defined as pieces less than 6 inches) were treated as 1-inch clumps. All large pieces of Nukon (defined as pieces larger than 6 inches) were treated as 6-inch pieces.	Sustained movement of a piece of debris all the way to the strainer would require a somewhat higher fluid velocity, particularly in cases where large debris quantities (including a mixture of sizes) would result in agglomeration of the debris on the containment floor	The quantity of small and large piece debris transported to the strainers is over-predicted
15.	Debris Transport	Small and large pieces of fiberglass debris have a high containment pool erosion fraction (10%)	Based on 30-day erosion test results, the erosion fraction for small pieces of fiberglass would be somewhat less than 10% and the erosion fraction for large pieces of fiberglass would be less than small pieces	The quantity of fines generated and subsequently transported to the strainers and core is over-predicted
16.	Strainer/Pump Failures	When determining the threshold break size based on strainer head loss acceptance criteria, a break is assumed to fail if the quantity of any one debris type for this break exceeds the test quantity	In many cases, one type of debris exceeds the tested quantity while other types of debris are significantly below the tested quantity	The breaks that fail the strainer acceptance criteria are over-predicted

Defense-in-Depth and Safety Margin

#	Topic	Conservatism Credited as Safety Margin	Realistic Conditions	Impact on Evaluation
17.	Strainer/Pump Failures	Miscellaneous debris (e.g., tags or labels) all transports to the strainers prior to any other debris and reduces the effective strainer area	It is likely that a large portion of the miscellaneous debris would not transport to the strainers, and any miscellaneous debris that does transport would tend to arrive along with or after other debris	The strainer surface area is under-predicted, and strainer head loss and debris limit failures are over-predicted
18.	Core Failures	The fiber penetration testing and correlation ignores effects of fiber and particulate interactions and accumulation of small and large pieces of fiberglass on the strainer	The penetration of fiberglass fines would be reduced by the accumulation of particulate and fiberglass fines and small pieces on the strainer	Fiber penetration (and subsequent accumulation within the reactor core) is over-predicted
19.	Core Failures	During penetration testing, every other module disc was removed from the test strainer modules to prevent bridging	Bridging, which is expected to occur at the plant strainer, would decrease penetration since some of the fiber debris cannot reach the perforated surfaces.	Core failures due to the accumulation of fiber debris are over-predicted
20.	Core Failures	Fiber limits associated with core blockage and boron precipitation are based on bounding tests and analyses from WCAP-17788	It is likely that significantly larger quantities of debris could accumulate inside the reactor core without full blockage	Core failures due to accumulation of fiber debris inside the reactor core are over-predicted

Defense-in-Depth and Safety Margin

#	Topic	Conservatism Credited as Safety Margin	Realistic Conditions	Impact on Evaluation
21.	Core Failures	All ECCS pumps (i.e., RHR, SIP, and CCP) are assumed to start taking suction from the sump at the time when the RHR pumps automatically switch over	This conservatively increased the time during which only the CS pumps are injecting from the RWST, and maximized the duration of time when all the penetrated fiber travels to the reactor	In-vessel fiber load is over-predicted
22.	Core Failures	All breaks were evaluated for in-vessel effects based on the hot leg break (HLB) debris limits	Debris accumulation in the core is significantly reduced for cold leg breaks (CLBs) and these breaks are less likely to fail the acceptance criteria	Core failures due to accumulation of fiber debris are over-predicted for CLBs
23.	Risk Quantification	All breaks larger than the threshold break size were assumed to result in core damage due to the effects of debris	Some of the breaks larger than the threshold break size generate less debris that would not exceed any of the GSI-191 acceptance criteria	Risk increase due to GSI-191 failures is over-predicted
24.	Risk Quantification	The threshold break size determined for the most limiting equipment configuration was assumed to be applicable to all equipment configurations. For head loss, single train failure was assumed while for in-vessel, failure of both containment spray pumps was assumed	Some equipment configurations would be significantly less likely to have debris related failures than others	Threshold break size is under-estimated, and risk increase due to GSI-191 failures is over-predicted

Defense-in-Depth and Safety Margin

4.0 References

1. **Regulatory Guide 1.174.** An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis. Revision 2.

2. **NRC (ML19228A011).** U.S. Nuclear Regulatory Commission Staff Review Guidance for In-Vessel Downstream Effects Supporting Review of Generic Letter 2004-02 Responses. September 2019.

3. **Regulatory Guide 1.54.** Guidance on Monitoring and Responding to Reactor Coolant System Leakage. May 2008. Revision 1.

4. **ASTM D5163-05A.** Standard guide for Establishing Procedures to Monitor the Performance of Coating Service Level 1 Coating Systems in an Operating Nuclear Power Plant.

5. **EPRI Document 1003102.** Guideline on Nuclear Safety Related Coatings. Revision 1.

6. **ANSI N45.2.6.** Qualification of Inspection, Examination and Testing Personnel for Nuclear Power Plants.

7. **Regulatory Guide 1.97.** Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions during and Following an Accident. December 1980. Revision 2.

8. **EA-12-049.** Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events. March 2012.

9. **ASME Section XI Code Case N-722-1.** Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated With Alloy 600/82/182 Materials.

10. **ASME Section XI Code Case N-770-2.** Alternative Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated With UNS N06082 or UNS W86182 Weld filler Material With or Without Application of Listed Mitigation Activities.

11. **NEI 03-08.** Guideline for the Management of Material Issues. Revision 2.

12. **Regulatory Guide 1.45.** Guidance on Monitoring and Responding to Reactor Coolant System Leakage. May 2008. Revision 1.

13. **WCN021-CALC-001.** *Wolf Creek Debris Transport Quantity Summary Calculation.* Revision 3 : February 25, 2019.

14. **WCN019-CALC-003.** *Wolf Creek Debris Generation Calculation.* Revision 4 : February 22, 2019.

15. **WCN019-CALC-006.** *Wolf Creek Debris Transport Calculation.* Revision 3 : February 22, 2019.

Wolf Creek Generating Station
Licensing Submittal for a Risk-Informed Resolution of Generic Letter 2004-02

Attachment X

Maintain GSI-191 Compliance (for Information Only)

Table of Contents

Maintain GSI-191 Compliance (for Information Only)

1.0 Risk-Informed GSI-191 Design Basis

With approval of the license amendment request (LAR) to use a risk-informed approach to address Generic Safety Issue (GSI)-191, the new design basis for the Wolf Creek Generating Station (WCGS) will be that the risk increase associated with post-accident debris effects is within Regulatory Guide (RG) 1.174 Region III (i.e., change in core damage frequency (ΔCDF) less than 1E-06 yr^{-1} and change in large early release frequency (ΔLERF) less than 1E-07 yr^{-1}) (Reference 1). Note that the ΔCDF guideline is more limiting for WCGS than the ΔLERF guideline because the calculated ΔLERF is more than four orders of magnitude lower than the calculated ΔCDF. Therefore, ΔLERF would not be exceeded without also exceeding ΔCDF.

Using log-linear interpolation of the 25-year geometric mean loss of coolant accident (LOCA) frequencies (Reference 2), failure of all breaks greater than or equal to 10 inches would result in a ΔCDF value of 6.6E-07 yr^{-1}. Therefore, the risk quantification would remain in RG 1.174 Region III (i.e., a ΔCDF less than 1E-06 yr^{-1}) even if all breaks larger than 10 inches fail, as long as none of the breaks smaller than this threshold fail. This is illustrated in Figure 1. Therefore, to maintain GSI-191 compliance, it is not necessary to define acceptable limits associated with breaks larger than 10 inches. However, it is necessary to define acceptable limits for breaks smaller than or equal to 10 inches to ensure that an identified issue would not cause any of these breaks to exceed the limits and potentially push the risk up into RG 1.174 Region II.

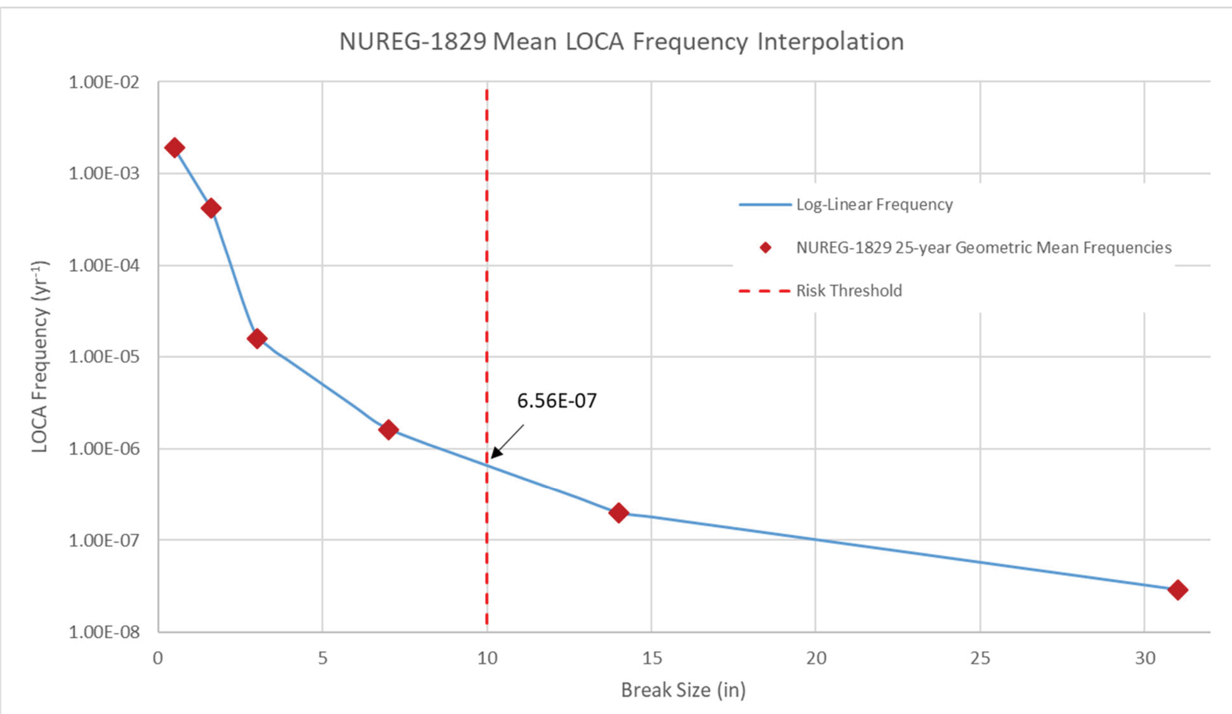

Figure 1: 25-Year Geometric Mean LOCA Frequencies with Log-Linear Interpolation

Maintain GSI-191 Compliance (for Information Only)

2.0 Available Debris Margins

Both the strainer and in-vessel debris limits must be addressed to show that breaks smaller than or equal to 10 inches do not fail. The 10-inch threshold break size was determined for strainer head loss and in-vessel effects using their most conservative equipment lineup respectively but was assumed to apply to all equipment configurations. The most conservative equipment configuration for strainer head loss is single train failure (e.g., loss of one train of residual heat removal (RHR) and containment spray (CS) pumps) because this maximizes the debris quantity that accumulates on the active strainer with the maximum flow rate through that strainer. The most conservative equipment configuration for in-vessel effects is two RHR pumps operating and failure of both CS pumps at the start of recirculation. This equipment configuration maximizes the strainer area available for penetration, maximizes drain down during the injection phase from the refueling water storage tank (RWST), and minimizes core bypass during the recirculation phase.

The acceptable debris limits based on the tested and analyzed debris quantities are shown in Table 1. Fine fiber impacts both strainer head loss and in-vessel effects. The maximum transportable fiber fine quantity in the sump pool that does not fail in-vessel effects results in a smaller margin than that for strainer head loss, and is therefore more limiting. As a result, the limit for fiber fines shown in Table 1 is based on the in-vessel debris limit and for the total quantity of transportable fine fiber in the sump pool.

All other debris categories shown in Table 1 (e.g., total fiber, particulate and miscellaneous debris) only impact strainer head loss. Therefore, the limits for these debris categories are based on the strainer head loss debris limits and debris accumulation for single train operation.

Table 1: Containment Sump Debris Limits for Breaks ≤ 10 Inches

Debris Type	Acceptable Limit
Fiber Fines	144.1 lb_m (1)
Total Fiber Fines, Small Pieces, and Large Pieces	322.5 lb_m (2)
Latent Particulate	122.2 lb_m (2)
ThermoLag Particulate	0.50 ft^3 (2)
Coatings Particulate	2.43 ft^3 (2)
Degraded Paint Chips	158.4 ft^2
Miscellaneous Debris (Tags, Labels, etc.)	20 ft^2 (3)

(1) This is the maximum allowable transported fiber fine quantity in the sump pool that resulted in acceptable in-vessel fiber loads with both RHR pumps in operation and both CS pumps failed.

(2) These are the debris quantities used during the full debris load head loss test (see Table 3.f.7-1 of Attachment 8).

(3) This is the miscellaneous debris surface area assumed for strainer head loss testing (see the Response to 3.b.5 in Attachment 8), which resulted in a reduction in strainer surface area of 15 ft^2 (see the Response to 3.f.4 in Attachment 8).

Maintain GSI-191 Compliance (for Information Only)

Available margin for a given debris type can be determined based on the difference between its debris limit and largest transportable quantity for the breaks up to 10 inches. As noted above, fine fiber impacts both strainer head loss and in-vessel effects, and the in-vessel limit results in smaller margin for fiber fines, compared with strainer head loss. Therefore, the debris limit and available margin for fiber fines are based on the in-vessel analysis for transportable fiber fines in the sump pool. For all other debris sizes and types, the transportable quantities and limits are based on the head loss analysis for debris transported to one strainer. The resulting debris margins are shown in Table 2.

Table 2: Debris Margins for Breaks ≤ 10 Inches

Debris Type	Current Quantity	Limit	Available Margin
Fiber Fines (lb_m)	119.6[(1)]	144.1	24.5
Total Fiber Fines, Small Pieces, and Large Pieces (lb_m)	235.8[(2)]	322.5	86.7
Latent Particulate (lb_m)	54.2[(3)]	122.2	68.0
ThermoLag Particulate (ft^3)[(4)]	0.51	0.50	0
Coatings Particulate (ft^3)	1.67[(5)]	2.43	0.76
Degraded Paint Chips (ft^2)	0	158.4	158.4
Miscellaneous Debris (ft^2)	7.1	20.0	12.9

(1) This is the maximum transported fiber fine quantity in the pool for breaks up to 10 inches during two-train operation with the built-in margin for transported latent fiber subtracted.

(2) This is the maximum transported total fiber quantity for breaks up to 10 inches during single-train operation (see BB-01-S105-04 listed in Table 3.e.6-10 of Attachment 8) with the built-in margin for transported latent fiber subtracted.

(3) This is the maximum transported latent particulate debris during single-train operation without margin based on generated latent debris load (75 lbm x 85%, see the Response to 3.d.3 in Attachment 8) and a transport fraction of 85% (see the Response to 3.e.6 in Attachment 8).

(4) The maximum transported quantity of ThermoLag debris for breaks up to 10 inches is slightly greater than the tested quantity due to rounding differences and is reported as no margin. If necessary, conservatisms in the quantity of ThermoLag debris generated or margin in other particulate debris types could be used to address operability issues related to ThermoLag debris quantities.

(5) This is the maximum transported coatings debris load for breaks up to 10 inches (see Table 3.f.7-1 of Attachment 8).

3.0 Application of Debris Margins for Operability Evaluation

The values in Table 2 are similar to the debris limits and margins for a deterministic design basis. These values can be used to perform a prompt operability determination following discovery of an unanalyzed debris source. As discussed in Section 1.0, risk quantification for breaks larger than 10 inches is not necessary. Even if all of these breaks were to fail, the frequency of occurrence is so low that the risk contribution and resultant ΔCDF and ΔLERF remain in RG 1.174, Region III, which is acceptable. Evaluation of smaller breaks must be performed, however.

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For example, if it was determined that insulation previously thought to be reflective metallic insulation (RMI) was actually fiberglass insulation, the quantity of fiberglass could be compared to the available margin to ensure that the total quantity does not exceed the acceptable limit for breaks smaller than or equal to 10 inches. A very simplistic assessment could be performed with a conservative assumption that the entire quantity of unanalyzed fiberglass fails as fine debris and transports to the strainers. Alternatively, a more refined assessment could be performed to determine the quantity of insulation within a bounding zone of influence (ZOI) in the vicinity of the insulation (for a break up to 10 inches) and/or determine realistic transport fractions for the newly identified debris source.

If the quantity of additional debris does not exceed the available margin, the sump can be declared operable. During the next outage, the debris source could be removed, or the design basis calculation could be updated to reflect the reduction in available margin.

However, if the quantity of additional debris exceeds the available margin, the sump would be declared inoperable, and the new containment sump TS 3.6.8 Condition A would be entered. Required Action A.3 allows 90 days to restore the sump to an operable condition. This additional time can be used to refine the debris generation and transport analysis to show that the debris quantities are within the limits, or to revise the risk quantification and submit an exigent or emergency LAR if the risk is sufficiently low to justify continued operation. Note that any Table 2 debris limit exceedance for the applicable break sizes, even if the resulting ∆CDF and ∆LERF are still within RG 1.174 Region III, requires prior Nuclear Regulatory Commission (NRC) approval to accept the condition beyond 90 days (TS 3.6.8, Condition A).

In the unlikely situation where the risk quantification shows that the ∆CDF associated with GSI-191 is unacceptably high (i.e., within RG 1.174 Region I), it would be necessary to shut down and remove the problematic source of debris or otherwise correct the identified issue. This process is illustrated in Figure 2 for a newly identified source of debris.

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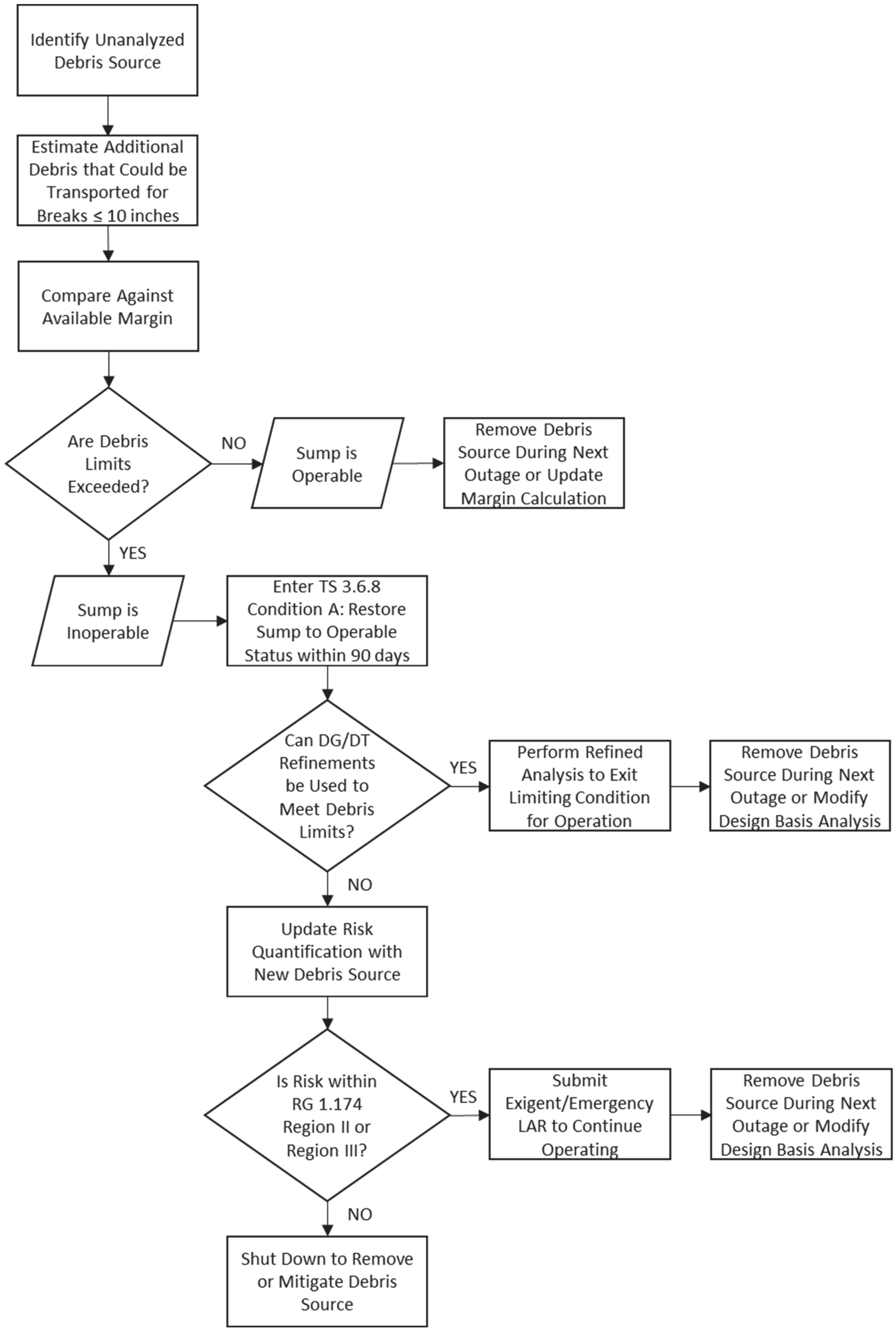

Figure 2: Illustration of operability evaluation for an unanalyzed debris source

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4.0 Application of Debris Margins for Future Plant Modifications

Future plant modifications will also be assessed for its potential impact on GSI-191 compliance using the debris margins shown in Table 2. The process is illustrated in Figure 3.

***Other GSI-191 inputs include:**

- Pump flow rates
- Pump NPSH available
- Pump NPSH required
- ECCS/CS Setpoints
- Sump water volume
- Sump pH
- Sump temperature
- Containment temperature
- Containment pressure
- Insulation type
- Buffer type
- Strainer geometry
- Strainer structural margin
- Rated core thermal power

Figure 3: Illustration of design modification process with respect to GSI-191 parameters

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5.0 References

1. Regulatory Guide 1.174, “An Approach for Using Probabilistic Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Revision 3, January 2018.
2. NUREG-1829 Volume 1, “Estimating Loss-of-Coolant Accident (LOCA) Frequencies through the Elicitation Process,” April 2008.