Vogtle Electric Generating Plant – Units 1 & 2
Supplemental Response to NRC Generic Letter 2004-02

Enclosure 1

Introduction and Overall Summary

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1.0 Introduction

In 2010, due to the ongoing challenges of resolving Generic Safety Issue (GSI)-191, the United States Nuclear Regulatory Commission (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, and these options were accepted by the NRC commissioners. In 2013, Southern Nuclear Operating Company (SNC) selected Option 2b (full risk-informed resolution path) for closure of NRC Generic Letter (GL) 2004-02 (Reference 24) at Vogtle Electric Generating Plant (VEGP) Units 1 and 2 (Reference 3).

The objective of GSI-191 is 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 for which recirculation is required (Reference 4). SNC has provided multiple responses to the NRC supporting the resolution of GSI-191. An in-depth history of the VEGP correspondences issued by or submitted to the NRC on the subject of GSI-191 is provided in Sections 1.0 and 2.0 of Enclosure 2, documenting VEGP's compliance with regulatory requirements. This submittal provides a complete summary of the risk-informed GSI-191 evaluation performed for VEGP Units 1 and 2, superseding all previous GL 2004-02 responses.

The VEGP GSI-191 submittal is organized as described below:

- Enclosure 1 provides a high-level summary of the other enclosures, and is organized with the same layout as draft Regulatory Guide (RG) 1.229 Section C (Reference 4).
- Enclosure 2 provides a detailed description of the plant-specific conditions and models related to GSI-191 (including some proprietary information). This enclosure is organized in accordance with the content guideline for GL 2004-02 responses (Reference 5). This enclosure also includes a response to each of the previous requests for additional information (RAIs) that VEGP had received on earlier GL 2004-02 submittals. Additionally, this enclosure contains attachments with affidavits for withholding the proprietary information.
- Enclosure 3 provides a description of the risk quantification using NARWHAL and the VEGP probabilistic risk assessment (PRA) model. This enclosure is organized with the same layout as draft RG 1.229 Appendix A (Reference 4). The enclosure explains how all of the individual parts are combined to quantify risk. It also provides discussion on uncertainty quantification.

- Enclosure 4 provides a summary of defense-in-depth and safety margin. This enclosure shows that the health and safety of the public are not adversely affected by debris-related failures of the strainers, pumps, downstream components, or core.
- Enclosure 5 is a duplicate of Enclosure 2 with the proprietary information redacted.

The overall evaluation for VEGP is based heavily on models that have been used in the past and accepted by the NRC for GSI-191 resolution. The results of this evaluation show with high confidence that the risk associated with GSI-191 is very low, as defined by RG 1.174 Region III (Reference 6). In addition, 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 6), 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.

The specific GSI-191 failure modes that were considered are:

1. Debris accumulation in an upstream flow path choke point (e.g., a refueling canal drain) exceeds blockage limits and reduces the available sump volume.
2. Strainer head loss exceeds the net positive suction head (NPSH) margin for the ECCS and CSS pumps when the strainer is fully submerged.
3. Strainer head loss exceeds half of the submerged strainer height when the strainer is partially submerged.
4. Strainer head loss exceeds the strainer structural margin.
5. Gas voids (i.e., water vapor due to flashing or air intrusion due to degasification or vortexing) downstream of the strainers exceed the acceptable void fraction limits of the ECCS and CSS pumps.
6. Debris penetration exceeds ex-vessel downstream effects limits for component wear or clogging.
7. Debris penetration exceeds in-vessel downstream effects limits for core blockage.
8. Buildup of oxides and other chemical precipitates on fuel cladding exceed heat transfer limits.
9. Boric acid concentration in the core exceeds the solubility limit resulting in boric acid precipitation.

Failure Modes 1, 6, and 8, as well as the vortexing portion of Failure Mode 5, have been addressed in a bounding manner for the range of possible breaks with no issues of concern (see Enclosure 2—Section 3.l for upstream blockage, Section 3.f.3 for vortexing, Section 3.m for ex-vessel downstream effects, and Section 3.n.1 for the LOCA deposition model (LOCADM) portion of the analysis of in-vessel effects) and were therefore not explicitly modeled in NARWHAL (a software tool that analyzes the GSI-191

phenomenological models in a self-consistent, time-dependent manner). The remaining failure modes were explicitly modeled. Note that core blockage (Failure Mode 7) and boric acid precipitation (Failure Mode 9) were addressed by using assumed debris limits.

Figure 1-1 shows the relationship between the various elements of the risk-informed GSI-191 analysis and documentation.

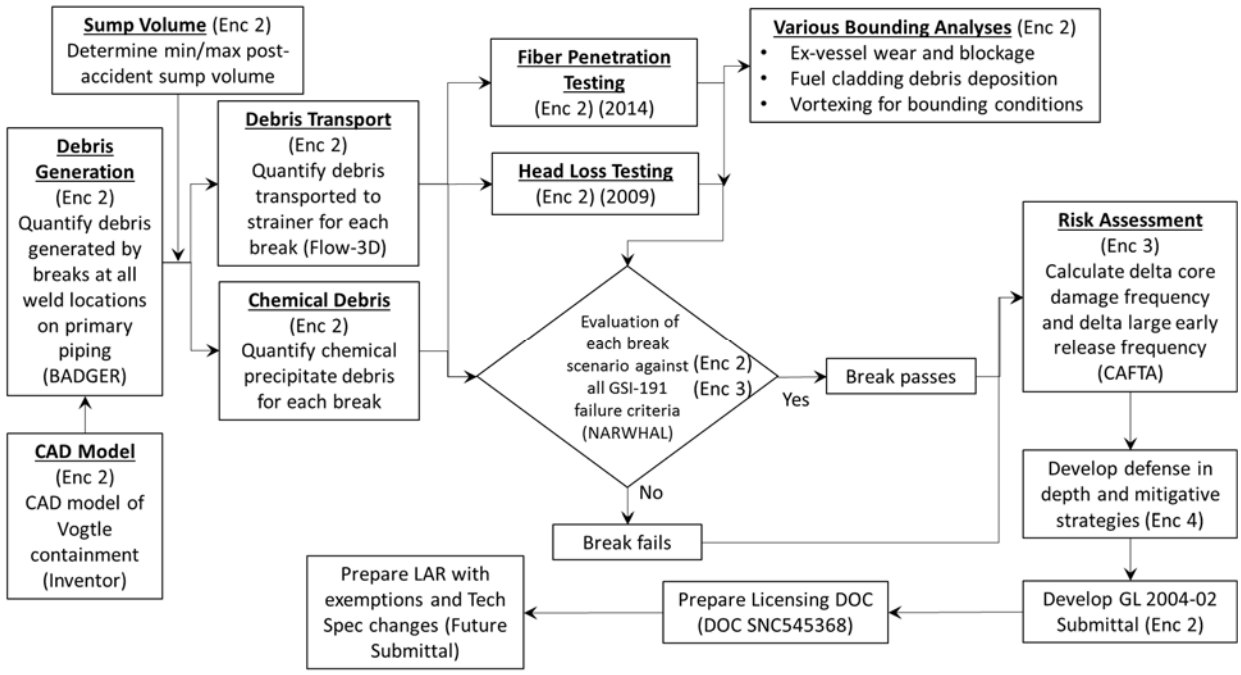

Figure 1-1 – Flow chart illustrating analysis and documentation elements

2.1 Hazards, Initiating Events, and Plant Operating Modes

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

A systematic process was used to determine the hazards, initiating events, and operating modes to be addressed in the Vogtle 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.

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

- transients,
- steam generator tube rupture,
- inadvertent safety injection,
- inadvertent or stuck-open power operated relief valves (PORVs) that discharge to the pressurizer relief tank (PRT),
- secondary side breaks outside containment, and
- interfacing system LOCAs that discharge outside containment

The internal initiating events that do 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.

Internal flood hazards 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 internal hazards and initiating events identified above that have the potential to generate debris inside containment may also require sump recirculation for mitigation of the event and result in debris transport to the containment sump. Therefore, the following events were included in the scope of the Vogtle GSI-191 analysis and considered with a detailed or conservative quantitative assessment or a qualitative evaluation:

1. Large, medium, and small LOCAs due to:
    i. Pipe breaks
    ii. Failure of non-piping components
    iii. Water hammer
2. Secondary side breaks inside containment (SSBI) that result in a consequential LOCA upon failure to terminate safety injection or a stuck open PORV, requiring sump recirculation
3. Seismically-induced LOCAs

These hazards and initiating events are discussed in more detail in Section 3.0.

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 RCS pressure and temperature (key inputs affecting the ZOI size) would either be approximately the same or significantly lower for Modes 2 through 6. In addition, 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 331.1.1.1.1(i)33), internal fire hazards were not assumed to result in pipe breaks. However, fire-induced LOCAs can occur, including spurious opening of a pressurizer PORV or safety valve, spurious reactor head vent, continuous letdown, spurious interfacing system LOCA, 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 safety valve, or spurious reactor head vent discharge to the PRT, which has negligible sources of debris in the vicinity of the rupture disk; continuous letdown and spurious interfacing system LOCAs discharge outside containment; therefore, these scenarios were all were screened from the analysis. The quantity of debris generated by an RCP seal LOCA is 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.

An evaluation of external hazards conducted for Vogtle concluded that in addition to internal flood, internal fire and seismic events, the only external hazards applicable to Vogtle are:

- aircraft impact,
- extreme winds and tornadoes,
- external flooding including intense local precipitation,
- industrial and military facility accidents,
- pipeline accidents,
- transportation accidents, and
- turbine-generated missiles

None of these external hazards listed have the potential to generate debris inside containment and were screened from the GSI-191 analysis. Therefore, seismic events are the only external hazard that affect this application of the Vogtle PRA.

2.2 Risk Attributable to Debris

The risk attributable to debris was quantified in terms of the change in core damage frequency (ΔCDF) and the change in large early release frequency (ΔLERF) compared to a hypothetical plant condition without any debris. This was done using a conservative approach that results in mean ΔCDF and ΔLERF values that are skewed high (as opposed to a best-estimate approach that would result in a more accurate prediction of the mean ΔCDF and ΔLERF values, or a bounding approach that would significantly over-predict the mean ΔCDF and ΔLERF values).

The risk quantification was performed using the NARWHAL software (to calculate the conditional failure probabilities (CFPs) associated with the effects of debris) and the VEGP internal events PRA model (with some modifications to represent the GSI-191 failure events accurately). The PRA model of record is referred to as the “base PRA model”, and the modified PRA model is referred to as the “GSI-191 PRA model”. The base PRA model has been peer reviewed against RG 1.200 (Reference 7) and is therefore appropriate to use for risk-informed applications. In order to support the detailed quantification of the GSI-191 risk impact, the base PRA model was modified to incorporate events for GSI-191 sump strainer and core blockage failures, along with the

LOCA initiating events and equipment configurations associated with each potential GSI-191 failure.

The risk evaluation relies on many engineering calculations and tests that have been developed and conducted for VEGP over the last several years to address GSI-191 and GL 2004-02. These calculations and tests are described in detail in Enclosure 2.

The GSI-191 risk quantification for VEGP shows that the overall risk associated with debris (CDF, LERF, ΔCDF, and ΔLERF) is very low as defined by Region III of RG 1.174 (Reference 6). Figure 1-2 and Figure 1-3 show the RG 1.174 risk guidelines.

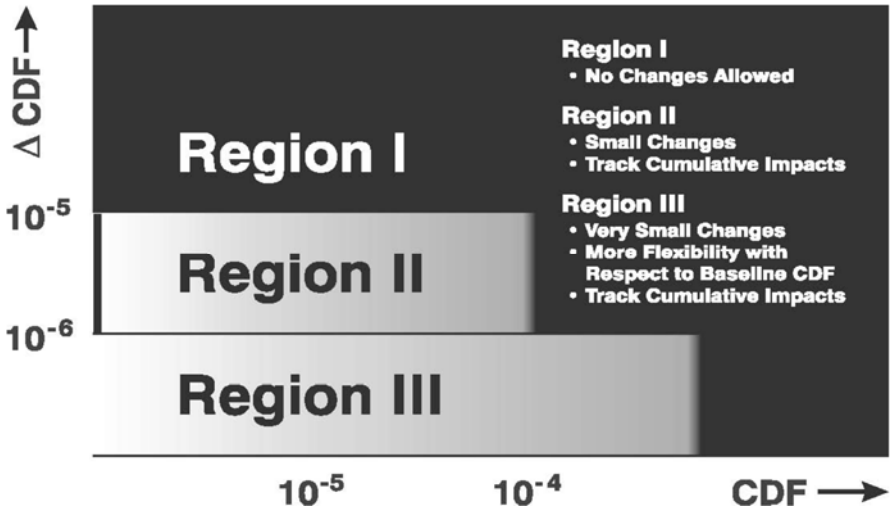

Figure 1-2 – RG 1.174 Risk Acceptance Guidelines for CDF and ΔCDF

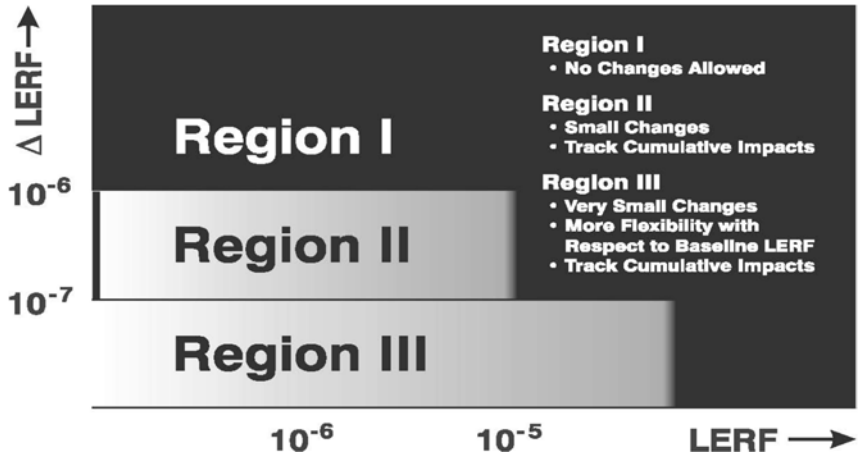

Figure 1-3 – RG 1.174 Risk Acceptance Guidelines for LERF and ΔLERF

As shown in Table 1-3 the total baseline risk (from internal events, internal fire, internal flood, and seismic events) for the VEGP Unit 1 PRA model is 4.39×10^{-5} per reactor-year

(ry^{-1}) for CDF and $1.73x10^{-6}$ ry^{-1} for LERF. The total baseline risk for the VEGP Unit 2 PRA model is $5.05x10^{-5}$ ry^{-1} for CDF and $1.90x10^{-6}$ ry^{-1} for LERF. The change in risk calculated using the VEGP GSI-191 PRA model is shown in Table 1-1. Note that the internal events and therefore the GSI-191 PRA models are identical for Units 1 and 2.

Table 1-1 – VEGP Total Risk Impact due to GSI-191 Failures

Case	ΔCDF (ry^{-1})	ΔLERF (ry^{-1})
Risk increase from GSI-191 failures for high-likelihood LOCA configurations	$2.32x10^{-8}$	$3.10x10^{-11}$
Bounding risk increase from GSI-191 failures for unlikely LOCA configurations	$1.41x10^{-9}$	$4.09x10^{-12}$
Risk increase from GSI-191 failures for seismically-induced LOCAs	$1.50x10^{-9}$	$1.50x10^{-10}$
Risk increase from GSI-191 failures for SSBIs	$1.39x10^{-9}$	$8.25x10^{-11}$
Total risk increase associated with GSI-191	**$2.75x10^{-8}$**	**$2.68x10^{-10}$**

Enclosures 2 and 3 provide a detailed description of the GSI-191 models and risk evaluation that were used to calculate these ΔCDF and ΔLERF values.

2.3 Technical Adequacy of Probabilistic Risk Assessment Results

The systematic risk assessment of debris for the resolution of GSI-191 at VEGP uses the applicable VEGP PRA models. This section provides information on the technical adequacy of the VEGP Internal Events (including internal flooding) and Seismic PRA model results in support of the resolution of GSI-191.

The guidance provided in RG 1.200 (Reference 7, Section 4.2) indicates that the following items be addressed in documentation submitted to the NRC to demonstrate the technical adequacy of the PRA:

1. Identification of permanent plant changes (such as design or operational practices) that have an impact on the PRA but have not been incorporated in the PRA.
2. The parts of the PRA used to produce the results are performed consistently with the PRA Standard as endorsed by RG 1.200.
3. A summary of the risk assessment methodology used to assess the risk of the application, including how the PRA model was modified to appropriately model the risk impact of the application.
4. Identifications of key assumptions and approximations in the PRA relevant to the results used in the decision making process.
5. A discussion of the resolution of peer review or self-assessment findings and observations that are applicable to the parts of the PRA required for the application.

6. Identification of parts of the PRA used in the analysis that were assessed to have capability categories less than that required for the application.

This section provides the information to address these items.

The VEGP PRA models are highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The quantification process used for the VEGP PRA models is based on the event tree / fault tree methodology, which is a well-known methodology in the industry.

The VEGP PRA models are controlled in accordance with the SNC procedure for PRA generation, maintenance and updates. The procedure defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operating experience, etc.), and for controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plants, the PRA maintenance procedure requires the following activities be routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model on an on-going basis.
- Reliability data, unavailability data, initiating events frequency data, human reliability data, and other such PRA inputs shall be reviewed approximately every two fuel cycles and updated as necessary to maintain the PRA consistent with the as-operated plant.

As demonstrated by the information presented in the following sections, the VEGP Units 1 and 2 Internal Events and Seismic PRA models are technically adequate for the systematic risk assessment of debris for the resolution of GSI-191.

2.3.1 Plant Changes Not Yet Incorporated into the PRA Model

As part of PRA model configuration control, SNC maintains a PRA model maintenance database that tracks all issues that have been identified that could impact the VEGP PRA model. Per the SNC procedure for PRA maintenance, the significance of the pending items in the database is evaluated to determine the impact on model results. Each pending item is prioritized for future model updates per its significance to model results. Based on a review of the VEGP PRA maintenance log, there are no significant outstanding changes that would impact the GSI-191 risk assessment.

A summary of plant changes implemented since the cutoff date of the VEGP Seismic PRA and a qualitative assessment of the likely impact of those changes is provided in Table 1-2.

Table 1-2 – Summary of Significant Plant Changes

Description of Plant Change	Impact on PRA Results
Safety-related battery chargers are no longer operated in a load-share configuration. Instead, a single charger will be in service and if it fails, the other charger will be placed in service by operator action.	An assessment of this change on the VEGP internal events PRA model indicated no significant impact. The battery chargers are modeled as seismically correlated. Thus, modeling the change would not affect the Seismic PRA results.
Permanently installed and portable FLEX equipment (other than low leakage RCP seals) have not been modeled in the PRA.	Credit for FLEX equipment is likely to improve the PRA results, but the impact is difficult to quantify without detailed modeling. The Westinghouse RCP shutdown seals have been installed at VEGP, and therefore are credited in the PRA model per the guidance in PWROG-14001-P (Reference 29) although the NRC has not yet issued a Safety Evaluation for this guidance.

2.3.2 Parts of the VEGP PRA Used

Version 5 of the VEGP Units 1 and 2 Internal Events PRA model is used for the GSI-191 risk assessment. The internal events PRA model is an at-power model (i.e., it addresses Modes 1 and 2 of reactor operation). The model includes both CDF and LERF from internal events, including internal flooding.

Version 2 of the VEGP Units 1 and 2 seismic PRA model is used for the assessment of GSI-191 risk from seismically-induced LOCAs. These versions are part of the current VEGP PRA model of record at the time of this analysis.

The latest CDF and LERF results for internal events (including internal flooding), fire, seismic, and other external hazards for VEGP Units 1 and 2 are provided in Table 1-3 - VEGP Units 1 and 2 Internal and External Events Summary

Table 1-3 - VEGP Units 1 and 2 Internal and External Events Summary

Event Type	Unit 1 CDF (per/year)	Unit 1 LERF (per/year)	Unit 2 CDF (per/year)	Unit 2 LERF (per/year)
Internal Events	2.52×10^{-6}	7.33×10^{-9}	2.52×10^{-6}	7.33×10^{-9}
Fire	3.86×10^{-5}	1.39×10^{-6}	4.52×10^{-5}	1.56×10^{-6}
Seismic	2.8×10^{-6}	3.3×10^{-7}	2.8×10^{-6}	3.3×10^{-7}
Other External	Screened out	Screened out	Screened out	Screened out
Total	**4.39×10^{-5}**	**1.73×10^{-6}**	**5.05×10^{-5}**	**1.90×10^{-6}**

It is noted that for VEGP Units 1 and 2, the Total CDF for internal and external events is less than 1.0×10^{-4}/year and the Total LERF is less than 1×10^{-5}/year, and therefore meets RG 1.174 total risk guidelines (Reference 6).

2.3.3 Summary of the Risk Assessment Methodology

The GSI-191 risk assessment methodology used for VEGP Units 1 and 2 involves quantifying the VEGP Units 1 and 2 internal events and seismic PRA models to determine the increase in risk from debris (i.e., the “risk attributable to debris”). The risk increase is defined as the difference in risk calculated considering debris effects and the risk calculated assuming debris is not present to determine both the increase in CDF (ΔCDF) and the increase in LERF (ΔLERF).

Enclosures 2 and 3 provide a detailed description of the GSI-191 models and risk evaluation that were used to calculate ΔCDF and ΔLERF.

2.3.4 Key Assumptions and Approximations in the PRA

Modeling uncertainties are considered in both the internal events PRA and the seismic PRA. Assumptions are made during the PRA development to address a modeling uncertainty because there is not a single definitive approach. The GSI-191 risk assessment methodology also incorporates various assumptions and approximations pertaining to modeling uncertainties. These assumptions and modeling uncertainties are reviewed to determine the impact on the GSI-191 risk assessment, as described in Enclosure 3, Section 14.2.3.

2.3.5 Assessment of PRA Model Technical Adequacy

<u>Internal Events PRA Model</u>

Numerous assessments of technical capability have been made for the VEGP Units 1 and 2 internal events PRA model:

- An independent PRA peer review was conducted under the auspices of the Westinghouse Owners Group (WOG) in 2001, following the industry PRA peer review

process described in NEI 00-02 (Reference 30). This peer review included an assessment of the PRA model maintenance and update process. All “Grade B” findings (there were no “Grade A” findings) were resolved in VEGP PRA model Revision 3.

- In 2005, the VEGP PRA model results were evaluated in the WOG PRA cross-comparison study performed in support of implementing the Mitigating Systems Performance Indicator (MSPI) process. After allowing for plant-specific features there were no MSPI cross-comparison outliers for the VEGP PRA.
- In 2006, MAAP (Modular Accident Analysis Program) evaluations performed for the VEGP PRA model were reviewed by an industry MAAP expert (from Fauske Associates, Inc., the company that developed the MAAP code) to check errors and reasonableness of the MAAP results. No significant issues were found from the review.
- In 2006, a gap analysis was performed for Revision 3 of the VEGP PRA model by an independent contractor against the 2005 addenda to the 2002 version of the ASME/ANS PRA Standard and the 2004 trial use version of RG 1.200. The major gaps related to documentation (especially system notebooks), the internal flooding PRA, and the treatment of uncertainty correlations were resolved in VEGP PRA model Revision 4 in 2009.
- In 2007, during the NRC review of severe accident mitigation alternative (SAMA) analysis for VEGP license renewal, the NRC issued RAIs for the VEGP PRA “L2UP” model related to dominant minimal cutsets for CDF, LERF, and other Level 2 release categories, questions about PRA quality, and Level 2 methodology. SNC provided responses to the RAIs and no additional RAIs were received from the NRC.
- In 2008 and 2009, as a part of MSPI margin improvement, the VEGP PRA Level 1 model was reviewed by an independent contractor (Westinghouse) to identify any excessive conservatism in the PRA model. The review concluded that there were no significant issues or excessive conservatism in the VEGP PRA that needed to be revised or refined.
- In 2008, a gap analysis was performed by an independent contractor (ERIN) for the VEGP internal flooding PRA. Issues from the gap analyses were resolved before finalizing the internal flooding PRA update.
- In 2008, a gap analysis was performed by an independent contractor (Scientech) for the human reliability analysis (HRA) and dependency analyses for post-initiator human failure events. No significant issues were found.
- An industry peer review was performed in May 2009. The Pressurized Water Reactor Owners Group (PWROG) peer review was based on the 2007 addenda to 2002 version of the ASME/ANS PRA standard and RG 1.200 Revision 1. The VEGP PRA was found to meet Capability Category II (CC-II) or better for most of the supporting requirements (SRs) in the PRA standard. The outstanding issues primarily pertained to documentation. A total of 46 facts and observations (F&Os) were identified, 11 of which were categorized as “Findings” (which were related to documentation). Seven of the F&Os recognized areas of strength in the PRA.
- In 2011 the VEGP PRA model was reviewed along with F&Os from the 2009 peer review to determine if model changes were necessary to be able to assess the risk

impact of the proposed surveillance frequency change per NEI 04-10, "Risk Informed Method for Control of Surveillance Frequencies". Open F&Os related to the systems of interest or that could potentially impact the results of the assessment were dispositioned as having no impact, incorporated into the model, or addressed with sensitivity analyses. PRA modeling changes were identified and incorporated into the model. Several components were added to the VEGP PRA model during this task.

- In 2011 the VEGP internal events PRA model (including flooding) was reviewed (along with the fire PRA model) to determine the technical capability for use in supporting the license amendment request to implement NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines". The review demonstrated and documented that the VEGP at-power internal events PRA model (including flooding) and the fire PRA model conform to the PRA standard at CC-II which satisfies the guidance of RG 1.200, Revision 2. In addition, the VEGP PRA model complies with all requirements for technical adequacy of the baseline PRA as defined in NEI 06-09.
- In 2013 a review and update of the MAAP parameter file was performed by an independent contractor (Fauske) to check for errors and reasonableness of MAAP results. A significant upgrade in MAAP capabilities was initiated.
- In 2014 a review was performed by an independent contractor (Reliability and Safety Consulting Engineers, Inc.) for initiating events and data update.
- In 2014 a review by an independent contractor (Scientech) was performed for the HRA and dependency analysis for post-initiator human failure events. Human error probabilities (HEPs) were reviewed, scenario timing verified, and a new dependency analysis was implemented.
- In 2014 a review was performed by an independent contractor (Nuenergy) which focused on the model of record and plant interface.

Attachment 1 presents the finding-level F&Os from the 2009 VEGP internal events PRA peer review F&Os, based on the 2007 addenda to the PRA standard. The resolution for each finding is described and the manner of that resolution is referenced. All VEGP internal events PRA model peer review findings are resolved.

Attachment 2 presents the additional/revised requirements associated with the most recent PRA standard (issued in 2009) as amended by RG 1.200, Revision 2. This table also describes the VEGP PRA model and documentation changes that assure consistency with the latest endorsed versions of the PRA standard and RG 1.200.

<u>Seismic PRA Model</u>

Version 2 of the VEGP Unit 1 and 2 seismic PRA reflects the as-built and as-operated plant as of August 31, 2015. The VEGP seismic PRA model has been assessed against RG 1.200, Revision 2. Specifically, the model was subject to a self-assessment and a peer review conducted by the PWROG in November 2014. A total of 73 F&Os were identified, 46 of which were categorized as findings and 27 as suggestions.

The peer review team determined that the VEGP seismic PRA model is of good quality and integrates the seismic hazard, the seismic fragilities, and the systems-analysis aspects appropriately to quantify CDF and LERF. The general conclusion of the peer review was that the VEGP seismic PRA is judged to be suitable for use for risk-informed applications. After the seismic PRA peer review, the peer review finding-level F&Os were appropriately dispositioned, and the seismic PRA model was updated to reflect these dispositions and further refine several fragility values. The seismic PRA peer review conclusions, the disposition of the finding-level F&Os, and the discussion below demonstrate that the VEGP seismic PRA is technically adequate for all aspects of this submittal.

Attachment 3 provides a summary of VEGP seismic PRA peer review finding-level F&Os and their disposition, as well as sensitivity analyses performed to address issues identified in the findings.

2.3.6 Capability Categories for Parts of the PRA

The 2007 version of the PRA standard used for the May 2009 peer review of the VEGP internal events PRA contains a total of 327 SRs in nine technical elements and one configuration control element. Eleven of the SRs represent deleted requirements (IE-A8, IE-A9, SC-A3, SY-A9, SY-B9, HR-G8, IF-A2, IF-B4, IF-D2, IF-E2, and QU-D2), and 20 were determined to be not applicable to the VEGP PRA. Thus, a total of 296 SRs was applicable to the VEGP internal events PRA. Among the 296 applicable SRs, 99% met Capability Category II or higher, as shown in Table 1-4.

Table 1-4 - Summary of VEGP Internal Event PRA Capability Categories

Capability Category Met	No. of SRs	% of total applicable SRs
CC-I/II/III (or SR Met)	210	70.9%
CC-II/III	24	8.1%
CC-I/II	14	4.7%
CC-III	7	2.4%
CC-II	38	12.8%
CC-I	0	0%
SR Not Met	3	1.0%
Total	**296**	**100%**

Three SRs were judged to be “Not Met”. These were HR-G6, QU-D3, and LE-G5. Supporting requirement HR-G6 was characterized as Not Met because the reasonableness check of the HRA was done for the previous revision of the PRA and not the latest revision. Supporting requirement QU-D3 was characterized as Not Met because the SR requires the PRA results to be compared with those from similar plants. The VEGP PRA report cited the MSPI benchmark report as evidence of meeting this requirement, which was an outdated comparison. Supporting requirement LE-G5 was characterized as Not Met because limitations of the LERF calculations that could impact risk-informed applications were not identified. Resolution of the Findings HR-G6-01,

QU-D3-01, and LE-G5-01 resulted in SRs HR-G6, QU-D3, and LE-G5 being met at a Capability Category I/II/III. Thus, the VEGP internal events PRA (including flood) meets the requirements of RG 1.200.

The 2013 addenda to the 2008 version of the PRA Standard used for the November 2014 peer review of the seismic PRA contains a total of 77 SRs in three technical elements. Of the 77 SRs, a total of 67 (87%) were met at Capability Category II or higher. The 10 SRs that were judged to be "Not Met" are listed in Table 1-5, along with the associated finding-level F&Os (16 total).

Table 1-5 – VEGP Seismic PRA Not Met/CC-I SRs and Associated Findings

SR	Findings
SHA-C4	12-18, 12-36
SHA-H1	12-18, 12-36
SHA-I1	12-15
SHA-I2	12-15
SHA-J1	12-1, 12-2, 12-11, 12-16
SHA-J3	12-8
SFR-A2	14-1, 14-7, 14-10
SPR-B2	16-4, 16-6
SPR-B4	16-1
SPR-F1	12-31, 16-5

As the table indicates, the following 10 SRs were assessed at less than CC-II.

- Six of the SRs are related to the seismic hazard analysis (SHA), for which the seven findings pertain to: (a) inadequate documentation of the hazard analysis; (b) demonstration that sufficient consideration has been given to more recent geologic events and associated modeling; and (c) sensitivity calculations for the models and parameters used in the site hazard. The documentation items have been addressed, as noted in Attachment 3 for the affected Findings listed in Table 1-5.
- One of the SRs is related to the seismic fragility analysis (SFR). Two of the three findings associated with this SR deal with conservatisms that have now been addressed within the analytical methodology. The remaining finding is associated with a specific polar crane fragility issue, and has also been addressed within the reviewed methodology, as noted in Attachment 3 for the affected findings listed in Table 1-5.
- Three of the SRs are related to the seismic plant response (SPR) model. Three of the five findings associated with this SR are related to implementation of the seismic performance shaping factor approach in the HRA. Those findings have been addressed and implemented in the seismic PRA model, without significant impact on the results. One finding is related to the relay chatter evaluation, and is resolved in the latest model update. The last finding is related to the SPR documentation, which has been updated to resolve the issue.

The information provided in this section demonstrates that the VEGP internal events and seismic PRA models meet the technical adequacy requirements of RG 1.200, Revision 2 and is of sufficient quality and level of detail to support the risk informed approach for GSI-191.

3.0 Initiating Event Frequencies

The initiating events relevant to GSI-191 at VEGP are LOCAs and SSBIs. The LOCA and SSBI frequencies from the base PRA model were used (with some modifications for the GSI-191 evaluation as described below). The initiating event frequencies used in the VEGP base PRA model are consistent with the requirements of the ASME/ANS PRA Standard (Reference 10) as endorsed by RG 1.200 (Reference 7), and confirmed by the VEGP base PRA model peer review.

The LOCA frequencies in the base PRA model are based on the geometric mean aggregation in NUREG-1829 (Reference 8) for medium and large LOCAs and the NRC initiating event database for small LOCAs (Reference 9). Although the small break LOCA frequency is nearly an order of magnitude lower than that produced from NUREG-1829 data, it has been demonstrated that the GSI-191 risk impact is not sensitive to the initiating event frequency for small LOCAs, because these breaks are not predicted to generate enough debris to cause strainer or core failures at VEGP (see Enclosure 3, Section 14.1). The parametric uncertainty associated with using the mean frequency was addressed by a sensitivity analysis using the 5th and 95th percentile frequency to calculate the GSI-191 CFPs and ΔCDF (see Enclosure 3, Section 14.2.2). In addition, the uncertainty associated with the use of the geometric aggregation for medium and large LOCA frequencies was assessed by performing a sensitivity analysis using the arithmetic mean aggregation (see Enclosure 3, Section 14.2.3).

The SSBI frequency in the base PRA model is based on updated industry initiating event data (Reference 11). For the GSI-191 evaluation, the SSBI frequency was separated for main steam line breaks (MSLBs) and feedwater line breaks (FWLBs) due to the significant difference in debris quantities that could be generated by these breaks. Based on a closer review of the FWLB frequencies, two industry events between 1987 and 1995 actually occurred outside containment, and therefore the FWLB frequency contribution was recalculated using a Jeffrey's non-informative distribution for the GSI-191 evaluation.

3.1 SSBI Frequency Allocation

As discussed in Enclosure 3, Section 14.1, the CFPs for SSBIs were calculated independently for MSLBs and FWLBs. This evaluation conservatively assumed that all SSBI breaks were double-ended guillotine breaks (DEGBs). Therefore, the MSLB frequency was split evenly among all of the breaks analyzed on the main steam lines to calculate the MSLB GSI-191 CFP, and the FWLB frequency was split evenly among all of the breaks analyzed on the feedwater lines to calculate the FWLB GSI-191 CFP.

3.2 LOCA Frequency Allocation

For the calculation of small, medium, and large LOCA CFPs, the LOCA frequencies were allocated to individual pipe welds using a top-down distribution methodology. The top-down LOCA frequency allocation methodology essentially treats all breaks of a similar size as having an equivalent LOCA frequency regardless of the weld size and associated degradation mechanisms. For specific break sizes within the small, medium, and large LOCA categories, the PRA model LOCA frequencies were interpolated using a semi-log interpolation scheme (i.e., linear interpolation between break sizes and logarithmic interpolation between frequencies).

The uncertainty associated with the top-down LOCA frequency allocation was assessed using a sensitivity analysis with different weld-specific LOCA frequency allocation weighting schemes. For this sensitivity, welds were classified as having a high, medium, or low rupture probability based on the weld-specific degradation mechanisms, and the frequency allocations were weighted accordingly (see Enclosure 3, Section 14.2.3).

Pipe LOCAs were postulated at weld locations. As described in Enclosure 2 Section 3.a.1, a range of break sizes and orientations were evaluated for all in-service inspection (ISI) welds in the unisolable portion of the Class 1 pressure boundary. The use of Class 1 ISI welds as 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 as shown in Figure 3.a.1-1 of Enclosures 2 and 5. 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). 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 8, p. xviii).

Non-pipe LOCAs were not explicitly evaluated. Non-pipe components whose failure would result in a LOCA include nozzles, component bodies, pressurizer heater sleeves, manways, control rod drive mechanism penetrations, safety relief valves, reactor coolant pump seals, the reactor vessel, the pressurizer vessel, the steam generator vessels, welded caps on retired lines, and other components. It was reasonably assumed that breaks at any of these non-piping 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, all of 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 reactor coolant pumps, 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 reactor vessel, the pressurizer vessel, and the steam generator 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. Non-pipe components associated with the reactor vessel such as control rod drive penetrations, manways, and instrument lines connected to the reactor vessel, etc., are located away from the hot and cold leg nozzles and are not near modeled pipe weld locations. However, any quantity of debris generated by non-pipe component welds located in the reactor cavity will be bounded by a reactor vessel nozzle break.

3.3 Seismically Induced LOCAs

In order to evaluate the risk impact from GSI-191 due to seismically-induced LOCAs, the VEGP Internal Events PRA model that was modified to perform the risk-informed GSI-191 evaluation was used as a guide to make corresponding modifications to the VEGP seismic PRA model. The GSI-191 risk impact presented in Table 1-1 therefore includes the risk increase from seismically-induced LOCAs. Enclosure 3 provides a description of the method used to calculate the ΔCDF and ΔLERF values for seismically-induced LOCAs.

3.4 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 pipe rupture in the break locations that can produce and transport problematic debris.

The portions of the VEGP 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 is 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. The pressurizer is also instrumented to monitor for indications of RCS leakage that would contribute to creating a water hammer condition and the VEGP Technical Specifications (TS) impose limits on RCS operational leakage.

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 VEGP, the only systems that flow into the primary loop piping are the safety injection (SI)

system, the residual heat removal (RHR) system, and charging from the chemical and volume control system (CVCS) (References 13, 14, and 15).

The potential for gas accumulation in the ECCS, which includes the CVCS, RHR, and SI sub-systems, is addressed under VEGP's response to GL 2008-01 (Reference 13). To address GL 2008-01, VEGP 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 CSS (References 14, 15, 25, and 26). VEGP'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 in these systems. (References 27 and 28).

Lastly, VEGP performed a search of corrective action program data for water hammer and found no issues in systems related to GSI-191 locations of concern. 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 events, the relevance of water hammer events in the context of GSI-191 is deemed insignificant. Therefore, LOCA frequencies are not impacted for VEGP Units 1 & 2 due to water hammer considerations in these systems.

4.0 Defense-in-Depth and Safety Margin

As described in RG 1.174 (Reference 6), sufficient defense-in-depth and safety margin must be maintained. Both of these aspects were evaluated in detail as described in Enclosure 4.

5.0 Uncertainty

Uncertainty quantification is a key requirement in RG 1.174 for a risk-informed evaluation (Reference 6). As defined in RG 1.174 and explained in more detail in NUREG-1855 (Reference 16) and two corresponding EPRI reports (References 17 and 18), there are three types of uncertainty that should be addressed:

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

Parametric uncertainty refers to the variability in input parameters that are used in the risk assessment. Due to the wide range of plant-specific post-LOCA conditions related to GSI-191 phenomena, this is a very important aspect for understanding the overall uncertainty.

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 16). For example, the use of a spherical zone of influence (ZOI) to model the debris quantity generated by a high energy break is a consensus model that has been widely adopted and accepted by the NRC (References 19 and 20). In general, the VEGP GSI-191 evaluation has been performed using standard models that have been widely accepted for deterministic evaluations (e.g., accepted ZOI sizes and prototypical strainer module testing). By using these consensus approaches, the effort to address model uncertainty is minimized.

Completeness uncertainty refers to a) the uncertainty associated with scenarios or phenomena that are excluded from the risk evaluation, and b) the uncertainty associated with unknown phenomena. Although it may not be practical to quantify the uncertainty associated with factors that are not explicitly modeled, their potential impact can be qualitatively assessed. Uncertainties associated with unknown phenomena, on the other hand, cannot even be qualitatively assessed. Uncertainties associated with unknown phenomena are the reason that it is important to maintain defense-in-depth and safety margins (see Enclosure 4).

Because all of the cases that were evaluated for model uncertainty and parametric uncertainty resulted in a ΔCDF less than 1×10^{-6} (see Section 5.1 and Section 5.2), it can be concluded with high confidence that the risk associated with GSI-191 is very low as defined by the acceptance guidelines in RG 1.174 (Reference 6).

5.1 Parametric Uncertainty

The parametric uncertainties associated with the VEGP risk-informed GSI-191 evaluation were evaluated in a very conservative manner by analyzing the worst case combinations of input parameters associated with strainer and core failures. Although the scenario is hypothetically possible, the probability of all of the worst-case conditions occurring simultaneously is extremely unlikely. As described in Enclosure 3, Section 14.2.3, the results of this evaluation showed that the parametric uncertainties are low (i.e., the resulting ΔCDF still falls within RG 1.174 Region III even under the worst-case combination of input parameter values).

5.2 Model Uncertainty

The model uncertainties were quantified using sensitivity analysis for models where no consensus exists. Sensitivities were run for the following models:

- Break model (continuum vs. DEGB-only)
- LOCA frequencies (VEGP PRA vs. NUREG-1829 arithmetic mean)
- LOCA frequency allocation to individual welds (top-down vs. degradation mechanism probability weighting)
- Containment spray (CS) actuation (CS actuates for hot leg breaks larger than 15 inches vs. CS actuating for more or fewer breaks)

- Aluminum metal release equation (UNM vs. WCAP-16530)
- Fiber bed thickness required to apply chemical head loss (0.45 inches vs. 0 inches)
- LBLOCA size range discretization (base case allocation of frequencies vs. allocations biased to smaller break sizes and larger break sizes)
- Chemical product generation (different aluminum release equations, solubility equations and sump/containment temperature profiles)
- NARWHAL time step size (different time step sizes for the first 24 hours)

As described in Enclosure 3, Section 14.2.3, the uncertainty associated with each of these models is low (i.e., the resulting ΔCDF still falls within RG 1.174 Region III for each sensitivity that was evaluated).

5.3 Completeness Uncertainty

Completeness uncertainty was qualitatively determined to be low. As described below, the VEGP 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 and consequential LOCAs from SSBIs were directly evaluated in the risk quantification as described in Section 2.2.
  - The SSBI evaluation included an analysis of DEGBs spaced no more than 5 ft apart along each of the main steam and feedwater lines.
  - The LOCA evaluation included pipe breaks on every 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 component failures rates (Reference 34), the probability of a normally open valve failing to close is less than 4E-04, and the probability of a large leak or spurious operation of an isolation valve is on the order of 1E-07 or less. Therefore, the conditional failure probabilities for breaks outside the first isolation valve would be orders of magnitude smaller than the conditional failure probabilities for equivalent breaks inside the first isolation valve.
  - Non-pipe LOCAs were shown to be reasonably represented or bounded by adjacent pipe breaks as described in Section 3.2.

  - High likelihood equipment configurations were explicitly evaluated.
  - Low likelihood equipment configurations were addressed using a bounding approach.
- The risk of seismic and water hammer-induced LOCAs was shown to be low as described in Sections 3.3 and 3.4.
- As described in Section 1.0 and Enclosure 2, all known GSI-191 phenomena and debris failure mechanisms were evaluated either in a bounding manner for phenomena not explicitly included in the VEGP risk model or in a reasonably conservative manner for phenomena that were included in the risk model.

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 35 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.

6.0 Monitoring Program

VEGP 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. These provide the guidance to inspect the condition of the sump strainers and the ability to assess impacts to the inputs and assumptions used in the PRA and the associated engineering analysis that support the proposed change. Programmatic requirements ensure that the potential for debris loading on the sump does not materially increase. In addition, programs and procedures have been implemented to evaluate and control potential sources of debris in containment.

7.0 Quality Assurance

Most of the analysis and testing for the risk-informed GSI-191 evaluation was performed as safety related under vendor quality assurance (QA) programs compliant with 10 CFR 50 Appendix B. In addition, the NARWHAL and BADGER software packages, which were used for VEGP GSI-191 analyses, were developed and are maintained by ENERCON as safety related items in accordance with ENERCON's 10 CFR Part 50

Appendix B QA program. The only GSI-191 analyses and tests that were not performed under an Appendix B QA program are the following:

- The aluminum release equation used for the NARWHAL CFP calculation was developed through testing at the University of New Mexico (UNM). Although the testing was not performed under an Appendix B QA program, it was conducted using standard practices for academic research at the same facility where the NRC-sponsored integrated chemical effects testing (ICET) was conducted (Reference 21). The test results (including the aluminum release equation) were peer reviewed and published in a scientific journal (Reference 22). The UNM aluminum release model was qualified for safety related use at VEGP as described in Enclosure 2, Section 3.o.2.9. Finally, a sensitivity calculation was performed to address the model uncertainty associated with the use of the UNM aluminum release equation (see Enclosure 3, Section 14.2.3).
- VEGP has a relatively high containment pressure setpoint for actuating containment sprays. The design-basis calculations show that this setpoint would be exceeded for a design-basis accident. However, best-estimate thermal hydraulic calculations performed for VEGP by Texas A&M University (TAMU) showed that hot leg breaks smaller than or equal to 15 inches and cold leg breaks up to DEGBs would not initiate containment sprays. Although the TAMU thermal-hydraulics work was not performed under an Appendix B QA program, it was prepared and peer reviewed using standard practices for academic research. Note that the results of the TAMU thermal-hydraulic analysis were only used to define which breaks would initiate containment sprays, and were not used to define the pressure and temperature profiles used in the NARWHAL CFP calculation. In addition, sensitivity calculations were performed to address the model uncertainty associated with the breaks that initiate containment sprays (see Enclosure 3, Section 14.2.3).
- The fiber penetration equations used for the NARWHAL CFP calculation were developed through testing at Alden Research Laboratory. Alden has a 10 CFR 50 Appendix B QA program. Although the testing was not officially conducted under the Alden QA program, it was performed using most of the same processes, reviews, and procedures in the QA program. In addition, sensitivity calculations were performed to evaluate the sensitivity to the fiber penetration fraction (see Enclosure 3, Section 14.2.2).
- The LOCA frequencies were allocated to individual welds using a top-down approach (see Section 3.2). To address the model uncertainty, a sensitivity analysis was performed using an alternate weighting scheme based on weld-specific degradation mechanisms. This weighting scheme was derived in part using information contained in a LOCA frequency evaluation prepared by KNF Consulting Services. Although the KNF evaluation was not performed under an Appendix B QA program, it provides a reasonable set of inputs for the purpose of the sensitivity analysis.
- The GSI-191 PRA calculations are not safety related, but were prepared as safety significant under the SNC QA program.

8.0 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 should 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 should also include any new information on LOCA frequencies that may be developed.

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 VEGP PRA model consistent with the as-operated plant. In addition, existing procedures are in place for periodic updates of risk-informed applications.

9.0 Reporting and Corrective Actions

Licensees are required to make a report to the NRC and take corrective action in the event that the risk of debris exceeds the NRC acceptance criteria or in the event that defense-in-depth or safety margins have decreased from NRC-approved analysis. The risk of debris is defined in terms of ΔCDF and ΔLERF, and the acceptance criteria are defined as the upper threshold for RG 1.174 Region II (i.e., $1x10^{-5}$ for ΔCDF and $1x10^{-6}$ for ΔLERF) (Reference 6).

Defense-in-depth measures and safety margin are specifically defined for VEGP in Enclosure 4. Any unacceptable changes in risk or reductions in defense-in-depth or safety margins that are identified through the monitoring program (see Section 6.0), the periodic updates (see Section 8.0), or other means will be reported to the NRC. In addition, these issues will be entered and tracked to resolution in accordance with the SNC corrective action program.

10.0 License Application

The specific requirements for the license application described in RG 1.174 (Reference 6) will be addressed at a later date.

11.0 References

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15. NL-08-1921, "Vogtle Electric Generating Plant Unit 2 Nine-Month Supplemental (Post-Outage) Response to Nuclear Regulatory Commission Generic Letter 2008-01," January 21, 2009
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19. NEI 04-07, Revision 0, "Pressurized Water Reactor Sump Performance Evaluation Methodology, 'Volume 1 – Pressurized Water Reactor Sump Performance Evaluation Methodology'," December 2004

20. 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
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24. NRC Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,” September 13, 2004
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Vogtle Electric Generating Plant
Supplemental Response to NRC Generic Letter 2004-02

Enclosure 1

Introduction and Overall Summary

Attachment 1

Resolution of the VEGP Internal Events PRA Peer Review Findings

F&O Number	Review Element	CC	F&O Description	F&O Resolution	Reference
IE-A4-01	IE-A4	CC II Met	The SR requires a systematic evaluation of each system to assess the possibility of an initiating event occurring due to failure of the system. The reviewers could not find documentation of such a systematic review. Additional notes made by review team in response to SNC's comments: When the reviewers asked for the Initiating Events (IE) notebook (NB), they were told that Chapter 2 of the main report is the IE NB. Chapter 2 does not contain any evidence that a systematic evaluation of each system was performed. Nor does Chapter 2 contain a system failure modes and effects analysis (FMEA) as required by the Standard which would have been an acceptable alternate. The fact that a systematic evaluation was performed during the Individual Plant Examination (IPE), in of itself, is not sufficient. The evaluation performed for the IPE should have been reviewed and a statement to that extent should have been presented in the Chapter 2. In absence of such evidence, the review comment stays. As noted elsewhere in the report, it is very important to have good documentation.	SR IE-A4 is met at Capability Category II or equivalent level per Peer Review Report. There is no technical issue associated with this F&O. A systematic systems evaluation of each system, including support systems was performed to assess initiating event possibility due to system failure. The results of this evaluation are documented in Table 2 and 3 and Appendix E and F of V-RIE-IEIF-U00-001, VEGP Electric Generating Plant, Initiating Event Notebook, June 2014. Discussed in Appendix D, table D.1 – (page D.3) – Map of ASME Initiating Events (IE) SRs to the IE notebook. This F&O is resolved.	V-RIE-IEIF-U00-001, VEGP Electric Generating Plant, Initiating Event Notebook, June 2014, Table 2 and Table 3 and Appendix E and F.
IE-D1-01	IE-D1	CC I/II/III Met	The lack of a central place for all the information related to initiating events made it difficult for the review team to review this topic. Most plants have all this information stored in a separate IE notebook. The review team recommends that VEGP do the same. Additional notes in response to SNC's comments: The review team disagrees with SNC's comments. The Standard requires that the work be documented in a manner that facilitates	SR IE-D1 is met at Capability Category II or equivalent level per Peer Review Report. There is no technical issue associated with this F&O. A review and update of the VEGP initiating events was completed in June 2014. An IE Notebook was developed as RIE calculation V-RIE-IEIF-U00-001	V-RIE-IEIF-U00-001, VEGP Electric Generating Plant, Initiating Event Notebook, June 2014.

Attachment 1 - Resolution of the VEGP Internal Events PRA Peer Review Findings

F&O Number	Review Element	CC	F&O Description	F&O Resolution	Reference
			PRA applications, upgrades and peer review. The review team does not believe that the work was documented a manner to facilitate peer review. One could almost make a case for 'not met' categorization for this element as the documentation is the weakest link in this whole effort. The F&O stays as written.	Initiating Events. The updated analysis notebook includes all the information related to the initiating events. Documentation of special initiators and system dependency analysis is included in appendix E and F respectively. Appendix D provides mapping of ASME IE SRs to locations in the IE notebook. This F&O is resolved.	
AS-A11-01	AS-A11	CC I/II/III Met	Dependencies are not preserved for consequential ATWS for the SLOCA initiator and the SGTR initiator. The existing ATWS trees, based on a LOFW initiator, were developed for transients that do not include a loss of RCS inventory or operator actions to mitigate a SGTR. Note: The review team decided to leave the F&O as is after reviewing SNC's comments.	SR AS-A11 is met at Capability Category II or equivalent level per Peer Review Report. In the revised model, if ATWT occurred after a SLOCA or SGTR, the accident sequence is treated in a similar way as a case with a stuck open PZR safety valve(s) where inventory make up by high pressure injection and recirculation as well as secondary heat removal is required to prevent core damage. This F&O is resolved.	Documented in PRA-BC-V-07-003 Rev 4.0 VEGP Internal Event PRA model, Chapter 5, Section 5.2, item 27. (Reference 4)
SY-B3-01	SY-B3	CC I/II/III Met	The treatment of main or frontline system and supporting or mitigating system Common Cause Failure (CCF) event groupings do not appear to consistent in the current PRA documents. It appears that for systems considered non-risk significant, reviews for CCF groups may not have been undertaken since the IPE modeling. The updated standards require a systematic treatment of all systems, not just the main systems contributing to core damage. New CCF groups may be required or updated documentation as to why these groups are not	SR SY-B3 is met at Capability Category II or equivalent level per Peer Review Report. CCF documentation is revised. A separate CCF Notebook (V-RIE-IEIF-U013 Common Cause Factors) was created. Potential CCF groups (for both risk significant and non-risk significant systems) are considered and documented as identified.	V-RIE-IEIF-U00-013, VEGP Electric Generating Plant, Common Cause Factor Notebook.

Attachment 1 - Resolution of the VEGP Internal Events PRA Peer Review Findings

F&O Number	Review Element	CC	F&O Description	F&O Resolution	Reference
			required is needed. Note: The review team decided to leave the F&O as is after reviewing SNC's comments.	This F&O is resolved.	
HR-G6-01	HR-G6	Not Met	Check of consistency and review for reasonableness is missing in the Revision 4 updated HRA draft and the prior revision document information related to these items is not appropriate to use in light of the updates performed and changes to the results. Section 8 includes a table of human failure events (HFEs) and human error probabilities (HEPs) but does not include HEP reasonableness check, as is documented in Section 8.3 of the November 2005 HRA update for Revision 3.	This F&O is resolved. Reasonableness check for all HRAs for Revision 4 model was re-performed. All HRAs have been determined to be reasonable or have been appropriately revised. The reasonableness check is documented in Section 8.2.2 of PRA-BC-V-07-003, Human Reliability Analysis for VEGP PRA Model Rev. 4.0 (Reference 4). This F&O is resolved.	PRA-BC-V-07-003 Rev 4.0, Chapter 8, Human Reliability Analysis for VEGP PRA Model, Section 8.2.2.
DA-C2-01	DA-C2	CC I/II/III Met	Generic data alone was used for the probability that a PORV is blocked -- refer to Table 6.3.9. Since PORV availability is a critical plant feature with respect to ATWS pressure control, the use of generic data for this parameter is deemed a weakness.	SR DA-C2 is met at Capability Category II or equivalent level per Peer Review Report. VEGP specific data was used for the probability that a PORV is blocked – refer to Table 6.4-1 in Chapter 6, VEGP Data, PRA-BC-V-07-003 Rev. 4.0 VEGP Internal Events PRA model. This F&O is resolved.	PRA-BC-V-07-003 Rev. 4.0 VEGP Internal Events PRA model, Chapter 6, VEGP Data, Table 6.4-1.

Attachment 1 - Resolution of the VEGP Internal Events PRA Peer Review Findings

F&O Number	Review Element	CC	F&O Description	F&O Resolution	Reference
IF-C2a-01	IF-C2a	CC I/II/III Met	Because of a lack of well documented analysis, a lot of information had to be obtained by talking to the analyst who performed the analysis, which is the basis for the F&O. Original F&O: From a more detailed review of the IPE flood calculations (which are the main input to defining the flood events and consequences), it is noted that successful operator mitigation of ALL flood events is assumed to occur 30 minutes into any flood scenario and fully terminate the flood flow, and it appears to be based on assumptions only, as no detailed discussion of the actual ability of operators to perform such actions is given. This appears to be in direct conflict with the HFEs included (but not modeled in the PRA model) in the flooding report (assumed perfect response vs. HFE calculation). Also, the report lists hundreds of pages of a detailed analysis approach using screening criteria, flow calculations, etc. and only by locating very specific statements. There appear to be conflicts of the inputs to the flooding PRA and the subsequent discussions of operator mitigation as well as using the information from the IPE calculations for propagation assessments. This is more than an editorial finding and impacts the entire basis of using the older calculated results in the current analysis. Additional notes made in response to SNC's comments: The lengthy flooding methodology outlined in the report is not used in the current VEGP flooding results as mentioned in the original F&O. The previous IPE flooding analysis is used as inputs to the flooding targets and propagations for a bounding case estimation,	SR IE-C2a is met at Capability Category II or equivalent level per Peer Review Report. The VEGP Internal Flooding analysis uses the internal flooding assessment conducted by ABS Consulting, and PRA-BC-V-07-003, Rev.4.0 VEGP Internal Events PRA model, Chapter 5, Linked Fault Tree, Section 5.3 Internal Flooding IE Integration. These documents were combined into V-RIE-IEIF-U00-008, Rev. 1, Internal Flooding Notebook. Calculation V-RIE-IEIF-U00-008, Rev. 1, Internal Flooding Notebook resolves the comment of "lack of well documented analysis". Automatic and operator actions that have the ability to terminate or contain floods are identified in V-RIE-IEIF-U00-008, Rev. 1, Internal Flooding Notebook, Section 10, Evaluate Flood Mitigation Strategies. Actions to mitigate the flood are generally not credited. Bounding assumptions about flood heights, propagation, and impact on equipment have been made (V-RIE-IEIF-U00-008, Rev. 1, Internal Flooding Notebook, Section 9.1, Flood Characterization). Engineering calculations for design basis flood conditions have been performed for each flood area, but these calculations are not directly referenced in the flooding analysis. The flood water flow was successfully isolated at 30	V-RIE-IEIF-U00-008, Rev. 1, Internal Flooding Notebook VEGP Electric Generating Plant Internal Flooding PRA, Rev. 2, dated 01/2009, VEGP Design Manual # DC-1009 Flooding Interdiscipline, PRA-BC-V-07-003, Rev.4.0 VEGP Internal Events PRA model, Chapter 5, Linked Fault Tree, Section 5.3 Internal Flooding IE Integration, VEGP Level 1 PRA for At-Power and Internal Floods (NRC).

F&O Number	Review Element	CC	F&O Description	F&O Resolution	Reference
			and the more thorough analysis outlined in the report is not undertaken but is in place for use (as was explained by the SNC analyst in charge of the flooding project during the peer review and very briefly mentioned in the flooding report). Each and every IPE flooding calculation reviewed during the peer review contained the assumption that the flood water flow was successfully isolated at 30 minutes, and all calculations for flood volumes, propagations, etc. were done with the amount of water generated in this 30 minutes with considerations for system characteristics. If any IPE flooding calculations are done, which do not contain this assumption, they were not seen during the peer review. No change to the F&O is warranted. The problem with this scenario of using the IPE flooding calculations for inputs to the described methodology is the following: If the flooding analysis was performed in accordance to the methodology outlined in the current report, new flooding volumes and propagation assessments would be required that did not take into account successful isolation at 30 minutes (as the IPE calculations do) since another operator isolation assessment is outlined in the flooding report methodology for normal HFE calculations for flow isolation.	minutes, and all calculations for flood volumes, propagations, etc. were done with the amount of water generated in this 30 minutes with considerations for system characteristics". While these actions are not explicitly modeled, the flooding model uses bounding assumptions with respect to operator actions (V-RIE-IEIF-U00-008, Rev. 1, Internal Flooding Notebook, Section 9.1, Flood characterization). This resolves the comment "successful operator mitigation of ALL flood events is assumed to occur 30 minutes into any flood scenario and fully terminate the flood flow". The calculations were reviewed by ABS and RIE analysts and an independent plant walkdown was performed. These activities support that model assumptions are conservative. There is no technical issue associated with this F&O. This F&O is resolved.	
QU-D3-01	QU-D3	Not Met	Reviewer asked the VEGP staff to provide evidence of comparison of the VEGP results to those from similar plants. The VEGP staff presented the benchmark report for MSPI as evidence of comparison. Reviewers concluded that report is not sufficient evidence for demonstrating compliance to this SR.	A new comparison study was performed by comparing VEGP PRA results with two PWR PRAs (Callaway and Wolf Creek), which are considered relatively similar to VEGP. In addition to the comparison of PRA reports, a plant visit to Callaway was performed to identify more details of Callaway systems and PRA modeling.	V-RIE-IEIF-U00-001, Initiating Events Notebook, Section 2, Table 4

F&O Number	Review Element	CC	F&O Description	F&O Resolution	Reference
				The plant comparisons were again updated with the 2014 initiating event update and V-RIE-IEIF-U00-001 Initiating Events Notebook was updated to reflect the changes. The comparison showed that all plants have loss of offsite power (LOSP)/station blackout (SBO) as the most dominant contributors which indicated that the VEGP PRA results are not an outlier, as compared to similar PWRs. Differences in dominant CDF contributors were investigated, and it was found that those differences are due to differences in details of system configuration/ operation and physical barriers for internal flooding and in the sources for generic initiating event frequency data (VEGP PRA used the latest generic (2010 – 2013) initiating frequency and failure data along with VEGP specific experience data for its data update). This F&O is resolved.	
QU-F5-01:	QU-F5	CC I/II/III Met	In Chapter 10, there is insufficient documentation for the quantification process, which would impact application (only EOOS). Reviews conclude that the documentation currently in Chapter 10 is not sufficient to meet this SR fully.	SR QU-F5 is met at Capability Category II or equivalent level per Peer review Report. Chapter 10 of PRA-BC-07-003 (VEGP internal PRA model rev 4.) contains detailed information for quantification process and also identifies limitations which would affect application. The identified limitation in quantification process is that the average model assumed a specific plant system	PRA-BC-V-007-003, VEGP Internal Event PRA, Rev 4.0, Chapter 10, VEGP PRA Level 1 and Level 2 Evaluation, Recovery Analysis and Uncertainty Analysis.

F&O Number	Review Element	CC	F&O Description	F&O Resolution	Reference
				alignment configuration. This assumed system alignment configuration is changed to reflect actual system alignment configuration when configuration specific risk is evaluated. There is no other limitation in quantification process. This F&O is resolved.	
LE-G5-01	LE-G5	Not Met	Limitations in the LERF analysis that would impact applications are not identified. The LERF analysis documentation is incomplete because limitations in the LERF analysis that would impact applications, as required by SR LE-G5, are not identified.	A comparison of VEGP LERF scenarios with those in Table 4.5.9.3 of the ASME PRA standard revealed that the VEGP PRA included more potential LERF scenarios than as required for a large dry containment plant in ASME PRA standard. The LERF scenarios modeled in VEGP PRA include containment bypass core damage scenarios (steam generator tube rupture and Interfacing systems LOCA), thermally or pressure induced steam generator tube rupture after core damage, containment isolation failure with core damage and various early containment failure modes. This F&O is resolved.	PRA-BC-V-07-003, VEGP Internal Event PRA, Rev 4.0, Chapter 9, VEGP Level 2 PRA Modeling.
MU-B4-01	MU-B4	CC I/II/III Met	The VEGP plant procedures do not specifically call for a peer review after a PRA upgrade has been completed. But the plant has had this peer review and other peer reviews in the past. This change is required by the SR.	Procedure RIE-014, Configuration Management of PRA Models, Qualitative Models and Software outlining requirements dealing with PRA configuration control, as referenced in ASME/ANS RA-Sa-2009, Section 1-5, have been developed to comply with requirements of RITS Initiative 4b.	See procedure RIE-014, Configuration Management of PRA Models, Qualitative Models and Software, Section 4.2 See procedure

Attachment 1 - Resolution of the VEGP Internal Events PRA Peer Review Findings

F&O Number	Review Element	CC	F&O Description	F&O Resolution	Reference
				Procedure RIE-001, Generation and Maintenance of PRA Models and Associated Updates, Section 4.2.8 states the requirement for either a full or focused peer review. The PRA model update process is discussed in Section 5 of this licensing submittal. This F&O is resolved	RIE-001, Generation and Maintenance of PRA Models and Associated Updates, Section 4.2.8. ASME/ANS RA-Sa-2009, Section 2-3 and Appendix 1-A. NEI 00-02, PRA Peer Review Process Guidance.

Vogtle Electric Generating Plant
Supplemental Response to NRC Generic Letter 2004-02

Enclosure 1

Introduction and Overall Summary

Attachment 2

Comparison of RG 1.200 Revision 1 and Revision 2 Supporting Requirements

Attachment 2 - Comparison of RG 1.200 Revision 1 and Revision 2 Supporting Requirements

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
IE-C10: CC-I/II/III: … An example of an acceptable generic data sources is NUREG/CR-5750 [Note 1].	IE-C12: CC-I/II/III: … An example of an acceptable generic data sources is NUREG/CR-6928 [Note 1].	The sentences were clarifications provided in RG 1.200 Revision 1 and Revision 2, respectively. The updated SR cites a more recent example of an acceptable generic data source.	NUREG/CR-6928 is used as the source for generic data priors in Revision 5 of the VEGP internal events PRA.
SY-B15: CC-I/II/III: … (h) harsh environments induced by containment venting, **or** failure that may occur prior to the onset of core damage.	SY-B14: CC-I/II/III: … (h) harsh environments induced by containment venting, failure of the containment venting ducts, or failure of the containment boundary that may occur prior to the onset of core damage	The sentences were clarifications provided in RG 1.200 Revision 1 and Revision 2, respectively. The updated SR explicitly requires consideration of containment venting ducts and failure of the containment boundary prior to core damage.	As noted in Table 9.2-1 of the internal events PRA calculation, failure of the containment boundary due to venting is not applicable to the VEGP large, dry, sub-atmospheric containment.
DA-C1: CC-I/II/III: … Examples of parameter estimates and associated sources include: *(a)* component failure rates and probabilities: NUREG/CR-4639 [Note (1)], NUREG/CR-4550 [Note (2)], NUREG-1715 [Note 7]	DA-C1: CC-I/II/III: … Examples of parameter estimates and associated sources include *(a)* component failure rates and probabilities: NUREG/CR-4639 [2-7], NUREG/CR-4550 [2-3], NUREG-1715 [2-21], NUREG/CR-6928 [2-20]	Reference NUREG-1715 was added by RG 1.200 Revision 1; References NUREG-1715 and NUREG/CR-6928 were included in the 2009 version of the PRA Standard. The updated SR cites more recent examples of acceptable generic data sources.	NUREG/CR-6928 is used as the source for generic data priors in Revision 5 of the VEGP internal events PRA.
QU-A2a: CC-I/II/III: PROVIDE estimates of the individual sequences in a manner consistent with the estimation of total CDF …	QU-A2: CC-I/II/III: PROVIDE estimates of the individual sequences in a manner consistent with the estimation of total CDF (and LERF) …	The LERF requirement was added by RG 1.200 Revision 2. The updated SR explicitly requires consideration of LERF.	Section 10.3.2 of the internal events PRA calculation presents estimates for individual LERF sequence cutsets.

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SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
QU-A2b: CC-I: ESTIMATE the point estimate CDF from internal events. CC-II: ESTIMATE the mean CDF from internal events, accounting for the "state-of-knowledge" correlation between event probabilities [Note (1)]. CC-III: CALCULATE the mean CDF from internal events by propagating the uncertainty distributions, ensuring that the "state-of-knowledge" correlation between event probabilities is taken into account.	QU-A3: CC-I: ESTIMATE the point estimate CDF (**and LERF)**. CC-II: ESTIMATE the mean CDF **(and LERF)**, accounting for the state-of-knowledge correlation between event probabilities [Note (1)]. CC-III: CALCULATE the mean CDF (and LERF) by propagating the uncertainty distributions, ensuring that the state-of-knowledge correlation between event probabilities is taken into account.	The phrase, "from internal events", was deleted from the 2009 version of the PRA Standard. The LERF requirement was added by RG 1.200 Revision 2. The SR explicitly requires consideration of LERF. However, per the note in 2007 SR LE-E4 and LE-F3, LERF was addressed in applicable requirements of Table 4.5.8, which includes all QU SRs. Thus, the peer review using the 2007 version of the PRA Standard addressed these LERF requirements.	The peer review based on the 2007 version of the PRA Standard addressed these LERF requirements. Section 10.3.2 of the internal events PRA calculation presents the mean CDF LERF results.
QU-B6: CC-I/II/III: ACCOUNT for system successes in addition to system failures in the evaluation of accident sequences to the extent needed for realistic estimation of CDF. This accounting may be accomplished by using numerical quantification of success probability, complementary logic, or a delete term approximation and includes the treatment of transfers among event trees where the "successes" may not be transferred between event trees.	QU-B6: CC-I/II/III: ACCOUNT for system successes in addition to system failures in the evaluation of accident sequences to the extent needed for realistic estimation of CDF or LERF. This accounting may be accomplished by using numerical quantification of success probability, complementary logic, or a delete term approximation and includes the treatment of transfers among event trees where the "successes" may not be transferred between event trees.	The LERF requirement was added by RG 1.200 Revision 2. The SR explicitly requires consideration of LERF. However, per the note in 2007 SR LE-E4 and LE-F3, LERF was addressed in applicable requirements of Table 4.5.8, which includes all QU SRs. Thus, the peer review using the 2007 version of the PRA Standard addressed these LERF requirements.	The peer review based on the 2007 version of the PRA Standard addressed these LERF requirements. The Level 2 PRA event trees presented in Section 9.2 of the internal events PRA calculation explicitly account for system successes.

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SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
QU-E3: CC-I: ESTIMATE the uncertainty interval of the CDF results. **Provide a basis for the estimate consistent with the characterization parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15).** CC-II: ESTIMATE the uncertainty interval of the CDF results. **ESTIMATE the uncertainty intervals associated with parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15), taking into account the state-of-knowledge correlation.** CC-III: **PROPAGATE parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15)**….(no change)	QU-E3: CC-I: ESTIMATE the uncertainty interval of the CDF (and LERF) results. **Provide a basis for the estimate consistent with the characterization parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15).** CC-II: ESTIMATE the uncertainty interval of the CDF (and LERF) results. **ESTIMATE the uncertainty intervals associated with parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15), taking into account the state-of-knowledge correlation.** CC-III: **PROPAGATE parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15)**….(no change)	The LERF requirement was added by RG 1.200 Revision 2. The SR explicitly requires consideration of LERF. However, per the Note in 2007 SR LE-E4 and LE-F3, LERF was addressed in applicable requirements of Table 4.5.8, which includes all QU SRs. Thus, the peer review using the 2007 version of the PRA Standard addressed these LERF requirements.	The peer review based on the 2007 version of the PRA Standard addressed these LERF requirements. Section 10.4 of the internal events PRA calculation presents the uncertainty intervals for both CDF and LERF, with consideration of the state-of-knowledge correlation.

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SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
QU-E4: CC-I: **PROVIDE an assessment of the impact of the model uncertainties and assumptions on the results of the PRA.** CC-II: **EVALUATE the sensitivity of the results to model uncertainties and key assumptions using sensitivity analyses** [Note (1)]. CC-III: **EVALUATE the sensitivity of the results to uncertain model boundary conditions and other assumptions using sensitivity analyses except where such sources of uncertainty have been adequately treated in the quantitative uncertainty analysis [Note (1)].**	QU-E4: CC-I/II/III: For each source of model uncertainty and related assumption identified in QU-E1 and QU-E2, respectively, IDENTIFY how the PRA model is affected (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, introduction of a new initiating event).	Separate requirements for CC-I, II, and III were collapsed into a single requirement for CC-I/II/III in the 2009 version of the PRA Standard. The reference to Note 1 was deleted by RG 1.200 Revision 2. The updated SR assigns the same requirement to all three CCs. Meeting CC-II: in the 2007 version of the PRA Standard assures that the new SR is met.	No action, CC-II met for 2007 version of the PRA Standard.

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SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
LE-F2: CC-I: **PROVIDE a qualitative assessment of the key sources of uncertainty. Examples:** ***(a)* Identify bounding assumptions.** ***(b)* Identify conservative treatment of phenomena.** CC-II: **PROVIDE uncertainty analysis that identifies the key sources of uncertainty and includes sensitivity studies for the significant contributors to LERF.** CC-III: **PROVIDE uncertainty analysis that identifies the key sources of uncertainty and includes sensitivity studies.**	LE-F3: CC-I/II/III: IDENTIFY and CHARACTERIZE the LERF sources of model uncertainty and related assumptions, in a manner consistent with the applicable requirements of Tables 2-2.7-2(d) and 2-2.7-2(e).	Separate requirements for CC-I, II, and III were collapsed into a single requirement for CC-I/II/III in the 2009 version of the PRA Standard. The updated SR assigns the same requirement to all three CCs. Meeting CC-II: in the 2007 version of the PRA Standard assures that the new SR is met.	No action, CC-II met for 2007 version of the PRA Standard.

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SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
IF-F2: CC-I/II/III: DOCUMENT the process used to identify … flood areas, … For example, this documentation typically includes … *(b)* flood areas used in the analysis and the reason for eliminating areas from further analysis …	IFPP-B2: CC-I/II/III: DOCUMENT the process used to identify flood areas. For example, this documentation typically includes *(a)* flood areas used in the analysis and the reason for eliminating areas from further analysis *(b)* any walkdowns performed in support of the plant partitioning	The requirement to document walkdowns performed in support of plant partitioning was added to the 2009 version of the PRA Standard. The updated SR cites examples of acceptable documentation of the process to identify flood sources. Since documentation of walkdowns was not in the 2007 version of the PRA Standard, it was not reviewed as part of the peer review conducted using that version of the PRA Standard.	Section 5 and Appendix A of the internal flooding PRA document the walkdowns performed to validate information related to flood areas, flood sources, SSCs, mitigation and other flood related features in the flood areas.
IF-B1: CC-I/II/III: For each flood area, IDENTIFY the potential sources of flooding [Note (1)]. INCLUDE: (a) equipment (e.g., piping, valves, pumps) located in the area that are connected to fluid systems (e.g., circulating water system, service water system, fire protection system, component cooling water system, feedwater system, condensate and steam systems) …	IFSO-A1: CC-I/II/III: For each flood area, IDENTIFY the potential sources of flooding [Note (1)]. INCLUDE (a) equipment (e.g., piping, valves, pumps) located in the area that are connected to fluid systems (e.g., circulating water system, service water system, fire protection system, component cooling water system, feedwater system, condensate and steam systems, and reactor coolant system) …	The requirement to include the fire protection system in Item (a) as a potential flooding source was added by RG 1.200 Revision 1. This requirement was addressed in the peer review, which used the 2007 version of the PRA Standard amended by RG 1.200 Revision 1. The requirement to include the reactor coolant system in Item (a) as a potential flooding source was added to the 2009 version of the PRA Standard. Thus, it was not reviewed as part of the peer review conducted using that version of the PRA Standard.	Potential flood sources identified in Section 5 of the internal flooding PRA reviewed as part of 2009 peer review against 2007 version of the PRA standard amended by RG 1.200, Revision 1 include RCS-connected systems - chemical and volume control system (CVCS), containment spray (CS), residual heat removal (RHR), reactor coolant system drain tank (RCSDT), safety injection (SI), and reactor water makeup system (RMWS). The Containment Building (and RCS components therein) is not included in the scope of the internal flooding analysis.

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SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
IF-F2: CC-I/II/III: DOCUMENT the process used to identify applicable flood sources. For example, this documentation typically includes *(a)* flood sources identified in the analysis, rules used to screen out these sources, and the resulting list of sources to be further examined … *(f)* screening criteria used in the analysis … *(j)* calculations or other analyses used to support or refine the flooding evaluation …	IFSO-B2: CC-I/II/III: DOCUMENT the process used to identify applicable flood sources. For example, this documentation typically includes *(a)* flood sources identified in the analysis, rules used to screen out these sources, and the resulting list of sources to be further examined *(b)*screening criteria used in the analysis *(c)* calculations or other analyses used to support or refine the flooding evaluation *(d)* any walkdowns performed in support of the identification or screening of flood sources	The requirement to document walkdowns performed in support of the identification or screening of flood sources was added to 2009 version of the PRA Standard. The updated SR cites examples of acceptable documentation of the process to identify flood sources. Since documentation of walkdowns was not in the 2007 version of the PRA Standard, it was not reviewed as part of the peer review conducted using that version of the PRA Standard.	Section 5 and Appendix A of the internal flooding PRA document the walkdowns performed to validate information related to flood areas, flood sources, SSCs, mitigation and other flood related features in the flood areas.

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SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
IF-F2: CC-I/II/III: DOCUMENT the process used to identify applicable flood scenarios. For example, this documentation typically includes … *(c)* propagation pathways … … *(d)* accident mitigating features and barriers credited … … *(e)* assumptions or calculations used in the determination of … flood-induced effects on equipment operability … *(f)* screening criteria used in the analysis *(g)* flooding scenarios considered, screened, and retained *(h)* description of how the internal event analysis models were modified … … *(j)* calculations or other analyses used to support or refine the flooding evaluation …	IFSN-B2: CC-I/II/III: DOCUMENT the process used to identify applicable flood scenarios. For example, this documentation typically includes *(a)* propagation pathways … *(b)* accident mitigating features and barriers credited … *(c)* assumptions or calculations used in the determination of … flood-induced effects on equipment operability *(d)* screening criteria used in the analysis *(e)* flooding scenarios considered, screened, and retained *(f)* description of how the internal event analysis models were modified … *(g)* calculations or other analyses used to support or refine the flooding evaluation *(h)* any walkdowns performed in support of the identification or screening of flood scenarios	The requirement to document walkdowns performed in support of the identification or screening of flood scenarios was added to 2009 version of the PRA Standard. The updated SR cites examples of acceptable documentation of the process to identify flood scenarios. Since documentation of walkdowns was not in the 2007 version of the PRA Standard, it was not reviewed as part of the peer review conducted using that version of the PRA Standard.	Section 5 and Appendix A of the internal flooding PRA document the walkdowns performed to validate information related to flood areas, flood sources, SSCs, mitigation and other flood related features in the flood areas.

Attachment 2 - Comparison of RG 1.200 Revision 1 and Revision 2 Supporting Requirements

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
IF-F2: CC-I/II/III: DOCUMENT the process used to define the applicable internal flood accident sequences and their associated quantification. For example, this documentation typically includes … *(j)* calculations or other analyses used to support or refine the flooding evaluation … *(f)* screening criteria used in the analysis … *(i)* flooding scenarios considered, screened, and retained … *(k)* results of the internal flood analysis, consistent with the quantification requirements provided in HLR-QU-D	IFQU-B2: CC-I/II/III: DOCUMENT the process used to define the applicable internal flood accident sequences and their associated quantification. For example, this documentation typically includes *(a)* calculations or other analyses used to support or refine the flooding evaluation *(b)* screening criteria used in the analysis *(c)* flooding scenarios considered, screened, and retained *(d)* results of the internal flood analysis, consistent with the quantification requirements provided in HLR-QU-D *(e)* any walkdowns performed in support of internal flood accident sequence quantification	The requirement to document walkdowns performed in support of internal flood accident sequence quantification was added in 2009 version of the PRA Standard. The updated SR cites examples of acceptable documentation of the process to identify flood related features considered in flood sequence quantification. Since documentation of walkdowns was not in the 2007 version of the PRA Standard, it was not reviewed as part of the peer review conducted using that version of the PRA Standard.	Section 5 and Appendix A of the internal flooding PRA document the walkdowns performed to validate information related to flood areas, flood sources, SSCs, mitigation and other flood related features in the flood areas that are considered in flood sequence definition.

Vogtle Electric Generating Plant
Supplemental Response to NRC Generic Letter 2004-02

Enclosure 1

Introduction and Overall Summary

Attachment 3

Resolution of the VEGP Seismic PRA Peer Review Findings

Attachment 3 - Resolution of the VEGP Seismic PRA Peer Review Findings

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
11-3	SHA-E2	II/III	While variability in the mean base-case Vs profile is incorporated in the site response analysis, no epistemic uncertainty in the base-case profile is represented. Documentation of the justification for this assessment should be expanded. (This F&O originated from SR SHA-E2)	To maintain hazard-consistent ground motion hazard at the control point, the site response analysis needs to incorporate appropriate epistemic uncertainty and aleatory variability in its inputs. The Vs profile for the Vogtle Units 1&2 site is represented by a single Vs profile, indicating there is no epistemic uncertainty in the mean base-case profile. Documentation of this assessment needs to be expanded. Discussion with staff indicates that consideration of the combined data for the Vogtle site (Units 1&2, Units 3&4, ISFSI) provides sufficient confidence that a single mean base-case profile characterizes the site. This conclusion is based on the quantity and quality of the combined data and an evaluation showing the site is relatively uniform with respect to Vs. For some depth ranges, data from the nearby Savannah River Site (SRS) are used to support the profile interpretation. The documentation presents summaries of velocity data, but does not provide sufficient	Expand documentation to demonstrate that a single base-case Vs profile adequately represents the Units 1&2 site. Or if that is not the case, include epistemic uncertainty in the characterization of Vs profile and evaluate the impact on control point ground motions.	There is an abundance of site-specific Vs data from VEGP Units 3&4, which reduces epistemic uncertainty to an insignificant level. Additional discussion of the rationale for use of a single base-case Vs profile for the site has been included in the documentation. The added discussion demonstrates that a single base-case shear-wave velocity (Vs) profile adequately represents the Vogtle site, based on the availability of Vs data, which reduces the epistemic uncertainty for this particular parameter. This finding has been resolved with no significant impact to the SPRA results or conclusions.

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
				information to support the lack of epistemic uncertainty at the Units 1&2 site over the complete depth range of the Vs profile. This would typically require multiple measurements throughout the depth range that provide a consistent picture of natural variability about a single mean base-case profile. The technical basis and justification that a single base-case profile is appropriate should be provided in more detail. This should include the basis for applying conclusions from other Vogtle locations to the Units 1&2 site. [A related Suggestion 11-2 addresses specifically potential epistemic uncertainty in the Blue Bluff Marl stratum.]		
11-8	SHA-E2	II/III	Upper crustal site attenuation of ground motion (kappa) is, generally, an uncertain parameter. Thus, to maintain hazard-consistent ground motion at the control point, this uncertainty should be incorporated in the site response analysis, or the basis for not including it	The Vogtle Site Response Analysis notes that the damping associated with the base-case profile corresponds to a total kappa value for the soil column of 0.01 sec. The report does not address epistemic uncertainty in kappa. In discussion with staff during the peer review, it was noted that randomization of the damping associated with the profile layers represents both random variability and epistemic uncertainty. It	Provide a basis in the documentation for representing base-case kappa at the site by a single value. The basis might include sensitivity analyses to show the impact of epistemic uncertainty in kappa.	A discussion of the range of possible values of deep soil damping has been included in the documentation. A sensitivity study on the epistemic uncertainty of deep soil damping has been performed using median, lower range, and upper range alternatives for deep rock damping. Site response analysis was performed using 1E-4 HF and LF rock input motion.

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
			should be provided. In either case, the technical basis and justification should be documented. (This F&O originated from SR SHA-E2)	was also noted that kappa was expected to be small for the Vogtle site and uncertainties in that small value would not be expected to have a significant impact on site amplification. Staff also noted that the approach used had been reviewed by the NRC for the Vogtle ESP and COLA. The SPID provides guidance accepted by the NRC for response to NTTF 2.1 Recommendation: Seismic that indicates kappa is difficult to measure and thus subject to large uncertainty (SPID Section B-5.1.3.2). Documentation of the technical basis for kappa characterization should be expanded.		The resulting amplification functions and log-standard deviation were weight-averaged and compared to the original base case for each of BBM High PI and BBM Low PI soil columns. It was concluded that the inclusion of alternative base cases for deep soil damping to account explicitly for the epistemic uncertainty associated with site kappa does not have any significant effects on the resulting seismic hazard curves and UHRS. The sensitivity study has been added to the SPRA documentation. This finding has been resolved with no significant impact to the SPRA results or conclusions.
12-1	SHA-J1	Not Met	As part of the PSHA implementation, the analyst has different alternatives for modeling the earthquake occurrences in the calculations. The PSHA documentation does not describe the approach that was	The approach that was taken to model earthquakes in the PSHA calculation was not identified. There are two basic alternatives that can be used to model earthquake events; as extended fault ruptures, or as point sources. The approach that is used influences how the CEUS ground motion model is implemented. No documentation is provided on	Documentation should be provided that describes how seismic sources are modeled in the PSHA (i.e., how the SSC and GMMs) were implemented in the Vogtle PSHA.	A PSHA report has been prepared that describes how earthquake events were modeled for area sources in the PSHA calculations. This was by modeling each earthquake as a point source, and using correction factors for distance and ground motion uncertainty that modify the ground motion

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
			used to model earthquakes. (This F&O originated from SR SHA-J1)	either of these subjects (earthquake source modeling and use of the ground motion attenuation models). From questions posed to the PSHA analysts, it is our understanding that earthquakes were modeled as point sources and the appropriate ground motion aleatory uncertainty was used in the calculation.		estimate to include the effect of a closer distance to a fault rupture (because the rupture may be closer to the site than the single point used to represent that event) and the uncertainty in ground motion because the azimuth of the rupture is unknown. These correction factors were published by EPRI. This finding has been resolved with no significant impact to the SPRA results or conclusions.
12-11	SHA-J1	Not Met	As part of the PSHA implementation, the analyst has alternatives for modeling the earthquake occurrences in the calculations. The PSHA documentation does not describe the approach that was used to model earthquakes in RLME sources. (This F&O originated from SR SHA-J1)	The PSHA analysts were asked to describe the approach that was used to model earthquakes in the Charleston RLME seismic source. The response indicated that earthquakes in the Charleston RLME source were modeled using 'pseudo faults'. The PSHA report does not: 1. Describe that a 'pseudo fault' approach was used to model earthquakes in the Charleston RLME source. 2. Provide a definition of 'pseudo faults'. 3. Describe how the 'pseudo fault' approach was implemented for the Charleston RLME seismic source (e.g., what was the fault	Provide a description of the earthquake modeling approach that was used to model the Charleston RLME seismic source and how the approach was implemented.	A PSHA report has been prepared that describes how pseudo-faults were implemented to represent the Charleston RLME source. This includes: 1. A description of the pseudo-faults. 2. A definition of pseudo-faults as constructed faults that represent possible sources of future large earthquakes. 3. Implementation of the pseudo-faults including spacing and limits at the borders of the Charleston source. 4. Documentation of the rupture area, length, and width that were

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
				spacing that was used; how was the earthquake rate distributed to the faults, etc.). 4. Document the fault rupture model that was used. 5. Describe how earthquake events are distributed on the faults.		estimated for possible future earthquakes. 5. A description of how earthquake ruptures are distributed on the faults. This finding has been resolved with no significant impact to the SPRA results or conclusions.
12-15	SHA-I1, SHA-I2	Not Met	A screening assessment was performed for soil liquefaction and is described in seismic fragility calculation. A screening assessment was not performed for other potential seismic hazards. (This F&O originated from SR SHA-I1)	A screening analysis was not performed for hazards such as settlement, fault displacement, tsunami, seiche, etc. It is anticipated these other seismic hazards will be screened out.	A screening analysis for other seismic hazards should be performed and documented as part of the PSHA and SPRA. It is expected that information in the FSAR for Vogtle 1 & 2 and in the COLA for Units 3 & 4 can be used to support this requirement.	This evaluation was done for the Vogtle 3&4 COLA and is noted in the ESP SAR. The Vogtle 3&4 evaluation is applicable to, and has been cited in, the Vogtle 1&2 SPRA Fragility report. This finding has been resolved with no significant impact to the SPRA results or conclusions.
12-16	SHA-J1	Not Met	The Vogtle PSHA has gone through a number of changes and revisions since 2012 due to changes in models, input data, etc. As new calculations were performed and reports generated, sensitivity results, were not carried forward. As a result,	The documentation of the PSHA is provided in a collection of documents that were prepared in the 2012-2014 time frame. There does not exist a single document that contains a set of results that is based on the current PSHA model.	Prepare a complete and up-to-date PSHA document that includes all results, sensitivity calculations, deaggregation results, etc. that is based on the current model.	A PSHA report has been prepared that includes hazard results, uncertainties in hazard, and sensitivities to input uncertainties; this summarizes hazard results for the Vogtle site. This finding has been resolved with no significant impact to the SPRA results or conclusions.

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
			there does not exist a current report that includes all PSHA results, deaggregations, etc. that is based on the current PSHA model. (This F&O originated from SR SHA-J1)			
12-18	SHA-B2, SHA-C4, SHA-H1	I/II Not Met Not Met	The Vogtle PSHA is based on the CEUS SSC seismic source model which was completed in 2012. The SSC model was a developed at a regional scale that was based on data gathered up until about 2010. (Note, the date when data was gathered varied; for example the earthquake catalog was complete through 2008.) In the sense that the CEUS SSC model was not specifically performed as a site-specific PSHA for the Vogtle site. (This F&O originated from SR SHA-B2)	As part of a site-specific PSHA, there is a need to gather, review and evaluate new geological, seismological, or geophysical information or information that is defined at a scale that was not considered in the development of the CEUS SSC model. As part of the Vogtle SPRA, no effort was made to gather up-to-date and local (local to the Vogtle site) information to evaluate whether any new information has become available on active faulting and/or the development new seismic sources or the revision of sources in the CEUS SSC model in the vicinity of the Vogtle plant. Since up-to-data was not gathered, consideration of alternatives could not be addressed.	A data gathering effort should be undertaken to identify new information that post-dates the CEUS SSC data collection effort. The data gathering effort should also look for information local to the Vogtle site region that was not considered, or at a scale that was not addressed as part of the CEUS SSC regional evaluation. Some of this information may be available in the COLA for Vogtle Units 3 & 4.	A detailed study of new geological, seismological, and geophysical information was conducted, to determine if any information subsequent to the EPRI SSC model is available that should be incorporated into the seismic hazard results for Vogtle. This study is described in the SPRA documentation. While the area around the site continues to be studied by many earth scientists, there was no new information identified that would change the estimate of seismic hazard for Vogtle. This finding has been resolved with no significant impact to the SPRA results or conclusions.

Attachment 3 - Resolution of the VEGP Seismic PRA Peer Review Findings

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
12-2	SHA-J1	Not Met	The method that is used in the Vogtle PSHA to estimate the soil site hazard is not described or referenced. (This F&O originated from SR SHA-J1)	For soil sites, the soil hazard is generally (though not exclusively, since other methods could be used) determined in two steps; probabilistic rock hazard results are estimated which are then combined with probabilistic estimates of the site response. The method used in the Vogtle PSHA to estimate the soil hazard is not described.	The documentation should include a description of the methodology that is used to combine the rock hazard results and the site amplification factors to determine the soil hazard at the Vogtle site.	The methodology used for the surface hazard calculation has been described in detail, and a comparison made between the GMRS using the two approaches 2A and 3. Approach 2A was used for the calculation of SSI input motions at foundation elevations and Approach 3 was used for the calculation of surface hazard and GMRS at the ground surface, as defined in NUREG/CR-6728. It was concluded that the use of Approach 2A USHRS as input to the SSI analysis of the Vogtle plant is considered acceptable and does not present any significant inconsistency with the seismic hazard curve and GMRS at the ground surface, which were calculated using Approach 3. This finding has been resolved with no significant impact to the SPRA results or conclusions.

Attachment 3 - Resolution of the VEGP Seismic PRA Peer Review Findings

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
12-22	SHA-E2	II/III	Versions 1 and 2 of the site response Calculation do not describe a framework for evaluating and characterizing sources of aleatory and epistemic uncertainty and how the approach was implemented. (This F&O originated from SR SHA-E2)	The site response calculation does not present a clear description of how aleatory and epistemic uncertainties are identified and evaluated. As a result it is difficult to track the propagation of uncertainties is carried out in the site response analysis. It is worth noting that there is some epistemic site response uncertainty that is accounted for in the rock GMPEs.	A framework and approach for evaluating and modeling uncertainties in the site response should be developed and implemented. The site response calculation documentation should fully describe the methodology and its implementation.	A description of the methodology used to account for epistemic and aleatory uncertainties in soil hazard has been added to the documentation. This finding has been resolved with no significant impact to the SPRA results or conclusions.
12-23	SPR-E5	II	The quantification process has included the uncertainties in the seismic hazard, fragility and systems-analysis elements of the SPRA. The results presented are internally inconsistent and are inconsistent with the results reported in other sections for CDF and LERF, respectively. (This F&O originated from SR SPR-E5)	The documentation presents the results of three different uncertainty calculations for CDF and LERF. In addition, point estimates for CDF and LERF are calculated and reported in other sections. Thus the documentation reports two estimates of the mean CDF and LERF respectively from different uncertainty calculations and a 'Point Estimates' result for each. All of these results are different than the point estimate (approximate mean) reported in other sections for CDF and LERF, respectively. The documentation in the report does not describe the basis (inputs) for these calculations, or offer an interpretation of the results.	Develop and document an understanding of the earlier point estimate results for CDF and LERF and of uncertainty results.	Additional detail has been added to the SPRA Quantification report to document the uncertainty, importance, and sensitivity analyses and relate the uncertainty analysis mean CDF and LERF to the point estimate values. This finding has been resolved with no significant impact to the SPRA results or conclusions.
12-24	SPR-E5	II	The Quantification report does not	The uncertainty analysis is presented with the results	Provide documentation of the uncertainty analysis	Additional detail has been added to the

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			provide documentation of the uncertainty analysis results. (This F&O originated from SR SPR-E5)	reported. The report provides limited discussion of the results and the insights that might be gained from them. The two sets of results that are reported are not discussed in terms of their relationship to each other. For instance the mean values should be the same (but are not). The uncertainty estimates provide insight to the total uncertainty and the contribution of the basic event uncertainty to the total. In addition, neither the table of results or the discussion identifies what is the 'final' uncertainty result that includes the propagation of uncertainties of all elements of the SPRA to the estimates of CDF and LERF.	that describes the results, how they are being interpreted and the insights that are derived from them.	documentation of the seismic plant response model, model implementation, and quantification in the QU report. In addition, the uncertainty, importance, and sensitivity analyses are described in more detail. This finding has been resolved with no significant impact to the SPRA results or conclusions.
12-26	SPR-E5	II	There are differences in the results for CDF and LERF that are reported. A possible contributor to these differences may be due to the number of Monte Carlo simulations that were performed. (This F&O originated from SR SPR-E5)	The report does not present the results of sensitivity calculations with regard to the number of Monte Carlo simulations that are needed to produce stable results. It is our understanding from discussion with the PRA staff that these types of sensitivity calculations were performed.	Document the results of sensitivity calculations on the number of Monte Carlo simulations required to produce stable results.	Updated Monte Carlo uncertainty runs have been performed with 20,000 iterations for SCDF and SLERF. This is a sufficiently high number of simulations to produce a stable result. The SPRA documentation has been updated to clearly indicates the results. This finding has been resolved with no significant

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
						impact to the SPRA results or conclusions.
12-27	SPR-F2	Met	Documentation should be provided that describes how the plant model analysis is quantified. (This F&O originated from SR SPR-F2)	The current quantification document does not provide a clear description of the how the plant model is quantified. For example the discussion does not identify how calculations are performed, what the limitations of these quantifications are and how they affect the results.	Provide clear and complete documentation of the approach used to quantify the seismic plant response model, to perform the risk quantification, uncertainty analysis, and importance analysis.	The QU report documentation has been updated to describe the quantification process, including the technique for combining cutsets over the 14 acceleration intervals, and obtaining the importance measures. This finding has been resolved with no significant impact to the SPRA results or conclusions.
12-29	SPR-E2	Met	The Quantification report provides limited documentation of the process and methods that were used to perform the uncertainty analysis. (This F&O originated from SR SPR-E2)	There is limited documentation of the process and the numerical methods that were used to perform the uncertainty analysis. Based on the documentation that is provided and discussions with the PRA staff there is limited but not complete understanding of the methods that were used and the relationship of these methods to the results were obtained. In some cases (as described in the documentation) the results from the uncertainty analysis are not the same as the results reported in other sections of the documentation for CDF and LERF (though this connection is not	Document the process and methods that were used to perform the uncertainty analysis. Where appropriate document where consistencies and potential inconsistencies in results might be expected.	Additional detail has been added to the QU report to document the uncertainty, importance, and sensitivity analyses and relate the uncertainty analysis mean SCDF and SLERF to the point estimate values. This finding has been resolved with no significant impact to the SPRA results or conclusions.

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
				clearly stated in the report). However, it would seem the results should be internally consistent.		
12-31	SPR-F1	Not Met	The standard requires a level of documentation that provides an understanding of the seismic plant response model and the quantification. This requirement is not met. (This F&O originated from SR SPR-F1)	There is limited documentation that describes the seismic plant response analysis and quantification; how the model was implemented, how the quantification was performed and a discussion of the analysis results. To meet this requirement, the documentation must be in considerable detail in order to support the review process and future updates. Part of the documentation should include a detailed discussion of the results, sensitivity calculations, and the uncertainty analysis.	Documentation should be provided in sufficient detail that describes the seismic plant model, how it is implement and quantified.	Additional detail has been added to the documentation of the seismic plant response model, model implementation, and quantification in the QU report. In addition, the uncertainty, importance, and sensitivity analyses are described in more detail. This finding has been resolved with no significant impact to the SPRA results or conclusions.
12-32	SPR-F3	Met	The documentation of the sources of model uncertainty and a description of the analysis assumptions is not complete in the SPRA quantification report. In addition, there is not a clear description of the uncertainty analysis and the contributors to the total uncertainty beyond a	The purpose of this supporting requirement is that documentation should be presented that addresses the sources of epistemic (knowledge) uncertainty that are modeled and their contribution to the total uncertainty in CDF and LERF. In addition, the documentation should discuss elements of the seismic plant model where there may be latent sources of uncertainty that are not modeled and assumptions that are made in	Document and discuss the contribution of the different sources of uncertainty that are modeled in the SPRA.	The documentation of the uncertainty analysis has been expanded in the Quantification report. A discussion of sources of model uncertainty has been added to the report, and potentially important sources have been addressed in the sensitivity analysis. This finding has been resolved with no significant impact to the SPRA results

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
			simple report from UNCERT. (This F&O originated from SR SPR-F3)	performing the analysis.		or conclusions.
12-36	SHA-B3, SHA-C4, SHA-H1	I/II, Not Met Not Met	As part of a site-specific PSHA, an up-to-date earthquake catalog should be used. The CEUS SSC study involved the development of a comprehensive earthquake catalog based on data through 2008. The Vogtle site-specific PSHA should consider the impact SSC of any additional seismicity since 2008 up to the time the study started. (This F&O originated from SR SHA-C4)	As part of the Vogtle PSHA an effort was not made to gather data on earthquakes that occurred since 2008. As such, the analysts did not assess whether more recent seismicity is consistent with the characterization parameters estimated as part of the CEUS SSC study (NRC, 2012). We note that as part of the Vogtle PSHA, calculations were performed to recompute the seismic hazard at the site to take into account changes in the CEUS SSC earthquake catalog through 2008 that were made following the completion of the CEUS SSC study. These changes reflect the identification of reservoir induced seismicity earthquakes and the re-interpretation of the location of some earthquakes in the Charleston, SC area that occurred in the 1880's (EPRI, 2014). References EPRI (2014). Review of EPRI 1021097 Earthquake Catalog for RIS Earthquakes in the Southeastern U. S. and	An up-to-date earthquake catalog for the Vogtle site region should be developed to assess whether modifications to the seismic source recurrence parameters or required. The updated catalog, resources used in compiling the update and the results of the evaluation should be documented as part of the PSHA. If more recent seismicity is not consistent with the existing CEUS SSC seismic source parameters, the parameters should be updated and the PSHA should be updated.	An update to the earthquake catalog was prepared from the time of the CEUS SSC catalog (through 2008) through February 2016. The rate of occurrence of earthquakes within 320 km of the Vogtle site was compared to the rate of earthquakes represented by the CEUS SSC seismic source model for that same area, this comparison being made for M>2.9. It was found that the updated catalog implied a rate of earthquakes that is lower than the mean rate from the CEUS SSC seismic sources. Therefore, incorporating the effects of a updated catalog on the hazard at Vogtle would decrease the hazard slightly, and was not undertaken. This comparison is documented in the SPRA documentation. This finding has been resolved with no significant

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
				Earthquakes in South Carolina Near the Time of the 1886 Charleston Earthquake Sequence, transmitted by letter from J. Richards to R. McGuire on March 5, 2014.		impact to the SPRA results or conclusions.
12-8	SHA-J3	Not Met	A foundational element of PSHA as it has evolved over the past 30 years is the development and implementation of methods to identify, evaluate, and model sources of epistemic (model and parametric) uncertainty in the estimate of ground motion hazards. As such fairly rigorous analyses are carried out (SSHAC studies) to quantitatively address model uncertainties. At the same time there is within any analysis sources of uncertainty that are not directly modeled and assumptions that are made for	The documentation of the sources of model uncertainty analysis and a description of the analysis assumptions is not complete in the PSHA report in its current form such that a clear understanding of the contribution of individual sources of uncertainty to the estimate of hazard are understood. Limited information on the contribution of seismic sources to the total mean hazard is presented, but information on the contributors to the uncertainty is not provided. With respect to addressing model uncertainties and associated assumptions there are some examples that can be identified in the Vogtle PSHA. For example, in the site response analysis the assumption is made that the 1D equivalent linear model (SHAKE type) to estimate the site amplification and ground motion input to plant structures is appropriate.	The resolution to this finding could involve: 1. Documentation and discussion of the contribution of different sources of uncertainty that are modeled in the PSHA. The documentation of the contribution of different sources of uncertainty can be shown by means of 'tornado plots' that quantify the sensitivity of the hazard at different ground motion levels to the various branches in the logic tree. These plots show which sources of epistemic uncertainty are most important. It should include the source model uncertainty, ground motion model uncertainty, and site response uncertainty. Currently, the total uncertainty is shown	Sources of uncertainty in the seismic hazard analysis for Vogtle are discussed in the updated SPRA documentation. These include uncertainty in seismic source model (for background earthquake sources and for the Charleston RLME), in maximum magnitude for background seismic sources and for the Charleston RLME, in ground motion prediction equation, in smoothing assumptions for seismicity parameters in background sources, and in site amplification model. "Tornado plots" are included in the updated SPRA documentation that show the contribution to total uncertainty in seismic hazard from source model uncertainty, maximum magnitude uncertainty,

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			pragmatic or other reasons. There are also sources of model uncertainty that are embedded in the context of current practice that are 'accepted' and typically not subject to critical review. For instance, in the PSHA it is standard practice to assume that the temporal occurrence of earthquakes is defined by a Poisson process. This assumption is well accepted despite the fact that it violates certain fundamentally understanding of tectonic processes (strain accumulation). A second practice is the fact that earthquake aftershocks are not modeled in the PSHA, even though they may be significant events (depending on the size of the main		by the hazard fractiles, but it is not broken down to provide understanding as to what is most important. 2. Identification and discussion of model assumptions that are made.	ground motion prediction equation uncertainty, smoothing assumptions for seismicity parameters in background sources, and site response uncertainty. These plots are presented for 10 Hz and 1 Hz spectral acceleration, for ground motion amplitudes corresponding to mean annual frequencies of exceedance of 1E-4 and 1E-5. These "tornado plots" show that ground motion prediction equation is the major contributor to seismic hazard uncertainty for both 10 Hz and 1 hz spectral acceleration, and maximum magnitude of the Charleston RLME source is an important contributor for 1 Hz spectral acceleration. The use of equivalent linear one-dimensional site response analysis, and its associated assumptions, and its adequacy for the Vogtle site are documented in the hazard calculation. This finding has been resolved with no significant impact to the SPRA results or conclusions.

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			event). In the spirit of the standard it seems appropriate that sources of model uncertainty that are modeled as well as sources of uncertainty and associated assumptions as they relate to the site-specific analysis should be identified/ discussed and their influence on the results discussed. As SPRA reviews and the use of the standard has evolved, it would seem the former interpretation is reasonable, but potentially incomplete. It is reasonable from the perspective that documentation of the sources of model uncertainty and their contribution to the site-specific hazard results is a valuable product that			

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			supports the peer review process and assessments in the future as new information becomes available). Similarly, documenting assumptions provides similar support for peer reviews and future updates. The notion that model uncertainties and related assumptions that are not addressed in the PSHA is at a certain level an extreme requirement that may not be readily met and may not be particularly supportive of the analysis that is performed. For purposes of this review, the following approach is taken with regard to this supporting requirement: 1. The documentation should present			

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			quantitative results and discussion the sources of epistemic uncertainty that are modeled and their contribution to the total uncertainty in the seismic hazard. 2. The documentation should discuss elements of the PSHA model where their may be latent sources of model uncertainty that are not modeled and assumptions that are made in performing the analysis. (This F&O originated from SR SHA-J3)			
14-1	SFR-A2	I	The conservatisms that exist in structural demand were not properly accounted for in the estimation of component and structure fragilities. (This F&O originated from SR SFR-A2) .	SFR-A2 requires that seismic fragilities be based on plant-specific data and that they are realistic and median centered with reasonable estimates of uncertainty. The structural response factor used in all component fragilities reviewed is reported as 1.0. This factor will be greater than 1.0 because of the conservatism introduced in the demand through the structural analysis. Because of this, the component and	Account for conservatism in the building response analyses in the structure response factor for component fragility evaluations. Use clipped spectra for assessing anchorage capacities.	Evaluation of anchorage has been updated to include clipping of in-structure response spectra, and the methodology is documented in the fragility notebook. Structure response is dominated by the soft soil on which Vogtle 1 and 2 structures are founded. This would cause higher damping at lower hazard frequency levels and lead

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				structural fragilities are biased low. The fragilities developed for structures and components that are mounted in those structures will be biased low because the input structural demands include conservatisms. Time histories used for the SSI analysis have been processed such that each record envelopes the target UHRS. This will introduce some level of conservatism. The input motion at the control point has been scaled to produce resultant FIRS that envelopes the FIRS coming out of the site-consistent input motion analysis. In structure response spectra coming out of the SSI analyses were not peak clipped when computing anchorage demands. Structure response at the calculated equipment fragility levels is considerably higher than the 1E-4 UHRS considered in the building response analyses. The structure will have additional cracked shear walls and higher associated levels of damping at these higher ground motions.		to stress similar to the stress calculated for the buildings at 1E-4. As a result the structural response factor is close to 1 and is accounted for appropriately in the fragility evaluations. The input motion at the control point in the SSI analysis has been modified to reasonably match the corresponding 1E-4 UHRS from the site-consistent input motion analysis. This finding has been resolved.
14-10	SFR-A2	I	Significant conservatisms were noted in several sampled fragility calculations.	In the fragility calculations of heat exchangers (PRA-BC-V-14-009 Appendix A), nozzle loads significantly contribute to the seismic demands which form the	Realistic nozzle loads should be determined for fragility evaluation of heat exchangers.	The CCW and ACCW heat exchanger capacities have been updated to reflect realistic nozzle loads. The equipment fragilities have

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			(This F&O originated from SR SFR-A2)	basis for the median capacities. Based on in-plant walkdowns by the peer review teams and also noted in the walkdown report, the piping is well supported in all directions and will not impose significant nozzle loads during a seismic event. The CCW and ACCW capacities are below the 2.5g screening level and are significant contributors to risk so more realistic fragilities are required. Battery rack 11806B3BN3 in calculation PRA-BC-V-14-010 Appendix J2 is governed by GERS capacity. The GERS capacity is taken to be 1g, which corresponds to a frequency of 1 Hz. This is not realistic. The actual capacity is about 4g. The median capacity reported in the calculation is well below the 2.5g screening level and is not realistic. The median capacity reported for the Turbine Driven Auxiliary Feedwater Pump is reported in Calculation PRA-BC-V-14-008 as 1.56g. This fragility is based on the seismic qualification document. The frequency range of interest for the fragility evaluation should be centered around the fundamental frequency of the assembly and	The equipment capacity factor should be based on the frequency range of interest. That frequency range of interest is centered at the fundamental frequency of the pump, and considers some uncertainty in that frequency.	been updated to account for appropriate frequency, and uncertainty has been considered in these updates. This finding has been resolved.

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				not consider the entire frequency range.		
14-14	SFR-G2	Met	The iterative process used for developing realistic fragilities is not well documented. (This F&O originated from SR SFR-G2)	In review of the seismic fragility calculation for the safety features sequencer (11821U3001), it was discovered that an iterative process was used. The initial fragility is based on EPRI 6041 screening methodology and an equipment capacity factor that is equal to the EPRI 6041 median capacity divided by the peak in structure demand. If this value is less than the screening capacity (2.5g), then the fragility may be refined by examining the component fundamental frequency. The fragility may be further refined by examining component specific qualification test reports. However, the fragility used in the logic tree by the systems analyst is generally the highest of these computed. This is reasonable and appropriate, however, this process is not described in the fragility notebook or fragility calculations.	Add a description of the iterative process for computing the component fragilities in the SPRA documentation	The description of the iterative process for computing fragilities has been documented. This finding has been resolved with no significant impact to the SPRA results or conclusions.
14-17	SFR-D2	Met	Inconsistencies and errors in NSSS fragility development. (This F&O originated	Fragilities for the Vogtle 1&2 Nuclear Steam Supply System (NSSS) are based on the results of the Westinghouse analysis of record (AOR) associated with the safe shutdown earthquake (SSE). In general, fragilities are developed through scaling of the	Update SNC calculation no. PRA-BC-V-14-015 to incorporate corrections and enhancements.	The following changes have been made: NSSS fragility calculations have been updated to reflect Westinghouse-provided critical loads and support capacities represented in the critical failure modes;

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
			from SR SFR-D2)	SSE demands to the RLE and using the AOR seismic margins. Various deficiencies were noted in the development of the fragilities associated with these components. Basis: The NSSS Seismic fragility evaluation (SNC calculation no. PRA-BC-V-14-015) includes detail calculations for each of the major NSSS components. It indicates that the critical failure modes for the components are controlled by the support capacities. During the Peer Review, the team members discussed these issues with SNC staff to obtain insights and develop potential resolution paths. Key issues included: (a) Basis for assumption that the support capacities represented the critical failure mode was not documented. SNC indicated that this was based on input from Westinghouse and NUREG-3360 and will update the fragility evaluation of provide this information. (b) Inelastic energy absorption was not credited to increase the median capacities - this does not result in realistic median capacities (overly conservative). (c) Reactor Coolant Pump fragility was based on consideration of the failure of the attached CCW		the effect of inelastic energy absorption is factored in and documented in fragility calculation as appropriate; the Reactor Coolant Pump fragility has been updated to reflect the failure of the pump associated with LOCA; the reactor internals fragility has been updated in the calculation; and the new fragilities have been reflected in the updated SPRA model. This finding has been resolved.

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				piping, due to an assumption that a small-break/RCP seal LOCA was critical. It was learned during the Peer Review that failure in the system model was linked to a large-break LOCA, so the failure mode considered in the fragility evaluation is not consistent with the system model - SNC indicated that they will revise the fragility evaluation. (d) Reactor Internal fragility evaluation determined the demand based an average spectral acceleration over the range of 2 to 3 Hz, rather than using the peak acceleration in this range of the ISRS, and did not consider the contribution of higher modes. SNC indicated that this was done to avoid an overly conservative capacity, but agreed that the contribution of higher modes should be addressed, and will revise the calculation. (f) Control Rod Drive Mechanism fragility evaluation assumed that material stresses were the critical failure mode, and did not address the potential impact of deflections on rod drop. SNC indicated that information provided by Westinghouse (based on a Japanese testing program) indicated that the deflection levels associated with seismic loading does not impact rod drop, and		

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				agree to add this discussion to the calculation.		
14-20	SFR-E4, SPR-B9	Met Met	Seismic induced fire evaluations are not documented in the walkdown report or fragility calculations. (This F&O originated from SR SFR-E4)	The only mention for seismic induced fire evaluation is contained in the quantification notebook. Based on discussions during the peer review, it is understood that seismic induced fire was a key consideration during the walkdowns. However, detail of the walkdown procedure for fire following earthquake is missing. The write up should include team composition, methodology, screening criteria, and results,	Seismic induced fire is an important element of the fragility evaluation process and this should be clearly documented.	The seismic-induced fire and flood evaluations have been updated, and documented in the fragility and quantification report. This includes the details of the walkdown procedure used to evaluate the potential for seismically induced fires, including the methodology, screening criteria and results. This finding has been resolved with no significant impact to the SPRA results or conclusions.
14-4	SFR-D1	Met	A potential for sloshing induced inundation of the NSCW Pumps (11202P4007, 11202P408) and associated discharge motor operated valves (1HV11600, 11606, 11607, 11613) in the NSCW exists and was not identified either in the walkdowns or subsequent analysis.	SFR-D1 requires that realistic failure modes of structures and equipment that interfere with the operation of that equipment be identified. The potential for earthquake induced sloshing of the water within the NSCW tower exists. From field walkdowns of the NSCW it was observed that there is a potential for sloshing of contents to potentially splash onto or flood the pumps and or motor operated valves on the attached discharge piping.	Evaluate the potential for flood induced failure of the NSCW Pumps or NSCW discharge MOVs.	The evaluation for potential flood induced failure of the NSCW pumps or the NSCW discharge MOVs has been performed and documented in the fragility calculation for the NSCW tower. There was no significant impact on the pump or MOV fragilities. This finding has been resolved with no significant impact to the SPRA results or conclusions.

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			(This F&O originated from SR SFR-D1)			
14-5	SFR-D1	Met	The potential for seismically-induced differential settlements between structures was not addressed. (This F&O originated from SR SFR-D1)	Vogtle 1&2 is a soil site, with engineered fill from the rock interface to the finished grade. The in-scope Seismic Category I structures have foundations with varying embedment depths, ranging from surface founded (elev. 220 ft.) to a foundation embedment of 110 ft. (elev. 110 ft.). Since soils, including engineered fill, will consolidate/settle to some extent when subjected to high level earthquake ground motion, and the amount of settlement is proportional to the thickness of the soil layer under the foundation, the settlement of one structure relative to another structure is dependent on the depth of the foundation embedment. The Fragility Notebook (PRA-BC-V-14-025) does not address the potential differential settlement between buildings, or the potential effect on commodities (e.g., piping, electrical raceways, HVAC ducts, etc.) that cross the separation	Develop estimates of the differential settlements between adjacent structures and assess the fragility of commodities based on their ability to accommodate the associated differential displacements.	Documentation has been updated to include the effects of earthquake induced settlement; no significant differential settlements were computed between the structures. This finding has been resolved with no significant impact to the SPRA results or conclusions.

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				between adjacent structures. During the performance of the Peer Review, SNC personnel indicated that the consideration of differential settlements was not required, since the structures were founded on engineered fill.		
14-6	SFR-G2	Met	The results of the seismic gap/shake space walkdowns are not documented. (This F&O originated from SR SFR-G2)	The walkdown guidance provided in Appendix F (Checklists and Walkdown Data Sheets) of EPRI NP-6041 includes attributes of seismic gaps between structures which should be addressed in the performance of the walkdowns. These include the clearance between adjacent structures and the ability of any subsystems (e.g., piping, cable trays, HVAC ducts) spanning the gap to accommodate the differential seismic displacements. The Seismic Walkdown Report (PRA-BC-V-14-005) does not include documentation of the results/findings/observations associated with the inspection of the seismic gaps between structures or the subsystems spanning the gap. During the performance of the Peer Review, SNC personal indicated that inspection of the seismic gaps was included in the seismic walkdowns, but not explicitly described in the report. The ability of components to	Provide documentation of the results of the seismic gap walkdowns.	As noted in the Finding basis, inspection of the seismic gaps was included in the seismic walkdowns. Piping across seismic gaps is designed with adequate flexibility to accommodate building motions, and pipe sleeves provide adequate gaps for piping movement. The documentation has been updated to reflect the inspections performed during the walkdowns. This finding has been resolved with no significant impact to the SPRA results or conclusions.

Attachment 3 - Resolution of the VEGP Seismic PRA Peer Review Findings

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
				accommodate potential differential movement at the building separations is implied in the discussion of rugged components (piping, cable trays, and HVAC ducts) in the section on Rationale for Screening in the report. In addition, information from the Vogtle IPEEE Report indicated that the seismic gaps had been inspected during the IPEEE.		
14-7	SFR-A2, SFR-F4	I, Met	The fragility evaluation for the Containment Polar Crane (in fragility notebook) did not address the impact of variation in the fundamental frequency on the applicable seismic demand. (This F&O originated from SR SFR-A2)	The determination of the fundamental frequency of structures and components involves a certain degree of uncertainty. This uncertainty must be accounted for in the determination of the seismic accelerations from the applicable in-structure response spectra (ISRS). The section of the Polar Crane of the Fragility Notebook evaluates the polar crane as a potential seismic interaction source relative to the reactor vessel and other NSSS components inside the containment structure. In the determination of the vertical spectral acceleration applicable to the polar crane, the computed fundamental frequency falls within a valley in the applicable ISRS, on the low frequency side of the primary spectral peak.	Update the fragility evaluation for the polar crane to address potential uncertainty in the fundamental frequency and the contribution of higher modes.	The fragility evaluation of the polar crane has been updated to address potential uncertainty in the fundamental frequency and contribution of higher modes. This finding has been resolved.

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F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
				Uncertainty in the calculated frequency, and the contribution of high modes, could result in an increase in the applied vertical acceleration. During the performance of the Peer Review, SNC personnel provided a written response indicating that it is appropriate to increase the applied acceleration by 50%, which will result in a 20% decrease in the median capacity of the polar crane.		
14-8	SFR-F3	II/III	Relay fragility calculations include conservative assumptions. (This F&O originated from SR SFR-F3)	The relay evaluation for the turbine driven auxiliary feedwater pump control panel in calculation PRA-BC-V-14-008 is based on a generic capacity for motor starters and contactors (intended for motor control centers) and an amplification factor associated with center of door panel response. Based on walkdown observations the relay is not mounted on the door panel so is likely on an internal bracket. The median capacity of 0.627g is well below the screening level and is not realistic. The relay evaluations in calculation PRA-BC-V-14-009 are governed by response in the vertical direction, and the in-cabinet amplification factors used in the calculation are associated with horizontal	Perform more realistic relay fragility evaluations.	The relay fragilities have been updated using the appropriate response and in-cabinet amplification factors, and are realistic. This finding has been resolved.

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				response. The resulting median capacities of 0.762g (Appendix M1) and 1.026g (Appendix M2) are well below the screening level and are not realistic.		
14-9	SFR-D2	Met	The seismic walkdown report includes a number of open items that are not are not traceable to a resolution (This F&O originated from SR SFR-D2)	The summary of the seismic walkdowns documents a number of issues identified during the performance of the walkdowns that required follow-up actions (31). These include spatial interaction issues, housekeeping issues, anchorage issues, valves having configurations that do not meet the EPRI guidelines, configuration issues, installation errors, etc. The Seismic Walkdown Report does not document how the issues identified during the walkdowns have been addressed, either in the field (e.g., correction of installation errors, resolution of housekeeping issues) or in the fragility evaluations (e.g., valve configurations, anchorage issues). During the performance of the Peer Review, the Peer Review Team provided a list of the walkdown issues to SNC personnel, and SNC provided a summary of how they were addressed. Most issues had been adequately addressed during the development of the SPRA, but it was determined that	Perform resolution of open items and provide documentation of the resolution associated with each of the issues, either in the Fragility Notebook or the SPRA Database.	The noted walkdown issues have been evaluated and reflected in the revised documentation: - potential piping interaction; - the difference in inverter anchorage configuration; - potential interaction concerns with the overhead heater; this evaluation is in the fragility notebook. Valve operator heights & weights that were outside EPRI guidelines have been taken into account in the fragility analysis for these components. The Diesel Generator Exhaust Silencer was re-evaluated to the as-operated condition. The fragility analysis for these components has been completed for the as built condition. This finding has been resolved.

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
				the following would require further effort for resolution: (a) Potential interaction between piping and deluge valve - follow-up walkdowns required. (b) Anchorage configuration on inverter - follow-up revision to fragility evaluation required (c) Overhead heater poses potential interaction issue - follow-up walkdown required. (d) Valve operator heights/weights outside of EPRI guidelines - follow-up walkdown required. (e) Diesel Generator exhaust silencer anchor bolt nuts - not addressed in fragility evaluation, further evaluation required. (f) Valve operator heights outside of EPRI guidelines and potential lack of yoke support - these valves are part of the unfinished scope described in the Fragility Notebook, which will be completed in the future. (g) Valve operator heights outside of EPRI guidelines - further evaluation required.		
16-1	SFR-F3, SPR-B4, SPR-E5	II/III, Not Met, II	The model presented for peer review did not incorporate the effects of relay chatter as the analysis was not yet complete.	Relay chatter is consistently being observed as a significant contributor to risk profile in recently peer reviewed S-PRAs and it is therefore realistic to expect that relay chatter is a potential significant contributor. During the peer review it was discussed that the SPRA team	Complete the analysis and incorporate the effects of relay chatter and similar devices in the PRA logic model.	The approach to screening and modeling of seismically-induced relay failures and chatter was provided to the peer review team and determined to have been performed appropriately; only the incorporation into the

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			(This F&O originated from SR SPR-B4)	does not believe relays will be a significant contributors but it was also said that this conclusion/ expectation is based on potentially crediting operator actions. Thus, the effects of relay chatter per se may be significant (and provide some insights) while the combination of relays and a number of HEP may not be.		model of the impacts of relay chatter from unscreened relays was not complete. The final screening resulted in only 2 relays being incorporated into the model, with one having an operator action. Relay chatter fragilities and impacts have been incorporated into the seismic model, in a manner consistent with that used for other failures. This finding has been resolved.
16-10	SPR-B6	Met	The documentation about the walkdowns in support to seismic impact on HRA appear limited. (This F&O originated from SR SPR-B6)	There is only a short sentence supporting the discussion on alternative access pathways.	More detailed documentation is suggested to support the conclusion on accessibility, alternative route, availability of tools/keys, clear identification of equipment manipulated in each local action. Obviously, the goal of the enhanced documentation is not to convince the peer reviewer that the walkdowns were performed but rather to ensure that the analyst is fully convinced of the conclusions.	Walkdown documentation on accessibility for operator actions, including photos, has been improved. Potential failure of block walls has been reviewed and documented. Required tools and equipment, such as ladders, have been identified with locations when needed. The documentation supports the seismic HRA assumptions and modeling. This finding has been resolved with no significant impact to the SPRA results

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					Past SPRAs have shown examples of equipment needed for the HFE that was not in the SEL, or that has different actuators when manually actuated, or that needed ladders that were not easily accessible or that were close to block walls (or under ceiling that could collapse) that were not considered an issue because the block walls were not near safety related equipment (and therefore not addressed in the rest of the SPRA work). In this perspective, a more systematic documentation of the feasibility and accessibility analysis for each of the HFE credited in the SPRA is suggested.	or conclusions.

Attachment 3 - Resolution of the VEGP Seismic PRA Peer Review Findings

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
16-11	SPR-E2	Met	Missing review of the potential for additional dependencies introduced by the SPRA models (QU-C1&2) (This F&O originated from SR SPR-E2)	It is understood that the investigation performed in internal events to identify potential HFE dependency has been relied upon in the Vogtle SPRA. The SPRA logic may identify additional dependencies trends that were not identified in the internal events.	As this exercise was apparently performed for the Fire PRA (as discussed during the peer review), it is suggested that a review of the potential for unforeseen dependencies trends is performed. As it is understood that the plan is to transition to a different dependency analysis method (based on HRA calculator), this may be addressed within the same transition as it is realistic to expect that not too many (if any) new dependencies would be identified.	A detailed quantitative HRA dependency analysis based on using the HRA calculator was performed and documented. There was no significant impact on results since human actions are not significant contributors in the Vogtle SPRA. This finding has been resolved.
16-12	SPR-E2	Met	Missing documentation of the review of non significant cutsets QU-D5. (This F&O originated from SR SPR-E2)	It is an industry expectation (as discussed in NEI peer review task force meetings) that review of the non significant cutsets is explicitly documented. Based on discussion during the peer review, two reviews were performed to validate the overall model and cutsets. The first was a random review of cutsets at midpoints and low significance for each of the %Gxx initiators to verify that the cutsets are valid cutsets, and that the patterns are appropriate. That is, if one	It is understood that the SPRA documentation will be revised to incorporate explicitly the two reviews discussed in the basis for this F&O. It is also recommended to document the review of cutsets following guidance from the NEI peer review task force.	The QU report has been updated to document the review of both dominant cutsets and non-significant cutsets for both CDF and LERF. This finding has been resolved with no significant impact to the SPRA results or conclusions.

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				cutset is valid, then another cutset with slightly different seismic failures (or random failures) should also be nearby. The second review, more importantly, lowered the median seismic capacity for each of the seismic initiators and some of the other seismic failures to ensure that the model would properly generate valid cutsets. For example, the LLOCA fragility was reduced to 0.5g to generate LLOCA cutsets. For ATWT, the fragility of the CRDs and RV internals were reduced to 0.5g to verify that valid ATWT cutsets were generated.		
16-15	SPR-E6, SPR-F2	Met, Met	Documentation of LERF model applicability review. (This F&O originated from SR SPR-F2)	The current documentation does not explain what are the basis for retaining the LERF logic and analysis unchanged within the SPRA logic. During the peer review the following explanation was provided by the SPRA team: "The internal events Level 2 notebook was reviewed to ensure that the definition of LERF would be appropriate for seismic events. The LERF definition includes the use of a 12 hour time period for release after event initiation, to allow for evacuation. This time	Expand the documentation to ensure that the criteria used to retain the LERF analysis in the SPRA is explained so that the same applicability review can be performed following future potential revisions of the LERF modeling.	The LERF documentation in the QU report was expanded to describe the review of applicability of the internal events PRA LERF analysis to the seismic PRA. This finding has been resolved with no significant impact to the SPRA results or conclusions.

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				period is considered to be valid for Vogtle seismic events, particularly due to the very low population density in the area. Other characteristics, such as bypass and scrubbing, are the same for seismic as for internal events. The logic for the internal events LERF model is very straightforward, with sequences from the CDF model ANDed with the appropriate LERF fault tree. This logic is also appropriate for seismic events."		
16-18	SPR-B8	III	Very small LOCA have been screened from the analysis based on walkdowns but little documentation exists of such walkdowns. (This F&O originated from SR SPR-B8)	The DB has a specific entry for the incore thermocouples and provides pictures of them. Still, in-core thermocouple tubing is not the only possible source of very small LOCA that is envisioned and the only documentation of addressing the other potential sources is in the quantification notebook: "For Vogtle 1&2, the seismic walkdowns inspected and photographed a large sample of the small piping and tubing lines connected to the primary system in order to identify any weaknesses. The piping was judged to be rugged."	To the peer review team knowledge Vogtle is the only plant that has elected to perform dedicated walkdowns in support of not modeling very small LOCA. This would be a best practice but it also behooves to the SPRA team to provide detailed documentation of such walkdowns and how they supported a systematic evaluation of the potential sources of very small LOCA.	Additional information on the walkdown for very small LOCA has been added to fragility report to provide the basis for the VSLOCA screening. This finding has been resolved with no significant impact to the SPRA results or conclusions.
16-2	SFR-C1, SPR-E1	I/II, Met	Fragilities were not corrected to reflect the 2014 hazard	The 2014 hazard was only used as input to FRANX for the final quantification. It is understood	During the peer review the SNC staff answered a question on this topic by	The fragilities have been recalculated based on the 2014 hazard and the new

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			used for quantification. (This F&O originated from SR SPR-E1)	that the fragility estimates have been performed based on the 2012 hazard. While it is not expected nor recommended to regenerate all the fragility work with the new hazard, some consideration on the possible change in fragility due to the use of the newer hazard should be made.	performing an initial limited investigation of the effect on fragilities correction to reflect the 2014 hazard and concluded that the effect of this scaling is not insignificant (especially for LERF). It is recommended to continue and expand this investigation to make the quantification fully consistent with the fragility values.	values incorporated into the SPRA model and quantification. This finding has been resolved.
16-4	SPR-B2	Not Met	The effect of seismic impact on performance shaping factors is considered in the analysis by the usage of the Surry method. (This F&O originated from SR SPR-B2)	There is no assessment of the effect of changing the breaking points in the Surry method. The Surry method is based on methods used in the past at SONGS and Diablo Canyon and the 0.8g breaking point was developed for California earthquakes. In the Vogtle analysis there is no indications on whether the breaking point at 0.8g is also applicable to Vogtle. There are also no sensitivity analyses that would support whether a change in the breaking points is significant or not.	While it is recognized that the industry is still developing methods in support to this particular topic (e.g., recently published EPRI HRA method for external events), some additional considerations should be done to understand the effect of HEPs in the model rather than simply implementing the Surry method as is. Three examples for addressing this finding may be the following: 1. Perform sensitivities on the values of the multipliers and the g levels where the breaking	The methodology used for the seismic HRA analysis is based on defining PSFs as a function of seismic hazard level (bins), which is consistent with the EPRI seismic HRA guidance in EPRI 3002008093. The Integrated PSFs and bins (breaking points) have been updated with additional breaking points and integrated PSFs to reflect seismic binning applicable to Vogtle, in accordance with this finding and consistent with the EPRI guidance. The updated values have been applied to both internal events HFEs and seismic-unique HFEs

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					point happens. 2. Use a different multipliers method with more breaking points. 3. Apply the impact of seismic specific PSF at the individual PSF level (i.e., timing, stress, etc.) in the HRA calculator.	within the plant response model. There was no significant impact on the SPRA results. This finding has been resolved.
16-5	SPR-B1, SPR-F1	Met, Not Met	LOCA modeling and fragility selection not clearly documented. (This F&O originated from SR SPR-F1)	The selection of the fragility data used for all LOCA is discussed in Appendix B.2 of the quantification notebook but is confusing in the mapping of selected fragilities with specific failures. It appears that the fragility selected to represent LOCA sequences are coming from specific components but then they are used to represents sort of surrogate events for potential failures along the piping network. Using localized events as surrogate for pipe network failure is probably conservative and may not be fully consistent with the system success criteria and modeling in the internal events modeling. For example, the seismic-induced MLOCA fragility seems to be based on failure of the pressurizer surge line, which is a localized failure. The seismic-induced MLOCA initiator is mapped to the internal events	Documentation on the use of fragility in support to LOCA should be clarified to better represent the rationale selected and potentially addresses the modeling uncertainties associated with this selection. While this finding is expected to be addressed via documentation, some additional suggestions are provided, such as: 1. Perform a sensitivity to show that the modeling approach described is not significantly skew the results for seismic; 2. Modify the logic by mapping the seismic-induced MLOCA to a different position in the logic (e.g., a dummy event can be entered in	LOCA basis has been re-evaluated and updated. This was partially due to seismic fragility update and partially a matter of adding amplifying information to the LOCA basis. The quantification report includes updated documentation. Although LOCAs are a significant contributor to the SPRA results, the VEGP SCDF and SLERF are sufficiently small that further LOCA modeling sensitivity beyond what has been provided in the updated model quantification is not warranted. This finding has been resolved.

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
				MLOCA initiator. The internal events logic for MLOCA has a split fraction that divides MLOCA (and LLOCA) in four 25% contributors impacting all four CL/HL. Since the seismic-induced MLOCA is a localized failure, the internal events logic is not fully applicable (probably slightly conservative). Because the documentation is potentially leading to a misunderstanding of the selected approach (thus impacting ease on update), this F&O is considered a finding against the documentation SR.	the model to provide a target for the FRANX injection).	
16-6	SPR-B2	Not Met	The effect of seismic impact on performance shaping factors is not considered for any action that was explicitly added for the SPRA (e.g., flood isolation or DG output breaker closure). (This F&O originated from SR SPR-B2)	The Vogtle SPRA elected to use Integrated Performance Shaping Factors (IPSF) multipliers. While this approach was used for the HEPs that were carried over from internal events, it was systematically not done for all the actions explicitly added for seismic. Based on discussion during the peer review, the analyst believed that having designed these actions for specific scenarios following a seismic event, the impact of seismic specific PSF is already included. The objection to this conclusion is	Expand the IPSF approach to all the operator actions credited in the SPRA.	The methodology used for the seismic HRA analysis is based on defining PSFs as a function of seismic hazard level (bins), which is consistent with the EPRI seismic HRA guidance in EPRI 3002008093 [9]. The Integrated PSFs and bins (breaking points) have been updated to reflect seismic binning applicable to Vogtle, in accordance with this finding and consistent with the EPRI guidance. The updated values have been applied to both internal events HFEs and

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
				that the seismic specific PSF should realistically change with the magnitude of the event. This change addresses the change in the overall context of the plant when a small seismic event happens as opposed to when a very large seismic event happens. This seems not to be captured by the approach selected for the Vogtle SPRA. One example of this is that an action that has a 30 minute Tsw (S-OA-BKR-LOCAL) maintains an HEP of 1.60E-03 at all g levels, including the %G14 interval (i.e., >2g). It is understood that this is not expected to be quantitatively significant because failure of the recovered equipment is taken care by the logic model.		seismic-unique HFEs within the plant response model. There was no significant impact on the SPRA results. This finding has been resolved.
16-7	SPR-E2	Met	Base case seismic LERF does not meet the truncation requirements from QU-B3. (This F&O originated from SR SPR-E2)	Both CDF and LERF are truncated at 1.0E-09 with 1000 cutsets managed by ACUBE. This meets the QU-B3 requirement for CDF but not for LERF.	LERF at 1E-11 truncation meets the QU-B3 truncation requirement. Rename LERF at 1E-11 as the base case for LERF.	LERF truncation, which was already considered in sensitivity studies, has been revised appropriately to meet QU-B3. A new LERF truncation limit has been established consistent with the LERF results. Quantification is at 1E-12, which is a suitably low value. This finding has been resolved.

Attachment 3 - Resolution of the VEGP Seismic PRA Peer Review Findings

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
16-8	SPR-E2	Met	Missing documentation of cutsets review (cfr. QU-D1) (This F&O originated from SR SPR-E2)	There is a description of the most important scenarios but there is no cutset-by-cutset review.	While it is understood that the Draft. B version of the quantification notebook is still somewhat a work in process, it is expected that when the model reaches a more stable state documentation of the review of the cutsets is going to be part of the documentation.	The QU report has been updated to document the review of both dominant cutsets and non-significant cutsets for both CDF and LERF. This finding has been resolved with no significant impact to the SPRA results or conclusions.
16-9	SPR-B1, SPR-B4b	Met, Met	Screening values used for the HEPs that (at the time of the provided documentation) were in the most significant cutsets. (This F&O originated from SR SPR-B1)	At the time when the documentation was provided for peer review, the most significant operator actions (i.e., flood isolation of ACCW HX) were all screening values, which would only meet CCI for HR-G1 (directly called through SPR-B1). In addition, there is little documentation or supporting evidence to justify screening values as low as 3.00E-2	An appropriate resolution of this F&O is pending the current evolution of the model and the importance of operator actions in the SPRA. Given the expectation that operator actions will be needed to mitigate the importance of relay chatter (not yet included in the SPRA logic model) this F&O was provided to ensure care is used in the generation of HEPs if they appear in important cutsets and also to provide more justification for screening values less than 1.00E-1 because a low screening value may indeed skew the actual importance of the newly generated HEP.	The seismic HRA analysis has been revised to be consistent with the EPRI seismic HRA guidance in EPRI 3002008093. The original screening HEPs have been updated using the HRA Calculator, consistent with the approach used in the VEGP internal events PRA. The Documentation has been updated. Operator response to relay chatter has been addressed and evaluated within the same process, and not found to be important. This finding has been resolved.

Attachment 3 - Resolution of the VEGP Seismic PRA Peer Review Findings

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
17-1	SPR-B1	Met	The documentation does not specifically address the applicability of the internal events accident sequences and success criteria to the SPRA model, and does not properly document the accident sequences created specifically for the SPRA model. (This F&O originated from SR SPR-B1)	The modeling approach injected seismic fragilities into fault trees that were modified from the internal events PRA model. It can be inferred from this approach, and it was verified by discussions with the staff, that the internal events sequences and success criteria were considered to be applicable to the SPRA model. This was not specifically stated in the documentation. Further, several additional seismic flooding sequences were added to the fault tree. These sequences are not discussed from an accident sequence and success criteria perspective. Inspection of the fault tree and discussions with the staff indicate that the sequences were appropriately developed with specific success criteria that is different from other internal events sequences. The development of these sequences needs to be included in the documentation. Including event trees for these sequences would also aid in a reader's understanding.	A separate section in the documentation that specifically addresses accident sequences and success criteria is needed to collect the information in one logical place, and is needed to support effective peer reviews and future model updates.	The discussion of accident sequences and success criteria has been expanded, and specific descriptions of the flooding scenarios has been added. This finding is documentation only and does not impact Seismic PRA model results. This finding has been resolved with no significant impact to the SPRA results or conclusions.

Attachment 3 - Resolution of the VEGP Seismic PRA Peer Review Findings

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
17-2	SPR-E2, SPR-F2	Met, Met	The processes used to create the presented quantification results are not fully documented. (This F&O originated from SR SPR-F2)	Examples include: The top cutsets shown in the quantification report are produced by combining the cutsets from all the seismic interval cutsets in a process that is not documented. While the process used to obtain the importance measures is documented, discussions with the PRA staff indicated that importances for some of the basic events were obtained in a different manner (setting to one or zero and requantifying). This is not documented in the notebook.	Expand the documentation to clearly explain the post-processing of the results generated by CAFTA and FRANX. Examples include: - Explain how the cutsets generated by FRANX are combined into g-level-independent cutsets. - Explain the post-processing used to generate importance measures, especially focusing on the deviation from a normal practice that is currently only mentioned in the notebook.	Documentation for QU results has been im-proved to describe the processes used to ag-gregate results over the 14 hazard intervals. The importance calculations have been re-quantified and the method for presentation documented. This finding has been resolved with no significant impact to the SPRA results or conclusions.
17-3	SPR-B3, SPR-E4	I/II, I/II	Subdividing correlation groups based on weaker/stronger components resulted in retention of non-minimal cutsets in some cases, which could impact CDF/LERF results as well as model importance measures. The magnitude and	To account for similar equipment that has different fragilities due to different building locations, certain correlation groups were subdivided to assign a seismic capacity to a weaker component that only failed that component. The higher capacity was then assigned to both components, and was effectively the correlated failure of both components. This can result in the retention of non-minimal cutsets in some cases. For example, for the	The impact of the retention of these non-minimal cutsets on CDF/LERF and importance measures should be assessed and the results documented, or a method to remove the non-minimal cutsets should be devised. Each subdivided correlation group should be investigated for similar effects.	The non-minimal cutsets in the peer reviewed model were identified and reviewed for impact, and determined to be non-significant to risk. The results were very slightly conservative due to these non-minimal cutsets. The issue has been addressed in the updated model, such non-minimal cutsets no longer appear.

Attachment 3 - Resolution of the VEGP Seismic PRA Peer Review Findings

F&O Number	Review Element	CC	F&O Description	F&O Basis	F&O Resolution	Disposition for GSI-191 and Other Applications
			acceptability of these impacts was not documented. (This F&O originated from SR SPR-E4)	Containment Fan Cooler Units there are cutsets in which, due to other failures, only one containment fan cooler needs to seismically fail to cause core damage. Inspection of the cutsets shows that two otherwise identical cutsets are retained: one in which the 1Fan 'group' occurs, and one in which the 4Fans group occurs. The 4Fans cutset is not minimal, and should not be included in the results. Discussions with the staff indicated that these non minimal cutsets were noted during the quantification review process, but were thought to not greatly impact overall results. No formal assessment was done, however, and no record of the informal assessment was included in the documentation.		This finding has been resolved with no significant impact to the SPRA results or conclusions.
17-4	SPR-E6	Met	No quantitative analysis of the relative contribution to LERF from Plant Damage States and Significant LERF contributors was presented in the quantification results. (This F&O originated from SR SPR-E6)	A quantitative analysis is required to meet CCII for LE-F1 & LE-G3, which are directly called from SPR-E6.	Perform the analysis and include the results in the quantification notebook.	The quantitative analysis of significant LERF plant damage states and contributors has been performed. A table and associated discussion of plant damage states and significant contributors has been added to the LERF QU documentation to resolve this finding. This finding has been resolved.

Vogtle Electric Generating Plant
Supplemental Response to NRC Generic Letter 2004-02

Enclosure 1

Introduction and Overall Summary

Attachment 4

SNC Response to NRC Request Regarding Risk-Informed Methodology (Reference 32)

Attachment 4 SNC Response to NRC Request Regarding Risk-Informed Methodology (Reference 1.1.1.1.1(i)32)

NRC Information Request:

Regulatory Guide 1.200, Rev. 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (ADAMS Accession No. ML090410014) describes an approach for determining whether the technical adequacy of a PRA is sufficient to provide confidence in the results; it also describes information the NRC staff expects to be included in risk-informed submittals. As discussed in the guide, a risk informed submittal should contain discussions concerning peer review. If the peer review is not performed against the established standards, then information needs to be included in the submittal demonstrating that the different criteria used are consistent with the established standards. The risk-informed methodology described in the technical report should include the following:

- Description demonstrating that the criteria in ASME/ANS RA-Sb-2013 (Addendum B), which is not an established endorsed standard, are consistent with the ASME/ANS RA-Sa-2009 (Addendum A), which is an established endorsed standard, for the seismic probabilistic risk assessment peer review facts and observations.
- If the different criteria are not consistent with the established endorsed standard, an explanation demonstrating that the analogous Addendum A supporting requirements have been met.

SNC Response:

Regulatory Guide 1.200, Revision 2.0 (Reference 7) endorses ASME/ANS RA-Sa-2009 (Addendum A, Reference 10) but, as noted in an NRC letter to ASME, does not endorse PRA Standard ASME/ANS RA-Sb-2013 (Addendum B, Reference 31). The Vogtle Electric Generating Plant, Units 1 and 2 (Vogtle) Seismic Probabilistic Risk Assessment (SPRA) peer review was performed using the SPRA requirements in Addendum B.

Because the peer review was not performed against the NRC endorsed standard, information demonstrating that any different criteria used are consistent with the NRC endorsed standard is provided here. The following discussion addresses the differences relative to establishing the technical capability of the Vogtle SPRA.

The ASME/ANS PRA Standard Part 5 requirements for at-power SPRA (in all versions of the PRA Standard) are organized into the following three major technical elements:

- Seismic hazard analysis (technical element SHA)
- Seismic fragility analysis (technical element SFR), and
- Seismic plant-response modeling (technical element SPR)

Each technical element contains several high level requirements (HLR) and associated supporting requirements (SR). The following three tables (Table 1 for SHA, Table 2 for SFR and Table 3 for SPR) provide SR by SR comparisons between Addendum A and

Addendum B of the ASME/ANS PRA Standard. Differences between SR wordings are denoted in the following tables using bold type. Changes in the HLRs between Addendum A and Addendum B are denoted using strikethrough text. The comparison assessment focuses on capability category II (CCII); as such, SR text not related to CCII is not included in the following tables. This comparison assessment is performed by SR and not by HLR; peer reviewers assess technical quality by reviewing SRs (HLRs are not assigned technical capabilities). The following assessment conclusion categories are used in the SR comparison assessment:

- **Addendum B Assessment Equates to Addendum A**: This conclusion category is used when the SR wording of Addendum A and Addendum B are the same or effectively the same (i.e., re-wording that is for clarity and consistency and not intended to change the intent). This conclusion category is also used in cases where the wording differences are related to moving the requirements to another SR. In certain SRs the qualifier for a specific capability category range is assigned to indicate the focus on capability category II when there are differences in other capability categories. A peer reviewer using Addendum A or Addendum B would come to the same technical capability assessment on these topics.
- **Addendum B Envelopes Addendum A**: This conclusion category is used when the SR wording of Addendum A and Addendum B are different and the requirements of Addendum B envelope that of Addendum A (i.e., if the analysis meets Addendum B CCII then by definition it meets Addendum A CCII or higher). A peer reviewer using Addendum B would assess a technical capability on these topics that would be equal to or greater than what would be determined if Addendum A were used in the review.
- **Vogtle Conforms to Addendum A**: This conclusion category is used when the SR wording of Addendum A and Addendum B are different, or some of the requirements of Addendum A no longer exist in Addendum B, yet the analysis performed and documented in the Vogtle analysis conforms to the requirements of Addendum A as well as Addendum B.
- **Vogtle Conforms to Accepted Current Practices**: This conclusion category is used when the SR wording of Addendum A and Addendum B are different; or some of the requirements of Addendum A no longer exist in Addendum B because they were removed from the Standard as confusing or judged too specific “how-to”. For the one SR where this conclusion category applies, Vogtle does not follow some of the specifics of the Addendum A SR requirements but Vogtle does conform to accepted current practices.

While the Vogtle SPRA was peer reviewed against the ASME/ANS RA-Sb-2013 (Addendum B) requirements, the tables that follow demonstrate that most of the supporting requirements in Addendum B are consistent with the supporting requirements in ASME/ANS RA-Sa-2009 (Addendum A). For the few supporting requirements where differences are noted in the tables, the Vogtle SPRA model and documentation meet the analogous Addendum A supporting requirements. Therefore, the Vogtle SPRA meets the technical adequacy requirements of Regulatory Guide 1.200, Revision 2 and is of sufficient quality and level of detail to support the risk informed approach for GSI-191.

Table 1: Comparison of Supporting Requirements of Addendum A and Addendum B for SHA

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
HLR-SHA-A: The frequency of earthquakes at the site shall be based on a site-specific probabilistic seismic hazard analysis (existing or new) that reflects the composite distribution of the informed technical community. The level of analysis shall be determined based on the intended application and on site-specific complexity.					
SHA-A1	ASME/ANS RA-Sa–2009	<not printed here; not focus of this assessment>	In performing the probabilistic seismic hazard analysis (PSHA), BASE it on, and MAKE it consist of, the collection and evaluation of available information and data, evaluation of the uncertainties in each element of the PSHA, and a defined process and documentation to make the PSHA traceable.		**Addendum B Assessment Equates to Addendum A** The wording for this SR is the same for Addendum A and Addendum B.
	ASME/ANS RA-Sb–2013	Identical to Addendum A	Identical to Addendum A		
SHA-A2	ASME/ANS RA-Sa–2009	As the parameter to characterize both hazard and fragilities, USE the spectral accelerations, or the average spectral acceleration over a selected band of frequencies, or peak ground acceleration.		<not printed here; not focus of this assessment>	**Addendum B CC1-III Assessment Equates to Addendum A CCI-II** In response to EPRI 2011 comment, Addendum B removed the CCIII requirements. The Addendum B SR equates to Addendum A CCI-II.
	ASME/ANS RA-Sb–2013	Identical to Addendum A for CC I/II			
SHA-A3	ASME/ANS RA-Sa–2009	**In the selection of frequencies to determine spectral accelerations or average spectral acceleration, CAPTURE** the frequencies of those structures, systems, or components, **or a combination thereof** that are significant in the PRA results and insights.			**Addendum B Assessment Equates to Addendum A** Addendum B changed action verbs to be consistent with accepted verb usage across SRs, and also made minor changes in wordings. This wording edit does not change the capability category requirements of this SR.
	ASME/ANS RA-Sb–2013	**If spectral acceleration or average spectral acceleration over a band of frequencies is used, INCLUDE** the response frequencies of SSCs that are significant in the PRA results and insights.			

Table 1: Comparison of Supporting Requirements of Addendum A and Addendum B for SHA

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
SHA-A4	ASME/ANS RA-Sa–2009	In developing the probabilistic seismic hazard analysis results, whether they are characterized by spectral accelerations, peak ground accelerations, or both, EXTEND them to large-enough values (consistent with the physical data and interpretations) so that the truncation does not produce unstable final numerical results, such as core damage frequency, and the delineation and ranking of seismic-initiated sequences are not affected.			**Addendum B Assessment Equates to Addendum A** Addendum B added the phrase "for use in accident sequence quantification". This wording edit does not change the capability category requirements of this SR.
	ASME/ANS RA-Sb–2013	In developing the probabilistic seismic hazard analysis results **for use in accident sequence quantification,** whether they are characterized by spectral accelerations, peak ground accelerations, or both, EXTEND them to large-enough values (consistent with the physical data and interpretations) so that the truncation does not produce unstable final numerical results, such as core damage frequency, and the delineation and ranking of seismic-initiated sequences are not affected.			
SHA-A5	ASME/ANS RA-Sa–2009	SPECIFY a lower-bound magnitude (or probabilistically defined characterization of magnitudes based on a damage parameter) for use in the hazard analysis, such that earthquakes of magnitude less than this value are not expected to cause significant damage to the engineered structures or equipment.			**Addendum B Assessment Equates to Addendum A** The wording for this SR is the same for Addendum A and Addendum B.
	ASME/ANS RA-Sb–2013	Identical to Addendum A			
HLR-SHA-B: To provide inputs to the probabilistic seismic hazard analysis, a comprehensive up-to-date database including geological, seismological, and geophysical data; local site topography; and surficial geologic and geotechnical site properties shall be compiled. A catalog of historical, instrumental, and paleoseismicity information shall also be compiled.					
SHA-B1	ASME/ANS RA-Sa–2009	In performing the probabilistic seismic hazard analysis (PSHA), **BASE it on** available or developed geological, seismological, geophysical, and geotechnical data that reflect the current state of the knowledge and that are used by experts/analysts to develop		<not printed here; not focus of this assessment>	**Addendum B Assessment Equates to Addendum A** Addendum B changed action verbs to be consistent with accepted verb usage across SRs. This wording edit does not change the

Table 1: Comparison of Supporting Requirements of Addendum A and Addendum B for SHA

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
		interpretations and inputs to the PSHA.			capability category requirements of this SR.
	ASME/ANSRA-Sb–2013	In performing the probabilistic seismic hazard analysis (PSHA), **USE** available or developed geological, seismological, geophysical, and geotechnical data that reflect the current state of the knowledge and that are used by experts/analysts to develop interpretations and inputs to the PSHA.		<not printed here; not focus of this assessment>	
SHA-B2	ASME/ANSRA-Sa–2009	ENSURE that the database and information used are adequate to characterize all credible seismic sources that may contribute significantly to the frequency of occurrence of vibratory ground motion at the site, considering regional attenuation of ground motions and local site effects. If the existing probabilistic seismic hazard analysis (PSHA) studies are to be used in the seismic PRA, ENSURE that any new data or interpretations that could affect the PSHA are adequately incorporated in the existing data and analysis.		<not printed here; not focus of this assessment>	**Addendum B Assessment Equates to Addendum A** The wording for this SR is the same for Addendum A and Addendum B for CCI and CCII.
	ASME/ANSRA-Sb–2013	Identical to Addendum A		<not printed here; not focus of this assessment>	
SHA-B3	ASME/ANSRA-Sa–2009	<not printed here; not focus of this assessment>	As a part of data collection, COMPILE a catalog of historically reported, geologically identified, and instrumentally recorded earthquakes. USE reference [5-30]		**Vogtle Conforms to Addendum A** A catalog of historically reported, geologically identified, and instrumentally recorded

Table 1: Comparison of Supporting Requirements of Addendum A and Addendum B for SHA

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
			requirements or equivalent.		earthquakes for the entire CEUS was compiled by the 2012 CEUS SSC report. Following a SSHAC Level 3 process, the CEUS SSC report is a robust evaluation of available information on historical seismicity, paleoseismic data on large-magnitude recurrence rates, and state-of-the-knowledge of earthquake seismic sources as considered in the informed technical community. The 2012 CEUS SSC catalog followed a SSHAC Level 3 process and is applicable for risk informed applications. Compiling a new catalog will not be as rigorous as the SSHAC Level 3 process. The Addenda B SR requirement is appropriate for CC-II. The 2012 CEUS SSC report used an earthquake catalog which extended through 2008. Recent earthquake activity in the vicinity of the Vogtle site was assessed for its impact on hazard. The study was based on a temporal update of the earthquake catalog from 2009 through February 2016. The assessment concluded that the 2012 CEUS SSC report seismicity parameters are appropriate for evaluation of seismic hazard at Vogtle. Based on this, the Vogtle PSHA that was performed conforms to Addendum A.
	ASME/ANSRA-Sb–2013	INCLUDE an appropriate existing catalog of historically reported earthquakes, instrumentally recorded earthquakes, and earthquakes reported through geological investigations. USE reference [5-30] requirements or equivalent.		<not printed here; not focus of this assessment>	
HLR-SHA-C: To account for the frequency of occurrence of seismic ground motions in the site region, the probabilistic seismic hazard analysis shall examine all credible sources of potentially damaging earthquakes.					

Table 1: Comparison of Supporting Requirements of Addendum A and Addendum B for SHA

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
SHA-C1	ASME/ANS RA-Sa–2009	In the probabilistic seismic hazard analysis, **EXAMINE all potential** sources of earthquakes **that affect the probabilistic hazard at the site. BASE the identification and characterization of seismic sources on** regional and site geological and geophysical data, historical and instrumental seismicity data, the regional stress field, and geological evidence of prehistoric earthquakes.			**Addendum B Assessment Envelopes Addendum A** Addendum B added additional clarifications and requirements into the text of this SR, and also Addendum B changed action verbs to be consistent with accepted verb usage across SRs. The Addendum B SR wording envelopes that of Addendum A.
	ASME/ANS RA-Sb–2013	In the probabilistic seismic hazard analysis, **EVALUATE sources** of earthquakes that **have the potential to contribute significantly to the probabilistic hazard at the site. IDENTIFY and CHARACTERIZE** seismic **sources taking into account previous compilations of seismic sources, based on** regional and site geological and geophysical data, historical and instrumental seismicity data, and geological evidence of prehistoric earthquakes.			
SHA-C2	ASME/ANS RA-Sa–2009	ENSURE that any expert elicitation process used to characterize the seismic sources is compatible with the level of analysis discussed in Requirement HA-A, and **FOLLOW** a structured approach.			**Addendum B Assessment Equates to Addendum A** Addendum B changed action verbs to be consistent with accepted verb usage across SRs. This wording edit does not change the capability category requirements of this SR.
	ASME/ANS RA-Sb–2013	ENSURE that any expert elicitation process used to characterize the seismic sources is compatible with the level of analysis discussed in Requirement HLR-SHA-A, and **USE** a structured approach.			
SHA-C3	ASME/ANS RA-Sa–2009	<not printed here; not focus of this assessment>	The seismic sources are characterized by **source location** and geometry, maximum earthquake magnitude, and earthquake recurrence. INCLUDE the aleatory and epistemic uncertainties explicitly in these characterizations.		**Vogtle Conforms to Addendum A** Addenda B added additional clarification into the text of this SR, and also added a clause "where significant" at the end. The Addenda B SR requirement is appropriate for CC-II. Under the SSHAC Level 3 process the aleatory and epistemic uncertainties in

Table 1: Comparison of Supporting Requirements of Addendum A and Addendum B for SHA

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
	ASME/ANSRA-Sb–2013	<not printed here; not focus of this assessment>	The seismic sources are characterized by **alternative source representation** and source geometry, maximum earthquake magnitude, and earthquake recurrence. INCLUDE the aleatory and epistemic uncertainties explicitly in these characterizations, **where significant.**		seismic sources are characterized for source location and geometry, magnitude, and activity rate. Logic trees to account for the epistemic uncertainty were developed as part of the SSHAC Level 3 methodology implemented in the CEUS SSC report. The aleatory uncertainty was also accounted for in the PSHA framework of the Vogtle PSHA. For seismic sources representing repeated large magnitude earthquakes (RLMEs), uncertainties in location and geometry, magnitude model, activity rate, and maximum magnitude were explicitly included in the characterization. For background sources, uncertainty in geometry was represented with alternative sets of area sources, uncertainties in recurrence rates were represented with alternative rates, and uncertainties in maximum magnitude were represented with distributions of values. These uncertainties were documented in the 2012 CEUS SSC report and were included in the Vogtle PSHA. Based on this, the Vogtle PSHA that was performed conforms to Addendum A.
SHA-C4	ASME/ANSRA-Sa–2009	If an existing probabilistic seismic hazard analysis study is used, **SHOW** that any seismic sources that **were previously unknown** or uncharacterized are not significant, or INCLUDE them in a **revision** of the hazard estimates.			**Addendum B Assessment Equates to Addendum A** Addendum B added additional clarifications and modified the sentence structure to highlight the seismic source model aspect of the PSHA. In addition, Addendum B changed action verbs to be consistent with accepted
	ASME/ANSRA-Sb–2013	If an existing seismic source model is used, **DEMONSTRATE** that any new seismic sources that **have been identified** or were uncharacterized **when the existing models were developed** are not			

Table 1: Comparison of Supporting Requirements of Addendum A and Addendum B for SHA

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
		significant, or INCLUDE them in the **update** of the hazard estimates.			verb usage across SRs. This wording edit does not change the capability category requirements of this SR.
HLR-SHA-D: The probabilistic seismic hazard analysis shall examine mechanisms influencing vibratory ground motion that can occur at a site given the occurrence of an earthquake of a certain type (e.g., strike slip, normal, reverse) and magnitude, and at a certain location. Uncertainties shall be addressed in characterizing the ground motion propagation.					
SHA-D1	ASME/ANS RA-Sa–2009	**ACCOUNT in the probabilistic seismic hazard analysis for** (a) credible mechanisms governing estimates of vibratory ground motion that can occur at a site (b) **regional and site-specific geological, geophysical, and geotechnical data and** historical and instrumental seismicity data (including strong motion data) (c) current attenuation models in the ground motion estimates			**Addendum B Assessment Equates to Addendum A** Addendum B changed action verbs to be consistent with accepted verb usage across SRs. In addition, Addendum B modified item (b) by removing requirements that were redundant with SHA-B1. This wording edit does not change the capability category requirements of this SR.
	ASME/ANS RA-Sb–2013	**In the vibratory ground motion analysis, INCLUDE** (a) credible mechanisms governing estimates of vibratory ground motion that can occur at a site (b) **available** historical and instrumental seismicity data (including strong motion data) (c) current attenuation models for the ground motion estimates			
SHA-D2	ASME/ANS RA-Sa–2009	ENSURE that any expert elicitation process used to characterize the ground motion is compatible with the level of analysis discussed in Requirement SHA-A, and **FOLLOW** a structured approach.			**Addendum B Assessment Envelopes Addendum A** Addendum B added additional requirements into the text of this SR, and also Addendum B changed action verbs to be consistent with accepted verb usage across SRs. The Addendum B SR wording envelopes that of Addendum A.
	ASME/ANS RA-Sb–2013	ENSURE that any expert elicitation process used to characterize the ground motion **or any other elements of the ground motion analysis** is compatible with the level of analysis discussed in Requirement HLR-SHA-A, and **USE** a structured approach.			
SHA-D3	ASME/ANS RA-Sa–2009	<not printed here; not focus of this assessment>	**ADDRESS** both the aleatory and epistemic uncertainties in the ground motion characterization in accordance with the level of analysis identified for Requirement SHA-A.		**Addendum B Assessment Envelopes Addendum A** Addendum B changed action verbs to be

Table 1: Comparison of Supporting Requirements of Addendum A and Addendum B for SHA

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
	ASME/ANS RA-Sb–2013	<not printed here; not focus of this assessment>	**INCLUDE** both the aleatory and epistemic uncertainties **separately** in the ground motion characterization in accordance with the level of analysis identified for Requirement HLR-SHA-A.		consistent with accepted verb usage across SRs, and also Addendum B added additional requirement on handling aleatory and epistemic uncertainties separately. The Addendum B SR wording envelopes that of Addendum A.
SHA-D4	ASME/ANS RA-Sa–2009	If an existing **probabilistic seismic hazard analysis study** is used, **SHOW** that **any ground motion models or** new information that were previously unused or unknown are not significant, or INCLUDE them in a **revision** of the hazard estimates.			**Addendum B Assessment Equates to Addendum A** Addendum B added additional clarifications and modified the sentence structure to highlight the ground motion model aspect of the PSHA. In addition, Addendum B changed action verbs to be consistent with accepted verb usage across SRs. This wording edit does not change the capability category requirements of this SR.
	ASME/ANS RA-Sb–2013	If existing **ground motion models** are used, **DEMONSTRATE** that new information not previously used or which was unknown when the existing models were developed would not significantly affect the PSHA results, or INCLUDE it in the **update** of the hazard estimates.			
HLR-SHA-E: The probabilistic seismic hazard analysis shall account for the effects of local site response.					
SHA-E1	ASME/ANS RA-Sa–2009	<not printed here; not focus of this assessment>	**ACCOUNT in the probabilistic seismic hazard analysis for** the effects of site topography, surficial geologic deposits, and site geotechnical properties on ground motions at the site.		**Addendum B Assessment Equates to Addendum A** Addendum B changed action verbs to be consistent with accepted verb usage across SRs. This wording edit does not change the capability category requirements of this SR.
	ASME/ANS RA-Sb–2013	<not printed here; not focus of this assessment>	**In the probabilistic seismic hazard analysis, INCLUDE** the effects of site topography, surficial geologic deposits, and site geotechnical properties on ground motions at the site.		
SHA-E2	ASME/ANS RA-Sa–2009	<not printed here; not focus of this assessment>	**ADDRESS** both the aleatory and epistemic uncertainties in the local site response analysis.		**Addendum B Assessment Equates to Addendum A**

Table 1: Comparison of Supporting Requirements of Addendum A and Addendum B for SHA

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
	ASME/ANS RA-Sb–2013	<not printed here; not focus of this assessment>	**INCLUDE** both the aleatory and epistemic uncertainties in the local site response analysis.		Addendum B changed action verbs to be consistent with accepted verb usage across SRs. This wording edit does not change the capability category requirements of this SR.
HLR-SHA-F: Uncertainties in each step of the hazard analysis shall be propagated and displayed in the final quantification of hazard estimates for the site. *Addendum A only:* The results shall include fractile hazard curves, median and mean hazard curves, and uniform hazard response spectra. For certain applications, the probabilistic seismic hazard analysis shall include seismic source deaggregation and magnitude-distance deaggregation					
SHA-F1	ASME/ANS RA-Sa–2009	<not printed here; not focus of this assessment>	In the final quantification of the seismic hazard, INCLUDE **and DISPLAY the propagation of both aleatory and epistemic uncertainties.**		**Addendum B Assessment Equates to Addendum A** In response to NRC 2012 comment on Addendum B ballot that said the wording of this SR was vague, Addendum B revised the wording to be clear that CCI is Mean only and that CCII-III is a family of hazard curves. This wording edit does not change the capability category requirements of this SR.
	ASME/ANS RA-Sb–2013	<not printed here; not focus of this assessment>	In the final quantification of the seismic hazard, INCLUDE **uncertainties through a family of hazard curves.**		
SHA-F2	ASME/ANS RA-Sa–2009	In the probabilistic seismic hazard analysis, INCLUDE appropriate sensitivity studies and inter mediate results to identify factors that are important to the site hazard and that make the analysis traceable.			**Addendum B Assessment Equates to Addendum A** The wording for this SR is the same for Addendum A and Addendum B.
	ASME/ANS RA-Sb–2013	Identical to Addendum A			

Table 1: Comparison of Supporting Requirements of Addendum A and Addendum B for SHA

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
SHA-F3	ASME/ANSRA-Sa–2009	<not printed here; not focus of this assessment>	**DEVELOP** the following results as a part of the quantification process, compatible with needs for the level of analysis determined in (HLR-SHA-A): (a) fractile and mean hazard curves for each ground motion parameter considered in the probabilistic seismic hazard analysis (b) fractile and mean uniform hazard response spectrum	<not printed here; not focus of this assessment>	**Addendum B Assessment Equates to Addendum A** Addendum B changed action verbs to be consistent with accepted verb usage across SRs, and also Addendum B added additional clarification on the phrase "quantification process" by modifying it to "hazard quantification process". This wording edit does not change the capability category requirements of this SR.
	ASME/ANSRA-Sb–2013	<not printed here; not focus of this assessment>	**CALCULATE** the following results as a part of the **hazard** quantification process, compatible with needs for the level of analysis determined in Requirement HLR-SHA-A: …	<not printed here; not focus of this assessment>	
HLR-SHA-G: Addendum A: For further use in the seismic PRA, the spectral shape shall be based on a site-specific evaluation taking into account the					

Table 1: Comparison of Supporting Requirements of Addendum A and Addendum B for SHA

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
contributions of deaggregated magnitude-distance results of the probabilistic seismic hazard analysis. Broad-band, smooth spectral shapes, such as those presented in NUREG/CR-0098 [5-5] (for lower seismicity sites such as most of those east of the U.S. Rocky Mountains) are also acceptable if they are shown to be appropriate for the site. The use of existing uniform hazard response spectra (UHSs) is acceptable unless evidence comes to light that would challenge these uniform hazard spectral shapes. Reg Guide 1.200 Rev2 Clarification: For further use in the seismic PRA, the spectral shape shall be based on a site-specific evaluation taking into account the contributions of deaggregated magnitude-distance results of the probabilistic seismic hazard analysis. ~~Broad-band, smooth spectral shapes, ... that would challenge these uniform hazard spectral shapes.~~ Addendum B: For further use in the seismic PRA, the spectral shape shall be based on a site-specific evaluation taking into account results of the probabilistic seismic hazard analysis.					
SHA-G1	ASME/ANSRA-Sa–2009	<not printed here; not focus of this assessment>	**BASE** the **response** spectral shape used in the seismic PRA **on site-specific evaluations performed for the probabilistic seismic hazard analysis. REFLECT or BOUND the site-specific considerations.**	<not printed here; not focus of this assessment>	**Addendum B Assessment Equates to Addendum A** Addendum B deleted the last sentence of this SR to make the language more concise and remove redundancy. In addition, Addendum B changed the action verb to be consistent with accepted verb usage across SRs. This wording edit does not change the capability category requirements of this SR.
	ASME/ANSRA-Sb–2013	<not printed here; not focus of this assessment>	**ENSURE that** the spectral shape used in the seismic PRA **uses site-specific evaluations performed for the PSHA.**	<not printed here; not focus of this assessment>	
HLR-SHA-H: When use is made of an existing study for probabilistic seismic hazard analysis purposes, it shall be confirmed that the basic data and interpretations are still valid in light of established current information [*Addendum A only:* the study meets the requirements outlined in A through G above, and the study is suitable for the intended application.] Reg Guide 1.200 Rev 2 Clarification: ~~When use ... for the intended application.~~ **It shall be confirmed that basic data and interpretations**					

Table 1: Comparison of Supporting Requirements of Addendum A and Addendum B for SHA					
SR	**Standard Rev.**	**Capability Category I**	**Capability Category II**	**Capability Category III**	**Basis for Assessment (CC-II Focus)**
from an existing study are valid.					
SHA-H1	ASME/ANS RA-Sa–2009	<not printed here; not focus of this assessment>	**Use of existing studies allowed.**	<not printed here; not focus of this assessment>	**Addendum B Assessment Envelopes Addendum A** Addendum B added additional clarifications and requirements into the text of this SR. The Addendum B SR wording envelopes that of Addendum A.
	ASME/ANS RA-Sb–2013	**CONFIRM that the basic data and interpretations for any existing studies used remain valid in light of established current information, consistent with the Requirements HLR-SHA-A through HLR-SHA-G, and DESCRIBE the bases and methodology used.**			
HLR-SHA-I: A screening analysis shall be performed to assess whether, in addition to the vibratory ground motion, other seismic hazards, such as fault displacement, landslide, soil liquefaction, or soil settlement, need to be included in the seismic PRA [*Addendum A only:* for the specific application. If so, the seismic PRA shall address the effect of these hazards through assessment of the frequency of hazard occurrence or the magnitude of hazard consequences, or both.] Reg Guide 1.200 Rev 2 Clarification: ~~A screening analysis ... or the magnitude of hazard consequences, or both.~~ **The hazard analysis shall include hazards other than vibratory ground motion if necessary.**					
SHA-I1	ASME/ANS RA-Sa–2009	**(There are no supporting requirements here.)**			**Addendum B Assessment Envelopes Addendum A** There are no supporting requirements for HLR-SHA-I in Addendum A. Since the HLRs only say "do something" and the SRs establish what needs to be done, the addition of SRs in Addendum B establishes new requirements beyond Addendum A.
	ASME/ANS RA-Sb–2013	**DOCUMENT the bases and methodology used for any screening out of the seismic hazards other than vibratory ground motion.**			
SHA-I2	ASME/ANS RA-Sa–2009	**no SHA-I2 in Addendum A**			**Addendum B Assessment Envelopes Addendum A** There are no supporting requirements for HLR-SHA-I in Addendum A. Since the HLRs only say "do something" and the SRs establish what needs to be done, the addition of SRs in Addendum B establishes new
	ASME/ANS RA-Sb–2013	**For those hazards not screened out, INCLUDE their effect through assessment of the frequency of hazard occurrence and the magnitude of hazard consequences.**			

Table 1: Comparison of Supporting Requirements of Addendum A and Addendum B for SHA					
SR	**Standard Rev.**	**Capability Category I**	**Capability Category II**	**Capability Category III**	**Basis for Assessment (CC-II Focus)**
					requirements beyond Addendum A.
HLR-SHA-J: Documentation of the probabilistic seismic hazard analysis shall be consistent with the applicable supporting requirements					
SHA-J1	ASME/ANS RA-Sa–2009	DOCUMENT the probabilistic seismic hazard analysis in a manner that facilitates PRA applications, upgrades, and peer review.			**Addendum B Assessment Equates to Addendum A** The wording for this SR is the same for Addendum A and Addendum B.
	ASME/ANS RA-Sb–2013	Identical to Addendum A			
SHA-J2	ASME/ANS RA-Sa–2009	DOCUMENT the process used in the probabilistic seismic hazard analysis. For example, this documentation is typically consistent with reference [5-28] and includes a description of: (a) the specific methods used for source characterization and ground motion characterization, (b) the scientific interpretations that are the basis for the inputs and results, and (c) if an existing PSHA is used, documentation to ensure that it is adequate to meet the spirit of the requirements herein.			**Addendum B Assessment Equates to Addendum A** The wording for this SR is the same for Addendum A and Addendum B.
	ASME/ANS RA-Sb–2013	Identical to Addendum A			
SHA-J3	ASME/ANS RA-Sa–2009	DOCUMENT the sources of model uncertainty and related assumptions associated with the probabilistic seismic hazard analysis.			**Addendum B Assessment Equates to Addendum A** The wording for this SR is the same for Addendum A and Addendum B.
	ASME/ANS RA-Sb–2013	Identical to Addendum A			

Table 2: Comparison of Supporting Requirements of Addendum A and Addendum B for SFR					
SR	**Standard Rev.**	**Capability Category I**	**Capability Category II**	**Capability Category III**	**Basis for Assessment (CC-II Focus)**
HLR-SFR-A: Addendum A: The seismic-fragility evaluation shall be performed to estimate seismic fragilities of SSCs whose failure may contribute to core damage or large early release, or both. Addendum B: The seismic-fragility evaluation shall be performed to estimate plant-specific, realistic seismic fragilities of structures, or systems, or components, or a combination thereof whose failure may contribute to core damage or large early release, or both.					
SFR-A1	ASME/ANS RA-Sa–2009	**DEVELOP** seismic fragilities for all those structures, systems, or components, **or a combination thereof** identified by the systems analysis (see Requirement SPR-D1).			**Addendum B Assessment Equates to Addendum A** The changes from Addendum A to Addendum B for this SR included the replacement of "DEVELOP" with "CALCULATE." The change implements a more precise action verb. These wording edits do not change capability category requirements of this SR. The elimination of the phrase "a combination thereof" does not change the requirements, since it is redundant to other SRs requiring consideration of seismic correlation.
	ASME/ANS RA-Sb–2013	**CALCULATE** seismic fragilities for SSCs identified by the systems analysis (see Requirement SPR-D1).			
SFR-A2	ASME/ANS RA-Sa–2009	<not printed here; not focus of this assessment>	**BASE** the seismic fragilities on plant-specific data, and ENSURE that they are realistic (median with uncertainties). Generic data (e.g., fragility test data, generic seismic qualification test data, and earthquake experience data)	<not printed here; not focus of this assessment>	**Addendum B Assessment Equates to Addendum A** The changes from Addendum A to Addendum B involve the replacement of "BASE" with the more precise action verb "CALCULATE" and the replacement of "conservative" with "applicable" in the last sentence with respect to the use of generic fragility data. The latter change precludes confusion due to a contradiction, as the requirement states first that realistic fragilities

Table 2: Comparison of Supporting Requirements of Addendum A and Addendum B for SFR					
SR	**Standard Rev.**	**Capability Category I**	**Capability Category II**	**Capability Category III**	**Basis for Assessment (CC-II Focus)**
			MAY be used for screening of certain structures, systems, or components, or a combination thereof and for calculating their seismic fragilities by applying the requirements under (HLR-SFR-F), which permits use of such generic data under specified conditions. However, DEMONSTRATE that any use of such generic data is **conservative.**		are required but then the use of generic data is to be conservative. In the context of the requirement, the use of "applicable" is correct and does not change the capability category requirements of this SR.
	ASME/ANSRA-Sb–2013	<not printed here; not focus of this assessment>	**CALCULATE** the seismic fragilities based on plant-specific data, and ENSURE that they are realistic (median with uncertainties). Generic data (e.g., fragility test data, generic seismic qualification test data, and earthquake experience data) may be used for screening	<not printed here; not focus of this assessment>	

Table 2: Comparison of Supporting Requirements of Addendum A and Addendum B for SFR					
SR	**Standard Rev.**	**Capability Category I**	**Capability Category II**	**Capability Category III**	**Basis for Assessment (CC-II Focus)**
			of certain SSCs and for calculating their seismic fragilities by applying the Requirement HLR-SFR-F, which permits use of such generic data under specified conditions. However, DEMONSTRATE that any use of such generic data is **applicable.**		
HLR-SFR-B: If screening of high-seismic-capacity components is performed, the basis for the screening shall be fully described.					
SFR-B1	ASME/ANSRA-Sa–2009	If screening of high-seismic-capacity components is performed, DESCRIBE **fully** the basis for screening and supporting documents. For example, it is acceptable to apply guidance given in EPRI NP-6041-SL, Rev. 1, and NUREG/CR-4334 to screen out components with high seismic capacity. **However, CHOOSE** the screening level high enough that the contribution to core damage frequency and large early release frequency from the screened-out components is not significant.	<not printed here; not focus of this assessment>	**Addendum B Assessment Equates to Addendum A** Addendum B removes the sentence citing NP-6041-SL and NUREG/CR-4334 as examples of screening bases and replaces the action verb "CHOOSE" with the more precise verb "SELECT." These wording edits are non-substantive and do not change capability category requirements of this SR.

Table 2: Comparison of Supporting Requirements of Addendum A and Addendum B for SFR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
	ASME/ANSRA-Sb–2013	If screening of high-seismic-capacity components is performed, DESCRIBE the basis for screening and the supporting documents **and SELECT** the screening level high enough that the contribution to core damage frequency and large early release frequency from the screened-out components is not significant.		<not printed here; not focus of this assessment>	
SFR-B2	ASME/ANSRA-Sa–2009	**ASSESS and DOCUMENT the applicability of the screening criteria given in EPRI NP-6041-SL, Rev. 1 [5-3] and NUREG/CR-4334 [5-4] for the specific plant and specific equipment.**			**Addendum B Assessment Equates to Addendum A** Addendum B deleted this SR. The clarifying non-mandatory footnote for SFR-B1, which remains unchanged from Addendum A to Addendum B, reiterates that NP-6041-SL and NUREG/CR-4334 "may be used" and are not mandatory. SFR-B2 in Addendum A states the applicability of NP-6041-SL and NUREG/CR-4334 shall be assessed and documented. However, if NP-6041-SL and NUREG/CR-4334 are applied for screening, SFR-B1 in both Addendum A and Addendum B state that the basis for use shall be described and documentation is addressed under SFR-G2. Therefore, SFR-B2 is redundant and its removal does not change the requirements of the standard.
	ASME/ANSRA-Sb–2013	**No SFR-B2 in Addendum B**			
HLR-SFR-C: The seismic-fragility evaluation shall be based on seismic response that the SSCs experience at their failure levels					

Table 2: Comparison of Supporting Requirements of Addendum A and Addendum B for SFR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
SFR-C1	ASME/ANS RA-Sa–2009	ESTIMATE the seismic responses that the components experience at their failure levels **on a realistic basis** using site-specific earthquake response spectra in three orthogonal directions, anchored to a ground motion parameter such as peak ground acceleration or average spectral acceleration over a given frequency band. ENSURE that the spectral shape used **reflects or** bounds the site-specific conditions.		<not printed here; not focus of this assessment>	**Addendum B Assessment Envelopes Addendum A** The two changes from Addendum A to Addendum B involve the deletion of "site-specific" prior to "earthquake response spectra" in the first sentence and deletion of "reflects" in the last sentence. Removal of "site-specific" prior to earthquake response spectra is inconsequential as bounding by site-specific conditions by definition renders the earthquake response spectra site-specific. Removal of "reflects" renders Addendum B as more precise since the properties must "bound" site-specific conditions and not merely "reflect."
	ASME/ANS RA-Sb–2013	ESTIMATE the seismic responses that the components experience at their failure levels using input earthquake response spectra in three orthogonal directions, anchored to a ground motion parameter such as peak ground acceleration or average spectral acceleration over a given frequency band, and ENSURE that the spectral shape used bounds the site-specific conditions.		<not printed here; not focus of this assessment>	
SFR-C2	ASME/ANS RA-Sa–2009	If probabilistic response analysis is performed to obtain **realistic** structural loads and floor response spectra, ENSURE that the number of simulations done (e.g., Monte Carlo simulation and Latin Hypercube Sampling) is large enough to obtain stable median and 85% nonexceedance responses. **ACCOUNT for** the entire spectrum of input ground motion levels displayed in the seismic hazard curves.		<not printed here; not focus of this assessment>	**Addendum B Assessment Envelopes Addendum A** The changes from Addendum A to Addendum B are to implement the approved action verb "ACCOUNT" to the more precise verb "INCLUDE" and to eliminate "realistic" in the first sentence prior to "structural loads and floor response spectra." The qualifier

Table 2: Comparison of Supporting Requirements of Addendum A and Addendum B for SFR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
	ASME/ANS RA-Sb–2013	If probabilistic response analysis is performed to obtain structural loads and floor response spectra, ENSURE that the number of simulations done (e.g., Monte Carlo simulation and Latin Hypercube Sampling) is large enough to obtain stable median and 85% nonexceedance responses. **INCLUDE** the entire spectrum of input ground motion levels displayed in the seismic hazard curves.		<not printed here; not focus of this assessment>	realistic is removed to avoid confusion as it is unnecessary in the context of applying probabilistic response analysis to determine structural loads and response spectra.
SFR-C3	ASME/ANS RA-Sa–2009	If scaling of existing design response analysis is used, JUSTIFY it based on the adequacy of structural models, foundation characteristics, and similarity of input ground motion.		<not printed here; not focus of this assessment>	**Vogtle Conforms to Addendum A** The change from Addendum A to Addendum B involved the deletion of the word "design" from "existing design response analysis." However, Plant Vogtle did not perform scaling of any existing response analysis, and therefore the change is irrelevant and Vogtle conforms to Addendum A.
	ASME/ANS RA-Sb–2013	If scaling of existing response analysis is used, JUSTIFY it based on the adequacy of structural models, foundation characteristics, and similarity of input ground motion.		<not printed here; not focus of this assessment>	
SFR-C4	ASME/ANS RA-Sa–2009	When the design response analysis models are judged not to be realistic and state of the art, or when the design input ground motion is significantly different from the site-specific input motion, PERFORM new analysis to obtain realistic structural loads and floor response spectra.		<not printed here; not focus of this assessment>	**Addendum B Assessment Equates to Addendum A** Similar to SFR-C3, Addendum B in SFR-C4 eliminates "design" from "existing design response analysis" in the first sentence and adds the "for use in the seismic PRA" to the last sentence. The judgement of existing response analysis models as unrealistic and not state of the art as stated in Addendum B would include the judgement of existing design response analyses; therefore this change is non-substantive. The addition of "for use in the seismic PRA" is for clarification
	ASME/ANS RA-Sb–2013	When the existing response analysis models are judged not to be realistic and state of the art, or when the design input ground motion is significantly different from the site-specific input motion, PERFORM new analysis to obtain realistic structural loads and floor response spectra for use in the seismic PRA.		<not printed here; not focus of this assessment>	

Table 2: Comparison of Supporting Requirements of Addendum A and Addendum B for SFR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
					and also non-substantive.
SFR-C5	ASME/ANSRA-Sa–2009	If median-centered response analysis is performed, ESTIMATE the median response (i.e., structural loads and floor response spectra) and variability in the response using established methods.		<not printed here; not focus of this assessment>	**Addendum B Assessment Equates to Addendum A** The wording for this SR is the same for Addendum A and Addendum B.
	ASME/ANSRA-Sb–2013	Identical to Addendum A		<not printed here; not focus of this assessment>	
SFR-C6	ASME/ANSRA-Sa–2009	When soil-structure interaction (SSI) analysis is conducted, ENSURE that it is median centered using median properties, at soil strain levels corresponding to the input ground motions that **dominate** the seismically induced core damage frequency. ACCOUNT for the uncertainties in the SSI analysis **by varying the low strain soil shear modulus between the median value times (1 + Cv) and the median value divided by (1 + Cv), where Cv is a factor that accounts for uncertainties in the SSI analysis and soil properties. If adequate soil investigation data are available, ESTABLISH the mean and standard deviation of the low strain shear modulus for every soil layer. Then ESTABLISH the value of Cv so that it will cover the mean plus or minus one standard deviation for every layer. The**		<not printed here; not focus of this assessment>	**Vogtle Conforms to Accepted Current Practices** The changes in SFR-C6 involved the replacement of "ACCOUNT for" with the more precise action verb "INCLUDE", the non-substantive replacement of "dominate" with "contribute most" for PRA standard consistency, and the removal of how to perform SSI uncertainty analysis. The SSI uncertainty analysis method presented in Addendum A is derived from ASCE 4-98 (as indicated by the non-mandatory Note 5). Section 3.3.1.7 of ASCE 4-98 states that the use of (1 + Cv) to vary low strain soil shear moduli is an acceptable method *in lieu of probabilistic evaluation,* which Section C.3.3.1.7 further states is the

Table 2: Comparison of Supporting Requirements of Addendum A and Addendum B for SFR					
SR	**Standard Rev.**	**Capability Category I**	**Capability Category II**	**Capability Category III**	**Basis for Assessment (CC-II Focus)**
		minimum value of Cv is 0-5. When insufficient data are available to address uncertainties in soil properties, ENSURE that Cv is taken as no less than 1.0.			preferred approach. Plant Vogtle accounted for uncertainties in the SSI analysis by applying strain-compatible soil properties derived from probabilistic evaluation via the PSHA. Use of the ASCE 4-98 alternate approach using (1 + Cv) would render overly conservative and unrealistic response analysis results, which would invalidate other SRs. Therefore, the Addendum B assessment is considered appropriate.
	ASME/ANS RA-Sb–2013	When soil-structure interaction (SSI) analysis is conducted, ENSURE that it is median centered using median properties, at soil strain levels corresponding to the input ground motions that **contribute most** to the seismically induced core damage frequency. INCLUDE the uncertainties in the SSI analysis.		<not printed here; not focus of this assessment>	
HLR-SFR-D: The seismic-fragility evaluation shall be performed for critical failure modes of SSCs such as structural failure modes and functional failure modes identified through the review of plant design documents, supplemented as needed by earthquake experience data, fragility test data, generic qualification test data, and a walkdown.					
SFR-D1	ASME/ANS RA-Sa–2009	IDENTIFY realistic failure modes of structures and equipment that interfere with the operability of equipment during or after the earthquake through a review of the plant design documents and the walkdown.			**Addendum B Assessment Envelopes Addendum A** The change from Addendum A to Addendum

Table 2: Comparison of Supporting Requirements of Addendum A and Addendum B for SFR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
	ASME/ANS RA-Sb–2013	IDENTIFY realistic failure modes of structures **(e.g., sliding, overturning, yielding, and excessive drift)**, equipment **(e.g., anchorage failure, impact with adjacent equipment or structures, bracing failure, and functional failure), and soil (e.g., liquefaction, slope instability, and excessive differential settlement)** that interfere with the operability of equipment during or after the earthquake, through a review of the plant design documents and the walkdown.			B involved the move of example failure modes from SFR-D2 to SFR-D1.
SFR-D2	ASME/ANS RA-Sa–2009	**EXAMINE** all relevant failure modes of structures **(e.g., sliding, overturning, yielding, and excessive drift)**, equipment **(e.g., anchorage failure, impact with adjacent equipment or structures, bracing failure, and functional failure), and soil (i.e., liquefaction, slope instability, and excessive differential settlement)**, and EVALUATE fragilities for critical failure modes.			**Addendum B Assessment Envelopes Addendum A** The changes from Addendum A to Addendum B for this SR involved the aforementioned move of example failure modes to SFR-D1 and change in wording from "EXAMINE all relevant failure modes of structures..." to "EVALUATE relevant failure modes identified in SFR-D1." The latter change implements a more precise action verb in addition to providing more specificity for the evaluation of failure modes.
	ASME/ANS RA-Sb–2013	**EVALUATE all relevant failure modes identified in Requirement SFR-D1**, and EVALUATE fragilities for critical failure modes.			
HLR-SFR-E: The seismic-fragility evaluation shall incorporate the findings of a detailed walkdown of the plant focusing on the anchorage, lateral seismic support, and potential systems interactions.					
SFR-E1	ASME/ANS RA-Sa–2009	CONDUCT a detailed walkdown of the plant, focusing on equipment anchorage, lateral seismic support, spatial interactions, and potential systems interactions (both structural and functional interactions).			**Addendum B Assessment Equates to Addendum A** The wording for this SR is the same for Addendum A and Addendum B.
	ASME/ANS RA-Sb–2013	Identical to Addendum A			

Table 2: Comparison of Supporting Requirements of Addendum A and Addendum B for SFR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
SFR-E2	ASME/ANS RA-Sa–2009	DOCUMENT the walkdown procedures, walkdown team composition and its members' qualifications, walkdown observations, and conclusions.			**Addendum B Assessment Equates to Addendum A** The wording for this SR is the same for Addendum A and Addendum B.
	ASME/ANS RA-Sb–2013	Identical to Addendum A			
SFR-E3	ASME/ANS RA-Sa–2009	If components are screened out during or following the walkdown, DOCUMENT **anchorage calculations and PROVIDE the basis justifying such a screening.**			**Addendum B Assessment Envelopes Addendum A** The change implemented from Addendum A to Addendum B involves increased requirement to not only document anchorage screening but all screening (e.g. including functional or component structure).
	ASME/ANS RA-Sb–2013	If components are screened out during or following the walkdown, DOCUMENT **the basis, including any anchorage calculations that justify such a screening.**			
SFR-E4	ASME/ANS RA-Sa–2009	During the walkdown, **FOCUS** on the potential for seismically induced fire and flooding.			**Addendum B Assessment Envelopes Addendum A** The change from Addendum A to Addendum B involved changing "FOCUS" into the more precise action verb "EVALUATE" in addition to adding specific criteria pertaining to NUREG-1407.
	ASME/ANS RA-Sb–2013	During the walkdown, **EVALUATE** the potential for seismically induced fire and flooding **by focusing on the issues described in NUREG-1407 [5-7].**			
SFR-E5	ASME/ANS RA-Sa–2009	During the walkdown, **EXAMINE** potential sources of interaction (e.g., II/I issues, impact between cabinets, masonry walls, flammable and combustion sources, flooding, and spray) and consequences of such interactions on equipment contained in the systems model.			**Addendum B Assessment Equates to Addendum A** The change from Addendum A to Addendum B for this SR involved the replacement of "EXAMINE" with "EVALUATE." The action verb change was made to be consistent with accepted verb usage across SRs. These wording edits do not change capability category requirements of this SR..
	ASME/ANS RA-Sb–2013	During the walkdown, **EVALUATE** potential sources of interaction (e.g., II/I issues, impact between cabinets, masonry walls, flammable and combustion sources, flooding, and spray) and consequences of such interactions on equipment contained in the systems model.			

Table 2: Comparison of Supporting Requirements of Addendum A and Addendum B for SFR					
SR	**Standard Rev.**	**Capability Category I**	**Capability Category II**	**Capability Category III**	**Basis for Assessment (CC-II Focus)**
HLR-SFR-F: The calculation of seismic-fragility parameters such as median capacity and variabilities shall be based on plant-specific data or, if necessary, on earthquake experience data, fragility test data, and generic qualification test data. Use of such generic data shall be justified.					
SFR-F1	ASME/ANS RA-Sa–2009	**BASE** component seismic-fragility parameters such as median capacity and variabilities (logarithmic standard deviations reflecting randomness and uncertainty) on plant-specific data **supplemented as appropriate by** earthquake experience data, fragility test data, and generic qualification test data.		<not printed here; not focus of this assessment>	**Addendum B Assessment Envelopes Addendum A** Addendum B changes the action verb from "BASE" to the more precise and restrictive verb "CALCULATE" in keeping with the approved action verb list and adds more specificity to the use of sources beyond plant-specific data, including the provision to justify the appropriateness of generic fragility data.
	ASME/ANS RA-Sb–2013	**CALCULATE** component seismic-fragility parameters such as median capacity and variabilities (logarithmic standard deviations reflecting randomness and uncertainty) **based** on plant-specific data **or, if necessary, on** earthquake experience data, fragility test data, and generic qualification test data. ***Exception*: JUSTIFY the use of generic fragility for any SSC as being appropriate for the plant.**		<not printed here; not focus of this assessment>	
SFR-F2	ASME/ANS RA-Sa–2009	For all structures, or systems, or components, or a combination thereof (SSCs) that appear in the dominant accident cut sets, ENSURE that they have site-specific fragility parameters that are derived based on plant-specific information, such as anchoring and installation of the component or structure and plant-specific material test data. *Exception*: JUSTIFY the use of generic fragility for any SSC as being appropriate for the plant.		<not printed here; not focus of this assessment>	**Addendum B Assessment Equates to Addendum A** Addendum B changes the use of "dominant accident cut sets" in Addendum A to "significant accident sequences." This wording edit is for consistency with the convention used elsewhere in the PRA standard to be "software neutral" and does not change the capability category

Table 2: Comparison of Supporting Requirements of Addendum A and Addendum B for SFR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
	ASME/ANS RA-Sb–2013	For all SSCs that appear in the significant accident sequences, ENSURE that they have site-specific fragility parameters that are derived based on plant-specific information, such as anchoring and installation of the component or structure and plant-specific material test data. Exception: JUSTIFY the use of generic fragility for any SSC as being appropriate for the plant.		<not printed here; not focus of this assessment>	requirements of this SR.
SFR-F3	ASME/ANS RA-Sa–2009	<not printed here; not focus of this assessment>	**DEVELOP** seismic fragilities for relays identified to be essential and that are included in the systems-analysis model.		**Addendum B Assessment Equates to Addendum A** The change from Addendum A to Addendum B for this SR was the replacement of "DEVELOP" with "CALCULATE." The action verb change was made to be consistent with accepted verb usage across SRs. These wording edits do not change capability category requirements of this SR.
	ASME/ANS RA-Sb–2013	<not printed here; not focus of this assessment>	**CALCULATE** seismic fragilities for relays identified to be essential and that are included in the systems-analysis model.		
SFR-F4	ASME/ANS RA-Sa–2009	**DEVELOP** seismic fragilities for structures, or systems, or components, or a combination thereof that are identified in the systems model as playing a role in the large early release frequency part of the seismic PRA. (See Requirements SPR-A1 and SPR-A3.)			**Addendum B Assessment Equates to Addendum A** The changes from Addendum A to Addendum B for this SR included the replacement of "DEVELOP" with "CALCULATE." The action verb change was made to be consistent with accepted verb usage across SRs. Also, SSC abbreviation used instead. These wording edits do not change capability category requirements of this SR.
	ASME/ANS RA-Sb–2013	**CALCULATE** seismic fragilities for SSCs that are identified in the systems model as playing a role in the large early release frequency part of the seismic PRA. (See Requirements SPR-A1 and SPR-A3.)			

Table 2: Comparison of Supporting Requirements of Addendum A and Addendum B for SFR					
SR	**Standard Rev.**	**Capability Category I**	**Capability Category II**	**Capability Category III**	**Basis for Assessment (CC-II Focus)**
HLR-SFR-G: Documentation of the seismic-fragility evaluation shall be consistent with the applicable supporting requirements.					
SFR-G1	ASME/ANS RA-Sa–2009	DOCUMENT the seismic fragility analysis in a manner that facilitates PRA applications, upgrades, and peer review.			**Addendum B Assessment Equates to Addendum A** The wording for this SR is the same for Addendum A and Addendum B.
	ASME/ANS RA-Sb–2013	Identical to Addendum A			
SFR-G2	ASME/ANS RA-Sa–2009	DOCUMENT the process used in the seismic-fragility analysis. For example, this typically includes a description of (a) The methodologies used to quantify the seismic fragilities of structures, or systems, or components, or a combination thereof, together with key assumptions (b) The seismic fragilities of structures, or systems, or components, or a combination thereof (SSC) fragility values that includes the method of seismic qualification, the dominant failure mode(s), the source of information, and the location of the component (c) The fragility parameter values (i.e., median acceleration capacity, Br and Bu) and the technical bases for them for each analyzed SSC, and (d) the different elements of seismic-fragility analysis, such as (1) the seismic response analysis (2) the screening steps (3) the walkdown (4) the review of design documents (5) the identification of critical failure modes for each SSC, and (6) the calculation of fragility parameter values for each SSC modeled			**Addendum B Assessment Equates to Addendum A** The only change in this SR is related to use of "structures, systems, and components" versus its acronym "SSC." This change is considered editorial and does not change the capability category requirements of this SR.

Table 2: Comparison of Supporting Requirements of Addendum A and Addendum B for SFR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
	ASME/ANS RA-Sb–2013	DOCUMENT the process used in the seismic-fragility analysis. For example, this typically includes a description of (a) the methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions (b) the SSC fragility values that includes the method of seismic qualification, the dominant failure mode(s), the source of information, and the location of the component (c) the fragility parameter values (i.e., median acceleration capacity, Br and Bu) and the technical bases for them for each analyzed SSC, and (d) the different elements of seismic-fragility analysis, such as (1) the seismic response analysis (2) the screening steps (3) the walkdown (4) the review of design documents (5) the identification of critical failure modes for each SSC, and (6) the calculation of fragility parameter values for each SSC modeled			
SFR-G3	ASME/ANS RA-Sa–2009	**DOCUMENT the sources of model uncertainty and related assumptions associated with the seismic fragility analysis.**			**Vogtle Conforms to Addendum A** Addendum B deleted this SR. However, the Plant Vogtle 1&2 SPRA documentation describes in detail the sources of model uncertainty and related assumptions associated with the seismic fragility analysis. Therefore, the Vogtle SPRA conforms to Addendum A.
	ASME/ANS RA-Sb–2013	**Deleted.**			

Table 3: Comparison of Supporting Requirements of Addendum A and Addendum B for SPR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
HLR-SPR-A: The seismic-PRA systems model shall include seismic-caused initiating events and other failures including seismic-induced SSC failures, non–seismic-induced unavailabilities, and human errors, that give rise to significant accident sequences and/or significant accident progression sequences.					
SPR-A1	ASME/ANS RA-Sa–2009	ENSURE that earthquake-caused initiating events that give rise to significant accident sequences and/or significant accident progression sequences are included in the seismic-PRA system model using a systematic process.			**Addendum B Assessment Equates to Addendum A** The wording for this SR is the same for Addendum A and Addendum B. A minor clarification (i.e., very low magnitude earthquakes can be excluded from the quantification process) was added in Addendum B to the clarifying non-mandatory footnote for this SR; this footnote adjustment does not change the capability category requirements of this SR.
	ASME/ANS RA-Sb–2013	Identical to Addendum A			
SPR-A2	ASME/ANS RA-Sa–2009	In the initiating-event selection process, DEVELOP a hierarchy to ensure that every earthquake greater than a certain defined size produces a plant shutdown within the systems model.			**Addendum B Assessment Equates to Addendum A** The wording for this SR is the same for Addendum A and Addendum B. A minor clarification (i.e., event ordering by consequence severity) was added in Addendum B to the clarifying non-mandatory footnote for this SR; this footnote adjustment does not change the capability category requirements of this SR.
	ASME/ANS RA-Sb–2013	Identical to Addendum A			
SPR-A3	ASME/ANS RA-Sa–2009	USE **the event trees and fault trees** from the internal-event at-power PRA model as the basis for the seismic **event trees**.			**Addendum B Assessment Equates to Addendum A**

Table 3: Comparison of Supporting Requirements of Addendum A and Addendum B for SPR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
	ASME/ANS RA-Sb–2013	USE **the accident sequences and the systems logic model** from the at-power, internal-event PRA model as the basis for the seismic-**PRA model**.			Some of the terms in this SR were changed in Addendum B so that they would not be less specific to a certain type of PRA modeling software. These term clarifications do not change the capability category requirements of this SR.
SPR-A4 (Add. B)	ASME/ANS RA-Sb–2013	**This supporting requirement is new to Addendum B and not included in Addendum A.** Under special circumstances based on the judgment of the analyst, DEVELOP an ad hoc systems model tailored especially to the seismic-PRA configurations or issues being modeled, instead of starting with the internal-events model and adapting it, as in Requirement SPR-A3. If this approach is used, ENSURE that the resulting model is consistent with the internal-events systems model regarding plant response and the cause-effect relationships of the failures.			**Addendum B Assessment Equates to Addendum A** Addendum B added this SR to recognize that some utilities may build a separate stand-alone SPRA as opposed to adding seismic aspects into an existing internal events PRA (as described by SPR-A3). This new SR ensures that the technical capability of the base modeling would be consistent with the internal events technical capability requirements. For Vogtle, this specific SR is not applicable because the Vogtle 1&2 SPRA is built upon the internal events PRA per SR SPR-A3 and thus the Vogtle SPRA conforms to Addendum A.
SPR-A4 (Add. A) SPR-A5 (Add. B)	ASME/ANS RA-Sa–2009	**SPR-A4 of Addendum A:** ENSURE that the PRA systems models reflect earthquake-caused failures and nonseismically induced unavailabilities and human errors that give rise to significant accident sequences or significant accident progression sequences.			**Addendum B Assessment Equates to Addendum A** The wording for this SR is the same for Addendum A and Addendum B. Miscellaneous edits were made in Addendum B to the clarifying non-mandatory footnote for this SR; this footnote adjustment does not change the capability category requirements
	ASME/ANS RA-Sb–2013	**SPR-A5 of Addendum B is identical to SPR-A4 of Addendum A. There is no change in the supporting requirement from Addendum A to Addendum B, except that the insertion of a new SPR-A4 in Addendum B changes the number for this**			

Table 3: Comparison of Supporting Requirements of Addendum A and Addendum B for SPR					
SR	**Standard Rev.**	**Capability Category I**	**Capability Category II**	**Capability Category III**	**Basis for Assessment (CC-II Focus)**
		requirement.			of this SR.
HLR-SPR-B: The seismic-PRA systems model shall be adapted to incorporate seismic-analysis aspects that are different from corresponding aspects found in the at-power, internal-events PRA systems model.					
SPR-B1	ASME/ANSRA-Sa–2009	In each of the following aspects of the seismic-PRA systems-analysis work, SATISFY the corresponding requirements in Part 2, except where they are not applicable or where this Part includes additional requirements. **DEVELOP a defined** basis to support the claimed nonapplicability of any exceptions. The aspects governed by this requirement are (a) initiating-event analysis (b) accident-sequence analysis (c) success-criteria analysis (d) systems analysis (e) data analysis (f) human-reliability analysis (g) use of expert judgment **When the Part 2 requirements are used, FOLLOW the Capability Category designations in Part 2, and for consistency USE the same Capability Category in this analysis.**			**Vogtle Conforms to Addendum A** Addendum B removed the last sentence of this SR in response to an EPRI 2011 comment on the Addendum B ballot. The last sentence was removed in Addendum B because it was determined to be confusing as wells as inappropriate specificity to require all new aspects in the SPRA to meet the exact same CCs of Part 2 SRs. In addition, Addendum B changed the action verb to be consistent with accepted verb usage across SRs. The Addendum B SR requirement clarifications are appropriate. Regardless, the Plant Vogtle 1&2 SPRA builds upon the internal events PRA and uses the same

Table 3: Comparison of Supporting Requirements of Addendum A and Addendum B for SPR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
	ASME/ANS RA-Sb–2013	In each of the following aspects of the seismic-PRA systems-analysis work, SATISFY the corresponding requirements in Part 2, except where they are not applicable or where this Part includes additional requirements. **SPECIFY** a basis to support the claimed nonapplicability of any exceptions. The aspects governed by this requirement are (a) initiating-event analysis (b) accident-sequence analysis (c) success-criteria analysis (d) systems analysis (e) data analysis (f) human-reliability analysis (g) use of expert judgment			general methodologies as used for Part 2 where applicable; therefore, the Vogtle SPRA conforms to Addendum A.
SPR-B2	ASME/ANS RA-Sa–2009	**In the human reliability analysis (HRA) aspect, EXAMINE additional post earthquake stresses that can increase the likelihood of human errors or inattention, compared to the likelihood assigned in the internal-events HRA when the same activities are undertaken in no earthquake accident sequences. Whether or not increases in error probabilities are used, JUSTIFY the basis for this decision about what error rates to use.**			**Addendum B Assessment Envelopes Addendum A** In response to an EPRI 2011 comment on the Addendum B ballot, Addendum B added CCI-II and CCIII capability differentiation as well as the requirement to include the seismic HEP impacts on the internal events based HEPs, not simply to Examine them. It also expanded on the considerations to emphasize PSFs beyond just stress. The Addendum B SR wording envelopes that of Addendum A.
	ASME/ANS RA-Sb–2013	**INCLUDE the following seismic impacts on performance-shaping factors (PSFs) for the control room and ex-control room post-initiator actions as appropriate to the human reliability analysis (HRA) methodology used:** **(a) additional post-earthquake workload and stress that can increase the likelihood of human errors or inattention**		<not printed here; not focus of this assessment>	

Table 3: Comparison of Supporting Requirements of Addendum A and Addendum B for SPR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
		(b) seismic failures that impact access **(c) cue availability**			
SPR-B3 (Add. A) SPR-B4a (Add. B)	ASME/ANS RA-Sa–2009	**SPR-B3 of Addendum A:** **If any screening is performed, PERFORM it using defined criteria that are documented in the PRA.**			**Addendum B Assessment Equates to Addendum A** Addendum B changed the sentence structure to correct awkward wording and to use accepted verb. This wording edit does not change the capability category requirements of this SR.
	ASME/ANS RA-Sb–2013	**SPR-B4a of Addendum B:** **If screening out on the basis of seismic capacity is performed in the systems model, SPECIFY the screening criterion.**			
SPR-B4 (Add. A) SPR-B3 (Add. B)	ASME/ANS RA-Sa–2009	**SPR-B4 of Addendum A:** PERFORM an analysis of seismic-caused dependencies and correlations in a way so that any screening of SSCs appropriately accounts for those **dependencies and** correlations. USE bounding or generic correlation values and PROVIDE the basis for such use.		<not printed here; not focus of this assessment>	**Addendum B Assessment Equates to Addendum A** Addendum B deleted the phrase "dependencies and" because they are redundant to the "correlation" term so as to avoid reviewer potential confusion. This wording edit does not change the capability category requirements of this SR.
	ASME/ANS RA-Sb–2013	**SPR-B3 of Addendum B:** PERFORM an analysis of seismic-caused dependencies and correlations in a way so that any screening of SSCs appropriately accounts for those correlations. USE bounding or generic correlation values and PROVIDE the basis for such use.		<not printed here; not focus of this assessment>	
SPR-B5 (Add. A)	ASME/ANS RA-Sa–2009	**SPR-B5 of Addendum A:** ENSURE that any screening of human-error basic events and non–seismic-failure basic events does not significantly affect the PRA's results.			**Addendum B Assessment Equates to Addendum A** Addendum B deleted this SR because it is redundant to SR SPR-A5, which already requires modeling of operator actions that contribute to significant accident sequences.
	ASME/ANS RA-Sb–2013	**No SR of Addendum B directly matches with SPR-B5 of Addendum A**			

Table 3: Comparison of Supporting Requirements of Addendum A and Addendum B for SPR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
					Also, it is redundant in part to SR SPR-E3, which requires validation that any screening does not impact the results. Finally, Vogtle did not screen out any human-error or non-seismic events that were in the FPIE model and would propagate through seismic accident sequences, and so conforms to Addendum A.
SPR-B4b	ASME/ANSRA-Sa–2009	**Not included in Addendum A.**			**Addendum B Assessment Envelopes Addendum A** This SR is new for Addendum B to ensure that recoveries added to the SPRA (i.e., not already included in the internal events PRA) are assessed for appropriateness. Addendum B envelopes Addendum A with respect to this SR.
	ASME/ANSRA-Sb–2013	**SPR-B4b of Addendum B:** If post-earthquake recovery actions are included in the systems model, INCLUDE them on a documented basis.			
SPR-B6 (Add. A) SPR-B4 (Add. B)	ASME/ANSRA-Sa–2009	<not printed here; not focus of this assessment>	SPR-B6 of Addendum A: **EXAMINE** the effects of the chatter of relays and similar devices.		**Addendum B Assessment Envelopes Addendum A** In response to an EPRI 2011 comment on the Addendum B ballot, Addendum B added the requirement to include the effects of relay chatter in the model, not simply to examine them. The differentiation for "low ruggedness relays" is deleted in Addendum B because the issue of chatter fragility is addressed in SFR SRs. The Addendum B SR wording envelopes that of Addendum A.
	ASME/ANSRA-Sb–2013	**SPR-B4 of Addendum B:** **INCLUDE** the effects of the chatter of relays and similar devices in the systems model.			

Table 3: Comparison of Supporting Requirements of Addendum A and Addendum B for SPR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
SPR-B7 (Add. A) SPR-B5 (Add. B)	ASME/ANS RA-Sa–2009	**SPR-B7 of Addendum A:** In the systems-analysis models, for each basic event that represents a seismically caused failure, INCLUDE the complementary "success" state where applicable to a particular SSC.			**Addendum B Assessment Equates to Addendum A** Addendum B revised the wording of this SR to include a capability category distinction to explicitly model fragility complements for risk significant fragilities for CCI-II and for all modeled fragilities for CCIII. The Vogtle 1&2 SPRA uses the SPRA CAFTA suite of codes with the ACUBE module addressing the fragility complement modeling and so meets the intent of both Addenda A and B since the quantification accounts mathematically for the success states.
	ASME/ANS RA-Sb–2013	**SPR-B5 of Addendum B:** **In the systems-analysis models, for each basic event that represents a significant seismically caused failure, INCLUDE the complementary "success" state where applicable to a particular SSC, and SPECIFY the criteria used for the term "significant" in this activity.**		<not printed here; not focus of this assessment>	
SPR-B8 (Add. A) SPR-B6 (Add. B)	ASME/ANS RA-Sa–2009	**SPR-B8 of Addendum A:** **EXAMINE** the possibility that a large earthquake can cause damage that blocks personnel access to safety equipment or controls, thereby inhibiting operator actions that might otherwise be credited.			**Addendum B Assessment Equates to Addendum A** Addendum B changed the action verb to be consistent with accepted verb usage across SRs. This wording edit does not change the capability category requirements of this SR.
	ASME/ANS RA-Sb–2013	**SPR-B6 of Addendum B:** **EVALUATE** the possibility that a large earthquake can cause damage that blocks personnel access to safety equipment or controls, thereby inhibiting operator actions that might otherwise be credited.			

Table 3: Comparison of Supporting Requirements of Addendum A and Addendum B for SPR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
SPR-B9 (Add. A) SPR-B7 (Add. B)	ASME/ANS RA-Sa–2009	<not printed here; not focus of this assessment>	**SPR-B9 of Addendum A: EXAMINE the likelihood that system recoveries modeled in the internal-events PRA may be more complex or even not possible after a large earthquake, and ADJUST the recovery models accordingly.**		**Addendum B CCII Assessment Equates to Addendum A CCII-III** In response to an EPRI 2011 comment on the Addendum B ballot, Addendum B added three capability differentiations. CCII requirements are effectively the same for Addendum A and Addendum B. The Addendum B SR adds the clarification that generic and/or conservative recovery values are acceptable. The Addendum B SR CCII wording effectively equates to Addendum A CCII-III given the state of practice at the time (i.e., detailed plant-specific system recovery probability calculations were not typical practice at the time of Addendum A).
	ASME/ANS RA-Sb–2013	<not printed here; not focus of this assessment>	**SPR-B7 of Addendum B: EVALUATE the likelihood that system recoveries modeled in the internal-events PRA may be more complex or even not possible after a large earthquake, and ADJUST the recovery models accordingly. It is acceptable to use generic or conservative recovery values.**	<not printed here; not focus of this assessment>	
SPR-B10 (Add. A) SPR-B8	ASME/ANS RA-Sa–2009	**SPR-B10 of Addendum A: EXAMINE the effect of including an earthquake-caused "small-small loss-of-coolant accident" as an additional fault within each sequence in the seismic-PRA model.**			**Addendum B Assessment Envelopes Addendum A** Addendum B added differentiation of

Table 3: Comparison of Supporting Requirements of Addendum A and Addendum B for SPR					
SR	**Standard Rev.**	**Capability Category I**	**Capability Category II**	**Capability Category III**	**Basis for Assessment (CC-II Focus)**
(Add. B)	ASME/ANSRA-Sb–2013	**SPR-B8 of Addendum B:** **ASSUME the existence of an earthquake-caused "very small loss-of-coolant accident" in the seismic-PRA accident sequences and system modeling, unless it is demonstrated that such a LOCA can be excluded, based on a walkdown or on another examination of the possible sources of such a LOCA.**		<not printed here; not focus of this assessment>	capability requirements to this SR that require modeling incorporation into the SPRA (if the topic requires it for a specific plant) whereas the EXAMINE verb in Addendum A did not require incorporation into the SPRA (i.e., it could be done as a sensitivity study). The Addendum B SR wording envelopes that of Addendum A.
SPR-B11 (Add. A) SPR-B9 (Add. B)	ASME/ANSRA-Sa–2009	**SPR-B11 of Addendum A:** **In the seismic PRA walkdown, INCLUDE the potential for seismically induced fires and flooding following the guidance given in NUREG-1407.**			**Addendum B Assessment Envelopes Addendum A** In response to an EPRI 2011 comment on the Addendum B ballot, Addendum B added the requirement to include risk significant scenarios into the SPRA models as opposed to only including them in the walkdown. Reference to NUREG-1407 was removed in Addendum B because this SR refers to SFR-E4 for input and SFR-E4 already includes the reference to NUREG-1407. The Addendum B SR wording envelopes that of Addendum A.
	ASME/ANSRA-Sb–2013	**SPR-B9 of Addendum B:** **If the seismic PRA walkdown (see Requirement SFR-E4) identifies the potential for seismically induced fires and flooding, INCLUDE potential significant contributions to accident sequences in the systems model.**			
HLR-SPR-C: The seismic-PRA systems model shall reflect the as-built and as-operated plant being analyzed.					
SPR-C1	ASME/ANSRA-Sa–2009	To ensure that the systems-analysis model reflects the as-built, as-operated plant, JUSTIFY any conservatisms or other distortions **introduced by demonstrating that the seismic-PRA's validity for applications is maintained.**			**Addendum B Assessment Equates to Addendum A** Addendum B changed the last clause of this

Table 3: Comparison of Supporting Requirements of Addendum A and Addendum B for SPR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
	ASME/ANS RA-Sb–2013	To ensure that the systems-analysis model reflects the as-built, as-operated plant, JUSTIFY any conservatisms or other distortions **that do not adequately reflect the as-built, as-operated plant.**			SR because the wording was awkward. This wording refinement does not change the capability category requirements of this SR.
HLR-SPR-D: The list of SSCs selected for seismic-fragility analysis shall include the SSCs that participate in accident sequences included in the seismic-PRA systems model.					
SPR-D1	ASME/ANS RA-Sa–2009	USE the seismic PRA systems model as the basis for developing the seismic equipment list, **which is the list of all SSCs to be considered by the subsequent seismic-fragility evaluation task.**			**Addendum B Assessment Envelopes Addendum A** Addendum B added additional clarifications and requirements into the text of this SR. The Addendum B SR wording envelopes that of Addendum A.
	ASME/ANS RA-Sb–2013	USE the PRA systems model as the basis for developing the seismic equipment list **to support the fragility analysis of 5-2.2. INCLUDE structures and passive components that may not be present in the internal-events model but that require consideration in the seismic PRA. SUPPLEMENT the list based on the review of industry seismic-PRA seismic equipment lists (SELs), if available.**			
HLR-SPR-E: The analysis to quantify core damage frequency and large early release frequency shall appropriately integrate the seismic hazard, the seismic fragilities, and the systems-analysis aspects					
SPR-E1	ASME/ANS RA-Sa–2009	In the quantification of core damage frequency and large early release frequency, PERFORM the integration using the seismic hazard, fragility, and systems analyses.			**Addendum B Assessment Equates to Addendum A** The wording for this SR is the same for Addendum A and Addendum B. Minor edits were made in Addendum B to the clarifying non-mandatory footnote for this SR; this footnote adjustment does not change the capability category requirements of this SR.
	ASME/ANS RA-Sb–2013	Identical to Addendum A			

Table 3: Comparison of Supporting Requirements of Addendum A and Addendum B for SPR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
SPR-E2	ASME/ANS RA-Sa–2009	<not printed here; not focus of this assessment>	In quantifying core damage frequency and large early release frequency, PERFORM the quantification on a cut-set-by-cut set or accident-sequence-by-accident-sequence basis (or for defined groups of these), as well as on a comprehensive/integrated basis.		**Addendum B Assessment Envelopes Addendum A** Addendum B revised this SR to back-reference to Part 2 requirements for the accident sequence quantification. This edit was made for consistency across the Std Parts to refer back to Part 2 SRs where applicable. The wording edit in Addendum B to this SR expands the capability category requirements beyond the requirements of Addendum A by incorporating many more SRs.
	ASME/ANS RA-Sb–2013	PERFORM seismic-sequence quantification **in accordance with the applicable requirements described in 2-2.7.**			
SPR-E3	ASME/ANS RA-Sa–2009	**In the analysis,** USE the quantification process to **ensure that any** screening of SSCs **does not affect the results, taking into account the various uncertainties.**			**Addendum B Assessment Equates to Addendum A** In addition to minor wording changes, Addendum B revised this SR to remove the clause regarding consideration of uncertainties. The use of the phrase "confirm and support" allows for the analyst to determine the best way to validate the screening, whereas Addendum A went beyond "what to do" into "how to do it." The end result is the same.
	ASME/ANS RA-Sb–2013	USE the quantification process **to confirm and support the** screening of SSCs (refer to Requirement SFR-B1).			
SPR-E4	ASME/ANS RA-Sa–2009	In the integration/quantification analysis, **ACCOUNT for** all significant dependencies and correlations that affect the results. It is acceptable to use generic correlation values. If used, **PROVIDE** the basis for such		<not printed here; not focus of this assessment>	**Addendum B Assessment Equates to Addendum A** Addendum B changed action verbs to be consistent with accepted verb usage across

Table 3: Comparison of Supporting Requirements of Addendum A and Addendum B for SPR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
		use.			SRs, and also deleted the redundant "dependencies" term. These wording edits do not change capability category requirements of this SR.
	ASME/ANS RA-Sb–2013	In the integration/quantification analysis, **INCLUDE** the significant correlations that affect the results. It is acceptable to use generic correlation values. If used, **SPECIFY** the basis for such use.		<not printed here; not focus of this assessment>	
SPR-E5	ASME/ANS RA-Sa–2009	<not printed here; not focus of this assessment>	Identical to Addendum B CCII except Addendum A uses ACCOUNT verb.		**Addendum B CCII Assessment Equates to Addendum A CCII-III** Addendum B changed action verbs to be consistent with accepted verb usage across SRs. Three Capability Category distinctions are instituted for Addendum B (Addendum B CCII corresponds to Addendum A CCII-III). These edits do not change the CCII capability category requirements of this SR.
	ASME/ANS RA-Sb–2013	<not printed here; not focus of this assessment>	In the integration/quantification analysis, **INCLUDE** in the uncertainties in core damage frequency and large early release frequency results that arise from each of the several inputs (the seismic hazard, the seismic fragilities, and the systems-analysis aspects).	<not printed here; not focus of this assessment>	

Table 3: Comparison of Supporting Requirements of Addendum A and Addendum B for SPR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
SPR-E6 (Add. A)	ASME/ANS RA-Sa–2009	PERFORM appropriate sensitivity studies to illuminate the sensitivity of the core damage frequency and large early release frequency results to the assumptions used about dependencies and correlations.			**Addendum B Assessment Envelopes Addendum A** In response to an EPRI 2011 comment on the Addendum B ballot, Addendum B deleted this SR because performance of sensitivity studies is already addressed by another SR through the new back-reference to meeting the Part 2 quantification requirements (see SPR-E2), which includes the requirement to conduct appropriate sensitivity studies to address sources of uncertainty. This new back-reference implies requirements well beyond just sensitivity to correlations, and so the Addendum B approach envelopes Addendum A, even with this deletion. The Vogtle 1&2 SPRA documentation includes a sensitivity case related to seismic fragility correlation modeling.
	ASME/ANS RA-Sb–2013	Deleted in Addendum B			
SPR-E6 (Add. B)	ASME/ANS RA-Sa–2009	New Requirement in Addendum B.			**Addendum B Assessment Envelopes Addendum A** Addendum B added this SR to back-reference to Part 2 requirements for the LERF analysis. This new SR was added for consistency across the Std Parts to refer back to Part 2 SRs where applicable. The
	ASME/ANS RA-Sb–2013	In the analysis of LERF, SATISFY the LERF requirements in 2-2.8, where applicable. Note 6 - Those aspects of LERF analysis that are common to internal-events PRA and seismic PRA are referred to here. Also, the discussion of LERF analysis in the last four paragraphs of 5-1.3 is			

Table 3: Comparison of Supporting Requirements of Addendum A and Addendum B for SPR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
		broadly applicable and should be referred to as background information.			Addendum B SR wording envelopes that of Addendum A given that Addendum A did not specifically cite this LERF back-referencing to Part 2.
HLR-SPR-F: The seismic-PRA analysis shall be documented in a manner that facilitates applying the PRA and updating it, and that enables peer review.					
SPR-F1	ASME/ANS RA-Sa–2009	DOCUMENT the seismic plant response analysis and quantification in a manner that facilitates PRA applications, upgrades, and peer review.			**Addendum B Assessment Equates to Addendum A** The wording for this SR is the same for Addendum A and Addendum B. Addendum B edited the non-mandatory footnote for this SR. This footnote edit does not change the capability category requirements of this SR.
	ASME/ANS RA-Sb–2013	Identical to Addendum A			
SPR-F2	ASME/ANS RA-Sa–2009	DOCUMENT the process used in the seismic plant response analysis and quantification. For example, this documentation typically includes a description of: (a) the specific adaptations made in the internal events PRA model to produce the seismic-PRA model, and their motivation, and (b) the major outputs of a seismic PRA, such as mean core damage frequency (CDF), mean large early release frequency (LERF), uncertainty distributions on CDF and LERF, results of sensitivity studies, significant risk contributors, and so on, are examples of the PRA results that are generally documented.			**Addendum B Assessment Equates to Addendum A** Addendum B moved the non-mandatory list of examples to the associated non-mandatory footnote. This edit does not change the capability category requirements of this SR.
	ASME/ANS RA-Sb–2013	DOCUMENT the process used in the seismic plant response analysis and quantification.			

Table 3: Comparison of Supporting Requirements of Addendum A and Addendum B for SPR

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
SPR-F3	ASME/ANS RA-Sa–2009	DOCUMENT the sources of model uncertainty and related assumptions associated with the seismic plant response model development.			**Addendum B Assessment Equates to Addendum A** The wording for this SR is the same for Addendum A and Addendum B. Addendum A had a missing non-mandatory footnote (not all uncertainties need to be assigned a numerical distribution in the documentation) that was re-inserted in Addendum B. This footnote does not change the capability category requirements of this SR.
	ASME/ANS RA-Sb–2013	Identical to Addendum A.			

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Risk Quantification

1.0 Introduction

Figure 3-1 shows a high-level summary of the GSI-191 risk quantification including the interface between the probabilistic risk assessment (PRA) evaluation, the phenomenological evaluation, and the risk guidelines from RG 1.174 (Reference 1).

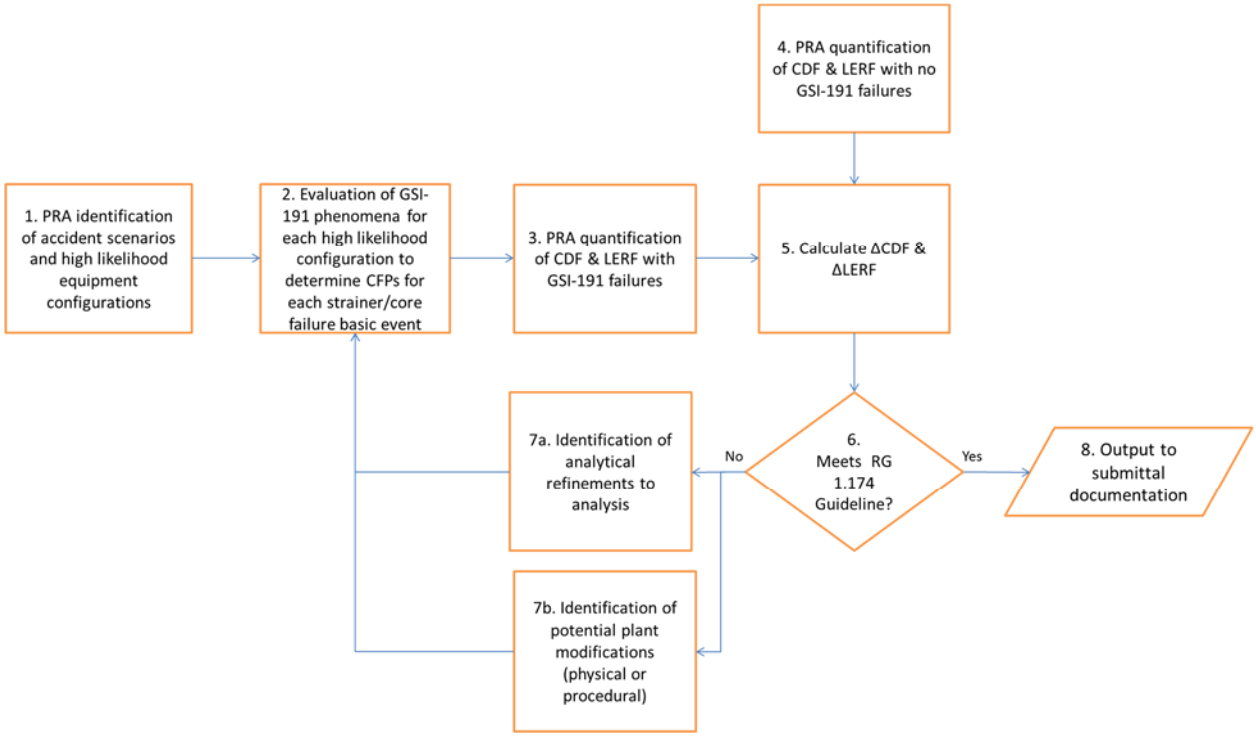

Figure 3-1 – Flow Chart Illustrating Risk-Informed GSI-191 Evaluation

The following steps are included in this flow chart:

- **Step 1: PRA identification of accident scenarios and high likelihood equipment configurations**

  Step 1 is an important part of the overall evaluation because it defines the scope of the GSI-191 problem. This step has two important parts: identification of accidents requiring recirculation through the emergency core cooling system (ECCS) strainers, and identification of high likelihood equipment configurations.

  As discussed in Enclosure 1, Section 2.1, the VEGP risk-informed GSI-191 evaluation considered all breaks that require strainer recirculation, which includes loss of coolant accidents (LOCAs) and secondary side breaks inside containment (SSBIs).

  For the accidents that need to be evaluated, it is important to identify the high likelihood equipment configurations. For example, because VEGP has two

100 percent capacity independent trains of ECCS and containment spray system (CSS) including a containment spray (CS) pump, a residual heat removal (RHR) pump, a safety injection (SI) pump, and a charging pump, there could be a large number of random failure combinations unrelated to GSI-191 issues (i.e., failure to start or failure to run). However, some equipment failure combinations may have sufficiently low probability that they can be addressed with a bounding analysis. The high likelihood equipment configurations are described in Section 6.3.

- **Step 2: Evaluation of GSI-191 phenomena for each high likelihood scenario to determine CFPs for each basic event**

  Step 2 is the process of using NARWHAL to calculate each of the conditional failure probabilities (CFPs) required by the PRA model. The details of this analysis are expanded upon in Section 13.0.

- **Step 3: PRA quantification of CDF and LERF with GSI-191 failures**

  Step 3 is a straightforward quantification of the core damage frequency (CDF) and large early release frequency (LERF) based on the GSI-191 failures for the as-built/as-operated plant.

  As discussed in Enclosure 1, Section 2.2, some minor modifications to the base PRA were necessary to perform this evaluation. Specifically, the GSI-191 PRA model includes basic events for both strainer failures and core failures to capture the GSI-191 CFPs. The modifications are described in more detail in Section 4.0.

- **Step 4: PRA quantification of CDF and LERF with no GSI-191 failures**

  Step 4 is essentially identical to Step 3, but all GSI-191 CFPs were set to zero to represent a hypothetical modification to the plant that would prevent any GSI-191 failures (e.g., removal of all sources of fiber debris).

- **Step 5: Calculate ΔCDF and ΔLERF**

  Step 5 is a simple comparison of the values calculated in Steps 3 and 4. The change in CDF (ΔCDF) and change in LERF (ΔLERF) values represent the risk associated with GSI-191.

- **Step 6: Comparison to RG 1.174 Guidelines**

  Step 6 is a comparison of the CDF, LERF, ΔCDF, and ΔLERF values to the acceptance guidelines defined in RG 1.174 (Reference 1). The ΔCDF limits are defined as $1x10^{-5}$/yr for the boundary between Regions I and II (low risk), and $1x10^{-6}$/yr for the boundary between Regions II and III (very low risk). The

ΔLERF limits are one order of magnitude lower than the ΔCDF limits ($1x10^{-6}$/yr for the boundary between Regions I and II, and $1x10^{-7}$/yr for the boundary between Regions II and III). Ideally, the evaluation would show that the risk associated with GSI-191 is in Region III. However, as described in RG 1.174, Region II is acceptable for a risk-informed submittal (Reference 1).

- **Step 7a: Identification of analytical refinements to analysis**

  In general, risk-informed evaluations should not contain significant conservatisms because conservatism skews the results in a way that might mask the true sources of risk. However, in practice, some level of bias that is generally accepted to be conservative is incorporated in risk-informed evaluations to avoid the extensive analysis and testing that would be required to develop more refined inputs and models. Therefore, if the results of the risk-informed evaluation are unacceptable, Step 7a can be used to identify specific conservatisms and implement refinements that may reduce the calculated risk to an acceptable level.

  Over the course of developing the VEGP risk-informed GSI-191 evaluation, a variety of refinements and simplifications were incorporated for different aspects of the model to balance realism and conservatism.

- **Step 7b: Identification of potential plant modifications (physical or procedural)**

  Step 7b is very similar to Step 7a, but is predicated on the analysis being as refined as practical (i.e., the inputs and analytical models used in the risk-informed evaluation are considered to be a reasonable representation of the post-accident conditions). In this case, if the risk is determined to be unacceptably high, it is necessary to make one or more plant modifications. The modifications could include physical changes (strainer replacement, insulation replacement, buffer change, degraded coatings remediation, etc.) and/or procedural changes (securing pumps earlier in the event, initiating hot leg recirculation earlier, etc.). If a plant modification is necessary, various options can be evaluated to determine the impact on risk, as well as the cost-benefit of implementing each option. Note that even if Step 6 shows that the risk is acceptably low, it may be beneficial to evaluate and implement some minor modifications that significantly reduce risk.

  For VEGP, a combination of physical and procedural modifications were determined to be beneficial. As described in Enclosure 2, Section 3.j, the RHR strainer height will be reduced by removing disks, and the procedures for switching over from refueling water storage tank (RWST) injection to sump recirculation are being modified to inject more water into containment for breaks that do not initiate containment sprays. These modifications will help ensure that the strainers are completely submerged for an increased number of

postulated LOCA scenarios, which significantly reduces the likelihood of debris-related failures.

- **Step 8: Output to submittal documentation**

  Once the risk has been determined to be acceptable, a licensing submittal can be prepared to document the risk quantification, defense-in-depth, safety margin, etc. This submittal is documented for VEGP in Enclosures 1 through 5.

2.0 Scope of Risk Assessment

As discussed in Enclosure 1, Sections 2.0 and 3.0, the scope of the risk model includes:

- Reactor coolant system (RCS) pipe breaks resulting in small, medium, and large LOCAs (includes 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 breaks due to seismically-induced LOCAs.
- SSBIs that result in a consequential LOCA upon failure to terminate safety injection or a stuck open power operated relief valve (PORV) (includes DEGBs at least every 5 ft on main steam line and feedwater line piping inside containment)

Water hammer LOCAs, non-piping LOCAs, and breaks past the first isolation valve were qualitatively addressed or screened out as described in Enclosure 1. In addition, as discussed in Enclosure 1, the risk model was evaluated for Mode 1, which is bounding for Modes 2 through 6.

3.0 Failure Mode Identification

The specific GSI-191 failure modes that were included in the risk model are:

- Strainer head loss exceeds the net positive suction head (NPSH) margin for the RHR and CS pumps when the strainer is fully submerged.
- Strainer head loss exceeds half of the submerged strainer height when the strainer is partially submerged.
- Strainer head loss exceeds the strainer structural margin.
- Gas voids downstream of the strainers exceed the acceptable void fraction limits of the RHR or CS pumps.
- Debris accumulation on the strainer exceeds tested debris limits. Although this by itself is not a unique failure mode, it represents an unknown head loss condition where any of the failure modes above may occur.
- Debris accumulation in the core exceeds debris limits for core blockage and boric acid precipitation.

Note that containment sprays are not required for containment cooling. Therefore, although CS pump failures were evaluated in the NARWHAL model, these failures were

not included in the CFP results. As discussed in Enclosure 1, Section 1.0, other failure modes (upstream blockage, vortexing, ex-vessel downstream effects, and the LOCA deposition model (LOCADM) portion of the analysis of in-vessel effects) were addressed in a bounding manner for the range of possible breaks with no issues of concern, and were therefore not explicitly modeled in NARWHAL. Note that no significant direct or indirect effects associated with these excluded failure modes have been identified with respect to the risk model.

4.0 PRA Model Changes

To perform the risk-informed GSI-191 evaluation, a few relatively minor changes were required for the base PRA model to incorporate the events for the GSI-191 sump strainer and core blockage failures, along with the associated LOCA initiating events and equipment configurations.

The following configurations were represented in the NARWHAL evaluations to determine CFPs for GSI-191 related strainer and core failures, and were included in the modifications to develop the VEGP GSI-191 PRA model:

- No equipment failed (all ECCS trains operating)
- One RHR train (RHPA or RHPB) failed
- One charging (CP) train (CPPA or CPPB) failed
- One SI train (SIPA or SIPB) failed
- One nuclear service cooling water (NSCW) train (NSCWA or NSCWB) failed, causing failure of one ECCS train
- One CS train (CSPA or CSPB) failed
- Two CS trains (CSPA and CSPB) failed

Each potential sump strainer and core blockage failure can be represented in the PRA model with a basic event that is combined with the appropriate LOCA initiating event and pump failure logic to represent the equipment configurations listed above. Some simplifications of the modeled configurations were performed based on the assumptions described below:

- The CS pumps were assumed not to actuate during a medium or small LOCA; thus, configurations with one or both CS trains failed were modeled for large LOCA initiating events, and were also modeled for the medium LOCA initiating events for sensitivity purposes only.
- CS is not required to be modeled in the VEGP PRA for containment cooling; therefore, the sump strainers for the CS pump recirculation suction, and associated GSI-191 failures, were also not included in the VEGP GSI-191 PRA model.
- Because the charging and SI pumps piggyback off the RHR pumps, failure of one charging or one SI train has the same impact on debris transport to the RHR sump strainer as the configuration with all ECCS trains operating; therefore, these configurations were grouped together.

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The VEGP base PRA model currently includes events to represent independent and common cause plugging of ECCS containment sumps A and B. For convenience, these existing sump-plugging events were combined under new logic gates NON-GSI-191-A and NON-GSI-191-B. These existing sump plugging events may ultimately be removed from the VEGP PRA model of record and replaced with the more detailed representation of GSI-191 sump strainer failure logic described below.

The logic developed for GSI-191 RHR sump strainer failures was modeled under new fault tree gates GSI-191-SUMP-A, GSI-191-SUMP-B, and GSI-191-SUMP-AB. Although the sump strainer CFPs may be zero for some of the scenarios included in the logic, these events were retained for potential sensitivity studies. The new logic gate for GSI-191 failure of RHR Strainer A (GSI-191-SUMP-A) was included under the existing PRA model gates for loss of flow from the ECCS containment sump A (ECCS-SUMP-A and ECCS-SUMP-A-ACR). Similarly, the new logic gate for GSI-191 failure of RHR sump strainer B (GSI-191-SUMP-B) was included under the existing PRA model gates for loss of flow from the ECCS containment sump B (ECCS-SUMP-B and ECCS-SUMP-B-ACR). Finally, the new logic gate for GSI-191 failure of both RHR sump strainers A and B (GSI-191-SUMP-AB) was included under the existing PRA model gates for loss of flow from sump A and from sump B (ECCS-SUMP-A, ECCS-SUMP-A-ACR, ECCS-SUMP-B, and ECCS-SUMP-B-ACR).

The logic developed for GSI-191 core blockage was modeled under new fault tree gate GSI-191-CORE. As with the sump strainer CFPs, the core blockage CFPs may be zero for some of the scenarios included in the logic; however, these events were also retained for potential sensitivity studies. The new logic gate for GSI-191 core blockage (GSI-191-CORE) was added under the existing PRA model top gate for core damage (CDF-TOTAL).

A GSI-191 core blockage logic gate was also added under the existing PRA model gates for LERF end states 01 through 08 (gates LERF-01 through LERF-08). End states LERF-01 through LERF-06 can result from small LOCAs only; therefore, the small LOCA core blockage gate (GSI-191-CORE-SL) was combined with the containment event tree logic associated with each of those end states (e.g., gate LERF01X…LERF06X). End state LERF-07 can result from medium or large LOCAs; therefore, the medium and large LOCA core blockage gates (GSI-191-CORE-ML “OR” GSI-191-CORE-LL) was combined with the containment event tree logic for end state LERF-07 (gate LERF07X). Finally, end state LERF-08 can result from small, medium, or large LOCAs; therefore, the core blockage gate for all LOCAs (GSI-191-CORE) was combined with the containment event tree logic for end state LERF-08 (gate LERF08X).

Proceduralized operator actions for switching from RWST injection to sump recirculation, switching from cold leg recirculation to hot leg recirculation, and securing containment sprays were all included in the risk-informed GSI-191 evaluation. However, no operator actions were credited to recover from the effects of debris-related failures on the strainer or the core in the ΔCDF and ΔLERF calculation.

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4.1 Seismic PRA Model

To quantify the contribution of seismically-induced LOCAs to GSI-191 risk, the initiating event frequency for such LOCAs was determined by considering the seismic fragilities for several RCS components whose failure could cause a LOCA. While LOCAs are generally related to piping failures for the internal events PRA, the piping fragility analyses for VEGP demonstrated that the RCS loop piping and the Nuclear Class 1 piping inside containment have high seismic capacity. Other systems with LOCA sensitive piping, including the chemical volume and control system (CVCS), RHR, SI, reactor vessel head vent, and reactor vessel level indication system (RVLIS), were also considered for the potential occurrence of a seismically-induced LOCA within the size range of concern. This ensured seismically-induced pressure boundary failure was considered for a range of components to identify the bounding median seismic capacity for potential LOCAs.

For development of a seismically-induced LOCA frequency, the seismic capacity of the reactor coolant pump (RCP) was selected. The RCP has the lowest seismic capacity among the nuclear steam supply system (NSSS) components. The selected bounding seismic capacity is based on an indirect seismically-induced LOCA due to the failure of the RCP column assembly support, which provides the least seismic margin of safety. Due to the flexibility and strength of the primary system piping and its supports, and the uncertainty attached to the postulated break size, this indirect seismically-induced failure could range in size from a small LOCA to a large LOCA. Therefore, the failure frequency is divided equally among the three LOCA initiating event sizes modeled in the VEGP PRA.

The VEGP internal events PRA model modifications made to perform the risk-informed GSI-191 evaluation were used as a guide to make corresponding modifications to the VEGP seismic PRA model. These modifications included the incorporation of the CFPs calculated by NARWHAL for strainer and core failures due to the effects of debris generated by a large break LOCA. It is assumed that these CFPs are also applicable to a (direct or indirect) seismically-induced LOCA occurring in one location. The indirect seismic LOCAs that are postulated are for a break in the cold leg or crossover leg piping next to any one of the RCPs. The RCP break location is postulated based on a collapse of the RCP support structure following a seismic event. The resulting break could be any size from a small break up to a DEGB of the cold leg or crossover leg.

The CFP values were calculated based on a range of break sizes (from ½ inch to a DEGB) postulated at all Class 1 ISI welds within the first isolation valve. The NARWHAL evaluation showed that none of the small and medium breaks generate sufficient debris quantities to cause GSI-191 failures (including small and medium breaks on the cold legs and crossover legs). Many large breaks also do not generate sufficient debris quantities to cause failures. The large break GSI-191 CFP values were calculated based on the LOCA frequencies for a random pipe break (with a higher frequency for 6-inch breaks and a much lower frequency for 31-inch DEGBs). Although the frequency as a function of break size would likely be different for a seismically-induced LOCA, it is

reasonably conservative to use the large break CFP values, because a seismic event could result in a small or medium break that would not cause any debris related failures.

A seismic event also has the potential to dislodge insulation from piping and components inside containment. The insulation is contained inside a fabric cover; therefore, any dislodged insulation would be expected to fail in relatively-large intact pieces that would be unlikely to transport to the sumps or have a significant effect on the strainers.

5.0 Sub-Model Development

The evaluation of failures related to the effects of debris was performed outside the VEGP PRA model using the NARWHAL software. The VEGP NARWHAL model includes the following sub-models:

- Post-accident conditions
  - Plant configuration
  - Plant states
  - Random equipment failures
  - Water volume and level
  - Flow rates
  - Pressure and temperature
  - pH
- Debris sources
  - Zone of influence (ZOI)-generated insulation, fire barrier, and qualified coatings debris
  - Unqualified coatings debris
  - Latent debris
  - Miscellaneous debris
- Chemical effects
  - Release via corrosion/dissolution
  - Solubility
  - Precipitate debris quantity
- Debris transport
  - Blowdown
  - Washdown
  - Pool fill
  - Recirculation
  - Erosion
- Strainer head loss
  - Debris groups
  - Clean strainer head loss (CSHL)
  - Conventional debris head loss
  - Chemical debris head loss
  - Head loss correction
  - Head loss extrapolation

- Strainer air intrusion
- Strainer and pump acceptance criteria
  - Strainer flashing
  - Strainer structural margin
  - Strainer partial submergence limit
  - Pump void fraction limit
  - Pump NPSH margin
  - Debris limits
- Strainer penetration
  - Fiber
  - Particulate
- Core acceptance criteria

These sub-models are described in more detail in the following sections.

6.0 Scenario Development

The post-accident conditions are described in the following sub-sections. These conditions include the plant configuration, plant state changes (due to automatic and manual operator actions), and various thermal-hydraulic parameters that were used in the VEGP NARWHAL model.

For all breaks evaluated, a 30-day mission time was used in the NARWHAL model. This is consistent with the mission time used for deterministic GSI-191 evaluations (Reference 2). Note that the VEGP PRA model uses the typical PRA mission time of 24 hours. Any RHR strainer and core failures predicted by NARWHAL (regardless of failure time) were included in the CFP values that were used in the GSI-191 PRA calculation. Although additional operator actions and compensatory measures (such as refilling the RWST) could be taken to mitigate late failures, these actions were not included in either the GSI-191 PRA model or the NARWHAL model.

6.1 Plant Configuration

The plant configuration determines the flow paths during the different phases of accident mitigation. At VEGP, there are a total of four separate sump strainer assemblies and two engineered safety features (ESF) trains. An ESF train consists of pumps associated with the ECCS and the CSS. A single train of ECCS consists of a high head centrifugal charging pump, a medium head SI pump, and a low head RHR pump. In addition to these pumps, each train of ECCS contains two SI accumulators. A single train of CSS contains a CS pump. There are two strainers dedicated to each ESF train, one for the ECCS pumps and one for the CS pump. Note that the strainers dedicated to the ECCS pumps are referred to as RHR strainers, and the strainers dedicated to the CSS are referred to as CS strainers.

Figure 3-2 shows the connection diagram from NARWHAL that was used to model VEGP. Valves are used in NARWHAL to define whether a flow path is currently active.

For example, each pump has a connection to the RWST and the strainer. During injection, the valves connecting the RWST to each of the pumps would be open and the valves connecting the pumps to the sump would be closed. The activity state and flow paths associated with each pump are discussed further in Section 6.2.

As shown in the figure, the equipment was arranged in a manner that is consistent with the plant-specific emergency operating procedures (EOPs). The ECCS pumps operate in parallel while taking suction from the RWST, and the charging and SI pumps take suction from the RHR discharge during recirculation. Additionally, these pumps can provide flow to the RCS through either the cold legs or the hot legs. The accumulators are aligned to provide rapid cooling through the cold legs. The CS pumps provide flow to the CS headers and can take suction from the RWST during injection and the sump during recirculation.

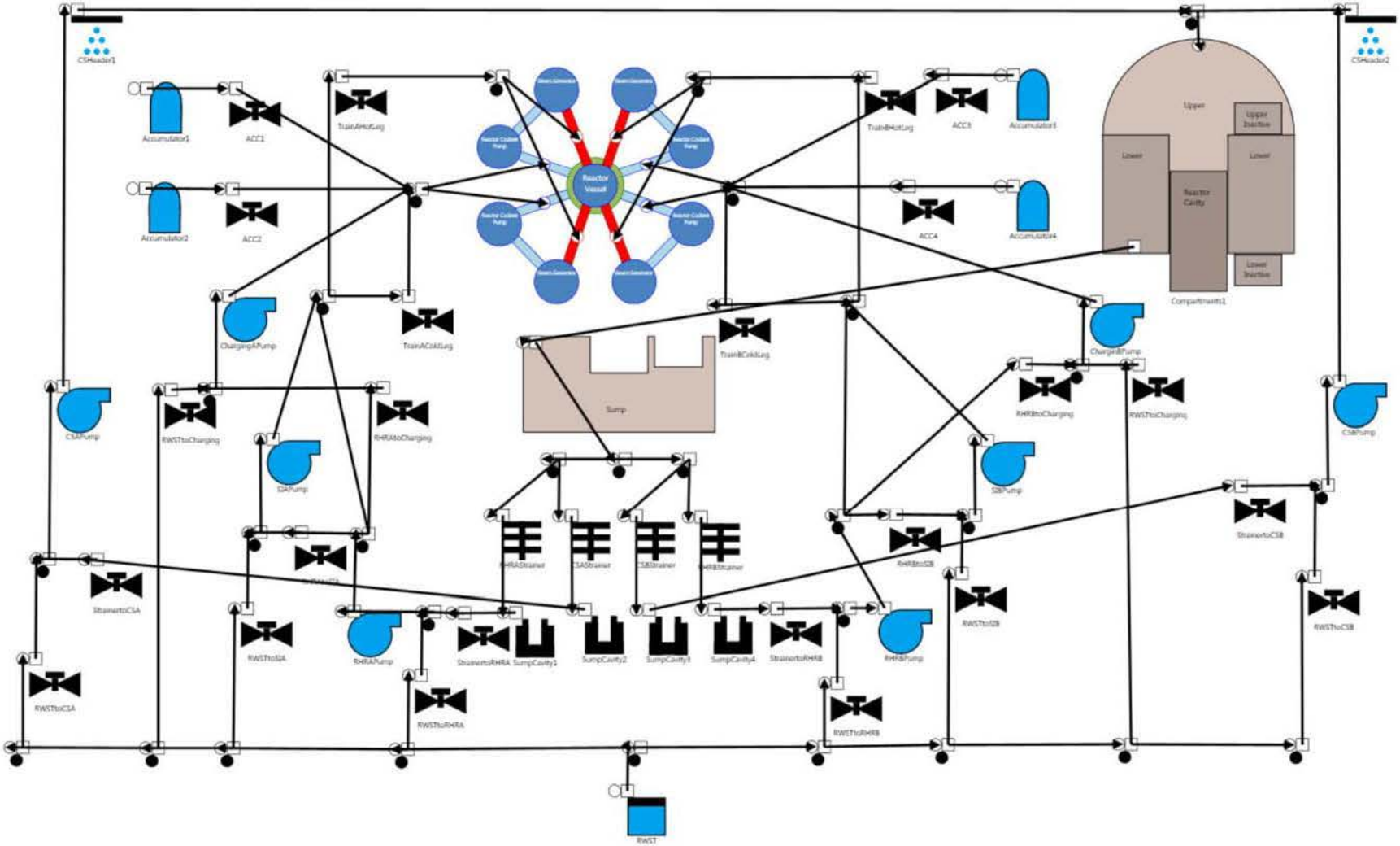

Figure 3-2 – VEGP Connection Diagram

6.2 Plant States

A plant state represents each of the different plant modes of post-accident operation and is defined by the activity state of the valves and pumps in the connection diagram. The plant states are consistent with the plant EOPs and are a function of the initiating event. For VEGP, two procedures mandate unique states as a function of break size and break side (i.e., hot leg vs. cold leg side). These procedures address SI accumulator injection and CS activation. As discussed in Section 13.2, the SI accumulators were modeled to only inject for breaks greater than or equal to 2 inches. Also as discussed in Section 13.2, the containment sprays were modeled to activate for hot leg breaks larger than 15 inches and operate for a duration of 24 hours. Note that different states are required for cold leg (CL) recirculation and hot leg (HL) recirculation. During HL recirculation, the RHR and SI pumps are aligned to discharge into the HLs, while the charging pumps continue to discharge into the CLs. Because the charging pump alignment after switchover to HL recirculation does not have any effect on the overall results, the NARWHAL model was simplified to align all ECCS pumps to discharge into the HLs during HL recirculation.

Accident mitigation at VEGP was modeled using the states shown in Table 3-1 and Table 3-2. Note that Table 3-1 describes the initiators that dictate which breaks enter a given state, and Table 3-2 describes the operating equipment during each state. In Table 3-1, the term “Full Recirc” is used to describe the state in which all active pumps are taking suction through the sump strainers (or in piggy-back mode).

Table 3-1 – Plant State Initiators

State Name	Min Break Size	Max Break Size	Break Side	Initiating Variable
Injection (small break no CS)	0	1.999	Both	Time = 0 minutes
Injection (2<Break<15, no CS)	1.999	15.001	Both	Time = 0 minutes
Injection (CL Break> 15, no CS)	15.001	31.001	Cold	Time = 0 minutes
Injection (CS)	15.001	31.001	Hot	Time = 0 minutes
RHR Recirc (CS)	15.001	31.001	Hot	ECCS Switchover Level in RWST (CS active)
Full Recirc (Break < 15, no CS)	0	15.001	Both	ECCS Switchover Level in RWST (CS inactive)
Full Recirc (CL Break > 15, no CS)	15.001	31.001	Cold	ECCS Switchover Level in RWST (CS inactive)
Full Recirc (CS)	15.001	31.001	Hot	CSS Switchover Level in RWST
Hot Leg Switchover (Break < 15, no CS)	0	15.001	Both	Time = 450 minutes

State Name	Min Break Size	Max Break Size	Break Side	Initiating Variable
Hot Leg Switchover (CL Break > 15, no CS)	15.001	31.001	Cold	Time = 450 minutes
Hot Leg Switchover (CS)	15.001	31.001	Hot	Time = 450 minutes
CS Termination	0	31.001	Both	Time = 1440 minutes

Table 3-2 – Plant State Component Activity

State Name	Component	Activity	Description
Injection (small break no CS)	RHR Pumps	Active	Suction from RWST
	SI Pumps	Active	Suction from RWST
	Charging Pumps	Active	Suction from RWST
	CS Pumps	Inactive	Not Operating
	SI Accumulators	Inactive	Not Operating
	Reactor Vessel Flow	CL	N/A
Injection (2<Break<15, no CS) & Injection (CL Break>15, no CS)	RHR Pumps	Active	Suction from RWST
	SI Pumps	Active	Suction from RWST
	Charging Pumps	Active	Suction from RWST
	CS Pumps	Inactive	Not Operating
	SI Accumulators	Active	Injection into CL
	Reactor Vessel Flow	CL	N/A
Injection (CS)	RHR Pumps	Active	Suction from RWST
	SI Pumps	Active	Suction from RWST
	Charging Pumps	Active	Suction from RWST
	CS Pumps	Active	Suction from RWST
	SI Accumulators	Active	Injection into CL
	Reactor Vessel Flow	CL	N/A
RHR Recirc (CS)	RHR Pumps	Active	Suction from Sump
	SI Pumps	Active	Suction from RWST
	Charging Pumps	Active	Suction from RWST
	CS Pumps	Active	Suction from RWST
	SI Accumulators	Inactive	N/A
	Reactor Vessel Flow	CL	N/A
Full Recirc (Break<15, no CS) & Full Recirc (CL Break>15, no CS)	RHR Pumps	Active	Suction from Sump
	SI Pumps	Active	Suction from RHR
	Charging Pumps	Active	Suction from RHR
	CS Pumps	Inactive	Not Operating
	SI Accumulators	Inactive	N/A

State Name	Component	Activity	Description
	Reactor Vessel Flow	CL	N/A
Full Recirc (CS)	RHR Pumps	Active	Suction from Sump
	SI Pumps	Active	Suction from RHR
	Charging Pumps	Active	Suction from RHR
	CS Pumps	Active	Suction from Sump
	SI Accumulators	Inactive	N/A
	Reactor Vessel Flow	CL	N/A
Hot Leg Switchover (Break<15, no CS) & Hot Leg Switchover (CL Break>15, no CS)	RHR Pumps	Active	Suction from Sump
	SI Pumps	Active	Suction from RHR
	Charging Pumps	Active	Suction from RHR
	CS Pumps	Inactive	Not Operating
	SI Accumulators	Inactive	N/A
	Reactor Vessel Flow	HL	N/A
Hot Leg Switchover (CS)	RHR Pumps	Active	Suction from Sump
	SI Pumps	Active	Suction from RHR
	Charging Pumps	Active	Suction from RHR
	CS Pumps	Active	Suction from Sump
	SI Accumulators	Inactive	N/A
	Reactor Vessel Flow	HL	N/A
CS Termination	RHR Pumps	Active	Suction from Sump
	SI Pumps	Active	Suction from RHR
	Charging Pumps	Active	Suction from RHR
	CS Pumps	Inactive	Not Operating
	SI Accumulators	Inactive	N/A
	Reactor Vessel Flow	HL	N/A

6.3 Random Equipment Failures

Random equipment failures are defined as failure to start or failure to run due to issues unrelated to GSI-191. The Vogtle PRA model was used to identify the high-likelihood equipment configurations that can occur in response to a LOCA, to focus the analysis of debris generation, transport, and resulting GSI-191 phenomena. The identification of high-likelihood equipment configurations followed a systematic process:

1. All possible combinations of system/train failures that affect the likelihood of debris-induced failure of ECCS following a LOCA were identified. The configurations considered were specific to each LOCA size (small, medium, or large). Examples of such configurations include successful operation of all ECCS equipment (e.g., RHR, containment spray, and containment coolers

following a large LOCA), failure of one train of RHR, failure of both trains of containment spray, etc.

2. For each possible configuration, the functional failure probability (FFP) was quantified for each system/train failed in that configuration, using the associated logic in the PRA model. Those FFP values were then used to calculate total probability for each equipment failure combination.
3. For each possible configuration identified for each LOCA size, the annual scenario frequency was calculated based on the associated initiating event frequency and total probability of the equipment failure configuration. The scenario frequencies for each LOCA size were summed and the significant scenarios were identified as those that comprised 95%, or individually contributed 1%, of the total frequency. The equipment failure combinations from those significant scenarios represent the high-likelihood configurations.

Based on symmetry and the inputs used in the VEGP NARWHAL model, pump failures in one train are analytically identical to the same pump failures in the other train. In addition, although there is a slight difference during the RWST injection phase, the VEGP NARWHAL CFP calculation showed that the GSI-191 CFP results (i.e., the breaks that fail) were identical for the cases with no equipment failures, a single charging pump failure, and a single SI pump failure, because the charging and SI pumps piggyback off of the RHR pumps during recirculation.

The switchover of the CS pumps from RWST injection to sump recirculation requires a manual operator action. Due to the human failure probability associated with this action, the probability of losing both CS pumps at the start of recirculation is higher than the probability of a single CS pump randomly failing to start or failing to run.

Table 3-3 shows the functional failure probabilities for the different equipment configurations from the VEGP GSI-191 PRA model. Note that these overall probabilities are based on the logic described above and have been normalized to 100 percent.

Table 3-3 – Functional Failure Probabilities

Equipment Configuration	Functional Failure Probability
No Equipment Failures	91.50%
2 CS Pump Failures	5.31%
1 RHR Pump Failure	1.46%
1 CS Pump Failure	1.26%
1 RHR Pump + 1 CS Pump Failures	0.39%
1 RHR Pump + 2 CS Pump Failures	0.07%
Total	**100%**

Enclosure 3

Risk Quantification

The high-likelihood equipment configurations were included in the detailed NARWHAL analysis for GSI-191 effects and were explicitly modeled in the PRA for quantification of the risk impacts due to GSI-191 phenomena.

The remaining low-likelihood configurations also have a risk impact from GSI-191 phenomena, but were not evaluated in detail. Instead, a bounding risk impact was calculated for the low-likelihood configurations. Calculation of the bounding risk impact for the low-likelihood configurations also followed a systematic process, as described below.

1. Because NARWHAL analysis was not performed to determine sump strainer or core cooling conditional failure probability (CFP) values for all low-likelihood configurations, a representative or bounding CFP was selected for each low-likelihood configuration using the following logic:

    i. Each low-likelihood configuration was binned based upon the impact on sump strainer debris accumulation, relative to the high-likelihood configurations evaluated in NARWHAL. For example, the failure of a containment cooler was captured by bounding temperature profiles used in the NARWHAL analysis, implicitly assuming the containment cooler failed. Therefore, low-likelihood configurations with containment cooler failure (e.g., RHR train A and containment cooler A failed) were considered equivalent to the high-likelihood configuration with only RHR train A failed.
    ii. Based on the binning performed, the appropriate CFP was selected to use as a representative or bounding value for the low-likelihood configuration. For example, the low-likelihood configuration with one RHR train failed and one CS train failed, has the same impact on sump strainer accumulation as the high-likelihood configuration with one ECCS train failed due to loss of nuclear service cooling water (NSCW). Therefore, the CFP calculated for one ECCS train failure was applied to the configuration with one RHR and one CS train failed.
    iii. If there was no similar high-likelihood configuration for a low-likelihood configuration, such as failure of one RHR train and both CS trains, a CFP of 1.0 was assumed.

2. For each low-likelihood configuration identified for each LOCA size, the bounding annual core damage frequency (CDF) was calculated based on the associated initiating event frequency, probability of the equipment failure configuration, and the selected representative or bounding CFP. The bounding CDF values from the low-likelihood configurations for each LOCA size were then summed. The base case CDF (assuming no GSI-191 failures) was neglected, so the sum of the bounding CDF from the low-likelihood configurations was assumed to be the risk increase (ΔCDF).

3. The bounding LERF due to low-likelihood configurations was calculated by multiplying the bounding CDF by the conditional large early release probability (CLERP) given core damage. The CLERP was obtained by dividing the base case (i.e., the case with no GSI-191 failures) LERF by the base case CDF. (Note that the CLERP for the high-likelihood configurations is identical to that for the base case with no GSI-191 failures, 2.9E-03, so this approach to calculate bounding LERF for the low-likelihood configurations is reasonable.) The bounding LERF from the low-likelihood configurations was assumed to be the risk increase (ΔLERF).

6.4 LOCA Frequencies

Table 3-4 shows the mean LOCA frequencies taken from the VEGP GSI-191 PRA model. As discussed in Enclosure 1, Section 3.0, the medium and large LOCA frequencies are based on the geometric aggregation from NUREG-1829 (Reference 3).

Table 3-4 – Mean LOCA Frequencies from GSI-191 PRA Model

Break Size	Exceedance Frequency (yr^{-1})
0.375	4.73E-04
2	1.39E-04
6	1.85E-06
31	1.50E-08

Note that the frequencies were used to calculate conditional failure probability as a function of PRA initiating event break size ranges. There are three break size ranges evaluated in the VEGP PRA. The small LOCA category comprises random breaks in the RCS in the range of 3/8-inch to 2-inch equivalent diameter. The medium LOCA initiating event is defined as a break in the RCS that is greater than or equal to 2 inches and less than 6 inches equivalent diameter. The large LOCA initiating event is defined as a break in the RCS that is greater than or equal to 6 inches up to a DEGB of the largest pipe (31 inches) in the RCS. Because the equivalent break size for a 31-inch DEGB is 43.8 inches ($31*\sqrt{2} = 43.84$), the frequency for each PRA size category is:

- $F_{SLOCA} = F_{0.375"} – F_{2"}$ = 4.73E-04 – 1.39E-04 = 3.34E-04 yr^{-1}
- $F_{MLOCA} = F_{2"} – F_{6"}$ = 1.39E-04 – 1.85E-06 = 1.37E-04 yr^{-1}
- $F_{LLOCA} = F_{6"} – F_{43.8"}$ [1] = 1.85E-06 – 0 = 1.85E-06 yr^{-1}

[1] Because the exceedance frequency for a 43.8-inch break is not available, the frequency was set to zero, which conservatively maximizes the frequency for large LOCAs.

6.5 Water Volume and Level

The height of the pool is a function of the total quantity of water in the sump. The pool level is calculated as a linear function of pool volume in gallons.

$$\mathrm{H_{pool}} = 1.20071 * 10^{-5} \left(\frac{\mathrm{ft}}{\mathrm{gal}}\right) * \mathrm{V_{pool}} - 0.058\ \mathrm{ft} \qquad \text{Equation 1}$$

Nomenclature:

H_{pool} = the height of the sump pool in ft
V_{pool} = the volume of the recirculation pool in gallons

At VEGP, three sources of water contribute to the recirculation pool inventory. These sources are the RWST, the SI accumulators, and the RCS.

- The total quantity of water delivered from the RWST is the difference between the initial and final levels. There are two final levels that are important in the analysis, the low-low level alarm (ECCS recirculation level) and the empty alarm (CS recirculation level). Note that this is different for breaks that do not activate containment sprays. The empty level alarm is the ECCS recirculation level for these breaks.
- Four SI accumulators provide immediate cooling to the core for breaks large enough to rapidly depressurize the RCS.
- A portion of the water initially in the RCS will be released as steam or spill to the pool through the break opening at the beginning of a LOCA.

Several hold-up volumes reduce the sump pool height:

- The amount of water held up as steam (vapor) in the containment atmosphere is a function of time.
- During recirculation, the total hold-up in the RCS is a function of break size and elevation.
- The containment spray falling through the containment building represents a transitory hold-up volume. Note that this hold-up was only applied while containment spray pumps were operating. After containment spray was terminated, this quantity was returned to the sump pool.
- The break flow falling through containment also represents a transitory hold-up volume that was applied to all breaks.
- The volume of the CS discharge piping is another hold-up that was applied for breaks that initiate containment sprays.
- Other miscellaneous hold-up volumes include the containment sump pits, the elevator pit, and the containment floor drains.

- The reactor cavity and in-core tunnel is one of the largest hold-up volumes. Due to the restricted flow paths into the reactor cavity, this hold-up volume was applied in a time-dependent manner based on the break location and whether containment sprays are activated.

The long-term water level is high enough to fully submerge the RHR and CS strainers for all breaks. For some reactor cavity breaks, there is a short period where the RHR strainers are not fully submerged just after the RHR pumps are switched to recirculation. However, for these breaks, the water level rises enough to submerge the strainers before the CS pumps finish drawing down the RWST.

6.6 Flow Rates

The pump flow rates that were used in the VEGP NARWHAL model are design flow rates for the SI pumps, charging pumps, and CS pumps. A flow rate approximately 20 percent higher than the design value was used for the RHR pumps.

The break flow rate is the sum of the flow through the RHR, SI, and charging pumps during RWST injection, and is the sum of the RHR pump flow rates during recirculation (when the charging and SI pumps are piggybacking off of the RHR pumps). No credit was taken for reduced RHR flow rates for smaller break sizes.

Note that for secondary side breaks in a feed and bleed scenario, only the charging and containment spray pumps were assumed to be active, which results in a significantly lower flow rate.

As discussed in Section 6.5, the reactor cavity and in-core tunnel have restricted flow paths. Because of the restricted flow paths and the fact that the volume is fairly large, a function was implemented to model the flow rate into the reactor cavity as a function of time. Note also that the flow rate into the reactor cavity is a function of break location.

For the breaks in the reactor cavity, the entire cavity is assumed to fill before the start of recirculation. In addition, the total reactor cavity hold-up volume is larger due to the flow rate inside of the cavity and the height of the flow paths that connect the cavity to the sump. For breaks outside the reactor cavity, the cavity would fill relatively slowly and would not be completely filled until after the start of recirculation.

The core boil-off flow rate used in the NARWHAL model was calculated based on the core power and the ANSI/ANS-5.1-1979 decay heat curve (Reference 4). An additional 20 percent margin was added to the boil-off flow rate.

6.7 Pressure and Temperature

The sump temperature, containment temperature, and containment pressure profiles were used in NARWHAL to determine time-dependent thermal-hydraulic properties.

Design basis temperature profiles were used for both sump temperature and containment temperature. These temperature profiles were based on a DEGB in the primary loop piping with minimum safeguards, and therefore represent the maximum temperature profiles. Although the actual temperature profiles would be significantly lower for smaller break sizes, the same temperature profiles were conservatively used for all break sizes. Note that there are competing factors associated with sump temperature, which could result in a lower temperature resulting in more failures. However, based on sensitivity analysis (see Section 14.2.2), it was determined that maximizing the temperature is more conservative for the VEGP model.

Containment accident pressure was not credited for NPSH margin calculations in the VEGP NARWHAL model. Because the technical specification minimum containment pressure is -0.3 psig at VEGP, the containment pressure profile was specified to be saturation pressure at pool temperatures above 210.96 degrees F, and 14.396 psia at pool temperatures below 210.96 degrees F.

For the purpose of degasification and flashing calculations, 3.5 psi of accident pressure was credited for the first 2.5 hours of the event. Both flashing and degasification are most problematic when the sump temperature is near or above 212 degrees F. The pool temperature is greater than 212 degrees F for approximately the first 120 minutes after a LOCA. Additional details are provided in Enclosure 2 Sections 3.f.14 and 3.g.14.

6.8 Sump and Spray pH

The maximum sump recirculation pH was conservatively rounded up to 7.8 for the VEGP NARWHAL model. Use of the maximum pH provides bounding chemical release quantities from submerged materials.

The minimum sump pH was rounded down to 7.0 for the VEGP NARWHAL model. The minimum sump pH was used to calculate the aluminum solubility limit in NARWHAL. Note that using different pH values to calculate release and solubility results in an over-prediction of the actual precipitate quantity.

The spray pH during RWST injection was specified to be 5.72, which is the maximum pH of the RWST. Using the maximum RWST pH maximizes chemical release from unsubmerged sources during the injection phase. After switchover to recirculation, the spray pH is equivalent to the sump pH.

7.0 Debris Sources

As described in Enclosure 2, Section 3.a.3, the types of debris in the NARWHAL model include Nukon fiberglass insulation, Interam fire barrier, and qualified epoxy and IOZ coatings debris generated inside the ZOI. The model also includes unqualified epoxy, IOZ, and alkyd coatings, latent dirt/dust and fiber, and miscellaneous debris, which are present in containment or generated by the post-accident environmental conditions.

The debris sources generated inside the ZOI range from essentially zero debris for the smallest break sizes up to 2,229 ft^3 of Nukon, 60 lbm of Interam, 2.1 ft^3 of qualified epoxy, and 0.3 ft^3 of qualified IOZ debris for the bounding breaks.

The other debris sources are independent of the break location and size and were therefore applied to all breaks. The debris quantities used in the NARWHAL model (including some operating margin) were 30.4 ft^3 for unqualified epoxy, 0.4 ft^3 for unqualified IOZ, 0.5 ft^3 for unqualified alkyd, 30 lbm for latent fiber, 170 lbm for latent dirt/dust, and 50 ft^2 for miscellaneous debris.

A four-category size distribution (fines, small pieces, large pieces, and intact blankets) was used for the Nukon debris based on the guidance in the appendices of NEI 04-07 Volume 2 (Reference 2). A more conservative two-category size distribution (fines and small pieces) was used for the Interam debris. All of the qualified and unqualified coatings debris was conservatively treated as fines. The latent debris was also treated as fines. The miscellaneous debris was simply treated as a reduction in the total strainer surface area. Note that 25 percent overlap of the miscellaneous debris was credited in the NARWHAL model, which is consistent with the guidance in NEI 04-07 Volume 2 (Reference 2).

Debris from all sources was essentially treated as being generated at the beginning of the event. The miscellaneous debris surface area reduction was applied prior to other debris transporting to the strainer. Unqualified coatings were modeled as failing after the pool fill phase (no transport to inactive cavities), but were available to transport at the start of recirculation. This is described further in Section 9.0.

8.0 Chemical Effects

As described in Enclosure 2 Section 3.o.2, the formation of chemical products was analyzed as a function of the temperature, pH, and pool volume inputs, as well as debris quantities and exposed aluminum and concrete surface areas. There are two parts to the chemical product generation model: the elemental chemical release from materials in containment, and chemical precipitate formation. Note that these processes are based on break-dependent conditions and were therefore analyzed separately for each postulated break.

8.1 Elemental Chemical Release

The Nukon and Interam debris both contribute to chemical release, which was quantified in NARWHAL using the WCAP-16530 release equations (Reference 6) and the break-specific debris quantities. Note that Interam debris only releases silicon. Therefore, the Interam has no effect on the chemical product generation because the only aluminum precipitate that is tracked in the VEGP NARWHAL model is sodium aluminum silicate (SAS) (see Section 8.2), and NARWHAL conservatively assumes an infinite source of silicon when SAS is the only aluminum precipitate tracked (Reference 7).

The amount of elemental chemical release from a given debris source is limited by the quantity of that element within the source. E-Glass (which includes Nukon) has 1.95 percent aluminum and 2.16 percent calcium by mass.

The exposed surfaces include aluminum metal and concrete, which would either be submerged in the containment pool or exposed to containment sprays. The same surface areas were analyzed for each break. The chemical release from exposed concrete was evaluated using the WCAP-16530 release equations (Reference 6), and the chemical release from aluminum was evaluated using the University of New Mexico (UNM) release equations (Reference 8).

8.2 Chemical Product Formation

The chemical precipitates analyzed in the VEGP NARWHAL model were SAS and calcium phosphate. The calcium phosphate was modeled in NARWHAL as precipitating immediately when calcium is released in solution (Reference 7). The SAS precipitates when the concentration of aluminum in the pool exceeds the aluminum solubility limit as calculated with the Argonne National Laboratory (ANL) solubility equation (Reference 7). Note that if precipitation of SAS was not predicted before 24 hours, then precipitation was forced at that time. Also note that aluminum was assumed not to remain dissolved in the pool after precipitation occurred (i.e., the aluminum solubility limit was only credited for precipitate timing). Forcing precipitation at 24 hours, as well as not taking credit for aluminum remaining dissolved in the pool, are conservative factors in the chemical product formation model.

The effects of the chemical precipitates on strainer head loss are described in Section 10.0, and the effects on core blockage are described in Section 12.0.

9.0 Debris Transport

As described in Enclosure 2, Section 3.e, debris transport includes the transport of debris during the blowdown, washdown, pool fill, and recirculation phases, as well as debris erosion. Transport can vary significantly as a function of flow rate, water level, etc. However, in the VEGP NARWHAL model, many of these parameters were not

explicitly modeled and were conservatively represented using bounding conditions (e.g., the bounding flow rates for large break conditions were used to calculate recirculation transport fractions that were applied to breaks of all sizes). The specific factors affecting transport that were included in the VEGP NARWHAL model include debris type and size, break location (i.e., breaks in the steam generator compartment on the Loop 1/3 side, breaks on the Loop 2/4 side, breaks in the reactor cavity, breaks in the pressurizer compartment, or breaks in the annulus), whether sprays are initiated or not, and whether one or two trains are operating.

The blowdown transport fractions are a function of the break location, as well as the size of debris. The only debris transported during the blowdown phase would be debris generated inside the ZOI. Fine debris was transported with the blowdown flow, with no credit for retention on structures. The transport of small and large pieces of Nukon and small pieces of Interam was dependent on the break location as well as the location of grating that debris would have to be blown through to reach upper containment or the containment floor.

The washdown transport fractions were based on containment spray initiation as well as the size of debris. If containment sprays were initiated, 100 percent of fine debris was modeled as being washed down to the containment floor. Some credit was taken for small pieces of Nukon and Interam being retained in upper containment, and most large pieces of Nukon that were transported to upper containment were modeled as being retained in upper containment. If containment sprays were not initiated, the transport would be significantly reduced. However, 10 percent of fine debris was still modeled as washing down from upper containment due to condensation drainage.

For pool fill transport, a relatively small fraction of debris was modeled as transporting to inactive cavities and the ECCS strainers as the sump cavities were filled. These transport fractions were applied to all debris that was in the containment pool at the end of the blowdown phase. This includes debris generated inside the ZOI as well as latent debris, but not unqualified coatings.

Recirculation transport fractions were developed based on computational fluid dynamics (CFD) modeling. Several simulations were run to determine the transport fractions for the various types and sizes of debris corresponding to different break locations, number of trains operating, and whether containment sprays were initiated.

Small and large pieces of Nukon and Interam debris that are retained in upper containment would be subject to erosion due to containment sprays. A one percent erosion fraction was used for this debris for breaks where containment sprays were initiated. Similarly, small and large pieces of Nukon and Interam debris in the containment pool would be subject to erosion and a 10 percent erosion fraction was used. Because the debris size is important with respect to penetration, NARWHAL applies the pool erosion for both the debris that settles in the pool as well as the debris

that transports to the strainers (Reference 7). This conservatively maximizes the quantity of fine debris.

Each of the transport processes described above were used to determine the total quantity of debris that reaches the strainer. However, the timing was not assumed to be instantaneous for all of these processes. Blowdown was treated as an instantaneous process at the beginning of the event. Washdown was treated as occurring after the inactive and sump cavities were filled during the pool fill phase, but before the start of recirculation. Debris that was transported to the ECCS strainers during the pool fill phase was modeled on the strainers at the start of recirculation (note that a fraction of the fine fiber debris penetrates prior to the start of recirculation as described in Section 11.0). For VEGP, failure time was not credited for unqualified coatings, and the unqualified coatings debris was treated as being in the pool at the start of recirculation. Although erosion is a time-dependent process, all of the erosion fines were treated as being generated at the start of recirculation. The actual transport to the strainers during the recirculation phase was modeled as a time-dependent process, where debris arrives on the strainers as a function of the pool turnover time (i.e., as a function of the pool volume and strainer flow rates). The debris accumulation on each strainer was proportional to the flow split to each strainer. This flow split was evaluated at each time step. Therefore, the relative accumulation of debris changes over time when various pumps were switched from RWST injection to sump recirculation, containment sprays were secured, etc.

It is likely that debris transport to the two RHR strainers could be asymmetric due to different flow conditions in the vicinity of the two strainers. However, assuming symmetric transport to the strainers is conservative because it maximizes the potential for both strainers to fail. With asymmetric transport, it is more likely for one strainer to fail, but less likely for the other strainer to fail. In other words, by assuming symmetric transport, the NARWHAL model over-predicts failure of both RHR trains and under-predicts failure of a single RHR train. This is conservative because the PRA success criteria only require operation of a single RHR pump for large breaks, and reducing the likelihood of both RHR strainers failing due to the effects of debris will decrease the CDF.

The NARWHAL evaluation provides CFPs that are conditioned on specific equipment configurations (e.g., all pumps running or single train failure), which are based on random (non-GSI-191 related) equipment failures. The effects of random pump failures are conservatively addressed in the NARWHAL model by assuming that the failures occur at the start of recirculation. This assumption maximizes accumulation of debris on the active strainers and increases the CFP values associated with those equipment configurations (see Assumption 15 in Section 13.2). Changing the model to assume asymmetric transport and/or random equipment failures later in the event would reduce the conservatism in the model and result in lower CDF values.

10.0 Strainer Debris Impact Evaluation

As discussed in Enclosure 2 Section 3.f.4, the head loss associated with debris accumulation on the strainer was determined using the results of prototypical strainer module testing.

There are a total of four separate sump strainer assemblies for each unit at VEGP. There are two for the CSS (i.e., CS strainers) and two for the ECCS (i.e., RHR strainers). Each RHR and CS strainer assembly consists of four parallel vertical stacks connected to a plenum installed over each sump pit. The modified height of each RHR strainer (not including the curb) will be approximately 3.77 ft, and the effective surface area will be 677.6 ft^2. The height of each CS strainer (not including the curb) is approximately 3.3 ft, and the effective surface area is 590 ft^2.

10.1 Strainer Head Loss

The total strainer head loss for each time step was calculated by combining the CSHL, appropriate debris head loss, and extrapolation constant (as necessary), as outlined below.

NARWHAL uses a rule-based approach to calculate debris head loss based on the results of testing with a prototypical strainer module. For the VEGP NARWHAL model, if the fiber debris load on the strainer was less than the tested quantity from the thin bed test (corresponding to a theoretical uniform bed thickness of 0.57 inches), then the maximum thin bed test conventional head loss (0.625 ft) was returned. If the fiber quantity is greater than what was tested in the thin bed test, then the maximum full load conventional head loss (5.46 ft) was returned. It should be noted that the volume of debris required to transition from the thin bed head loss to the full load head loss is dependent on the effective strainer area. The strainer area is adjusted to account for the contribution of miscellaneous debris (see Section 7.0) and brief periods of time when the strainer is not fully submerged for some reactor cavity breaks (see Section 6.5). Therefore, the debris volume required to form a 0.57-inch thick bed is scaled appropriately based on the reduced strainer surface area.

The head loss effects of calcium phosphate and SAS were each analyzed separately from the conventional debris head loss. Chemical head loss was only applied if the fiber debris quantity on the strainer was greater than 0.45-inch equivalent (see Enclosure 2, Section 3.f.10). If this condition was met, and any calcium phosphate accumulated on the strainer, the head loss corresponding to the full quantity of calcium phosphate debris (1.11 ft) was added. Similarly, given the accumulation of sufficient fiber and any SAS on the strainer, the head loss corresponding to the full SAS debris quantity (5.24 ft) was added.

The test results were extrapolated to 30 days in accordance with the March 2008 NRC guidance (Reference 9) to account for chemical head loss that had not leveled off by the

end of the test. For convenience, the 30-day head loss extrapolation value (which represents a gradual increase over time) was instantaneously applied at 7.5 hours in the VEGP NARWHAL model. In addition, because the extrapolation constant was associated with the chemical head loss, this constant was only applied if the fiber bed was thick enough for the chemical head loss to be added (i.e., greater than 0.45 inches).

The combined debris head loss, which may include conventional debris head loss, chemical debris head loss and extrapolation constant as appropriate, was corrected based on the sump thermal-hydraulic conditions compared to the test conditions. As specified in the March 2008 NRC guidance (Reference 9), flow sweep data was used to develop the flow and temperature scaling. This scaling was performed at each time step using the time-dependent approach velocity and temperature in the VEGP NARWHAL model.

The corrected debris head loss was then combined with the CSHL to determine the total strainer head loss. The bounding CSHL of 0.375 ft at 4,500 gpm was applied in the VEGP NARWHAL model for all time steps.

Because the head loss results are only applicable for debris quantities up to what was tested, debris limits were specified in the VEGP NARWHAL model corresponding to the tested quantities. Separate debris limits were specified for fiber, particulate, Interam, calcium phosphate, and SAS debris. If any one of these debris limits was exceeded, the strainer was assumed to fail.

10.2 Degasification and Flashing

NARWHAL calculates degasification and flashing based on the total strainer head loss and other important parameters (containment pressure, sump temperature, water level, etc.) (Reference 7). The strainer was automatically assumed to fail if any flashing occurs. The degasification calculation results in a gas void fraction that was compared against the pump limits (a void fraction limit of 0.02 was used for all pumps). If the void fraction exceeds a pump limit, the pump was assumed to fail.

For the degasification and flashing calculations for each break, 3.5 psi of accident pressure was credited for the first 2.5 hours. Note that no accident pressure was credited for NPSH in the VEGP NARWHAL model. The midpoint of the strainer was used to calculate the average degasification across the entire strainer. The elevation at the stop of the strainer was used to evaluate flashing.

10.3 Structural and NPSH Margin

The NPSH margin is the NPSH available (excluding strainer head losses) minus the NPSH required. The NPSH available was calculated in the VEGP NARWHAL model based on the time-dependent containment pressure (without crediting accident

pressure), sump temperature, water level, and major and minor losses in the pump suction piping. As discussed in Section 6.7, a bounding pressure and temperature profile was used, which conservatively minimizes NPSH available. The NPSH required was calculated as a function of the time-dependent flow rate based on the pump curves. The NPSH required was also adjusted as a function of the void fraction as described in RG 1.82 (Reference 10).

The strainer head loss was compared against the strainer structural margin (24.0 ft) and the pump NPSH margin at each time step to determine whether a failure occurred.

11.0 Debris Penetration Evaluation

As described in Enclosure 2, Section 3.n.1, fiber penetration correlations were developed based on testing with a prototypical strainer module. These equations were used to calculate the fine fiber quantity that passes through the strainer from both prompt penetration and longer-term shedding.

A penetration fraction of 0.48 was also applied to the fine fiber that transports to the strainers during pool fill. As shown in Enclosure 2, Section 3.n.1, this penetration fraction bounds the prompt penetration corresponding to clean strainer conditions.

12.0 Debris Penetration Effects

As discussed in Enclosure 1, Section 1.0 and Enclosure 2, Section 3.m, ex-vessel downstream effects were addressed in a bounding manner and therefore were not included in the VEGP NARWHAL model.

As described in Enclosure 2, Section 3.n.1, core blockage and boron precipitation were evaluated using assumed fiber debris limits and acceptance criteria. VEGP uses Westinghouse fuel and the reactor vessel has an upflow barrel/baffle design with pressure relief holes in the core plates that are not currently credited in the long-term core cooling analyses.

Any debris that does not accumulate in the reactor vessel was modeled as automatically returning to the sump pool where it would be available to transport and potentially pass through the strainers again.

12.1 Cold Leg Breaks

The fiber debris that penetrates the RHR strainers transports to the reactor vessel through the ECCS flow. During cold leg recirculation, a portion of the ECCS flow that enters the reactor vessel through the cold legs would travel through the core inlet as make-up for boil-off, while the rest of the cold-leg flow would spill from the break. The fraction of fiber debris that was caught on the core inlet was the ratio of the boil-off flow rate to the ECCS flow rate into the cold leg. Note that margin was added to the boil-off

flow rate (see Section 6.6). Once switchover to hot leg recirculation had occurred, debris no longer accumulated at the core inlet. Instead, any penetrated debris that entered the reactor vessel was captured incore. However, by the time hot leg recirculation was initiated, most of the fiber fines had been captured on the strainers, and very little additional fiber was transported to the core. Any fiber that was captured in the reactor vessel was assumed to remain there for the duration of the event.

For cold leg breaks, an in-vessel failure was recorded if the core inlet fiber load was greater than the specified limit (see Enclosure 2, Section 3.n.1).

12.2 Hot Leg Breaks

For hot leg breaks, a significant quantity of fiber could accumulate on both the core inlet as well as incore. Flow would either enter the core inlet or the alternate flow path based on the head loss due to debris-related resistance across the core inlet at the bottom core plate. To determine the flow split for the ECCS flow that entered the reactor vessel through the cold leg, several preliminary calculations were performed. Initially, a K_{split} variable was calculated based on an assumed function of ECCS flow.

Next, the current K_{factor} variable was calculated based on an assumed function of the fiber quantity on the core inlet and the presence of chemical precipitates. If chemicals had already precipitated, then the current K_{factor} was set to a very high value. Similarly, no matter the time, if the quantity of fiber on the core inlet was greater than an assumed threshold then the current K_{factor} was also set to a very high value. Note that in the context of core fiber accumulation, the chemical precipitation timing refers to the time at which aluminum precipitation occurs (see Section 8.2).

For all other conditions, the K_{factor} was calculated based on an assumed piecewise function of the current fiber debris load.

Using the K_{split} and K_{factor} values, an m_{split} variable was calculated. If the K_{factor} was less than K_{split}, then m_{split} was set to 0. If the K_{factor} was very high, then the m_{split} was set to 1. If neither of these conditions were met, then the m_{split} variable was calculated based on an assumed function of the K_{factor} and K_{split} values. Note that an assumed maximum curve fit was used, which results in the minimum m_{split} value.

With these variables calculated, the fraction of debris that was caught on the core inlet and within the core for a hot leg break was calculated using the following equations.

$$\text{Core Inlet (HL Break)} = \left(1 - m_{split}\right) * \frac{Q_{cold\ leg}}{Q_{hot\ leg} + Q_{cold\ leg}}$$

$$\text{Incore (HL Break)} = m_{split} * \frac{Q_{cold\ leg}}{Q_{hot\ leg} + Q_{cold\ leg}} + \frac{Q_{hot\ leg}}{Q_{hot\ leg} + Q_{cold\ leg}}$$

Following switchover to hot leg recirculation, all of the fiber carried with the ECCS flow was modeled as accumulating in the core.

For hot leg breaks, an in-vessel failure was recorded if any of the following failure criteria were met:

1. The calculated K_{factor} exceeded the specified limit before the specified t_{block} time.
2. The incore fiber load was greater than the specified limit.
3. The reactor vessel fiber load (sum of the incore and core inlet fiber quantities) was greater than the specified limit.

See Enclosure 2, Section 3.n.1 for additional details on the hot leg break failure criteria.

13.0 Sub-Model Integration

The following flow diagrams show an overview of the analytical models that were used to determine how water was transported, how the conventional debris was generated and transported, how chemical precipitates formed and transported, how the strainer failure criteria were analyzed, and how the core failure criteria were analyzed. All of these models were addressed holistically in a time-dependent manner.

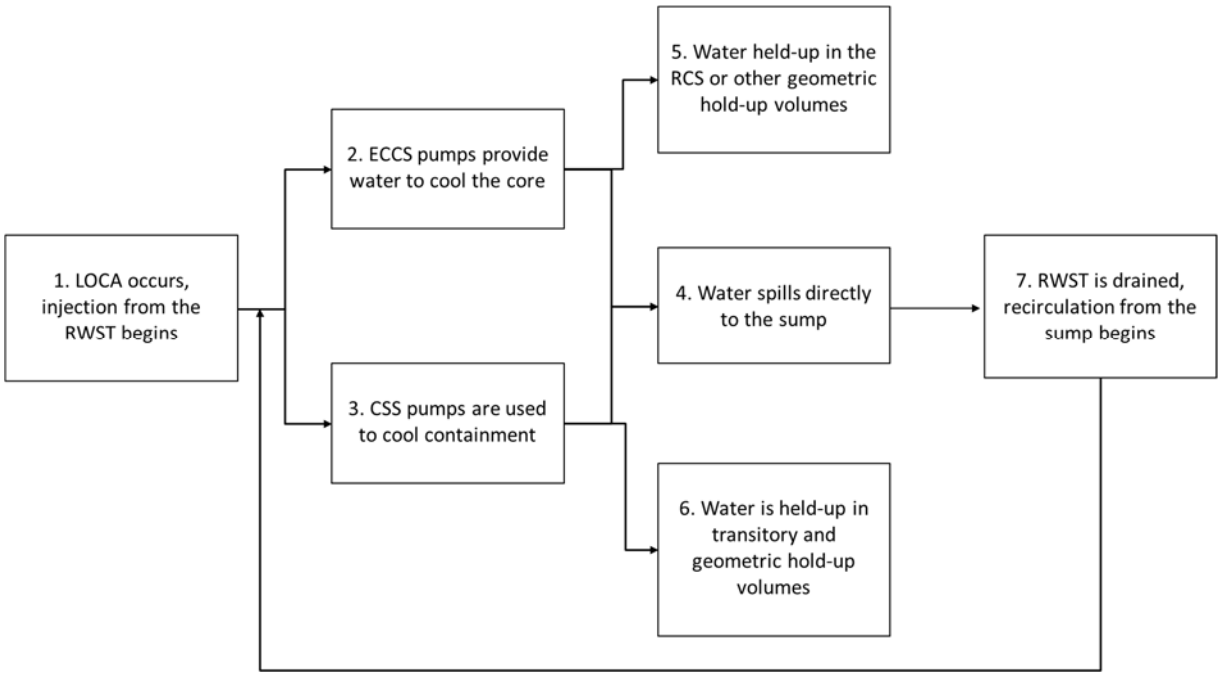

Figure 3-3 – Water Transport and Accumulation

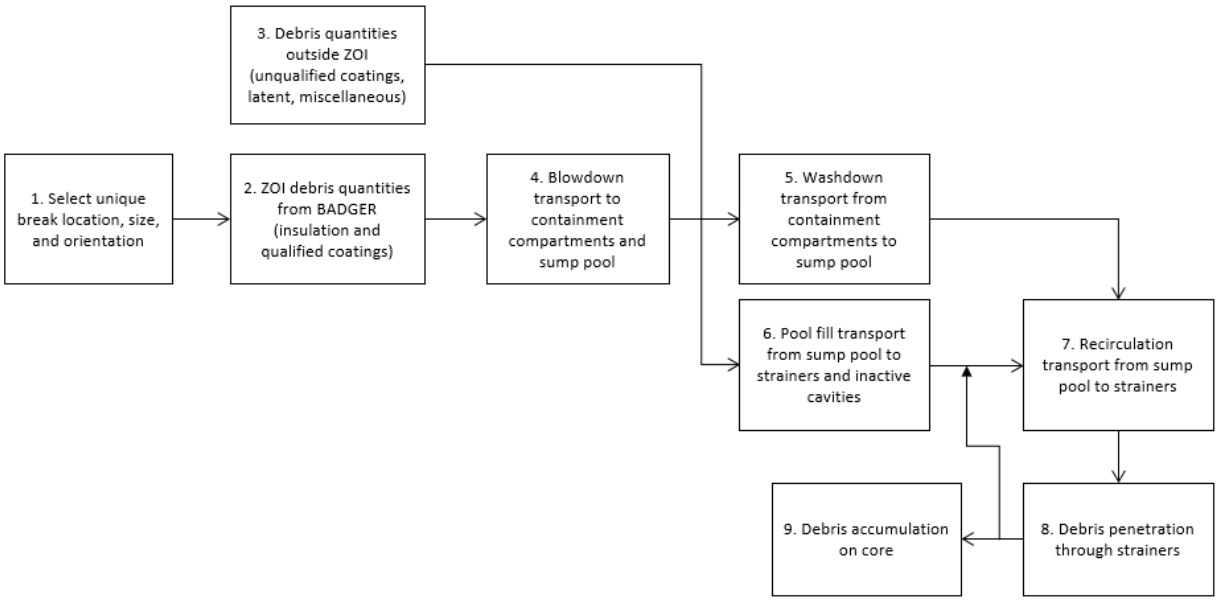

Figure 3-4 – Debris Generation and Transport

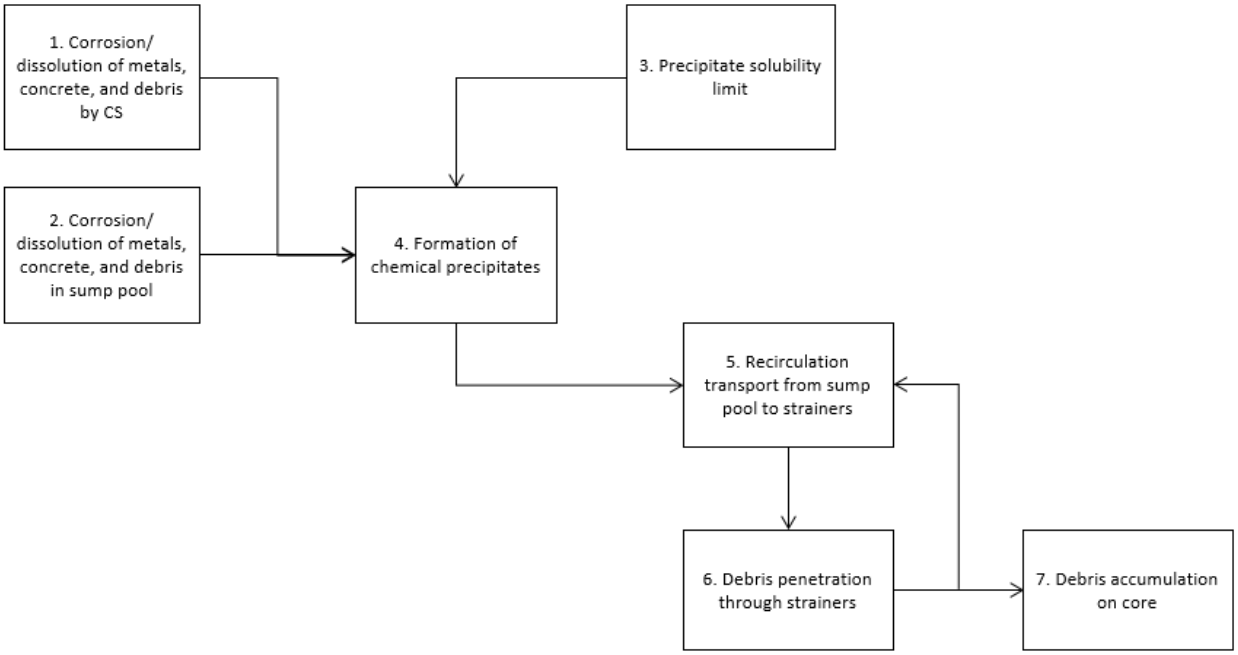

Figure 3-5 – Chemical Product Formation and Transport

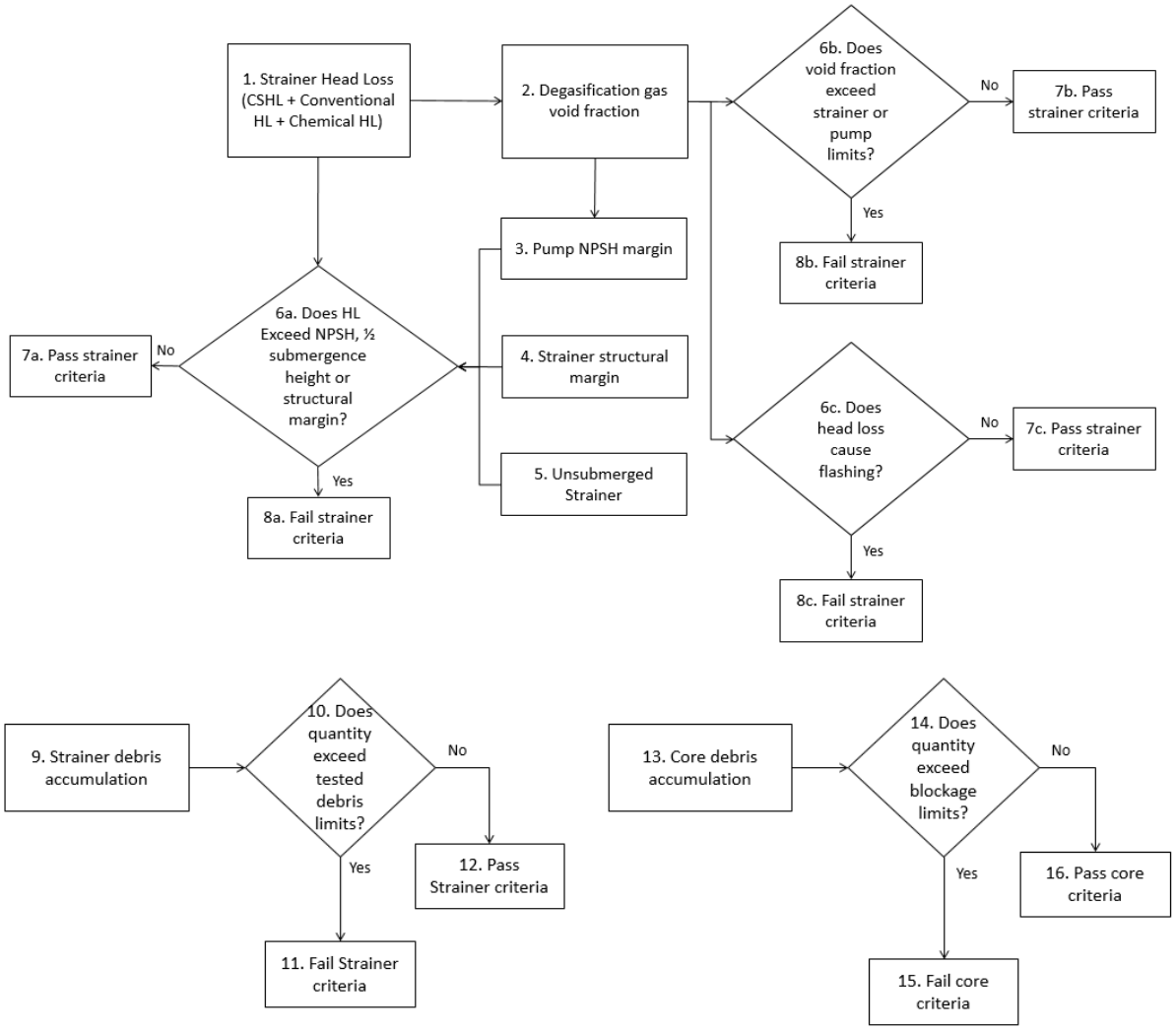

Figure 3-6 – Comparison of Strainer Head Loss and Core Debris Loads to Failure Criteria

Any failures that occurred were binned as strainer failures or core failures, and these results were used to calculate the GSI-191 CFP values as described in more detail below. The NARWHAL software, which was used to integrate all of the sub-models, is described in Section 13.1, and Section 13.2 provides a summary of the assumptions used in the VEGP NARWHAL evaluation.

In order to calculate the CFP values, the following steps were taken:

1. GSI-191 failures were computed for each break as described above. For VEGP, the failures for input into the PRA were computed in the following 12 categories (Strainer A and B correspond to the RHR strainers):
   a. Small breaks
      i. Core failures

   ii. Strainer A and Strainer B failures (without core failures)
   iii. Strainer A failures (without core or Strainer B failures)
   iv. Strainer B failures (without core or Strainer A failures)
b. Medium breaks
   i. Core failures
   ii. Strainer A and B failures (without core failures)
   iii. Strainer A failures (without core or Strainer B failures)
   iv. Strainer B failures (without core or Strainer A failures)
c. Large breaks
   i. Core failures
   ii. Strainer A and B failures (without core failures)
   iii. Strainer A failures (without core or Strainer B failures)
   iv. Strainer B failures (without core or Strainer A failures)

2. Overall plant-wide LOCA frequencies were allocated to individual welds and break sizes using a top-down LOCA frequency methodology.
   a. Plant-wide LOCA frequencies were based on the small, medium, and large break frequencies in the VEGP GSI-191 PRA model with log-linear interpolation for intermediate break sizes.
   b. The frequency for a given break size was allocated to individual welds (that can experience a break of that size).
3. The PRA model category for large breaks was broken up into size ranges to more accurately calculate the overall CFP. Smaller breaks within a given size range were assumed to have the same probability as larger breaks within the size range.
4. The CFP for a PRA category (e.g., large breaks) was calculated based on the combined CFP and LOCA frequency weight for each size range.

Figure 3-7 shows an example of how the size ranges were used to calculate the strainer CFP for the case with no random equipment failures. In this example, the overall frequencies result in a probability weight of 82.3 percent in Size Range 1, 15.1 percent in Size Range 2, and 2.6 percent in Size Range 3. The corresponding strainer CFP values are 0.0 percent for Size Range 1, 3.0 percent for Size Range 2, and 27.8 percent for Size Range 3. Therefore, the overall strainer CFP for large breaks is 1.2 percent as shown below:

$$\mathrm{CFP}_{\mathrm{Large}} = 0.823 \cdot 0.000 + 0.151 \cdot 0.030 + 0.026 \cdot 0.278 = 0.012$$

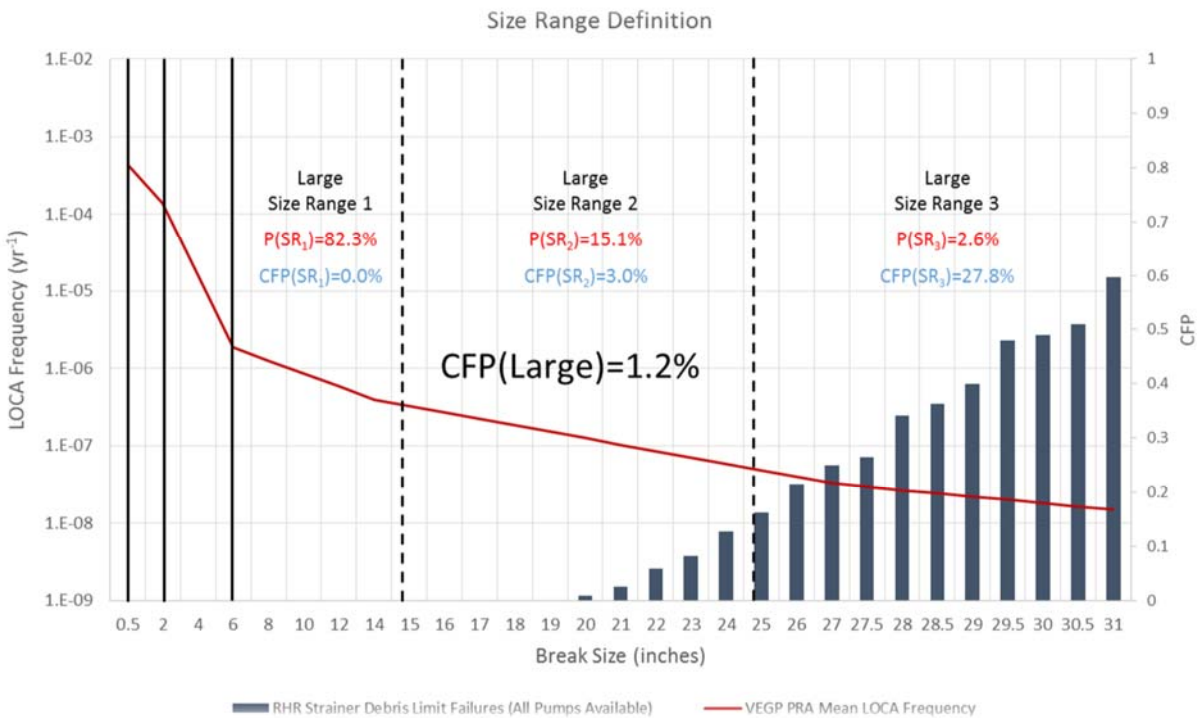

Figure 3-7 – Illustration of CFP Calculation for Large Breaks Based on Three Size Ranges

13.1 NARWHAL Software

NARWHAL is an object-oriented program that models the connections between important plant components (pumps, strainers, tanks, etc.) based on user-defined inputs. The software performs mass balance calculations that determine the time-dependent quantity of water, debris, and chemical solutes associated with each physical object. Using these time-dependent quantities along with other user-specified conditions, each aspect of GSI-191 can be evaluated in an integrated manner. The software can be used to simulate a single break or many thousands of breaks to evaluate the risk associated with GSI-191.

At any given time during the simulation, the state of the plant can be defined by a fixed set of parameters (i.e., the on/off states of components, the quantity of debris stored by components, etc.). This collection of parameters is called the state vector. NARWHAL updates the state vector by determining the amount of change in each variable given a change in time. The amount of change in each variable is determined using a series of algorithms called “marching algorithms”, which are graph traversal algorithms that maintain consistent flow through the plant system. For example, if a 10,000 gpm pump is fed by a pump that is limited to 1,000 gpm, the algorithms will determine that the high capacity pump can only pump 1,000 gpm. These algorithms are essentially a way for NARWHAL components to communicate information to one another.

Risk Quantification

A single NARWHAL simulation relies on a series of marching algorithms (Figure 3-8). The algorithms were designed to handle generic configurations of components, meaning that the user can design arbitrary networks as long as the configuration is valid (i.e., there is a source, active pumps, and a sink).

The first algorithm, the activity march, exists solely to determine the health of the network and its components. This algorithm is responsible for detecting valid and invalid paths of flow through the system. For instance, it is necessary to detect the failure of a pump if the strainer that feeds the pump fails. Conversely, this algorithm determines that a strainer will not receive flow if the pumps feeding from it shut down or fail.

The second algorithm, the source march, determines that flow through the network and its components is consistent. This is important when, for instance, pump flow rates are a function of other pumps (e.g., piggy-backing). It is also important when considering flow restrictions in the system, such as break size dependent flow rates.

The third algorithm, the water march, determines the flow rates and storage balances on all components in the network. This algorithm implements the mass balance equations inherent to the NARWHAL base component. After this algorithm has run, water is moved from the network sources, through all active components, to the network sinks.

The fourth and final algorithm, the debris march, uses information generated by the previous algorithms to determine the mass balance of debris and chemicals in the network. This includes determining the release of chemicals, the movement of debris, the capture of debris, and the formation of precipitates.

Figure 3-8 – NARWHAL's Basic Procedure

In each time step, after the marching algorithms have run and a new state vector has been calculated, a series of tests are run against a number of failure criteria (i.e., strainer structural margin, component debris limits, strainer submergence, etc.). If a failure occurs, it is noted in the results, but the simulation is allowed to continue running as if the failure had not occurred (e.g., a pump that fails due to loss of NPSH is allowed to continue running normally for the remainder of the simulation).

At the end of a simulation, NARWHAL reports the outcome of the simulation in one of three ways. If a single break simulation is being run, NARWHAL outputs a list of time-dependent vectors representing a number of variables including core debris quantities, strainer debris quantities, component failure states, and component flow rates. If a bulk simulation is being run, NARWHAL will not report time vectors. Instead, it reports summary variables such as failure times, total debris on each strainer at the end of the simulation, maximum head loss for each strainer, and total fiber on the core at switchover to hot leg recirculation. In addition, descriptive break information is reported including the break location and size. In this mode, each time a simulation for a given break is completed, a new record containing the summary information is automatically entered into the results file. If a probability or sensitivity simulation is being run, NARWHAL outputs the summary CFP values for each bulk simulation. Additionally, descriptive information about the variables being modified is entered into the results file.

In addition to calculating which breaks pass and fail the GSI-191 acceptance criteria, NARWHAL can also be used to post-process the results to calculate the CFP values based on the approach described in Section 13.0.

13.2 NARWHAL Model Assumptions

The following assumptions were used for the VEGP NARWHAL model evaluation:

1. It was assumed that breaks less than 2 inches do not result in rapid, full depressurization of the RCS. Therefore, injection by the SI accumulators is not required for these breaks. This assumption is consistent with the minimum water level calculation.
2. The containment sprays were assumed to only be activated for hot leg breaks greater than 15 inches. Note that this includes all partial breaks and DEGBs greater than 15 inches on the hot legs; however, no failures on the cold or intermediate legs are assumed to actuate containment sprays. Although there is some uncertainty in which breaks initiate containment sprays, the relatively high containment pressure required to actuate sprays (21.5 psig) significantly reduces the likelihood that most breaks will exceed the set point and actuate containment sprays. The assumption that only very large hot leg breaks will initiate containment sprays is consistent with the results of best-estimate thermal-hydraulic modeling for a range of potential break sizes on the hot and cold leg piping. This modeling showed that a hot leg DEGB initiated containment sprays, while all other evaluated breaks (including a cold leg DEGB and partial 15-inch breaks on both the hot and cold legs) did not. Assuming that hot leg breaks greater than 15 inches activate containment sprays reasonably represents what was learned in the best-estimate thermal hydraulic modeling.
3. It was assumed that containment sprays would be secured at 24 hours. Although there is some uncertainty in the spray duration, this is a reasonably conservative assumption because sprays are required to operate for at least 2 hours once they are initiated, and running sprays longer than 2 hours would significantly increase the quantity of aluminum released from unsubmerged sources.
4. The RHR flow rate used in the NARWHAL CFP calculation was assumed to be 3,700 gpm. The design flow rate for the RHR pumps is 3,000 gpm. However, it is not expected that the actual plant flow rate would be this low. The use of a higher flow rate is generally conservative in terms of recirculation timing, flashing calculations, and head loss correction. The use of 3,700 gpm for the RHR flow rate is consistent with what was used in the deterministic NPSH calculation and the single train value used in the ECCS system head curve.
5. The minimum sump recirculation pH was assumed to be 7.0. This conservatively bounds the calculated minimum value of 7.12 and is consistent with the minimum acceptable pH documented in the buffer verification calculation. Note that the minimum sump recirculation pH was used to calculate the aluminum solubility limit in the NARWHAL model.
6. The spray pH during injection was assumed to be 5.72 as determined from the conservative maximum pH of the RWST. This maximizes chemical release from unsubmerged sources during injection.

7. It was assumed that the breaks downstream of the first isolation valve do not need to be addressed in this quantification because they are not risk significant. This is a reasonable assumption 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 known quantities of 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.
8. The amount of latent debris documented in the debris generation calculation is 60 lbm. This value was conservatively increased to a total of 200 lbm in containment for operating margin.
9. The amount of miscellaneous debris documented in the debris generation calculation is 4 ft^2. This value was conservatively increased to a total quantity of 50 ft^2 in containment for operating margin.
10. It was assumed that there is a total quantity of 926.6 ft^2 of unsubmerged aluminum metal and 348.4 ft^2 of submerged aluminum metal in containment. This includes some margin for future additions of aluminum, as described in the chemical product generation calculation.
11. The total amount of exposed concrete that would be submerged in the pool was assumed to be 10,000 ft^2. This is a conservatively large surface area used to maximize the potential for chemical release and is consistent with the quantity used in the chemical product generation calculation. Note that the chemical product generation calculation did not evaluate the chemical effects associated with unsubmerged concrete due to the conservative quantity used for submerged concrete.
12. VEGP requires modifications to the existing RHR strainers to maintain long-term submergence during recirculation for all breaks. Two disks per disk stack will be removed from each of the RHR strainer assemblies to achieve submergence.
13. The containment pressure was assumed to be saturation pressure when the pool temperature is greater than 210.96 degrees F, and 14.396 psia after the pool temperature has dropped below 210.96 degrees F. The pressure of 14.396 psia was calculated based on the minimum containment pressure of -0.3 psig per VEGP Technical Specification 3.6.4 and an atmospheric pressure of 14.696 psia. The temperature of 210.96 degrees F is the corresponding saturation temperature at 14.396 psia determined from linear interpolation. This is a conservative assumption because it only credits the accident pressure necessary to keep the pool as a liquid. Note that a small amount of containment accident pressure was credited for degasification and flashing calculations (see Assumption 16).
14. It was assumed that head loss due to chemical precipitates is only applied once the theoretical fiber bed is greater than 0.45 inches. This is supported by the 2009 thin bed head loss test.
15. It was assumed that all random equipment failures evaluated for the different NARWHAL CFP evaluations occurred at the beginning of recirculation. This is a conservative assumption because it results in a quicker switchover to recirculation when compared to failure at the beginning of the event. Additionally, for CS pump

and/or RHR pump failure cases, it results in more debris accumulation on the remaining active strainers.

16. For the purpose of degasification and flashing calculations, it was assumed that 3.5 psi of accident pressure would be available for the first 2.5 hours of event. As described in Assumption 13, no credit was taken for accident pressure in the NPSH available calculations. The basis for this assumption is described in more detail in Enclosure 2 Sections 3.f.14 and 3.g.14.
17. It was assumed that the assumed maximum m_{split} curve fit for core blockage calculations should be used in the NARWHAL CFP evaluation. This is a conservative assumption because it maximizes the quantity of fiber that accumulates on the core inlet. Although the curve fit results in a higher m_{split} value initially, it results in a lower m_{split} value for greater resistances across the core inlet. This allows more overall fiber to accumulate at the core inlet for breaks that challenge the core inlet fiber limit.
18. The accumulators were assumed to not inject for any secondary side break. This is a reasonable assumption because secondary side breaks do not result in rapid depressurization of the RCS, which would trigger accumulator injection.
19. Containment sprays were assumed to be initiated in both trains for secondary side breaks. This is a generally conservative assumption because if containment spray is not initiated, a large fraction of debris would be retained in upper containment. In addition, several different equipment configurations were evaluated, including a single CS pump failing and both CS pumps failing.
20. For a secondary side break, the total flow to the ECCS was assumed to be provided by both charging pumps at a flow rate of 230 gpm/pump. This is a reasonable assumption because the PORV water-relieving capacity is 230 gpm per PORV. Note that recirculation for secondary side breaks would occur due to overflow of the pressurizer relief tank through the two PORVs. The use of the maximum flow rate through the PORVs is conservative because it maximizes the flow split to the RHR strainers during recirculation.
21. All secondary side breaks were assumed to accumulate fiber in-vessel in the same way as a hot leg break. This is a reasonable assumption because the RCS would be bled through the pressurizer relief tank during a feed and bleed scenario. The pressurizer relief tank is connected downstream of the hot leg.
22. The general inputs used to calculate the CFPs for LOCAs were assumed to be applicable for secondary side breaks. These inputs include the following:
    - Thermal hydraulic inputs (water sources, flow rates, temperature profiles, pH, etc.)
    - Debris inputs (unqualified coatings quantities, debris transport fractions, latent debris, etc.)
    - Chemical precipitate debris
    - Strainer geometry
    - Strainer head loss
    - Strainer failure options
    - Strainer penetration equations
    - Core blockage equations

This provides a reasonable set of conditions to evaluate the risk impact of secondary side breaks.

23. It was assumed that a LOCA that occurs during full power operation (i.e., Mode 1) is equivalent or bounding compared to the other operating modes. This is a reasonable assumption because the RCS pressure and temperature (key inputs affecting the ZOI size) would either be approximately the same or significantly lower for Modes 2 through 6. In addition, the flow rate required to cool the core (a key input affecting core blockage) would be reduced significantly for low power or shutdown modes.

13.3 Time Step Sensitivity for NARWHAL Modeling

Each break evaluated in NARWHAL was run for a duration of 30 days (43,200 hours). The first 24 hours (1,440 minutes) were evaluated with a time step size of 1 minute. After the first day, the time step is increased to 60 minutes. This is a reasonable application since the majority of the transient occurs within the first few hours and long-term effects can be effectively analyzed with a coarser time step.

The most important time-dependent phenomena in the NARWHAL calculations are the chemical release/precipitation and the overall debris mass balance (including accumulation at the strainers, penetration through the strainers, and accumulation at the core). These models, and especially the mass balance calculations, have been extensively tested as part of the NARWHAL V&V. In addition, Vogtle has developed safety-related hand calculations for both chemical effects (including release and precipitation of each type of precipitate) and in-vessel effects (including fiber mass balance and strainer penetration). Although the smallest time increment that can be specified in NARWHAL is 1 minute, the hand calculations had no limit on the minimum time step size.

The hand calculation for in-vessel effects included a time step sensitivity analysis that varied the time step size for the first hour from 60 seconds (equivalent to the time step size used in NARWHAL) down to 5 seconds for a given debris load, flow conditions, etc. The results of this analysis (see Table 3-4a) showed that, as the time step decreases, the quantity fiber that accumulates at the core inlet for a cold leg break is reduced by approximately 19%, and the total quantity that accumulates in the reactor vessel for a hot leg break is reduced by approximately 5%. This indicates that the time step size has some effect on penetrated fiber mass and core fiber accumulation. However, using a larger time-step size is conservative with respect to the core failure criteria.

Table 3-4a – Time Step Size Sensitivity for In-Vessel Effects Analysis

Time Step (secs)	CLB Core Inlet Load at HLSO (g/FA)	HLB Total RV Fiber Load (g/FA)
60	9.64	89.74
20	8.29	86.35
10	7.96	85.51
5	7.81	85.10

For chemical effects, the hand calculation used variable time-step sizes to capture the available Vogtle thermal-hydraulic data with time steps less than half a second early in the event and less than 30 seconds for the first 4 hours. The maximum quantity of calcium phosphate and SAS were calculated to be 63.2 kg (139 lbm) and 40.2 kg (89 lbm), respectively. In the NARWHAL calculation, the maximum quantities of calcium phosphate and SAS calculated for the base case conditions were 116 lbm (17% less than the hand calculation) and 87 lbm (2% less than the hand calculation), respectively. As summarized in Table 3-4b, additional sensitivity runs confirmed that the variability between the NARWHAL results and the hand calculation is attributed to different sump pool volumes used, which is a key variable for chemical effects. The hand calculation used a constant, maximum water volume while NARWHAL base case calculated a time-dependent sump pool volume based on minimum injection volumes and maximum holdup volumes. Therefore, the time-step sizes used in NARWHAL are sufficiently accurate.

Table 3-4b – Time Step Size Sensitivity for Chemical Effects Analysis

Case	Total Water (lbm)	Calcium Phosphate (lbm)	SAS (lbm)
Hand calculation – Maximum Precipitate Case	6,675,858	139	89
NARWHAL – Base Case	5,485,739	116	87
Hand calculation – Sensitivity Case	3,597,241	79	79

As discussed in Enclosure 5, a rule-based approach was used to evaluate strainer head loss, resulting in simple step changes for the head loss values in NARWHAL. These head loss values are unaffected by the numerical integration parameters (i.e., time step size). However, the quantity of fiber, SAS, and calcium phosphate on the strainer were used to determine which head loss value is applicable. Therefore, the effect of time step size on accumulation of debris on the strainer was investigated using NARWHAL sensitivity cases. The overall ΔCDF was also calculated for these sensitivity cases to show any effect when all pertinent GSI-191 phenomena are integrated in a single model. These sensitivity cases are described in more detail below.

A total of six sensitivity cases were run in NARWHAL using different time-step sizes for the first 24 hours: 1 minute, 2 minutes, 3 minutes, 4 minutes, 5 minutes, and 15 minutes. Note that after 24 hours (1,440 minutes), the time-step size remained at 60 minutes for all sensitivity cases. As shown in Table 3-4c, there is some minor variation in the overall results as a function of the time-step size. However, this variation is insignificant for time steps smaller than 5 minutes.

Table 3-4c – NARWHAL Time-Step Sensitivity ΔCDF Results

Time step size	ΔCDF
Base Case (1 minute)	2.47E-08
2 minutes	2.48E-08
3 minutes	2.45E-08
4 minutes	2.49E-08
5 minutes	2.41E-08
15 minutes	4.47E-08

Based on the above comparisons, it can be concluded that a one-minute time-step size for the first 24 hours is small enough to sufficiently capture the transient GSI-191 phenomena and accurately quantify the risk associated with GSI-191.

14.0 Systematic Risk Assessment

As described in Section 4.0, the VEGP GSI-191 PRA model includes the necessary modifications to calculate the risk impact associated with the effects of debris on the strainers and in the core. No new human failure events due to the effects of debris were identified, and no human actions to mitigate the effects of debris were credited in the risk calculation.

The only common inputs that were used in both the NARWHAL model and the GSI-191 PRA model are the LOCA frequencies and the equipment configurations. As discussed in Section 6.4, the GSI-191 CFPs were calculated using the same LOCA frequencies that were used in the GSI-191 PRA model. In addition, as discussed in Section 6.3, the high likelihood equipment configurations were explicitly evaluated in NARWHAL to calculate separate sets of CFPs for each high likelihood configuration. These CFPs were entered for the appropriate equipment configurations in the GSI-191 PRA model. Therefore, the VEGP NARWHAL model and GSI-191 PRA model are consistent.

14.1 VEGP NARWHAL CFP Evaluation

The VEGP NARWHAL CFP calculation showed that there were no small or medium break LOCAs that fail for any equipment configurations. Table 3-5 shows the NARWHAL CFP results for large break LOCAs for each equipment configuration.

Table 3-5 – NARWHAL CFP Results for Large Break LOCAs

Equipment Configuration	Core	Strainer A and B	Strainer A only	Strainer B only
No Equipment Failure	0	0.0118	0	0
RHR Pump B Failure	0	N/A	0.0679	N/A
Charging Pump B Failure	0	0.0118	0	0
SI Pump B Failure	0	0.0118	0	0
Train B Failure	0	N/A	0.0736	N/A
CS Pump B Failure	0	0.0139	0	0
Both CS Pumps Failure	0	0.0177	0	0

Table 3-6 and Table 3-7 show the CFP results for DEGBs on the feedwater line piping and main steam line piping, respectively. None of the feedwater line breaks produced a sufficient quantity of debris to fail, and main steam line breaks only failed under the equipment configuration where both CS pumps fail due to issues unrelated to debris.

Table 3-6 – Feedwater Line DEGB CFP Results

Equipment Configuration	Core	Strainer A and B	Strainer A only	Strainer B only
No Equipment Failure	0	0	0	0
Charging Pump B Failure	0	N/A	0	N/A
CS Pump B Failure	0	0	0	0
Both CS Pumps Failure	0	0	0	0

Table 3-7 – Main Steam Line DEGB CFP Results

Equipment Configuration	Core	Strainer A and B	Strainer A only	Strainer B only
No Equipment Failure	0	0	0	0
Charging Pump B Failure	0	N/A	0	N/A
CS Pump B Failure	0	0	0	0
Both CS Pumps Failure	0	0.475	0	0

The total baseline risk (from internal events, internal fire, internal flood, and seismic events) for the VEGP Unit 1 PRA model is 4.39×10^{-5} per reactor-year (ry^{-1}) for CDF and 1.73×10^{-6} ry^{-1} for LERF. The total baseline risk for the VEGP Unit 2 PRA model is 5.05×10^{-5} ry^{-1} for CDF and 1.90×10^{-6} ry^{-1} for LERF. Using the CFP results described above, the change in risk calculated with the VEGP GSI-191 PRA model is shown in Table 3-8.

Table 3-8 – VEGP Total Risk Impact due to GSI-191 Failures

Case	ΔCDF (ry^{-1})	ΔLERF (ry^{-1})
Risk increase from GSI-191 failures for high-likelihood LOCA configurations	2.32×10^{-8}	3.10×10^{-11}
Bounding risk increase from GSI-191 failures for unlikely LOCA configurations	1.41×10^{-9}	4.09×10^{-12}
Risk increase from GSI-191 failures for seismically-induced LOCAs	1.50×10^{-9}	1.50×10^{-10}
Risk increase from GSI-191 failures for SSBIs	1.39×10^{-9}	8.25×10^{-11}
Total risk increase associated with GSI-191	**2.75×10^{-8}**	**2.68×10^{-10}**

These CDF, LERF, ΔCDF, and ΔLERF values fall well within the RG 1.174 Region III guidelines. Therefore, the effects of debris have very low risk at VEGP.

Table 3-9 provides a summary of information on each Class 1 ISI weld inside the first isolation valve. The results shown in this table (specifically the maximum transported fiber and whether a failure was observed at the weld) are based on the single train failure equipment configuration.

Table 3-9 – Weld Information List

Weld ID	Break Side	Weld Identifier	Compartment	Max Transported Fiber to RHR A Strainer (ft^3)	Degradation Mechanism	Failure ?
29	hot	11201-004-6-RB	SG_1-4	639.75	D&C	yes
29	hot	11201-001-5-RB	SG_1-4	641.70	D&C	yes
29	hot	11201-001-3-RB	SG_1-4	624.20	D&C	yes
29	hot	11201-004-4-RB	SG_1-4	605.83	D&C	yes
29	hot	11201-003-5-RB	SG_2-3	482.44	D&C	yes
29	hot	11201-002-5-RB	SG_2-3	476.86	D&C	yes
29	hot	11201-002-3-RB	SG_2-3	473.54	D&C	yes
29	hot	11201-003-3-RB	SG_2-3	462.00	D&C	yes
31	cold	11201-008-1-RB	SG_1-4	196.91	D&C	yes
31	cold	11201-005-1-RB	SG_1-4	195.89	D&C	yes
31	cold	11201-008-2-RB	SG_1-4	186.47	D&C	yes

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Weld ID	Break Side	Weld Identifier	Compartment	Max Transported Fiber to RHR A Strainer (ft^3)	Degradation Mechanism	Failure ?
31	cold	11201-005-2-RB	SG_1-4	187.35	D&C	yes
31	cold	11201-006-1-RB	SG_2-3	168.92	D&C	yes
31	cold	11201-005-3-RB	SG_1-4	168.82	D&C	yes
31	cold	11201-007-1-RB	SG_2-3	167.20	D&C	yes
31	cold	11201-008-3-RB	SG_1-4	163.00	D&C	yes
31	cold	11201-006-2-RB	SG_2-3	158.19	D&C	yes
31	cold	11201-007-2-RB	SG_2-3	156.75	D&C	yes
31	cold	11201-006-3-RB	SG_2-3	150.86	D&C	yes
29	hot	11201-001-1-RB	Reactor Cavity	150.42	D&C	yes
29	hot	11201-004-1-RB	Reactor Cavity	148.06	D&C	yes
29	hot	11201-V6-001-W37-RB	Reactor Cavity	148.06	D&C	yes
29	hot	11201-V6-001-W36-RB	Reactor Cavity	147.50	D&C	yes
29	hot	11201-003-1-RB	Reactor Cavity	142.41	D&C	yes
29	hot	11201-V6-001-W33-RB	Reactor Cavity	141.36	D&C	yes
29	hot	11201-002-1-RB	Reactor Cavity	141.13	D&C	yes
29	hot	11201-V6-001-W40-RB	Reactor Cavity	139.03	D&C	yes
31	cold	11201-005-4-RB	SG_1-4	137.98	D&C	yes
31	cold	11201-008-4-RB	SG_1-4	137.57	D&C	yes
31	cold	11201-007-3-RB	SG_2-3	136.78	D&C	yes
31	cold	11201-006-4-RB	SG_2-3	131.70	D&C	yes
31	cold	11201-008-8-RB	SG_1-4	124.64	D&C	yes
31	cold	11201-005-8-RB	SG_1-4	125.83	D&C	yes

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Weld ID	Break Side	Weld Identifier	Compartment	Max Transported Fiber to RHR A Strainer (ft^3)	Degradation Mechanism	Failure ?
31	cold	11201-005-7-RB	SG_1-4	123.84	D&C	yes
31	cold	11201-008-7-RB	SG_1-4	122.60	D&C	yes
27.5	cold	11201-009-1-RB	SG_1-4	122.60	D&C	yes
27.5	cold	11201-012-1-RB	SG_1-4	122.34	D&C	yes
31	cold	11201-007-4-RB	SG_2-3	114.04	D&C	yes
31	cold	11201-006-7-RB	SG_2-3	103.71	D&C	yes
31	cold	11201-007-7-RB	SG_2-3	103.73	D&C	yes
31	cold	11201-006-8-RB	SG_2-3	100.48	D&C	yes
31	cold	11201-007-8-RB	SG_2-3	100.22	D&C	yes
27.5	cold	11201-011-1-RB	SG_2-3	98.15	D&C	yes
27.5	cold	11201-010-1-RB	SG_2-3	97.11	D&C	yes
12.814	hot	11201-053-2-RB	SG_1-4	54.67	TF, D&C	yes
12.814	hot	11201-053-3-RB	SG_1-4	44.09	TF, D&C	yes
27.5	cold	11201-009-8-RB	Reactor Cavity	42.37	D&C	yes
27.5	cold	11201-V6-001-W35-RB	Reactor Cavity	41.10	PWSCC[1], D&C	no
27.5	cold	11201-009-9-RB	Reactor Cavity	41.10	D&C	no
27.5	cold	11201-012-8-RB	Reactor Cavity	40.78	D&C	no
27.5	cold	11201-010-6-RB	Reactor Cavity	39.42	D&C	no
27.5	cold	11201-V6-001-W38-RB	Reactor Cavity	39.28	PWSCC[1], D&C	no
27.5	cold	11201-012-9-RB	Reactor Cavity	39.28	D&C	no
27.5	cold	11201-010-7-RB	Reactor Cavity	38.45	D&C	no
27.5	cold	11201-V6-001-W34-RB	Reactor Cavity	38.44	PWSCC[1], D&C	no

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Weld ID	Break Side	Weld Identifier	Compartment	Max Transported Fiber to RHR A Strainer (ft^3)	Degradation Mechanism	Failure ?
27.5	cold	11201-011-7-RB	Reactor Cavity	37.78	D&C	no
27.5	cold	11201-011-8-RB	Reactor Cavity	36.71	D&C	no
27.5	cold	11201-V6-001-W39-RB	Reactor Cavity	36.71	PWSCC[1], D&C	no
12.814	hot	11201-053-1-RB	SG_1-4	36.37	TF, D&C	Yes[2]
12.814	hot	11201-004-2-RB	SG_1-4	35.50	TF, D&C	no
12.814	hot	11201-053-4-RB	SG_1-4	26.42	TF, D&C	no
8.75	cold	11204-124-16-RB	SG_1-4	26.21	D&C	no
8.75	cold	11204-124-17-RB	SG_1-4	25.65	D&C	no
8.75	cold	11204-127-20-RB	SG_1-4	25.49	D&C	no
10.5	hot	11201-036-5-RB	SG_1-4	25.01	D&C	no
10.5	hot	11201-036-6-RB	SG_1-4	24.65	D&C	no
8.75	cold	11204-127-21-RB	SG_1-4	24.63	D&C	no
10.5	hot	11201-049-1-RB	SG_1-4	24.35	D&C	no
10.5	hot	11201-004-3-RB	SG_1-4	24.32	D&C	no
10.5	hot	11201-036-4-RB	SG_1-4	24.31	D&C	no
8.75	cold	11204-124-18-RB	SG_1-4	24.28	D&C	no
8.75	cold	11204-126-16-RB	SG_2-3	23.81	D&C	no
10.5	hot	11201-001-2-RB	SG_1-4	23.44	D&C	no
8.75	cold	11204-126-17-RB	SG_2-3	23.43	D&C	no
8.75	cold	11204-127-22-RB	SG_1-4	23.41	D&C	no
10.5	hot	11201-036-1-RB	SG_1-4	22.82	D&C	no
8.75	cold	11201-009-6-RB	SG_1-4	22.71	D&C	no

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Weld ID	Break Side	Weld Identifier	Compartment	Max Transported Fiber to RHR A Strainer (ft^3)	Degradation Mechanism	Failure ?
8.75	cold	11204-126-18-RB	SG_2-3	22.28	D&C	no
8.75	cold	11204-125-16-RB	SG_2-3	22.00	D&C	no
8.75	cold	11201-012-6-RB	SG_1-4	22.00	D&C	no
10.5	hot	11201-049-2-RB	SG_1-4	21.53	D&C	no
8.75	cold	11204-125-17-RB	SG_2-3	21.48	D&C	no
10.5	hot	11201-049-6-RB	SG_1-4	20.97	D&C	no
8.75	cold	11201-011-5-RB	SG_2-3	20.40	D&C	no
8.75	cold	11204-125-18-RB	SG_2-3	20.35	D&C	no
10.5	hot	11201-036-3-RB	SG_1-4	20.25	D&C	no
10.5	hot	11201-036-2-RB	SG_1-4	20.18	D&C	no
10.5	hot	11204-021-27-RB	SG_1-4	20.08	D&C	no
10.5	hot	11201-049-3-RB	SG_1-4	20.05	D&C	no
11.188	hot	11201-053-5-RB	SG_1-4	20.00	TF, D&C	no
10.5	hot	11204-021-28-RB	SG_1-4	19.83	D&C	no
10.5	hot	11201-049-4-RB	SG_1-4	19.60	D&C	no
10.5	hot	11201-049-5-RB	SG_1-4	19.36	D&C	no
8.75	cold	11201-010-4-RB	SG_2-3	18.69	D&C	no
11.188	hot	11201-V6-002-W22-RB	Pressurizer Compartment	16.54	D&C	no
5.189	cold	11201-058-6-RB	Pressurizer Compartment	14.95	D&C	no
5.189	cold	11201-V6-002-W17-RB	Pressurizer Compartment	14.94	D&C	no
5.189	cold	11201-059-7-RB	Pressurizer Compartment	14.88	D&C	no
5.189	cold	11201-V6-002-W18-RB	Pressurizer Compartment	14.79	D&C	no

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Weld ID	Break Side	Weld Identifier	Compartment	Max Transported Fiber to RHR A Strainer (ft^3)	Degradation Mechanism	Failure ?
5.189	cold	11201-059-6-RB	Pressurizer Compartment	14.76	D&C	no
5.189	cold	11201-056-5-RB	Pressurizer Compartment	14.74	D&C	no
5.189	cold	11201-V6-002-W19-RB	Pressurizer Compartment	14.74	D&C	no
5.189	cold	11201-059-2-RB	Pressurizer Compartment	14.70	D&C	no
5.189	cold	11201-V6-002-W20-RB	Pressurizer Compartment	14.70	D&C	no
5.189	cold	11201-059-5-RB	Pressurizer Compartment	14.67	D&C	no
5.189	cold	11201-059-4-RB	Pressurizer Compartment	14.59	D&C	no
5.189	cold	11201-057-2-RB	Pressurizer Compartment	14.57	D&C	no
5.189	cold	11201-059-8-RB	Pressurizer Compartment	14.54	D&C	no
5.189	cold	11201-057-7-RB	Pressurizer Compartment	14.54	D&C	no
5.189	cold	11201-058-2-RB	Pressurizer Compartment	14.49	D&C	no
5.189	cold	11201-059-3-RB	Pressurizer Compartment	14.49	D&C	no
5.189	cold	11201-056-2-RB	Pressurizer Compartment	14.42	D&C	no
5.189	cold	11201-058-5-RB	Pressurizer Compartment	14.40	D&C	no
5.189	cold	11201-058-7-RB	Pressurizer Compartment	14.33	D&C	no
5.189	cold	11201-058-3-RB	Pressurizer Compartment	14.31	D&C	no
5.189	cold	11201-057-6-RB	Pressurizer Compartment	14.30	D&C	no
5.189	cold	11201-057-8-RB	Pressurizer Compartment	14.29	D&C	no
5.189	cold	11201-059-9-RB	Pressurizer Compartment	14.28	D&C	no
5.189	cold	11201-057-3-RB	Pressurizer Compartment	14.22	D&C	no
5.189	cold	11201-056-4-RB	Pressurizer Compartment	14.15	D&C	no
5.189	cold	11201-058-4-RB	Pressurizer Compartment	14.15	D&C	no

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Weld ID	Break Side	Weld Identifier	Compartment	Max Transported Fiber to RHR A Strainer (ft^3)	Degradation Mechanism	Failure ?
5.189	cold	11201-059-10-RB	Pressurizer Compartment	14.01	D&C	no
5.189	cold	11201-056-6-RB	Pressurizer Compartment	14.00	D&C	no
5.189	cold	11201-057-5-RB	Pressurizer Compartment	13.92	D&C	no
5.189	cold	11201-057-4-RB	Pressurizer Compartment	13.88	D&C	no
3.438	cold	11201-012-4-RB	SG_1-4	13.69	D&C	no
5.189	cold	11201-056-3-RB	Pressurizer Compartment	13.68	D&C	no
3.438	cold	11201-030-1-RB	SG_1-4	13.58	D&C	no
3.438	cold	11201-009-4-RB	SG_1-4	13.57	D&C	no
3.438	cold	11201-029-1-RB	SG_1-4	13.53	D&C	no
3.438	cold	11201-V6-002-W21-RB	Pressurizer Compartment	13.49	D&C	no
5.189	cold	11201-059-11-RB	Pressurizer Compartment	12.98	D&C	no
5.189	cold	11201-059-12-RB	Pressurizer Compartment	12.85	D&C	no
5.189	cold	11201-059-13-RB	Pressurizer Compartment	12.84	D&C	no
5.189	cold	11201-060-1-RB	Pressurizer Compartment	12.83	D&C	no
5.189	cold	11201-060-2-RB	Pressurizer Compartment	12.83	D&C	no
2.626	cold	11201-009-5-RB	SG_1-4	12.81	VF,TF, D&C	no
2.626	cold	11201-011-4-RB	SG_2-3	12.79	VF, D&C	no
2.626	cold	11208-009-6-RB	SG_1-4	12.78	VF,TF, D&C	no
2.626	cold	11201-048-1-RB	SG_2-3	12.74	VF, D&C	no
2.626	cold	11201-012-5-RB	SG_1-4	12.68	VF,TF, D&C	no
2.626	cold	11201-008-5-RB	SG_1-4	12.64	D&C	no
2.626	cold	11201-005-5-RB	SG_1-4	12.64	D&C	no

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Weld ID	Break Side	Weld Identifier	Compartment	Max Transported Fiber to RHR A Strainer (ft^3)	Degradation Mechanism	Failure ?
2.626	cold	11201-006-5-RB	SG_2-3	12.63	D&C	no
2.626	cold	11201-007-5-RB	SG_2-3	12.62	D&C	no
5.189	hot	11204-023-20-RB	SG_1-4	12.54	D&C	no
2.626	cold	11201-011-6-RB	Reactor Cavity	12.49	TF, D&C	no
2.626	cold	11201-012-7-RB	Reactor Cavity	12.48	TF, D&C	no
5.189	hot	11204-023-21-RB	SG_1-4	12.48	D&C	no
3.438	cold	11201-029-2-RB	SG_1-4	12.46	D&C	no
2.626	cold	11204-246-36-RB	Reactor Cavity	12.45	TF, D&C	no
2.626	cold	11201-005-9-RB	SG_1-4	12.45	D&C	no
2.626	cold	11201-006-9-RB	SG_2-3	12.44	D&C	no
2.626	cold	11201-008-9-RB	SG_1-4	12.43	D&C	no
2.626	cold	11201-007-9-RB	SG_2-3	12.42	D&C	no
2.626	cold	11204-245-33-RB	Reactor Cavity	12.38	TF, D&C	no
5.189	cold	11201-030-29-RB	Pressurizer Compartment	12.38	D&C	no
5.189	cold	11201-030-30-RB	Pressurizer Compartment	12.36	TF, D&C	no
5.189	cold	11201-030-34-RB	Pressurizer Compartment	12.33	TF, D&C	no
5.189	cold	11201-030-33-RB	Pressurizer Compartment	12.33	TF, D&C	no
3.438	cold	11201-029-3-RB	SG_1-4	12.31	D&C	no
2.626	cold	11208-007-6-RB	SG_1-4	12.30	VF,TF, D&C	no
3.438	cold	11201-029-3 A-RB	SG_1-4	12.27	D&C	no
5.189	cold	11201-030-28-RB	Pressurizer Compartment	12.25	D&C	no
3.438	cold	11201-029-8-RB	SG_1-4	12.24	D&C	no

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Weld ID	Break Side	Weld Identifier	Compartment	Max Transported Fiber to RHR A Strainer (ft^3)	Degradation Mechanism	Failure ?
3.438	cold	11201-029-5-RB	SG_1-4	12.22	D&C	no
3.438	cold	11201-029-6-RB	SG_1-4	12.18	D&C	no
3.438	cold	11201-029-9-RB	SG_1-4	12.17	D&C	no
3.438	cold	11201-029-7-RB	SG_1-4	12.16	D&C	no
2.626	cold	11201-048-2-RB	SG_2-3	12.15	VF, D&C	no
5.189	cold	11201-030-35-RB	Pressurizer Compartment	12.14	TF, D&C	no
5.189	cold	11201-030-32-RB	Pressurizer Compartment	12.14	TF, D&C	no
5.189	cold	11201-030-31 A-RB	Pressurizer Compartment	12.12	TF, D&C	no
2.125	cold	11201-011-3-RB	SG_2-3	12.12	D&C	no
2.125	cold	11201-009-3-RB	SG_1-4	12.12	D&C	no
2.125	cold	11201-012-3-RB	SG_1-4	12.12	D&C	no
2.125	cold	11201-010-3-RB	SG_2-3	12.12	D&C	no
3.438	cold	11201-029-4-RB	SG_1-4	12.10	D&C	no
3.438	cold	11201-030-20-RB	Pressurizer Compartment	12.05	D&C	no
3.438	cold	11201-030-19-RB	Pressurizer Compartment	12.04	D&C	no
2.626	cold	11201-048-3-RB	SG_2-3	12.02	VF, D&C	no
2.626	cold	11201-009-7-RB	SG_1-4	12.01	TF, D&C	no
2.626	cold	11201-010-5-RB	SG_2-3	11.98	TF, D&C	no
2.626	cold	11204-243-34-RB	SG_1-4	11.98	TF, D&C	no
3.438	cold	11201-030-21-RB	Pressurizer Compartment	11.94	D&C	no
2.626	cold	11204-244-28-RB	SG_2-3	11.93	TF, D&C	no
5.189	hot	11204-023-19-RB	SG_1-4	11.91	D&C	no

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Weld ID	Break Side	Weld Identifier	Compartment	Max Transported Fiber to RHR A Strainer (ft^3)	Degradation Mechanism	Failure ?
2.626	cold	11208-007-5-RB	SG_1-4	11.90	VF,TF, D&C	no
2.626	cold	11208-007-4-RB	SG_1-4	11.87	VF,TF, D&C	no
3.438	cold	11201-030-18-RB	Pressurizer Compartment	11.86	D&C	no
3.438	cold	11201-030-17-RB	Pressurizer Compartment	11.84	D&C	no
3.438	cold	11201-030-12-RB	SG_1-4	11.84	D&C	no
3.438	cold	11201-029-25-RB	Pressurizer Compartment	11.83	D&C	no
3.438	cold	11201-030-14-RB	SG_1-4	11.82	D&C	no
3.438	cold	11201-030-13-RB	SG_1-4	11.82	D&C	no
3.438	cold	11201-030-11-RB	SG_1-4	11.82	D&C	no
2.626	cold	11208-009-5-RB	SG_1-4	11.80	VF,TF, D&C	no
3.438	cold	11201-030-22-RB	Pressurizer Compartment	11.79	D&C	no
3.438	cold	11201-030-5-RB	SG_1-4	11.79	D&C	no
2.626	cold	11208-009-4-RB	SG_1-4	11.78	VF,TF, D&C	no
1.689	cold	11201-009-2-RB	SG_1-4	11.78	D&C	no
3.438	cold	11201-030-8-RB	SG_1-4	11.78	D&C	no
1.689	cold	11201-011-2-RB	SG_2-3	11.78	D&C	no
1.689	cold	11201-012-2-RB	SG_1-4	11.78	D&C	no
1.689	cold	11201-010-2-RB	SG_2-3	11.78	D&C	no
3.438	cold	11201-029-24-RB	Pressurizer Compartment	11.77	D&C	no
3.438	cold	11201-030-9-RB	SG_1-4	11.77	D&C	no
3.438	cold	11201-030-4-RB	SG_1-4	11.77	D&C	no
1.689	cold	11201-007-6-RB	SG_2-3	11.77	D&C	no

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Risk Quantification

Weld ID	Break Side	Weld Identifier	Compartment	Max Transported Fiber to RHR A Strainer (ft^3)	Degradation Mechanism	Failure ?
1.689	cold	11201-005-6-RB	SG_1-4	11.77	D&C	no
1.689	cold	11208-012-3-RB	Annulus	11.77	VF, D&C	no
1.689	cold	11201-006-6-RB	SG_2-3	11.77	D&C	no
1.689	cold	11201-008-6-RB	SG_1-4	11.77	D&C	no
3.438	cold	11201-030-7-RB	SG_1-4	11.76	D&C	no
1.689	cold	11208-012-5 B-RB	Annulus	11.76	VF, D&C	no
1.689	cold	11201-031-1-RB	SG_1-4	11.76	D&C	no
3.438	cold	11201-029-22-RB	Pressurizer Compartment	11.75	D&C	no
3.438	cold	11201-029-18-RB	SG_1-4	11.75	D&C	no
3.438	cold	11201-030-6-RB	SG_1-4	11.75	D&C	no
1.689	cold	11201-011-9-RB	SG_2-3	11.75	D&C	no
1.689	cold	11201-010-8-RB	SG_2-3	11.75	D&C	no
1.689	cold	11201-009-10-RB	SG_1-4	11.75	D&C	no
1.689	cold	11201-012-10-RB	SG_1-4	11.75	D&C	no
3.438	cold	11201-029-19-RB	SG_1-4	11.75	D&C	no
3.438	cold	11201-030-3-RB	SG_1-4	11.74	D&C	no
1.689	cold	11201-042-1-RB	SG_2-3	11.74	D&C	no
1.689	cold	11201-051-1-RB	SG_1-4	11.73	D&C	no
1.689	cold	11201-046-1-RB	SG_2-3	11.73	D&C	no
1.689	cold	11208-012-5 A-RB	Annulus	11.73	VF, D&C	no
1.689	cold	11201-042-2-RB	SG_2-3	11.73	TF, D&C	no
1.689	cold	11201-051-2-RB	SG_1-4	11.73	TF, D&C	no

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Risk Quantification

Weld ID	Break Side	Weld Identifier	Compartment	Max Transported Fiber to RHR A Strainer (ft^3)	Degradation Mechanism	Failure ?
1.689	cold	11201-046-2-RB	SG_2-3	11.73	TF, D&C	no
1.689	cold	11208-012-4-RB	Annulus	11.73	VF, D&C	no
1.689	cold	11201-031-2-RB	SG_1-4	11.73	TF, D&C	no
1.689	cold	11208-012-5-RB	Annulus	11.72	VF, D&C	no
3.438	cold	11201-030-2-RB	SG_1-4	11.72	D&C	no
2.626	cold	11208-007-3-RB	SG_1-4	11.72	VF,TF, D&C	no
3.438	cold	11201-029-26-RB	Pressurizer Compartment	11.71	D&C	no
3.438	cold	11201-029-17-RB	SG_1-4	11.70	D&C	no
2.626	cold	11208-009-3-RB	SG_1-4	11.69	VF,TF, D&C	no
1.689	cold	11201-046-3-RB	SG_2-3	11.69	TF, D&C	no
5.189	cold	11201-030-38-RB	Pressurizer Compartment	11.69	TF, D&C	no
1.689	cold	11201-051-3-RB	SG_1-4	11.69	TF, D&C	no
1.689	cold	11201-042-3-RB	SG_2-3	11.69	TF, D&C	no
3.438	cold	11201-029-12-RB	SG_1-4	11.69	D&C	no
1.689	cold	11201-031-3-RB	SG_1-4	11.69	TF, D&C	no
3.438	cold	11201-029-13-RB	SG_1-4	11.69	D&C	no
2.626	cold	11201-060-3-RB	Pressurizer Compartment	11.68	D&C	no
3.438	cold	11201-029-14-RB	SG_1-4	11.68	D&C	no
3.438	cold	11201-029-23-RB	Pressurizer Compartment	11.67	D&C	no
3.438	cold	11201-029-11-RB	SG_1-4	11.67	D&C	no
3.438	cold	11201-029-16-RB	SG_1-4	11.66	D&C	no
2.626	cold	11201-048-5-RB	SG_2-3	11.66	VF,TF, D&C	no

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Risk Quantification

Weld ID	Break Side	Weld Identifier	Compartment	Max Transported Fiber to RHR A Strainer (ft^3)	Degradation Mechanism	Failure ?
2.626	cold	11201-048-4-RB	SG_2-3	11.65	VF,TF, D&C	no
2.626	cold	11201-060-5-RB	Pressurizer Compartment	11.65	D&C	no
3.438	cold	11201-029-15-RB	SG_1-4	11.65	D&C	no
3.438	cold	11201-030-24-RB	Pressurizer Compartment	11.65	D&C	no
3.438	cold	11201-030-23-RB	Pressurizer Compartment	11.65	D&C	no
5.189	cold	11201-030-37-RB	Pressurizer Compartment	11.65	TF, D&C	no
2.626	cold	11201-059-14-RB	Pressurizer Compartment	11.65	D&C	no
2.626	cold	11201-060-4-RB	Pressurizer Compartment	11.64	D&C	no
5.189	cold	11201-030-36-RB	Pressurizer Compartment	11.64	TF, D&C	no
2.626	cold	11201-048-6-RB	SG_2-3	11.62	VF,TF, D&C	no
2.626	cold	11201-059-15-RB	Pressurizer Compartment	11.61	D&C	no
5.189	hot	11201-003-2-RB	SG_2-3	11.61	D&C	no
2.626	cold	11201-060-6-RB	Pressurizer Compartment	11.61	D&C	no
3.438	cold	11201-030-25-RB	Pressurizer Compartment	11.61	D&C	no
3.438	cold	11201-030-26-RB	Pressurizer Compartment	11.60	D&C	no
2.626	cold	11201-059-16-RB	Pressurizer Compartment	11.60	D&C	no
3.438	cold	11201-030-27-RB	Pressurizer Compartment	11.60	D&C	no
1.689	cold	11201-051-4-RB	SG_1-4	11.57	D&C	no
2.626	cold	11201-048-7-RB	SG_2-3	11.56	VF,TF, D&C	no
2.626	cold	11201-059-17-RB	Pressurizer Compartment	11.55	D&C	no
1.338	cold	11204-246-35-RB	Reactor Cavity	11.54	TF, D&C	no
3.438	cold	11201-030-39-RB	Pressurizer Compartment	11.54	TF, D&C	no

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Risk Quantification

Weld ID	Break Side	Weld Identifier	Compartment	Max Transported Fiber to RHR A Strainer (ft^3)	Degradation Mechanism	Failure ?
3.438	cold	11201-030-16-RB	Pressurizer Compartment	11.53	D&C	no
3.438	cold	11201-030-15-RB	Pressurizer Compartment	11.52	D&C	no
2.626	cold	11201-048-10-RB	SG_2-3	11.52	VF,TF, D&C	no
1.689	cold	11201-051-6-RB	SG_1-4	11.51	D&C	no
2.626	cold	11201-048-8-RB	SG_2-3	11.51	VF,TF, D&C	no
2.626	cold	11201-048-9-RB	SG_2-3	11.51	VF,TF, D&C	no
1.689	cold	11201-051-5-RB	SG_1-4	11.50	D&C	no
3.438	cold	11201-029-21-RB	Pressurizer Compartment	11.50	D&C	no
1.338	cold	11204-245-32-RB	Reactor Cavity	11.50	TF, D&C	no
3.438	cold	11201-029-20-RB	Pressurizer Compartment	11.50	D&C	no
3.438	cold	11201-030-40-RB	Pressurizer Compartment	11.48	TF, D&C	no
1.689	cold	11201-042-4-RB	SG_2-3	11.47	D&C	no
3.438	cold	11201-030-41-RB	Pressurizer Compartment	11.46	TF, D&C	no
3.438	cold	11201-030-44-RB	Pressurizer Compartment	11.46	TF, D&C	no
3.438	cold	11201-030-42-RB	Pressurizer Compartment	11.46	TF, D&C	no
3.438	cold	11201-030-43-RB	Pressurizer Compartment	11.46	TF, D&C	no
2.626	cold	11201-060-7-RB	Pressurizer Compartment	11.46	D&C	no
1.689	cold	11201-030-31-RB	Pressurizer Compartment	11.45	VF,TF, D&C	no
1.689	cold	11208-012-6-RB	Pressurizer Compartment	11.44	VF, D&C	no
2.626	cold	11201-059-18-RB	Pressurizer Compartment	11.44	D&C	no
1.689	cold	11201-042-5-RB	SG_2-3	11.44	D&C	no
1.689	cold	11201-031-5-RB	SG_1-4	11.44	D&C	no

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Risk Quantification

Weld ID	Break Side	Weld Identifier	Compartment	Max Transported Fiber to RHR A Strainer (ft^3)	Degradation Mechanism	Failure ?
1.689	cold	11201-031-6-RB	SG_1-4	11.43	D&C	no
1.338	cold	11204-246-34-RB	Reactor Cavity	11.43	TF, D&C	no
1.338	cold	11204-243-33-RB	SG_1-4	11.43	TF, D&C	no
1.338	cold	11204-245-31-RB	Reactor Cavity	11.42	TF, D&C	no
1.689	cold	11201-042-6-RB	SG_2-3	11.42	D&C	no
1.689	cold	11201-046-5-RB	SG_2-3	11.42	D&C	no
1.689	cold	11201-046-4-RB	SG_2-3	11.42	D&C	no
1.689	cold	11201-031-4-RB	SG_1-4	11.42	D&C	no
1.16	cold	11201-030-10-RB	SG_1-4	11.41	D&C	no
2.626	cold	11201-059-19-RB	Pressurizer Compartment	11.41	D&C	no
1.16	cold	11201-029-10-RB	SG_1-4	11.41	D&C	no
1.338	cold	11204-244-27-RB	SG_2-3	11.41	TF, D&C	no
1.338	cold	11204-246-33-RB	Reactor Cavity	11.41	TF, D&C	no
1.16	cold	11201-030-46-RB	SG_1-4	11.41	D&C	no
1.16	cold	11201-029-27-RB	SG_1-4	11.41	D&C	no
2.626	cold	11201-060-8-RB	Pressurizer Compartment	11.41	D&C	no
1.338	cold	11204-245-30-RB	Reactor Cavity	11.40	TF, D&C	no
2.626	cold	11201-060-10-RB	Pressurizer Compartment	11.39	D&C	no
1.338	cold	11204-243-32-RB	SG_1-4	11.38	TF, D&C	no
2.626	cold	11201-059-20-RB	Pressurizer Compartment	11.38	D&C	no
2.626	cold	11201-060-9-RB	Pressurizer Compartment	11.38	D&C	no
1.338	cold	11204-243-30-RB	SG_1-4	11.37	D&C	no

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Risk Quantification

Weld ID	Break Side	Weld Identifier	Compartment	Max Transported Fiber to RHR A Strainer (ft^3)	Degradation Mechanism	Failure ?
1.338	cold	11204-243-31-RB	SG_1-4	11.37	TF, D&C	no
1.338	cold	11204-245-29-RB	SG_2-3	11.37	D&C	no
1.338	cold	11208-045-14 A-RB	SG_2-3	11.36	VF, D&C	no
1.338	cold	11208-045-14 B-RB	SG_2-3	11.36	VF, D&C	no
1.338	cold	11208-045-14-RB	SG_2-3	11.36	VF, D&C	no
1.338	cold	11208-043-5 A-RB	SG_2-3	11.36	VF, D&C	no
1.338	cold	11208-043-5 B-RB	SG_2-3	11.36	VF, D&C	no
1.338	cold	11208-024-5 A-RB	SG_1-4	11.36	VF, D&C	no
1.338	cold	11208-024-5 B-RB	SG_1-4	11.36	VF, D&C	no
1.338	cold	11208-047-14 A-RB	SG_1-4	11.36	VF, D&C	no
1.338	cold	11208-047-14 B-RB	SG_1-4	11.36	VF, D&C	no
1.338	cold	11204-244-26-RB	SG_2-3	11.35	TF, D&C	no
1.338	cold	11208-047-14-RB	SG_1-4	11.35	VF, D&C	no
1.338	cold	11208-024-5-RB	SG_1-4	11.35	VF, D&C	no
1.338	cold	11208-043-5-RB	SG_2-3	11.35	VF, D&C	no
2.626	cold	11201-059-21-RB	Pressurizer Compartment	11.35	D&C	no
1.338	cold	11208-045-13 A-RB	SG_2-3	11.34	VF, D&C	no
1.338	cold	11204-244-25-RB	SG_2-3	11.34	TF, D&C	no
1.338	cold	11208-045-9-RB	SG_2-3	11.34	VF, D&C	no
1.338	cold	11208-045-8-RB	SG_2-3	11.34	VF, D&C	no
1.338	cold	11208-045-9 A-RB	SG_2-3	11.34	VF, D&C	no
1.338	cold	11208-045-16-RB	SG_2-3	11.34	VF, D&C	no

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Risk Quantification

Weld ID	Break Side	Weld Identifier	Compartment	Max Transported Fiber to RHR A Strainer (ft^3)	Degradation Mechanism	Failure ?
1.338	cold	11208-045-12-RB	SG_2-3	11.34	VF, D&C	no
1.338	cold	11208-045-15-RB	SG_2-3	11.34	VF, D&C	no
1.338	cold	11208-024-3-RB	SG_1-4	11.34	VF, D&C	no
1.338	cold	11208-045-9 B-RB	SG_2-3	11.33	VF, D&C	no
1.338	cold	11208-045-13-RB	SG_2-3	11.33	VF, D&C	no
1.338	cold	11208-045-10-RB	SG_2-3	11.33	VF, D&C	no
1.338	cold	11208-045-11-RB	SG_2-3	11.33	VF, D&C	no
1.338	cold	11208-024-4 B-RB	SG_1-4	11.33	VF, D&C	no
1.338	cold	11204-246-32-RB	SG_1-4	11.33	TF, D&C	no
1.338	cold	11208-047-9 C-RB	SG_1-4	11.33	VF, D&C	no
1.338	cold	11208-047-9 B-RB	SG_1-4	11.33	VF, D&C	no
1.338	cold	11208-047-16-RB	SG_1-4	11.33	VF, D&C	no
1.338	cold	11208-024-4 A-RB	SG_1-4	11.32	VF, D&C	no
1.338	cold	11208-024-7-RB	SG_1-4	11.32	VF, D&C	no
1.338	cold	11208-043-6-RB	SG_2-3	11.32	VF, D&C	no
1.338	cold	11208-047-9 A-RB	SG_1-4	11.32	VF, D&C	no
1.338	cold	11208-043-3 A-RB	SG_2-3	11.32	VF, D&C	no
1.338	cold	11208-043-3 B-RB	SG_2-3	11.32	VF, D&C	no
1.338	cold	11208-043-3-RB	SG_2-3	11.32	VF, D&C	no
1.338	cold	11208-043-3 C-RB	SG_2-3	11.32	VF, D&C	no
1.338	cold	11208-043-3 D-RB	SG_2-3	11.32	VF, D&C	no
1.338	cold	11208-047-5 B-RB	SG_1-4	11.31	VF, D&C	no

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Risk Quantification

Weld ID	Break Side	Weld Identifier	Compartment	Max Transported Fiber to RHR A Strainer (ft^3)	Degradation Mechanism	Failure ?
1.338	cold	11208-047-15-RB	SG_1-4	11.31	VF, D&C	no
1.338	cold	11208-043-4 A-RB	SG_2-3	11.31	VF, D&C	no
1.338	cold	11208-043-7-RB	SG_2-3	11.31	VF, D&C	no
1.338	cold	11208-024-3 A-RB	SG_1-4	11.31	VF, D&C	no
1.338	cold	11208-024-3 B-RB	SG_1-4	11.31	VF, D&C	no
1.338	cold	11208-024-4-RB	SG_1-4	11.31	VF, D&C	no
1.338	cold	11208-024-6-RB	SG_1-4	11.31	VF, D&C	no
1.338	cold	11208-045-3-RB	SG_2-3	11.31	VF, D&C	no
1.338	cold	11208-045-4-RB	SG_2-3	11.31	VF, D&C	no
1.338	cold	11208-045-5 A-RB	SG_2-3	11.31	VF, D&C	no
1.338	cold	11208-045-5 B-RB	SG_2-3	11.31	VF, D&C	no
1.338	cold	11208-045-5-RB	SG_2-3	11.31	VF, D&C	no
1.338	cold	11208-045-6-RB	SG_2-3	11.31	VF, D&C	no
1.338	cold	11208-045-7-RB	SG_2-3	11.31	VF, D&C	no
1.338	cold	11208-047-3-RB	SG_1-4	11.31	VF, D&C	no
1.338	cold	11208-047-5 A-RB	SG_1-4	11.31	VF, D&C	no
1.338	cold	11208-043-4 B-RB	SG_2-3	11.31	VF, D&C	no
1.338	cold	11204-244-24-RB	SG_2-3	11.31	TF, D&C	no
1.338	cold	11208-024-4 AA-RB	SG_1-4	11.31	VF, D&C	no
1.338	cold	11208-043-4-RB	SG_2-3	11.31	VF, D&C	no
5.189	hot	11204-025-24-RB	SG_2-3	11.28	D&C	no
5.189	hot	11201-002-2-RB	SG_2-3	11.26	D&C	no

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Risk Quantification

Weld ID	Break Side	Weld Identifier	Compartment	Max Transported Fiber to RHR A Strainer (ft^3)	Degradation Mechanism	Failure ?
5.189	hot	11204-024-19-RB	SG_2-3	11.23	D&C	no
5.189	hot	11204-024-14-RB	SG_2-3	11.20	D&C	no
5.189	hot	11204-021-18-RB	SG_1-4	11.12	D&C	no
5.189	hot	11204-025-23-RB	SG_2-3	11.09	D&C	no
5.189	hot	11204-025-20-RB	SG_2-3	11.06	D&C	no
5.189	hot	11204-021-19-RB	SG_1-4	10.97	D&C	no
5.189	hot	11204-021-17-RB	SG_1-4	10.89	D&C	no
5.189	hot	11204-024-18-RB	SG_2-3	10.84	D&C	no
5.189	hot	11204-025-22-RB	SG_2-3	10.83	D&C	no
5.189	hot	11204-025-21-RB	SG_2-3	10.82	D&C	no
5.189	hot	11204-024-17-RB	SG_2-3	10.67	D&C	no
5.189	hot	11204-021-26-RB	SG_1-4	10.41	D&C	no
5.189	hot	11204-021-22-RB	SG_1-4	10.41	D&C	no
5.189	hot	11204-024-16-RB	SG_2-3	10.41	D&C	no
5.189	hot	11204-024-15-RB	SG_2-3	10.33	D&C	no
5.189	hot	11204-023-17-RB	SG_1-4	10.27	D&C	no
5.189	hot	11204-021-25-RB	SG_1-4	10.18	D&C	no
5.189	hot	11204-021-20-RB	SG_1-4	10.15	D&C	no
5.189	hot	11204-021-23-RB	SG_1-4	10.15	D&C	no
5.189	hot	11204-023-18-RB	SG_1-4	10.03	D&C	no
5.189	hot	11204-021-24-RB	SG_1-4	10.03	D&C	no
5.189	hot	11204-023-16-RB	SG_1-4	9.93	D&C	no

Weld ID	Break Side	Weld Identifier	Compartment	Max Transported Fiber to RHR A Strainer (ft^3)	Degradation Mechanism	Failure ?
5.189	hot	11204-021-21-RB	SG_1-4	9.84	D&C	no
2.125	hot	11201-003-4-RB	SG_2-3	9.15	D&C	no
2.125	hot	11201-004-5-RB	SG_1-4	9.15	D&C	no
2.125	hot	11201-002-4-RB	SG_2-3	9.14	D&C	no
2.125	hot	11201-001-4-RB	SG_1-4	9.14	D&C	no
5.189	hot	11204-025-25-RB	SG_2-3	8.48	D&C	no

[1] Primary water stress corrosion cracking (PWSCC) degradation was assumed for four cold leg welds since these are the only ASME Code Section XI Category B-F welds with nominal pipe size (NPS) of 4 inches or larger that have not been mitigated for PWSCC. Assuming the PWSCC degradation for a weld directly affects the likelihood of a break on that weld but may not necessarily result in GSI-191 failures if the breaks at that weld do not exceed any of the acceptance criteria.

[2] The failure of the 12.814-inch break at weld 11201-053-1-RB was caused by the calcium phosphate debris limit being exceeded. Note that the total quantity of generated fiber debris (including large and intact pieces) was used to calculate calcium release. This resulted in the high calcium phosphate debris load for this break although it has less transportable fibrous debris than some of the larger breaks.

14.2 NARWHAL Uncertainty and Sensitivity

The purpose of this section is to describe the sensitivity analysis and uncertainty quantification associated with the GSI-191 phenomenological models evaluated using NARWHAL.

14.2.1 Simplified Risk Estimation Methodology

For the purposes of sensitivity analysis and uncertainty quantification, a simplified method was used to estimate ΔCDF in NARWHAL. The simplification includes a reduction in the number of equipment configurations explicitly evaluated for each sensitivity, and also directly calculates ΔCDF in NARWHAL using the LOCA frequencies and the equipment functional failure probabilities (along with the NARWHAL calculated CFP results).

The GSI-191 risk can be reasonably estimated for the purpose of sensitivity analysis without explicitly modeling all six of the configurations listed in Section 6.3. The scenario with a combined failure of one RHR pump and two CS pumps has a very low

probability resulting in a negligible impact on the risk quantification. Although it is not necessarily bounded by any of the other cases, it was assumed to have the same conditional failure probability as the scenario with a combined failure of one RHR pump and one CS pump. The scenario with one RHR pump failure is bounded by the scenario with a combined failure of one RHR pump and one CS pump. In addition, the scenario with one CS pump failure is bounded by the scenario with two CS pump failures. Therefore, the functional failure probabilities listed in Section 6.3 were combined for the purpose of sensitivity analysis and uncertainty quantification as shown in Table 3-10.

Table 3-10 – Combined Functional Failure Probabilities

Equipment Configuration	Functional Failure Probability
No Equipment Failures	91.50%
2 CS Pump Failures	6.57%
1 RHR Pump + 1 CS Pump Failures	1.92%
Total	**100%**

For the VEGP NARWHAL model, the CFPs were reported for each PRA size category and success criterion. To estimate the ΔCDF, the CFPs for each PRA success criterion were summed within each PRA size category. Therefore, a single CFP value was calculated for each PRA size category. The following equation was then used to estimate ΔCDF.

$$\Delta\mathrm{CDF} = \sum_{i=0}^{i=N}\sum_{j=0}^{j=X} \mathrm{IEF_i} * \mathrm{CFP_{ij}} * \mathrm{FFP_j}$$

Nomenclature:

- i = Each PRA size category
- j = Each equipment configuration
- IEF = Initiating event frequency for each PRA size category
- CFP = Conditional failure probability for each PRA size category and each equipment configuration
- FFP = Functional failure probability for each equipment configuration

Note that initiating event frequency of each PRA size category was defined by the LOCA frequencies from Section 6.4.

If there are no medium or small breaks that fail, the equation can be simplified as shown below:

$$\Delta CDF = IEF * \sum_{j=0}^{j=X} CFP_j * FFP_j$$

Nomenclature:

j = Each equipment configuration
IEF = Initiating event frequency for large LOCAs
CFP = Large LOCA conditional failure probability for each equipment configuration
FFP = Functional failure probability for each equipment configuration

Given a LOCA frequency for large breaks of 1.85×10^{-6} yr^{-1} (see Section 6.4), the functional failure probabilities shown in Table 3-10, and the conditional failure probabilities shown in Table 3-5, ΔCDF can be estimated as shown below:

$$\Delta CDF = 1.85 \times 10^{-6} \cdot (0.9150 \cdot 0.0118 + 0.0657 \cdot 0.0177 + 0.0192 \cdot 0.0736) = 2.47 \times 10^{-8}$$

This value is nearly identical to the ΔCDF value that was calculated using the GSI-191 PRA model (2.46×10^{-8} yr^{-1} excluding the SSBI contribution in Table 3-8). Therefore, this method provides an efficient and accurate ΔCDF estimate.

Because the base CDF and LERF values at VEGP are well within the RG 1.174 acceptance guidelines (Reference 1), and ΔLERF is more than two orders of magnitude lower than ΔCDF (see Table 3-8), the risk sensitivity was evaluated by comparing ΔCDF to the ΔCDF acceptance guideline. As shown in Enclosure 1, Section 2.2, the risk associated with GSI-191 is considered to be small for a mean ΔCDF below 1×10^{-5} yr^{-1} and very small for a mean ΔCDF below 1×10^{-6} yr^{-1}.

14.2.2 Sensitivity Analysis

Parametric sensitivity analysis was performed to identify which inputs have the greatest impact on the risk quantification results. The parametric sensitivity analysis includes the process of identifying input variables to evaluate, selecting minimum, nominal, and maximum values for each variable, quantifying risk in terms of ΔCDF as a common output that can be compared for each sensitivity, and using the ΔCDF results to rank the sensitivity of each input variable.

The VEGP NARWHAL model includes numerous inputs that could have been included in the sensitivity analysis. However, some of these input parameters are directly correlated to other parameters (and therefore should not be independently analyzed), some parameters were pre-screened as having an insignificant effect on the results, and some parameters do not require an independent analysis because they would have the same type of effect as other similar parameters that are evaluated.

Enclosure 3

Risk Quantification

A consistent methodology was used to determine the minimum, nominal, and maximum values for each of the parametric sensitivity inputs. Consistency is important because using a very large range for one parameter and a very small range for another parameter may mask the true sensitivity of the second parameter and indicate that the first parameter has a much greater effect on the results. However, selecting consistent minimum and maximum values is challenging due to practical considerations. For example, debris transport fractions can vary between 0 percent and 100 percent, the initial RWST level may vary between the technical specification minimum limit and the high level alarm, and debris head loss may vary from 0 ft at the low end to an unknown value at the high end. In addition, some parameters are not fixed values and may be determined as a function of time (e.g., pool temperature) or as a correlation based on other calculated parameters (e.g., penetration fraction). The following methodology provides an approach for evaluating the various input parameters in a consistent manner.

- The nominal value was defined as the input value used in the NARWHAL base case. The base case NARWHAL model that was used for sensitivity analysis and uncertainty quantification was identical to the model from the NARWHAL CFP calculation.
- The minimum and maximum values for each sensitivity input depend on the nominal value and the available information. If the nominal value was conservatively skewed toward the minimum direction, the minimum value used for the parametric sensitivity was 10 percent lower than the nominal value. Similarly, if the nominal value was conservatively skewed toward the maximum direction, the maximum value used for the parametric sensitivity was 10 percent higher than the nominal value.
- For all other cases, the minimum and maximum values were determined by the available information. Design limits were used preferentially if they were available. If a range of values was determined analytically, the minimum or maximum from the range was used if design limits were not available.
- If no information was available for the range of a given input, then the minimum or maximum value was assumed to be ± 25 percent of the nominal value.

The results of the parametric sensitivity analysis were used to rank each input parameter. This was done using a tornado diagram, which illustrates how sensitive the chosen output metric (ΔCDF) is to changing an input variable's value from nominal to maximum (or minimum). The tornado diagram was created by first running NARWHAL with all inputs set at nominal conditions, and recording the output metric. One variable was then changed to its maximum value (with all others held constant), the software was re-run, the output metric was recorded, and the results were compared to the nominal case. This process was repeated with each variable being independently modified to the maximum and minimum values. The output responses were then sorted by magnitude and shown from highest output response (most risk-sensitive parameter) to lowest output response (least risk-sensitive parameter).

The minimum and maximum values used in the sensitivity analysis are shown in Table 3-11. Based on the methodology described above, some of the parametric sensitivity cases consider input values that are outside the plant operating conditions (e.g., the RWST volume used for the NARWHAL base case corresponds to the technical specification minimum level, so the minimum volume used for the sensitivity is less than the technical specification minimum). Therefore, these sensitivity results are only intended to provide insights into the relative importance of each input parameter.

The ΔCDF results of the sensitivity analysis are shown in Table 3-12, and the difference in ΔCDF (compared to the NARWHAL base case value of 2.47E-08 yr^{-1}) was plotted in the tornado diagram shown in Figure 3-9. Note that the "Containment Spray Duration of 1439 Minutes" parametric sensitivity case is not shown in Figure 3-9 since it is bounded by the "Containment Spray Duration" parametric sensitivity case."

Table 3-11 – Maximum and Minimum Parametric Sensitivity Inputs

Input Parameter	Units	Minimum Input	Maximum Input
Simulation Time	minutes	32,400	54,000
Initial RWST Level	lbm	5,062,577	6,025,079
RHR Pump Flow Rate	gpm	2,775	4,500
CS Pump Flow Rate	gpm	1,950	3,374
Pressure and Temperature Profiles		75% *Design Basis	110% * Design Basis
Sump pH		7.1	8.1
ZOI Debris Quantity		75% * Base Case	110% * Base Case
Latent Debris Quantity			
Particulate	lbm	51 (0.303 ft^3)	187 (1.109 ft^3)
Fiber	ft^3	3.75	13.75
Miscellaneous Debris Quantity	ft^2	2	55
Submerged Aluminum Surface Area	ft^2	278.7	383.2
Unsubmerged Aluminum Surface Area	ft^2	741.3	1,019.3
Debris Head Loss			
Conventional for Fiber ≤ 3.1 ft^3	ft of H_2O	0.47	0.78
Conventional for Fiber > 3.1 ft^3	ft of H_2O	3.50	6.83
Calcium Phosphate	ft of H_2O	0.83	2.25
SAS	ft of H_2O	3.24	6.55
Strainer Debris Limits			
Fiber	ft^3	9.927	13.79
Particulate	ft^3	1.643	2.282
Fire Barrier	lbm	26.24	36.45
Calcium Phosphate	lbm	4.77	6.63
SAS	lbm	8.046	11.18
Containment Accident Pressure	psi	3.15	4.375
Strainer Penetration Fractions[1]		75% * Correlation Results	125% * Correlation Results
Containment Spray Duration	minutes	120	43,200
Reactor Vessel Hot Leg Break Fine Fiber Limit	g/FA	50	125
Reactor Vessel Cold Leg Break Fine Fiber Limit	g/FA	11.25	18.75
Geometric LOCA Frequency Values		5th Percentile	95th Percentile

Input Parameter		Units	Minimum Input	Maximum Input
Unqualified Coatings Quantities				
Unqualified Epoxy	UC	ft^3	13.253	19.438
	LC	ft^3	9.533	13.982
Unqualified IOZ	UC	ft^3	0.088	0.129
	LC	ft^3	0.199	0.292
Unqualified Alkyd	UC	ft^3	0.000	0.000
	LC	ft^3	0.387	0.568

[1] A sensitivity run was performed using a spray duration of 1,439 minutes, which is the nominal spray duration of 24 hours (or 1,440 minutes) minus 1 minute. This was to show that the small accumulation of SAS on the spray strainers, due to a one-minute overlap between the formation of SAS and containment spray operation, had a negligibly small effect on the risk quantification.

Table 3-12 – Results of Parametric Sensitivity Analysis

Input Parameter	ΔCDF at Minimum Input	ΔCDF at Maximum Input
Simulation Time	2.46E-08	2.47E-08
Initial RWST Level	1.23E-08	2.47E-08
RHR Pump Flow Rate	2.21E-08	8.06E-08
CS Pump Flow Rate	2.73E-08	2.25E-08
Pressure and Temperature Profiles	1.09E-08	2.47E-08
Sump pH	1.08E-08	2.56E-08
ZOI Debris Quantity	8.54E-09	3.33E-08
Latent Debris Quantity	2.30E-08	2.48E-08
Miscellaneous Debris Quantity	2.22E-08	2.47E-08
Submerged Aluminum Surface Area	2.47E-08	2.47E-08
Unsubmerged Aluminum Surface Area	2.47E-08	2.47E-08
Debris Head Loss	2.47E-08	2.53E-08
Strainer Debris Limits	3.47E-07	5.31E-08
Containment Accident Pressure	2.47E-08	2.47E-08
Strainer Penetration Fractions	2.50E-08	2.98E-08
Containment Spray Duration	2.85E-08	2.44E-08
Containment Spray Duration of 1439 Minutes	2.47E-08	---
Reactor Vessel Hot Leg Break Fine Fiber Limit	1.09E-07	2.47E-08
Reactor Vessel Cold Leg Break Fine Fiber Limit	2.47E-08	2.47E-08
Geometric LOCA Frequency Values	4.16E-11	6.25E-08
Unqualified Coatings Quantity	2.47E-08	2.47E-08

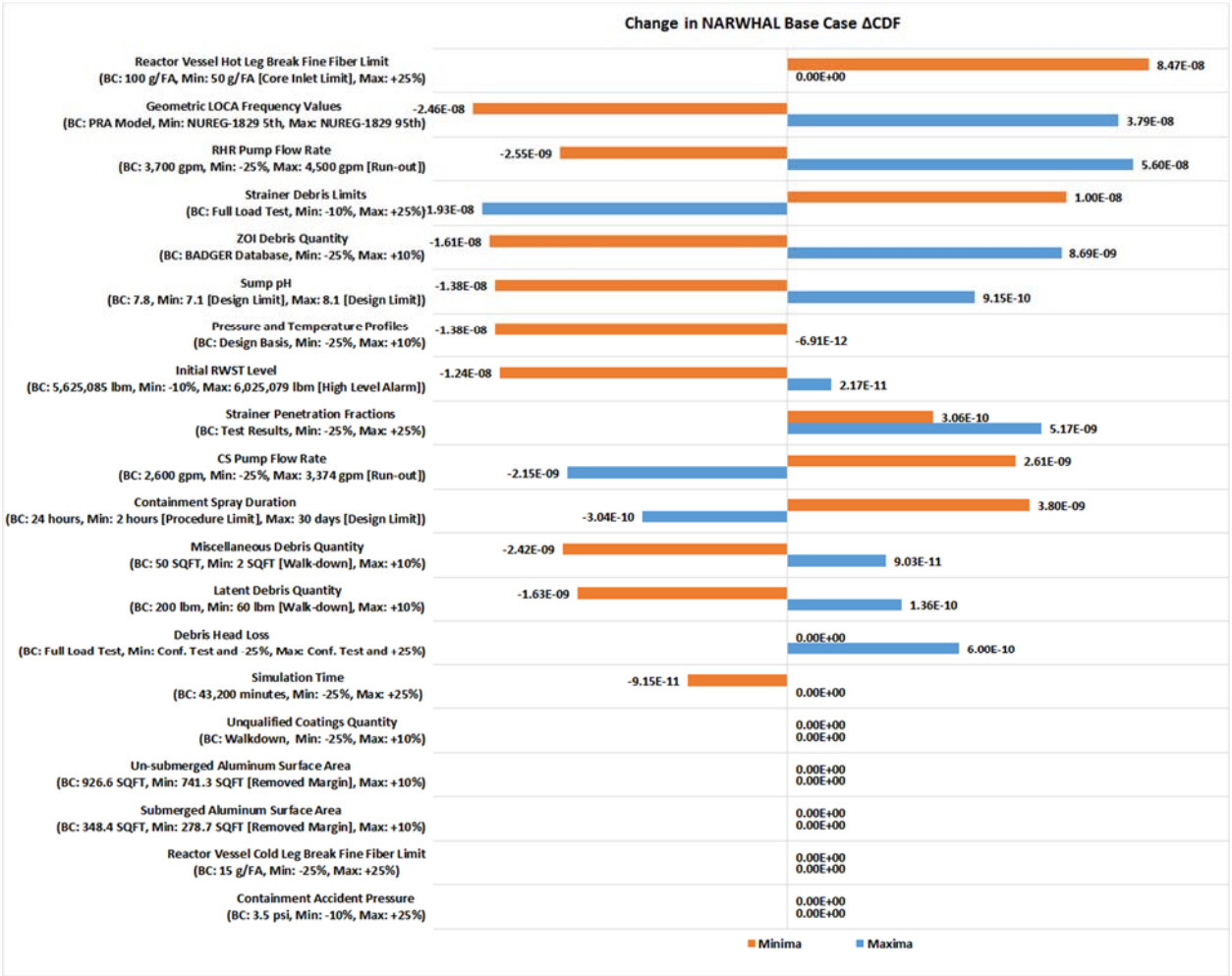

Figure 3-9 – Tornado Diagram Showing Risk Sensitivity Ranking

The parameters that have the most significant effect on ΔCDF are the in-vessel fiber limit for hot leg breaks, the LOCA frequency values, and pump flow rates. Reducing the total fine fiber limit in the reactor vessel for hot leg breaks affects the number of core failures. Note that for the NARWHAL base case, no core fiber limit failures were observed. However, when reducing the total fiber limit to 50 g/FA, some failures were observed.

Adjusting the LOCA frequencies to the 5th or 95th percentiles affects the results in two ways—first, it is an input for calculating the CFP values as described in Section 13.0, and second, it is a direct input for calculating ΔCDF as described in Section 14.2.1. The effect on the CFP values is relatively minor and is driven by the change in frequency as a function of break size. The direct effect on ΔCDF is driven by the magnitude of the LOCA frequency, which has a significant effect because the 5th and 95th percentiles are in some case orders of magnitude different from the mean LOCA frequencies.

The flow rates of the RHR and CS pumps also affect the ΔCDF. Generally, a higher RHR flow rate and a lower CS flow rate results in more fiber debris deposited on the RHR strainers. A lower RHR flow rate and higher CS flow rate results in more fiber being deposited on the CS strainers.

The strainer penetration fraction resulted in a higher ΔCDF for the minimum input and the maximum input when compared to the base case. The maximum penetration fractions increased ΔCDF because it resulted in an increased quantity of fine fiber deposition in the reactor vessel. This led to core failures, which were not seen in the NARWHAL base case. The minimum penetration fractions increased ΔCDF because it resulted in an increased quantity of fine fiber deposited on the strainers. At each time step, less of the fine fiber that arrives penetrated the strainer, and less fine fiber that was previously on the strainer was shed. Although this does not have a large effect, it led to a few more fiber debris limit failures than were seen in the NARWHAL base case.

Additionally, the containment temperature and pressure profiles parameter resulted in a lower ΔCDF for both the minimum and maximum values. The minimum temperature and pressure profiles result in a lower ΔCDF because the quantity of chemical precipitates that form is reduced, which leads to less debris limit failures. The maximum temperature and pressure profiles result in a negligibly lower ΔCDF than when compared to the base case. This is because the higher temperature results in a larger volume of water in the containment pool, which affects the rate at which debris accumulates on the strainer. For a handful of breaks, this variation in the rate of debris accumulation resulted in success instead of failure due to the debris limit failure criterion.

14.2.3 Uncertainty Quantification

As described in Enclosure 1, Section 5.0, uncertainty quantification includes parametric uncertainty, model uncertainty, and completeness uncertainty. The parametric and model uncertainties were quantified by running NARWHAL sensitivity cases.

Note that the parametric uncertainty evaluation has a different purpose than the parametric sensitivity analysis described in Section 14.2.2. The purpose of the parametric sensitivity analysis was to determine the effect of one-at-a-time changes in various input parameters to understand the independent effect of each parameter on the results. In many cases, the parameter changes went outside the bounds of the realistic plant-specific conditions. However, the purpose of the parametric uncertainty quantification was to quantify the overall uncertainties associated with the input parameters. Therefore, the effect of simultaneous variations in multiple input parameters was considered, but none of the inputs were shifted beyond the bounds of realistic plant-specific conditions.

The parametric uncertainties were quantified using a series of sensitivities with a bounding set of input parameters with respect to a) strainer failures and b) core failures.

In cases where the bounding direction for a given input parameter (e.g., pool volume/level) could not be determined, both the minimum and maximum values were run. Table 3-13 shows the worst case conditions for strainer failures, and Table 3-14 shows the worst case conditions for core failures.

Table 3-13 – Worst Case Conditions for Strainer Failure

Parameter	Bounding Direction	NARWHAL Base Case Input	Sensitivity Case Input
Fiber Insulation Debris Quantity	Maximum	Consensus (Maximum)	Same as NARWHAL Base Case
Qualified Coatings Debris Quantity	Maximum	Consensus (Maximum)	Same as NARWHAL Base Case
Fire Barrier Debris Quantity	Maximum	Consensus (Maximum)	Same as NARWHAL Base Case
Unqualified Coatings Debris Quantity	Maximum	Consensus (Maximum)	Same as NARWHAL Base Case
Latent Debris Quantity	Maximum	Consensus (Maximum)	Same as NARWHAL Base Case
Miscellaneous Debris Quantity	Maximum	Consensus (Maximum)	Same as NARWHAL Base Case
Debris Transport Fractions	Maximum	Consensus (Maximum)	Same as NARWHAL Base Case
Pool Volume/Level	Minimum or Maximum	Minimum	Minimum and Maximum
Containment Pressure	Minimum	Consensus (Minimum)	Same as NARWHAL Base Case
Pool Temperature	Minimum or Maximum	Design Basis (Maximum)	Same as NARWHAL Base Case (Maximum is Conservative Based on Parametric Sensitivity Results)
ECCS Flow Rate	Maximum	Design (based on comparison of the pump curve and system resistance)	Maximum
CS Flow Rate (assuming sprays initiate)	Minimum	Design	Minimum
ECCS/CS Switchover Time	Minimum	Function of Water Volume and Flow Rates	Function of Water Volume and Flow Rates
Hot Leg Switchover Time	N/A	Procedural Step (Minimum)	Same as NARWHAL Base Case

Parameter	Bounding Direction	NARWHAL Base Case Input	Sensitivity Case Input
Secure CS Time	Minimum or Maximum	Midpoint	Minimum and Maximum
Boil-off Flow Rate	N/A	Consensus (Maximum)	Same as NARWHAL Base Case
pH	Minimum or Maximum	Maximum for release, Minimum for solubility	Same as NARWHAL Base Case
Head Loss	Maximum	Consensus (Maximum)	Same as NARWHAL Base Case
Structural Margin	Minimum	Design (Minimum)	Same as NARWHAL Base Case
NPSH Margin	Minimum	Consensus (Minimum)	Same as NARWHAL Base Case
Pump Void Fraction Limit	Minimum	Consensus (Minimum)	Same as NARWHAL Base Case
Penetration	Minimum	Consensus (Maximum)	Minimum (no penetration)
Core Fiber Limit	N/A	Consensus (Minimum)	Same as NARWHAL Base Case
LOCA Frequency	Maximum	Nominal (mean)	Maximum (95th percentile)

Table 3-14 – Worst Case Conditions for Core Failure

Parameter	Bounding Direction	NARWHAL Base Case Input	Sensitivity Case Input
Fiber Insulation Debris Quantity	Maximum	Consensus (Maximum)	Same as NARWHAL Base Case
Qualified Coatings Debris Quantity	N/A	Consensus (Maximum)	Same as NARWHAL Base Case
Fire Barrier Debris Quantity	N/A	Consensus (Maximum)	Same as NARWHAL Base Case
Unqualified Coatings Debris Quantity	N/A	Consensus (Maximum)	Same as NARWHAL Base Case
Latent Debris Quantity	Maximum	Consensus (Maximum)	Same as NARWHAL Base Case
Miscellaneous Debris Quantity	N/A	Consensus (Maximum)	Same as NARWHAL Base Case
Debris Transport Fractions	Maximum	Consensus (Maximum)	Same as NARWHAL Base Case
Pool Volume/Level	Minimum or Maximum	Minimum	Minimum and Maximum

Parameter	Bounding Direction	NARWHAL Base Case Input	Sensitivity Case Input
Containment Pressure	N/A	Consensus (Minimum)	Same as NARWHAL Base Case
Pool Temperature	Minimum or Maximum	Design Basis (Maximum)	Same as NARWHAL Base Case (Maximum is Conservative Based on Parametric Sensitivity Results)
ECCS Flow Rate	Minimum or Maximum	Design (based on comparison of the pump curve and system resistance)	Minimum and Maximum
CS Flow Rate (assuming sprays initiate)	Minimum	Design	Minimum
ECCS/CS Switchover Time	Minimum	Function of Water Volume and Flow Rates	Function of Water Volume and Flow Rates
Hot Leg Switchover Time	Maximum	Procedural Step (Minimum)	Maximum
Secure CS Time	Minimum	Midpoint	Minimum
Boil-off Flow Rate	Maximum	Consensus (Maximum)	Same as NARWHAL Base Case
pH	Minimum or Maximum	Maximum for release, Minimum for solubility	Same as NARWHAL Base Case
Head Loss	N/A	Consensus (Maximum)	Same as NARWHAL Base Case
Structural Margin	N/A	Design (Minimum)	Same as NARWHAL Base Case
NPSH Margin	N/A	Consensus (Minimum)	Same as NARWHAL Base Case
Pump Void Fraction Limit	N/A	Consensus (Minimum)	Same as NARWHAL Base Case
Penetration	Maximum	Consensus (Maximum)	Same as NARWHAL Base Case
Core Fiber Limit	Minimum	Consensus (Minimum)	Same as NARWHAL Base Case
LOCA Frequency	Maximum	Nominal (mean)	Maximum (95th percentile)

Because minimum and maximum inputs were considered for both the pool volume and CS duration inputs, a 2x2 matrix of simulations was required for the bounding strainer

failure cases. Similarly, because minimum and maximum inputs were considered for both the pool volume and RHR flow rate inputs, a separate 2x2 matrix of simulations was required for the bounding core failure cases. The difference in the bounding strainer and core failure cases compared to the NARWHAL base case are summarized below:

1. Strainer Failure Cases (2x2 Matrix)
   a. Water Volume
      i. Minimum (NARWHAL Base Case Inputs)
      ii. Maximum (Approximately 500,000 lbm Additional Water)
   b. Maximum RHR Flow Rate (4,500 gpm)
   c. Minimum CS Flow Rate (1,950 gpm)
   d. CS Duration
      i. Minimum (120 minutes)
      ii. Maximum (43,200 minutes)
   e. Minimum Penetration (0 percent)
   f. Maximum LOCA Frequency (95th Percentile)
2. Core Failure Cases (2x2 Matrix)
   a. Water Volume
      i. Minimum (NARWHAL Base Case Inputs)
      ii. Maximum (Approximately 500,000 lbm Additional Water)
   b. RHR Flow Rate
      i. Minimum (2,775 gpm)
      ii. Maximum (4,500 gpm)
   c. Minimum CS Flow Rate (1,950 gpm)
   d. Maximum hot leg switchover (HLSO) Time (563 minutes)
   e. Minimum CS Duration (120 minutes)
   f. Maximum LOCA Frequency (95th Percentile)

Table 3-15 shows the ΔCDF for each of the parametric uncertainty cases. Figure 3-10 illustrates the change in ΔCDF for each of the parametric uncertainty cases in comparison to the NARWHAL base case.

Table 3-15 – Results of Parametric Uncertainty Quantification

Sensitivity Case	Description	ΔCDF	Change in ΔCDF from NARWHAL Base Case
Strainer Case 1	Min water volume and min CS duration	1.79E-07	1.55E-07
Strainer Case 2	Min water volume and max CS duration	1.76E-07	1.51E-07
Strainer Case 3	Max water volume and min CS duration	1.09E-07	8.39E-08
Strainer Case 4	Max water volume and max CS duration	1.08E-07	8.30E-08
Core Case 1	Min water volume and min RHR flow rate	7.11E-08	4.65E-08
Core Case 2	Min water volume and max RHR flow rate	1.63E-07	1.39E-07
Core Case 3	Max water volume and min RHR flow rate	7.23E-08	4.76E-08
Core Case 4	Max water volume and max RHR flow rate	9.39E-08	6.92E-08

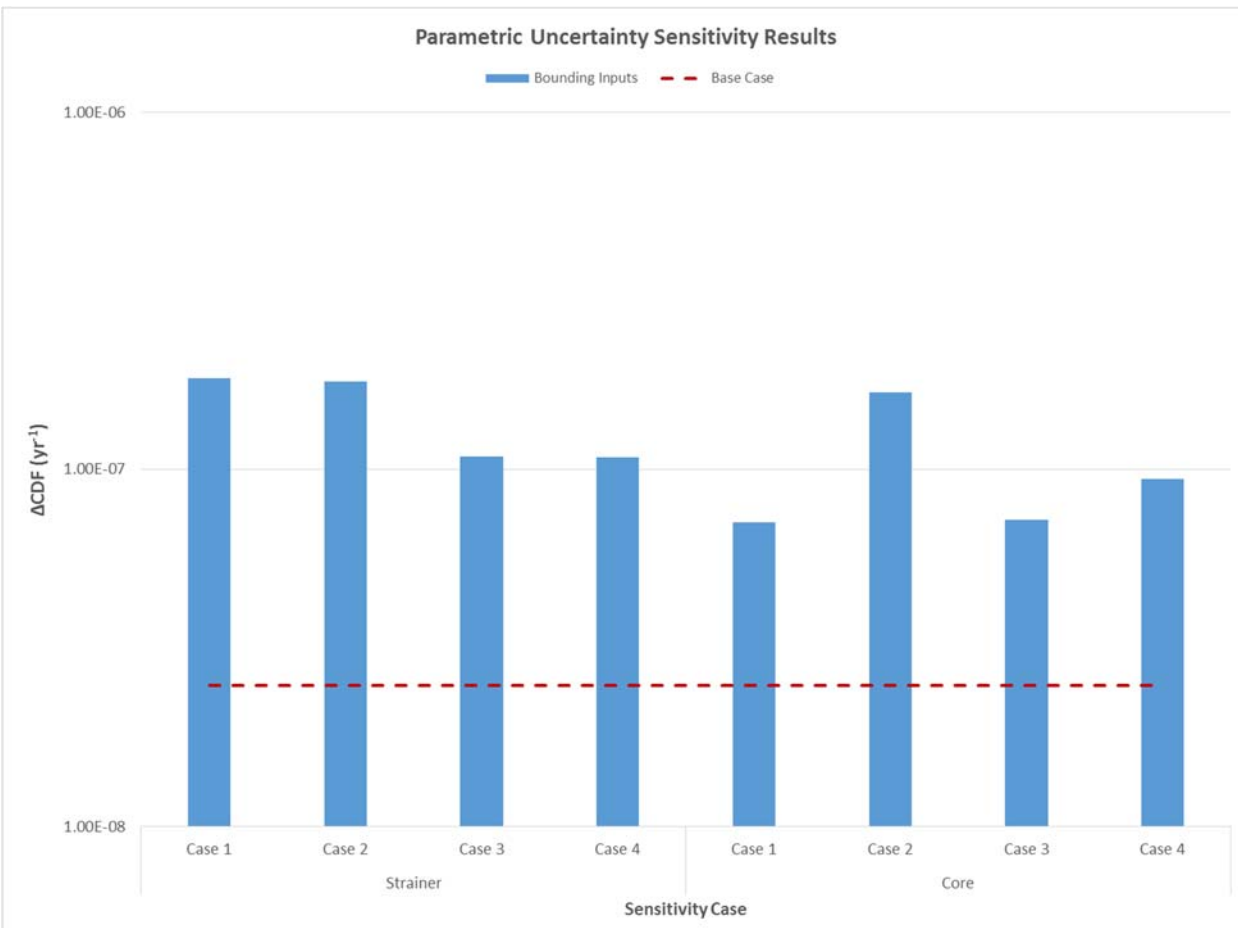

Figure 3-10 – Comparison of Parametric Uncertainty Sensitivity Cases to the NARWHAL Base Case

Risk Quantification

The process described above evaluates the parametric uncertainty in a very conservative manner by analyzing the worst-case combinations of input values. Although the scenario is hypothetically possible, the probability of all of the worst-case conditions occurring simultaneously is extremely unlikely. The overall results of this evaluation show that the parametric uncertainty is low.

To meet the guidance in NUREG-1855 (Reference 11), model uncertainty must be addressed for any models or approaches for which no consensus exists. As discussed in Enclosure 1, Section 5.0, most of the GSI-191 models used for the VEGP evaluation are consensus models that have been widely used by the industry and accepted by the NRC. However, the following models used for VEGP are not consensus models and therefore were included in the model uncertainty quantification evaluation:

- Break model
- LOCA frequencies
- LOCA frequency allocation to individual welds
- CS actuation
- Aluminum metal release equation
- Fiber bed thickness required for chemical head loss
- LBLOCA size range discretization
- Chemical product generation
- NARWHAL time step size

To address the uncertainty in these models, alternative models were evaluated as shown in Table 3-16.

Table 3-16 – Alternative Models Used to Quantify Model Uncertainty

Model	NARWHAL Base Case	Sensitivity Case(s)
Break model	Continuum break model	DEGB-only model
LOCA frequencies	VEGP PRA frequencies (derived from NUREG-1829 geometric mean)	NUREG-1829 arithmetic mean frequencies
LOCA frequency allocation	Top-down allocation	Hybrid allocation with multiple options (based on weld degradation mechanism probability weighting)
CS actuation	Hot leg breaks larger than 15 inches	Multiple options including no breaks and all breaks larger than 2 inches
Aluminum metal release	UNM release equation	WCAP-16530 release equation
Fiber bed thickness required for chemical head loss	0.45 inches	0 inches

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Risk Quantification

Model	NARWHAL Base Case	Sensitivity Case(s)
LBLOCA size range discretization	(6-15, 15-25, and 25-43.84 inches)	Multiple options with a biased allocation of frequencies to smaller break sizes and larger break sizes
Chemical product generation	• UNM aluminum metal release equation • ANL solubility equation • Maximum sump recirculation pH and design basis temperature profiles	<u>Case 1</u> • WCAP-16530 release equation for aluminum • Howe solubility equation • Best estimate sump pH (7.42) • Best estimate sump and containment temperature profiles <u>Case 2</u> Increased calcium phosphate debris limit by a factor of 2.5 to eliminate the failures observed in the base case. <u>Case 3</u> • WCAP-16530 release equation for aluminum • Howe solubility equation • Allowed aluminum to remain in solution after precipitation occurs; precipitation not forced at 24 hours.
NARWHAL time step size	1 minute for the first 24 hours	Multiple cases with time steps of 2, 3, 4, 5 and 15 minutes for the first 24 hours

Risk Quantification

Table 3-17 shows the ΔCDF for each of the model uncertainty cases. Figure 3 and 3-12 illustrate the change in ΔCDF for each of the model uncertainty cases in comparison to the NARWHAL base case.

Table 3-17 – Results of Model Uncertainty Quantification

Model with No Consensus	Sensitivity Case	ΔCDF	Change in ΔCDF from NARWHAL Base Case
Continuum Break Model	DEGB-Only Model	8.10E-08	5.63E-08
Top-Down LOCA Frequency Allocation with Values from the VEGP PRA	Top-Down LOCA Frequency Allocation with NUREG-1829 Arithmetic Mean Values	5.29E-07	5.04E-07
Methodology to Allocate LOCA Frequency to Welds	Hybrid Allocation Methodology: Skewed to High Rupture Probability Welds	4.91E-11	-2.46E-08
	Hybrid Allocation Methodology: Skewed to High and Medium Rupture Probability Welds	3.55E-09	-2.11E-08
	Hybrid Allocation Methodology: Spread Equally Across all Welds (top-down)	2.47E-08	0.00E+00
Breaks Activating Containment Sprays	All Breaks >15"	2.42E-08	-4.46E-10
	All Breaks >6"	2.39E-08	-7.61E-10
	All Breaks >2"	2.39E-08	-7.61E-10
	No Breaks	2.70E-08	2.39E-09
UNM Aluminum Metal Release Equation	WCAP-16530 Equation	2.57E-08	9.98E-10
0.45-inch Fiber Thickness Required for Chemical Head Loss	0-inch Fiber Thickness Required for Chemical Head Loss	6.74E-08	4.27E-08
LBLOCA Size Range Discretization (6-15, 15-25, and 25-43.84 inches)	Bias 1 (6-10, 10-15, and 15-43.84 inches)	5.13E-08	2.67E-08
	Bias 2 (6-20, 20-27, and 27-43.84 inches)	2.33E-08	-1.31E-09

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Risk Quantification

Model with No Consensus	Sensitivity Case	ΔCDF	Change in ΔCDF from NARWHAL Base Case
Chemical Effects Sensitivities	Case 1 (Howe Solubility, WCAP-16530 Aluminum Metal Release, Best Estimate pH, Best Estimate Temperature Profiles)	1.43E-08	-1.04E-08
	Case 2 (No Calcium Phosphate Debris Limit Failure)	8.87E-09	-1.58E-08
	Case 3 (Howe Solubility, WCAP-16530 Aluminum Metal Release, Full Solubility)	2.49E-08	2.33E-10
NARWHAL Time Step Size	2 minutes	2.48E-08	1.00E-10
	3 minutes	2.45E-08	-1.31E-10
	4minutes	2.49E-08	2.69E-10
	5 minutes	2.41E-08	-5.53E-10
	15 minutes	4.47E-08	2.01E-08

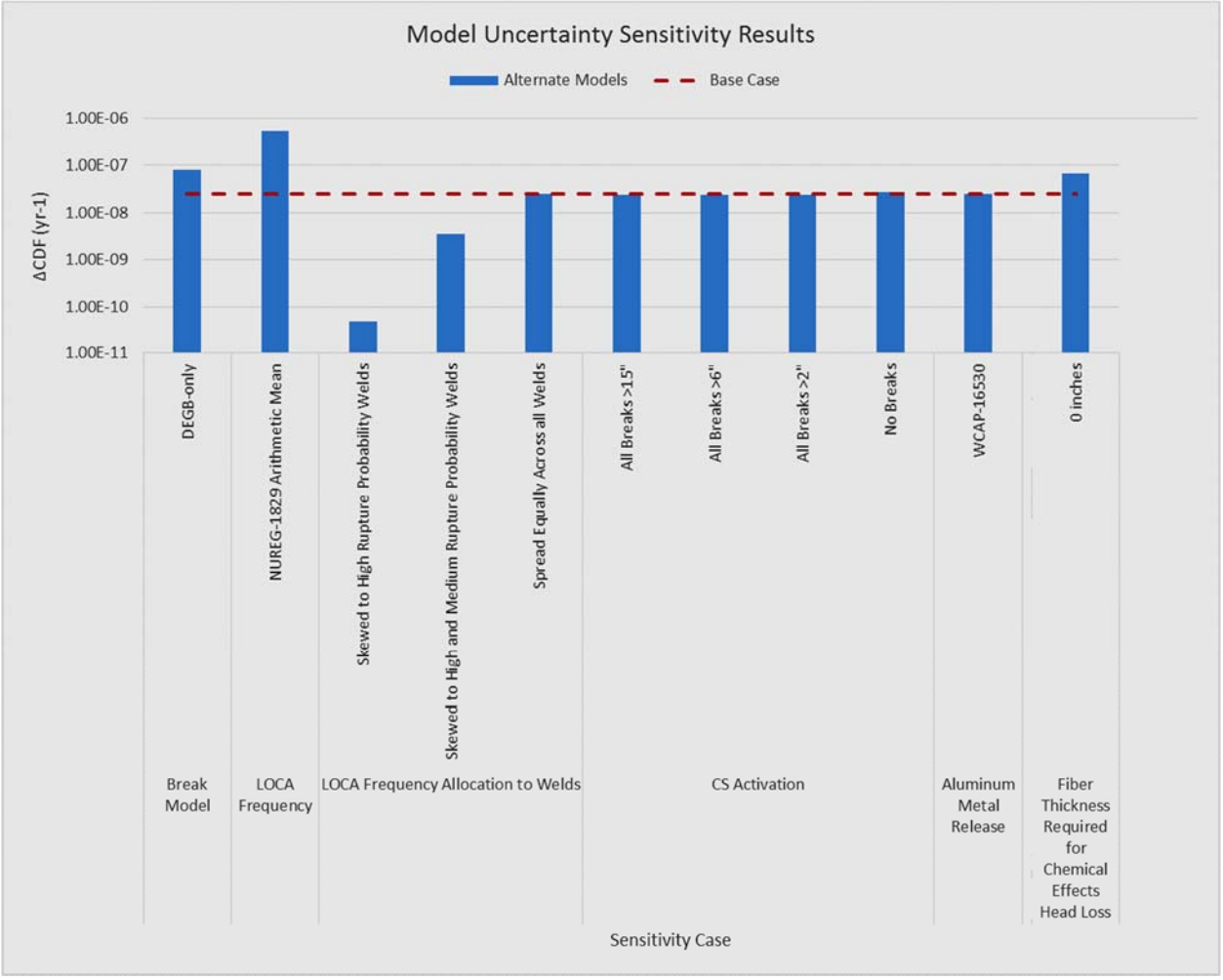

Figure 3-11 – Comparison of Model Uncertainty Sensitivity Cases to the Base Case (1 of 2)

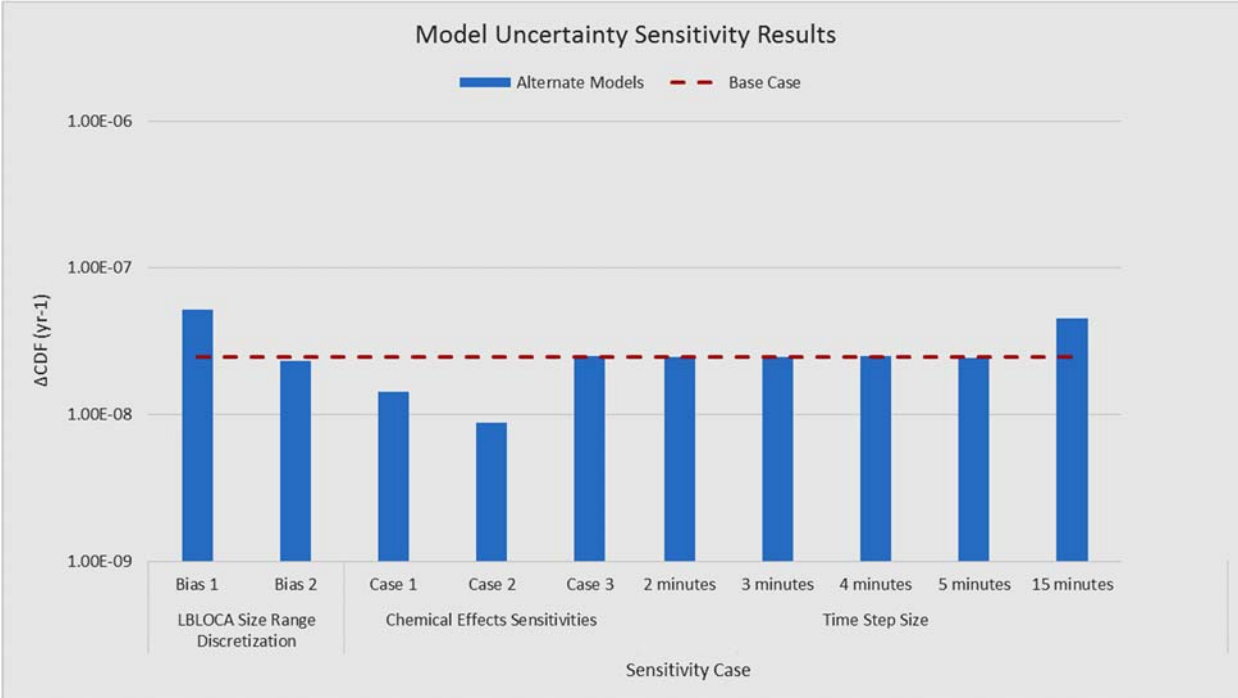

Figure 3-12 – Comparison of Model Uncertainty Sensitivity Cases to the Base Case (2 of 2)

The results of this evaluation show that the model uncertainty is low (i.e., the resulting ΔCDF for each of the model uncertainty cases is within Region 3 as defined in RG 1.174).

Because all of the cases that were evaluated for model uncertainty and parametric uncertainty resulted in a ΔCDF less than $1x10^{-6}$, it can be concluded with high confidence that the risk associated with GSI-191 is very low as defined by the acceptance guidelines in RG 1.174 (Reference 1).

14.3 PRA Model Uncertainty and Sensitivity

The purpose of this section is to address the impact of PRA modeling epistemic uncertainty on the GSI-191 risk assessment. The baseline internal events and seismic PRA models document assumptions and sources of uncertainty, and these have been reviewed during the model peer reviews. Therefore, the approach taken was to review these PRA models and documentation to identify those items that may be directly relevant to the GSI-191 risk assessment, perform sensitivity analyses where appropriate, and discuss the results with dispositions for the uncertainties.

14.3.1 Internal Events PRA Model Uncertainty

The epistemic uncertainty analysis approach described below applies to the Internal Events PRA.

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Risk Quantification

The baseline Internal Events PRA model uncertainty report was developed based on the guidance in NUREG-1855 (Reference 11). As described in NUREG-1855, sources of uncertainty include "parametric" uncertainties, "modeling" uncertainties, and "completeness" (or scope and level of detail) uncertainties. (The epistemic uncertainties unique to the seismic PRA are addressed in a later section.)

Parametric uncertainty was addressed as part of the VEGP baseline PRA model aleatory uncertainty analysis.

Assumptions are made during the PRA development as a way to address a particular modeling uncertainty because there is not a single definitive approach. The assumptions are defined consistent with the definition provided in NUREG-1855 (Reference 11). Plant-specific assumptions made for each of the VEGP Internal Events PRA technical elements are noted in the individual PRA notebooks. These assumptions were collected from each notebook and evaluated to determine if they are related to source of modeling uncertainty, and if so that uncertainty was characterized. In addition, EPRI TR-1016737 (Reference 12) compiled a listing of generic sources of modeling uncertainty for each PRA technical element, which were also considered.

Completeness uncertainty addresses scope and level of detail of the PRA model. Uncertainties associated with scope and level of detail are documented in the PRA, but are only considered for their impact on a specific application.

From the characterization of potential sources of uncertainty in the baseline Internal Events PRA model and of supplementary issues from EPRI TR-1016737 (Reference 12), the following items may impact the internal events PRA results. Sensitivity analyses are included to further evaluate these items as a source of uncertainty.

Table 3-18 provides a summary of the evaluation assessing the impact of the identified sources of model (epistemic) uncertainty on the GSI-191 risk assessment. For each of the sources of uncertainty, the potential impact on the GSI-191 risk assessment is addressed, either qualitatively or by an appropriate sensitivity case, to determine the impact on the GSI-191 application.

Table 3-18 – Assessment of VEGP Internal Events PRA Epistemic Uncertainty Impacts

Source of Epistemic Uncertainty	Related Assumptions	Sensitivity Case	Disposition
High RCS pressure impacts the potential for induced steam generator tube rupture (SGTR). Medium and large LOCA and reactor vessel rupture are treated as low RCS pressure scenarios. All other core damage sequences are considered high pressure sequences where the induced SGTR failure mode is possible. Therefore, the baseline PRA model may overestimate the contribution of induced SGTR to LERF.	Scenarios with significant RCP seal leakage, a stuck open pressurizer valve, or a pressurizer PORV open for feed-and-bleed cooling are conservatively considered high RCS pressure scenarios.	Sensitivity cases were performed for the baseline Internal Events PRA by reclassifying the identified scenarios as low RCS pressure to determine impact on LERF.	The GSI-191 risk assessment demonstrates that only large LOCAs could result in debris related failures. Therefore, the possible overestimation of induced SGTR for high pressure scenarios has no impact on the GSI-191 risk assessment.
Certain initiating events can be affected by seasonal variations (e.g., loss of offsite power (LOSP), loss of service water (SW), etc.) and baseline PRA does not address seasonal variations.	The generic industry frequency for the LOSP event developed in NUREG/CR-6890 is applicable to the VEGP site. The NSCW cooling towers are not required during cold weather months.	None	The GSI-191 phenomena are of concern for initiating events that could generate debris from insulation materials and coatings inside containment, which could then be transported to the containment sump and fail the ECCS sump suction strainers during the recirculation phase needed

Source of Epistemic Uncertainty	Related Assumptions	Sensitivity Case	Disposition
			to maintain core cooling. For VEGP, the initiating events that meet these criteria are LOCAs and SSBI. Therefore, seasonal variations of certain other initiating events have no impact on the GSI-191 risk assessment.
The method of calculation of human error probabilities (HEPs) for the Human Reliability Analysis (HRA) may introduce uncertainty based on the particular methodology applied.	Detailed evaluations of HEPs are performed for the risk significant, pre- and post-initiator human failure events (HFEs) using industry consensus methods. The Technique for Human Error Rate Prediction (THERP) method is applied for pre-initiator HFEs. The Cause-Based Decision Tree Method (CBDTM) is used for cognitive errors and THERP for execution errors for post-initiator HFEs.	The overall modeling uncertainty associated with the general basis for HEPs is addressed by the standard baseline PRA HEP sensitivity cases for the internal events PRA.	Since the VEGP PRA model is based on industry consensus modeling approaches for its HEP calculations, and there are no additional HFEs added for the GSI-191 risk assessment, this is not considered a significant source of epistemic uncertainty and therefore has no impact on the GSI-191 risk assessment.
The VEGP PRA medium LOCA frequency is based upon data from NUREG/CR-6928, which is an order of magnitude higher than the previous	None	A sensitivity was performed for the Internal Events to determine the impact of the increased medium LOCA frequency from	The LOCA frequency values in NUREG/CR-6928 are in turn based on the LOCA frequency data from NUREG-1829. NUREG-1829 data are used to develop

Source of Epistemic Uncertainty	Related Assumptions	Sensitivity Case	Disposition
data used from NUREG/CR-5750.		NUREG/CR-5750. A more than 10% increase in CDF and nearly 9% increase in LERF occurs due to the updated data.	LOCA frequencies for the GSI-191 risk assessment. The GSI-191 risk impact, however, is not sensitive to the initiating event frequency for medium LOCAs. No medium LOCAs result in sump strainer or core failures due to the effects of debris.
Steam generator (SG) tube condition affects the probabilities of induced SGTR. The current VEGP 3-18SG tube condition is pristine.	If SG tube condition degrades, the induced SGTR probability during secondary side break or anticipated transient without scram for pressure- or thermal-induced SGTR in the LERF analysis would increase.	A sensitivity analysis was performed with average vs. pristine SG tube conditions. CDF increased by slightly more than 1%, while LERF nearly tripled.	The GSI-191 risk assessment demonstrates that only large LOCA could result in sump strainer failure. Therefore, the possible under-estimation of induced SGTR for high pressure scenarios has no impact on the GSI-191 risk assessment.
The presence of water in the reactor cavity at the time of vessel breach would affect the probability of containment failures (early release due to steam explosion and late release due to base mat melt through).	The base internal events VEGP Level 2 PRA assumes a dry reactor cavity condition.	A sensitivity study was performed for a wet reactor cavity. CDF increased by less than 2%, and LERF increased by more than 12%.	The risk increase for large early release due to GSI-191 is nearly three orders of magnitude below the RG 1.174 Region III risk acceptance criteria. A 12% increase in the large early release frequency would still be well below the Region III threshold.

14.3.2 Seismic PRA Model Uncertainty

The ASME/ANS PRA Standard (Reference 13) and RG 1.200 (Reference 14) include a number of requirements related to identification and evaluation of the impact of assumptions and sources of uncertainty on the PRA results. NUREG-1855 (Reference 11) and EPRI TR-1016737 (Reference 12) provide guidance on assessment of uncertainty for applications of a PRA. Sources of uncertainty within the VEGP seismic PRA model are addressed as follows:

- Parametric uncertainty was addressed as part of the VEGP seismic PRA model quantification.
- Modeling uncertainties and associated assumptions specific to the VEGP seismic PRA technical elements are noted in the seismic PRA documentation and were subject to peer review.
- Completeness uncertainty addresses scope and level of detail. Uncertainties associated with scope and level of detail are documented in the seismic PRA. No specific completeness issues were identified in the VEGP seismic PRA peer review.

A summary of potentially important sources of uncertainty in the VEGP seismic PRA is provided in Table 3-19.

Table 3-19 – Assessment of VEGP Seismic PRA Uncertainty Impacts

PRA Element	Summary of Treatment of Sources of Uncertainty	Potential Impact on Seismic PRA Results
Seismic Hazard	The VEGP seismic PRA peer review team noted that both the aleatory and epistemic uncertainties were addressed by characterizing the seismic sources. The review team commented that the site response analysis did not fully evaluate and model aleatory and epistemic uncertainties in the site response analysis.	With regard to aleatory and epistemic uncertainties in the site response analysis, there is an abundance of site-specific data from VEGP Units 3 and 4 that reduces epistemic uncertainty to an insignificant level. The characterization of the seismic hazard reasonably reflects sources of uncertainty.
Seismic Fragilities	The seismic PRA peer review team had no comments on sources of uncertainty pertaining to fragilities.	Several sensitivity studies evaluate the impact of changes to fragilities on the seismic PRA results as one means of assessing the impact of fragilities uncertainties on the seismic PRA results. No changes to the model were recommended based on these results.
Seismic PRA Model	The seismic PRA peer review team commented that the VEGP seismic PRA team relied on the UNCERT code for the propagation of the parametric uncertainties in the seismic PRA with little explanation or documentation of the meaning of the uncertainties results.	The seismic PRA quantification report includes a discussion of sources of model uncertainty, and potentially important sources have been addressed in the sensitivity analysis. No changes to the model were recommended based on these results.

15.0 References

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2. NEI 04-07 Volume 2, 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

3. NUREG-1829, “Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process,” April 2008
4. ANSI/ANS-5.1-1979, “American National Standard for Decay Heat Power in Light Water Reactors,” August 1979
5. WCAP-17788-P, Revision 0, “Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090),” July 2015
6. WCAP-16530-NP-A, “Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191,” March 2008
7. NARWHAL-SUM-02, Revision 1, “NARWHAL Version 2.1 Software User’s Manual,” September 9, 2016
8. Howe, Kerry J., ET. Al, “Corrosion and solubility in a TSP-buffered chemical environment following a loss of coolant accident: Part 1 – Aluminum,” Nuclear Engineering and Design, Volume 292, October 2015: 296–305
9. ML080230038, “NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Strainer Head Loss and Vortexing,” March 2008
10. Regulatory Guide 1.82, Revision 4, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident,” March 2012
11. NUREG-1855, Revision 1, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking,” March 2017
12. EPRI Report 1016737, “Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments,” December 2008
13. 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
14. Regulatory Guide 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” March 2009

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Enclosure 4

Defense-in-Depth and Safety Margin

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Defense-in-Depth and Safety Margin

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Enclosure 4

Defense-in-Depth and Safety Margin

1.0 Introduction

For the purpose of this VEGP risk-informed 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 includes mitigative design features and actions that address protection of the public from radiation due to sequences that go to failure (e.g., containment integrity, emergency plans, operator actions not credited in the GSI-191 evaluation, use of FLEX, etc.).

Similarly, safety margin is defined as the response to the question of what aspects of the analysis increase 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 DID evaluation shows that there is adequate system capability to provide assurance that public health and safety are protected in the event that a 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 design for the VEGP containment buildings.

The safety margin evaluation identifies many conservatisms throughout the evaluation, which provides high confidence that successful end states are truly successful, and that many end states that are assumed to fail in reality would also be successful.

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 (LB) 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 VEGP 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. Finally, the evaluation shows if and how the proposed changes affect the barriers for release of radioactivity and emergency plan actions.

2.1 Evaluation for RG 1.174 DID Philosophy

VEGP 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:

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Defense-in-Depth and Safety Margin

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. VEGP has performed various physical and procedural changes, for example, installation of new strainers with increased surface areas and a reduced opening size, increased RWST inventory to sump pool, removal of problematic insulation materials, procedural changes to delay isolation of RHR pumps from RWST, and program controls to ensure the debris load limits are not exceeded. Additional changes are being planned, for example, modifying the height of the RHR strainers and sump recirculation initiation sequence. These changes reduced the risk associated with the effects of LOCA-generated debris. 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 ΔCDF and ΔLERF 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, 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). As discussed above, the modifications made as part of the proposed licensing basis change do not change the redundancy, independence, and diversity of the ECCS or containment spray system. These systems have been fully analyzed relative to their contribution to nuclear safety through plant-specific 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 Enclosure 3, Section 14.4, the uncertainties in the risk-informed approach were examined. Although the use of alternate models or variations in inputs can in some cases result in higher calculated ΔCDF values, all uncertainty quantification cases showed risk results in Region III of RG 1.174 (Reference 1), which provides high confidence that there are no risk outliers.

- 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 not affected by the physical and procedural changes. 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 RHR pumps take suction from the containment recirculation sumps and supply flow back to the reactor directly and/or through the CCPs and 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 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 VEGP DID measures for detecting and mitigating adverse conditions in order to support the VEGP 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 NPSH margin associated with the high head losses across the sump strainers and debris bed. For VEGP, 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.

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 VEGP emergency operating procedures (EOPs) provide the operators with guidance on monitoring sump strainer blockage (e.g., Procedures 19010-C “E-1 Loss of Reactor or Secondary Coolant”, 19013-C “ES-1.3 Transfer to Cold Leg Recirculation”, and 19111-C “ECA-1.1 Loss of Emergency Coolant Recirculation”). If sump blockage is detected, Procedure 19113-C “ECA-1.3 Recirculation Sump Blockage” provides actions that operators should take to respond to the condition.
- VEGP EOPs incorporated the Bulletin 2003-01 training and procedural guidance to expedite plant cooldown in response to a small break LOCA.
- For small to medium break LOCAs, depletion of the RWST can be delayed by following Procedure 19012-C “ES-1.2 Post LOCA Cooldown and Depressurization". This procedure 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 686,000 gallons, and the low-level alarm setting is 696,448 gallons. The RWST level is normally maintained above the low-level alarm setting, and the nominal volume of the tank is 715,000 gallons.

Several measures are in place to control the debris sources inside the VEGP containment buildings.

- 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.
- For the Technical Requirements Manual Surveillance Requirement (TRS) 13.5.1.1, VEGP has implemented Procedure 14900-C, “Containment Exit Inspection” in conjunction with 00303-C, “Containment Entry”. Per these procedures, prior to entering Mode 4 (Hot Shutdown) from Mode 5, several walk-downs are required to be performed by station management and operations personnel to ensure the containment buildings are 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 (SR) 3.5.2.7, VEGP has implemented procedures 14903-1/2 both titled “Containment Emergency Sump Inspection”, to verify by visual inspection that the suction inlets are not restricted by debris and that the sump strainers are correctly configured according to plant design and show no structural distress or abnormal corrosion. These procedures also ensure that the protective covers for the TSP baskets and sump strainer are removed. This inspection is required on an 18-month frequency in accordance with the Surveillance Frequency Control Program.
- VEGP Procedure 00309-C is used 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.
- Inspections of the coatings in containment are part of a protective coatings program complying with Regulatory Guide 1.54 (Reference 2) and ANSI N101.4-1972 (Reference 3), to ensure that coatings do not adversely affect safety-related systems, structures, or components.

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. VEGP 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 Regulatory Guide 1.97 Revision 2 (Reference 4). VEGP has the following methods for detection of sump strainer blockage conditions.

- VEGP has indications in the control room for SI, RHR, and CS pump flows and SI and CS pump discharge pressures. Instrumentation is available to provide the operators with indications of pump cavitation, such as erratic flow or low discharge pressure.
- VEGP has core exit thermocouple (CET) and reactor vessel level indications 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.

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 VEGP EOPs 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.
- VEGP Procedure 19111-C “ECA-1.1 Loss of Emergency Coolant Recirculation" minimizes the pumps required depending on plant conditions and directs shutting down all pumps as applicable. Procedure 19113-C “ECA-1.3 Recirculation Sump Blockage” also contains steps to shut down SI pumps and CCPs that piggyback off a potentially cavitating RHR pump during recirculation.
- VEGP Procedure 19113-C “ECA-1.3 Recirculation Sump Blockage” 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 5), “Mitigation Strategies for Beyond-Design-Basis External Event

(BDBEE)", VEGP developed diverse and flexible coping strategies (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 (RMWST), 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 VEGP plant design has simultaneous hot leg and cold leg injection 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 7.5 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 core exit thermocouple (CET) temperature indication and reactor water level, as monitored by the reactor vessel level instrumentation system (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, VEGP's functional restoration procedure would attempt to establish injection flow of clean water from the RWST. CETs

are monitored during EOP usage as well as for status tree functional restoration entries and the safety parameter display system (SPDS).

- Reactor vessel water level is also monitored and a decreasing water level could indicate a lower core region flow blockage. VEGP 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 or density 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 EOPs 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 to mitigate an inadequate reactor core flow condition, as laid out in Procedures 19221-1/2 “FR-C.1 Response to Inadequate Core Cooling” and 19222-1/2 “FR-C.2 Response to Degraded Core Cooling”. Upon identification of an inadequate RCS inventory or an inadequate core heat removal condition, the EOPs 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 RCPs and open pressurizer PORVs

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 RCP(s) associated with the available idle SG(s). 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.

VEGP 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 SGs 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 VEGP 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 inservice inspection (ISI) program provides rules for the examination and repair of piping and other RCS components, and plays an important role in the prevention of pipe breaks. The integrity of the Class 1 welds, piping, and components are maintained at a high level of reliability through the ASME Section XI inspection program (Reference 6). VEGP ISI procedures also ensure that inspections are performed in accordance with the schedule requirements of the code. The ASME Code Section XI Category B-F welds on the hot legs near the RPV have been mitigated for PWSCC by application of mechanical devices that reverse the stress fields. Stainless steel safe ends have been used to eliminate susceptibility to PWSCC for the B-F welds on the hot legs and cold legs at the steam generators. All B-F welds attached to the pressurizer (including the surge line and pressurizer spray and relief valve piping) have been mitigated by application of a full structural weld overlay. For all of the RCS piping, there are only 4 B-F Alloy 82/182 welds of 4 inches and larger that have not been mitigated. These welds are on the cold leg, as identified in Table 3-9 of Enclosure 3. No flaws have been detected in any of these unmitigated welds.
- 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 (Reference 7).
- The leak detection program at VEGP is capable of early identification of RCS leakage in accordance with RG 1.45 (Reference 8) 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. The VEGP RCS leakage detection system is described in Section 5.2.5 of the FSAR. No changes to the system have been made since April 2015.
- Some of the operator actions outlined in the VEGP SAMG procedures can help maintain integrity of the RCS when directed by the emergency operating procedures. Such actions include injection into SGs and RCS, depressurization of RCS, makeup to RWST, realignment to injection from RWST and flooding containment.

2.3.3 Reactor Containment Integrity

The VEGP containment buildings are 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, CSS, 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 buildings remain 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 VEGP 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.

3.0 Safety Margin

There are numerous conservatisms used throughout the risk-informed GSI-191 evaluation for VEGP. However, not all of these conservatisms were classified as safety margin. Some conservatisms were included to provide future operating margin (i.e., margin added to the current plant conditions to allow for future changes, and flexibility in conducting maintenance or inspections). The key distinction between safety margin and operating margin is that safety margin cannot be reduced without approval from the NRC (Reference 1), whereas operating margin can be modified if necessary based on plant changes.

Table 4-1 describes the safety margins included in the risk-informed GSI-191 evaluation. As noted in this table, there are many conservatisms throughout the evaluation, which provide high confidence that successful end states are truly successful, and that many end states that are assumed to fail in reality would also be successful. Note that in several places, the effect of conservatism on the model is

described as over-predicting or under-predicting a specific physical phenomena or failure. These terms are generically used to refer to either a change in the actual value predicted by the model or an increase in margin. For example, flashing is not predicted to occur. However, because of conservatism in specific portions of the model, the potential for flashing is over-predicted (i.e., there is more real margin to prevent flashing than is predicted by the model).

Defense-in-Depth and Safety Margin

Table 4-1 – Description of Safety Margin

Topic	Conservatism Credited as Safety Margin	Realistic Conditions	Impact on Evaluation
Scenario Frequency	All frequency associated with secondary side breaks is allocated to DEGBs	Smaller breaks on main steam and feedwater piping are much more likely than DEGBs, and would generate significantly lower debris quantities	Overall likelihood of failure is over-predicted for secondary side breaks
Scenario Frequency	Random pump failures are assumed to occur at switchover to recirculation	Random pump failures can occur at the beginning of the event, at the start of recirculation or any time during the event	Failures at the start of the event would delay switchover to recirculation, failures later in the event would result in distribution of debris across more strainers
Thermal-Hydraulics	Initial containment pressure is at technical specification (TS) minimum of -0.3 psig	Containment pressure would be above TS minimum	NPSH margin is under-predicted and degasification and flashing are over-predicted
Thermal-Hydraulics	No credit taken for containment accident pressure in NPSH calculations and minimal credit taken for degasification and flashing calculations	The post-LOCA containment pressure would be significantly higher than the saturation pressure	NPSH margin is under-predicted and degasification and flashing are over-predicted

Defense-in-Depth and Safety Margin

Topic	Conservatism Credited as Safety Margin	Realistic Conditions	Impact on Evaluation
Thermal-Hydraulics	Design basis containment and sump temperature profiles used for all break sizes	Containment and sump temperature profiles would be significantly lower for smaller break sizes	Chemical release, precipitate quantities, degasification, and flashing are over-predicted, and aluminum solubility, corrected strainer head loss, and NPSH margin are under-predicted (all impacts are conservative with the exception of under-predicted aluminum solubility and corrected strainer head loss; however, the sensitivity analysis described in Enclosure 3, Section 14.3 showed that the conservative effects outweigh the non-conservative effects)
Debris Generation	With the exception of shadowing by concrete walls, no credit was taken for structures or restraints that would limit the quantity of debris generated within a break ZOI	Full offset of pipe DEGBs (especially on the primary loop piping) would be significantly limited due to physical restraints; also, insulation and qualified coatings would not be completely destroyed within a given ZOI due to the shielding effects of equipment and other structures	Quantity of debris generated inside the ZOI is over-predicted
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

Defense-in-Depth and Safety Margin

Topic	Conservatism Credited as Safety Margin	Realistic Conditions	Impact on Evaluation
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
Debris Generation	Unqualified coatings fail at the start of the accident	Unqualified coatings would fail gradually and may not fail until much later in the event	Unqualified coatings that fail in upper containment after sprays are secured would not transport
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	Precipitate quantity is over-predicted and precipitates would form later than predicted
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
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 and calcium release from insulation are over-predicted, resulting in an over-prediction of aluminum and calcium precipitate quantities

Defense-in-Depth and Safety Margin

Topic	Conservatism Credited as Safety Margin	Realistic Conditions	Impact on Evaluation
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 (Reference 10).	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
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
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
Debris Transport	Small and large pieces of fiberglass 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 (Reference 11))	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

Defense-in-Depth and Safety Margin

Topic	Conservatism Credited as Safety Margin	Realistic Conditions	Impact on Evaluation
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
Strainer/Pump Failures	A strainer is assumed to fail any time the accumulated debris quantities exceed the tested debris quantities	In many cases, one type of debris (e.g., calcium phosphate) exceeds the tested quantity while other types of debris (e.g., sodium aluminum silicate) are significantly below the tested quantity; also, most breaks have available margin to accommodate higher head losses	The breaks that fail the strainer acceptance criteria are over-predicted
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

Defense-in-Depth and Safety Margin

Topic	Conservatism Credited as Safety Margin	Realistic Conditions	Impact on Evaluation
Strainer/Pump Failures	Debris head loss was conservatively calculated using a rule-based approach (i.e., if the accumulation of a given debris type exceeds a certain threshold, a bounding head loss is automatically applied)	Head loss would increase gradually as debris accumulates and most breaks would not accumulate enough debris to reach the head losses that were applied	The strainer head loss for both conventional and chemical debris is over-predicted
Strainer/Pump Failures	Strainer head loss testing was conservatively performed using a strainer module with fewer disks and scaled up to the full height strainers based on the area ratios	Because flow preferentially passes through the lower disks, it is likely that a larger quantity of debris could accumulate on the full height strainers than predicted using a simple area ratio	The strainer head loss for both conventional and chemical debris is over-predicted
Strainer/Pump Failures	Calcium phosphate head loss was applied for all breaks that generate and transport a sufficient quantity of fiber debris	Based on observations from autoclave tests described in Enclosure 2 Section 3.o.2.9 and other tests representative of VEGP conditions, calcium phosphate precipitation is either unlikely to occur or the actual precipitates would have a negligible effect on head loss	The strainer head loss is over-predicted for calcium phosphate debris
Strainer/Pump Failures	The chemical head loss was extrapolated to 30 days and the extrapolation constant was applied 450 minutes after the start of the event	The head loss associated with the full chemical debris load is not likely to continue increasing over 30 days, and even if it did, additional NPSH margin would be available later in the event as the pool temperature drops	The strainer head loss is over-predicted early in the event

Defense-in-Depth and Safety Margin

Topic	Conservatism Credited as Safety Margin	Realistic Conditions	Impact on Evaluation
Strainer/Pump Failures	Strainer failure is assumed in all cases where the head loss meets or exceeds the structural margin of the strainer	It is likely that the strainer could withstand higher head losses than predicted, and even if a structural failure occurs, it may not result in a complete loss of functionality	Strainer structural failures are over-predicted
Strainer/Pump Failures	All gas voids formed by degasification were assumed to transport to the pumps	Due to the relatively low Froude number, gas voids are likely to accumulate in the strainer and vent back to the pool when the buoyancy of the accumulated air exceeds the strainer head loss	Gas void fractions at the pumps are over-predicted
Strainer/Pump Failures	Pump NPSH required was adjusted for gas voids based on very conservative guidance (Reference 13)	Small gas void fractions would likely have a much smaller effect on NPSH required	Pump NPSH required is over-predicted and Pump NPSH margin is under-predicted when gas voids are present
Core Failures	The fiber penetration correlation ignores effects of fiber and particulate interactions and accumulation of pieces of fiberglass	The penetration of fiberglass fines would be reduced by the accumulation of particulate and fiberglass pieces on the strainer	Fiber penetration (and subsequent accumulation on the core) is over-predicted
Core Failures	The WCAP-16793-NP methodology for evaluating peak cladding temperature and total deposition thickness due to accumulation of debris on the fuel rods is conservative (Reference 14)	The peak cladding temperature and total deposition thickness would be lower	The peak cladding temperature and total deposition thickness are over-predicted

Defense-in-Depth and Safety Margin

Topic	Conservatism Credited as Safety Margin	Realistic Conditions	Impact on Evaluation
Core Failures	Maximum boil-off flow rate with additional 20% margin used to calculate debris accumulation on core inlet for cold leg breaks	Debris transport to core inlet would be proportional to boil-off flow rate and actual boil-off flow rate is likely to be lower	Debris accumulation on core inlet is over-predicted for cold leg breaks
Core Failures	Fiber limits associated with core blockage and boron precipitation are based on bounding tests and analyses	It is likely that significantly larger quantities of debris could accumulate in the core without resulting in core damage	Core failures due to the accumulation of fiber debris are over-predicted.

Defense-in-Depth and Safety Margin

For information, Table 4-2 shows the operating margin included in the analysis for various types of debris and exposed aluminum surface areas.

Table 4-2 – Description of Operating Margin

Item	Actual Value	Value Used	Operating Margin
Epoxy Unqualified Coatings	30.131 ft^3	30.382 ft^3	0.251 ft^3
Alkyd Unqualified Coatings	0.265 ft^3	0.516 ft^3	0.251 ft^3
IOZ Unqualified Coatings	0.131 ft^3	0.382 ft^3	0.251 ft^3
Latent Debris	60 lb_m	200 lb_m	140 lb_m
Miscellaneous Debris	4 ft^2	50 ft^2	46 ft^2
Unsubmerged Aluminum Metal	741.3 ft^2	926.6 ft^2	185.3 ft^2
Submerged Aluminum Metal	278.7 ft^2	348.4 ft^2	69.7 ft^2

4.0 References

1. Regulatory Guide 1.174, Revision 2, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” May 2011
2. Regulatory Guide 1.54, Revision 2, “Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants,” October 2010
3. ANSI N101.4-1972, “Quality Assurance for Protective Coatings Applied to Nuclear Facilities,” American National Standards Institute, Washington, DC
4. Regulatory Guide 1.97, Revision 2, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions during and Following an Accident,” December 1980
5. EA-12-049, “Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events,” March 12, 2012
6. ASME Boiler and Pressure Vessel Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” 2010 Edition, July 1, 2010
7. ASME, Boiler and Pressure Vessel Code, Section III, “Rules for Construction of Nuclear Power Plant Components,” 2010 Edition, July 1, 2010
8. Regulatory Guide 1.45, Revision 1, “Guidance on Monitoring and Responding to Reactor Coolant System Leakage,” May 2008
9. Not used
10. NUREG/CR-7172, “Knowledge Base Report on Emergency Core Cooling Sump Performance in Operating Light Water Reactors,” January 2014
11. NUREG/CR-6772, “GSI-191: Separate-Effects Characterization of Debris Transport in Water,” August 2002
12. Not Used
13. Regulatory Guide 1.82, Revision 4, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident,” March 2012
14. WCAP-16793-NP-A, “Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid”, Revision 2, July 2013.

Vogtle Electric Generating Plant – Units 1 & 2
Supplemental Response to NRC Generic Letter 2004-02

Enclosure 5

Supplemental Response to NRC Generic Letter 2004-02 (Non-Proprietary)

Vogtle Electric Generating Plant
Supplemental Response to NRC Generic Letter 2004-02

Enclosure 5

Supplemental Response to NRC Generic Letter 2004-02 (Non-Proprietary)

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1.0 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 1.0:

This submittal proposes a risk-informed methodology for determining the design requirements 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. The risk-informed analysis covers a full spectrum of postulated LOCAs, including double-ended guillotine breaks (DEGBs), for all pipe sizes up to and including the design-basis accident (DBA) LOCA, to provide assurance that the most severe postulated LOCAs are evaluated. Vogtle Electric Generating Plant (VEGP) conservatively relegates to failure the individual breaks that can generate and transport debris that are not bounded by VEGP analyzed limits. The results of the evaluation in Enclosure 3 show that the risk from the failures related to LOCA-generated debris is “very small” as the risk falls in Region III of RG 1.174. The methodology includes conservatisms in the plant-specific testing and in the assumption that all breaks that exceed the tested debris quantities are relegated to failure. Conservatisms in the VEGP approach are discussed in Enclosure 4, Defense-in-Depth and Safety Margin.

Southern Nuclear Operating Company (SNC) is utilizing a risk-informed approach to the effects of LOCA debris for VEGP. The risk-informed approach replaces the existing deterministic approach described in the VEGP licensing basis and consequently requires an amendment to the VEGP Units 1 and 2 operating licenses to incorporate the revised methodology 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 will be described in the future license amendment request (LAR). Exemptions to the overall requirements associated with 10 CFR 50.46(a)(1), GDC 35, GDC 38, and GDC 41 are required due to the change in methodology. The requests for exemption will be provided in the future LAR.

In addition, SNC proposes to amend the VEGP Unit 1 and Unit 2 operating licenses to revise the Technical Specifications (TSs) for the ECCS and CSS. The proposed TS changes detailed in the future LAR will align the TSs with the risk-informed

methodology change. The licensing discussion is continued in the Response to 3.p of this enclosure.

In the resolution of Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized Water Reactor (PWR) Sump Performance," VEGP implemented (or will implement) the following changes:

- The refueling water storage tank (RWST) High level was increased and the Low-Low level (initiation of semi-automatic switchover to recirculation) decreased to provide increased submergence of the sump strainers while maintaining adequate net positive suction head (NPSH) for the ECCS and containment spray (CS) pumps; allowing sufficient time for completion of operator actions for switchover to recirculation.

- To improve existing margins until all corrective actions can be implemented, VEGP installed larger sump strainers that increased the available screen area for each of the residual heat removal (RHR) strainers and the CS strainers. The hole diameters of the strainer perforated plates were reduced to lessen the potential for debris passing through the strainer and causing plugging and/or wear of the downstream ECCS and CS piping and equipment, and reactor vessel. In addition, Min-K insulation was removed from the containment bioshield area.

- Orifices were installed in the intermediate- and high-head ECCS lines, and the associated throttle valves were adjusted to operate at a minimum internal clearance greater than the size of debris that could pass through the strainers. The opening size was increased to ensure that adequate clearance in the valves will prevent debris from causing excessive wear or plugging.

- Procedural and program controls are in place to ensure materials used in the containments will not result in an increase of the debris loading beyond the analyzed values. This includes controls for containment coatings, labels, and insulation.

- Extensive analysis has been performed in accordance with Nuclear Energy Institute (NEI) 04-07 guidance (Reference 2), the associated United States Nuclear Regulatory Commission (NRC) safety evaluation (SE) (Reference 3), and other industry documents reviewed by the NRC. With few exceptions, VEGP has followed this guidance. Technical justification is available and provided for the few cases where other approaches were utilized.

- The emergency operating procedures are being revised to delay operator action to isolate the RHR pumps from the RWST. This ensures that water level in the RWST is drawn down to the Empty alarm level for all scenarios (it previously only reached the Empty alarm level for scenarios that actuated CS) and prevents most scenarios from resulting in partially submerged strainers.

- To ensure full submergence for an increased number of postulated break scenarios, the Unit 1 and Unit 2 RHR strainers require a reduction in height. The Unit 1 and Unit 2 design packages have been prepared.

Correspondence Background

The following discussion contains correspondences issued by or submitted to the NRC prior to December 31, 2007, on the subject of GSI-191. The title of each letter is provided in the reference section of this enclosure.

The NRC issued Bulletin 2003-01 on June 9, 2003 (Reference 5), asking for a 60-day response providing a description of any interim compensatory measures that have been implemented, or that will be implemented, to reduce the risk which may be associated with potentially degraded or nonconforming ECCS and CSS recirculation functions until an evaluation to determine compliance is complete. SNC provided the 60-day response in a letter dated August 7, 2003 (Reference 6). Supplemental letters dated October 29, 2004 (Reference 7), and July 22, 2005 (Reference 8), were provided by SNC in response to requests for additional information.

The NRC issued Generic Letter (GL) 2004-02 on September 13, 2004 (Reference 1), requesting an initial 90-day response, a 12-month response, and for the guidance of the GL to be met by December 31, 2007. In December 2004, NEI issued NEI 04-07 (Reference 2) providing an evaluation methodology for the industry. The NRC provided the associated SE (Reference 3) on December 6, 2004. The NRC had already issued RG 1.82 Revision 3 (Reference 25) in November 2003.

SNC provided the initial response for VEGP in a letter dated February 25, 2005 (Reference 10). SNC provided a follow-up response on August 31, 2005 (Reference 11), providing more details on how SNC would meet the GL 2004-02 requirements.

The NRC issued a request for additional information on February 9, 2006 (Reference 12), with a 60-day response time. NEI worked with the NRC and recognized that much of the information needed to address the RAIs would not be available until ongoing testing activities were completed. The NRC-issued letter dated March 28, 2006 (Reference 13), identified that the Request for Additional Information (RAI) answers could be provided as part of the supplemental response by the end of December 2007. An NRC letter dated January 4, 2007 (Reference 18), provided clarification that even if a licensee had an extension for modifications past 2007, the supplemental response was still due by December 31, 2007.

SNC submitted an extension request in a letter dated June 22, 2006 (Reference 14), for modification/installation of the Unit 1 ECCS flow orifices and for chemical effects testing. In a teleconference on June 30, 2006, with the NRC staff reviewer of the

June 22, 2006, extension request, SNC was asked to provide an update of on-going activities and a clarification as to what activities are driving the extension request. SNC provided the requested information in a response dated July 28, 2006 (Reference 15), which also requested an extension from December 31, 2007, to the spring 2008 outage. This extension request was approved in an NRC letter dated September 7, 2006 (Reference 16).

The NRC issued a letter dated August 15, 2007, containing the content guide for the GL 2004-02 supplemental response due in December 2007. Additional information was provided by the NRC in a letter dated September 27, 2007, for chemical effects, protective coatings, and head loss testing. A revision to the content guide was issued by the NRC in a letter dated November 21, 2007 (Reference 108). The due date for the supplemental response was extended by an NRC letter dated November 30, 2007 to allow the supplemental response to be submitted by February 29, 2008. An NRC letter dated November 8, 2007, provided guidance for requesting plant-specific extensions. Additional information was also provided in an NRC letter dated November 13, 2007, on how GSI-191 would be closed and how the closure would be documented for each site.

SNC submitted a letter dated December 7, 2007 (Reference 19), requesting an extension for submittal of chemical effects testing results, downstream effects – components and systems, and downstream effects – fuel and vessel until June 30, 2008. This request was approved in an NRC letter dated December 19, 2007 (Reference 20).

2.0 General Description of and Schedule for Correction 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 generic letter. 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 2.0:

SNC has performed analysis to determine the susceptibility of the ECCS and CSS recirculation functions for VEGP to the adverse effects of post-accident debris blockage and operation with debris-laden fluids. These analyses conform, to the greatest extent practicable, to the NEI 04-07 methodology (Reference 2) as approved by the NRC SE dated December 6, 2004 (Reference 3). As of April 24, 2017, SNC has completed the following GL 2004-02 (Reference 1) actions, analyses, and modifications:

- Replaced Unit 1 and Unit 2 containment emergency sump screens during refueling outage 1R13 (Fall 2006), and refueling outage 2R12 (Spring 2007), respectively
- Installed ECCS flow orifices in the intermediate and high-head ECCS lines that allow the ECCS throttle valves to be opened greater than the maximum expected strainer bypass debris size while maintaining the capability to ensure ECCS flow balance, mitigating downstream effects (2008)
- Completed inspection of containment per NEI 02-01 (Reference 41), “Condition Assessment Guidelines: Debris Sources Inside PWR Containment”
- Performed latent debris sampling and characterization
- Participated in the PWR Owners Group (PWROG) program to evaluate downstream effects related to in-vessel long-term cooling with results documented in WCAP-16793-NP-A (Reference 22)
- Removed Min-K insulation in the original zone of influence (ZOI) analyzed for GL 2004-02 from VEGP’s containments based on preliminary head loss testing (without chemical effects) which determined that the removal of Min-K insulation resulted in a significant reduction in head loss across a debris-laden strainer

- Implemented programmatic and procedural changes to maintain acceptable configuration and to protect the newly established design and licensing basis
- Developed containment 3D CAD model of VEGP Unit 1 containment to include pipe welds, for both VEGP containments because Units 1 and 2 are virtually identical (CAD model of VEGP Unit 1 containment was used to determine a correlation between the containment pool volume and containment pool level)
- Completed detailed laser scans of the VEGP containments, which provide measurements for contingency insulation replacement for Units 1 and 2 (laser scans of both units were completed before January 1, 2013)
- Developed detailed debris generation and debris transport analyses and a computational fluid dynamics (CFD) model
- Developed a hydraulic model of the ECCS
- Performed detailed CS and RHR NPSH analysis
- Performed water level analysis
- Modified probabilistic risk assessment (PRA) to include strainer and core blockage events
- Quantified chemical precipitants using WCAP-16530 (with refinements)
- Performed chemical effects testing
- Completed RELAP5-3D/MELCOR modeling similar to South Texas Project (STP) model (however, the results are not used as input for the base case analysis)
- Performed strainer head loss and fiber debris penetration testing
- Participated in the PWROG Comprehensive Analysis and Test Program for GSI-191 Closure
- Performed downstream wear and blockage analysis to WCAP-16406-P-A, Revision 1 (Reference 21)
- Performed detailed structural analysis of strainers
- Assembled base case final inputs for quantifying the conditional failure probabilities related to GSI-191 using the software package Nuclear Accident Risk-Weighted Analysis (NARWHAL) (see Enclosure 3, Section 13.1 for general description of the software).
- Completed NARWHAL sensitivity analyses
- Integrated NARWHAL results into VEGP PRA model to determine changes in core damage frequency (ΔCDF) and changes in large early release frequency (ΔLERF)
- Revised operating procedures to ensure that the RHR strainers are completely submerged for an increased number of postulated LOCA scenarios (operator action to isolate the RWST from the RHR pumps was delayed until the Empty level is reached to ensure sufficient injection of RWST water for breaks that activate CS)

The following risk-informed resolution path activities are planned by SNC to address GL 2004-02 and support closure of GSI-191 for VEGP.

- SNC is planning to modify the VEGP Unit 1 and Unit 2 RHR sump strainers to reduce the overall height by removing the top two strainer disks per stack from each of the RHR strainer assemblies.

- SNC will submit an LAR for a risk-informed resolution to GL 2004-02 for VEGP within six months after receipt of the SE for WCAP-17788-P.
- SNC will submit any necessary revisions to the supplemental response to support closure of GL 2004-02 for VEGP Units 1 and 2 within six months after receipt of the SE for WCAP-17788-P.

Correspondence Background

The following discussion contains correspondences issued by or submitted to the NRC beyond December 31, 2007, on the subject of GSI-191. The correspondences document VEGP's compliance with regulatory requirements per Requested Information Item 2(b) and include reference to approved extension requests. The title of each letter is provided in the reference section of this enclosure.

SNC letter dated February 28, 2008, NL-07-1777 (Reference 95), provided SNC's supplemental response to GL 2004-02 for VEGP.

SNC letter dated May 21, 2008, NL-08-0670 (Reference 96), provided a revised transmittal of SNC's supplemental response to GL 2004-02 for VEGP based on NRC's questions regarding the proprietary nature of information provided in SNC letter dated February 28, 2008 (Reference 95).

SNC letter dated May 22, 2008, NL-08-0818 (Reference 97), requested an extension for the final response to GL 2004-02 for the completion of WCAP-16406-P-A and WCAP-16793-NP-A evaluations, and chemical effects testing and evaluation of test results. An extension was granted to SNC by the NRC to July 31, 2008, in a letter dated May 29, 2008, as stated in NL-08-1155 (Reference 98).

SNC letters dated July 31, 2008, NL-08-1155 (Reference 98) and NL-08-1195 (Reference 99), provided the downstream effects results for components and in-vessel analyses and requested an extension for the GL 2004-02 supplemental response for chemical effects, respectively.

SNC letter dated August 22, 2008, NL-08-1228 (Reference 100), provided the GL 2004-02 response for chemical effects. The letter also contained a revised answer to question 3.g.15 originally submitted in SNC letter dated May 21, 2008 (Reference 96).

NRC letter dated September 17, 2008, NL-08-1497 (Reference 101), provided RAIs from a partial review of prior SNC responses to GL 2004-02 pertaining to the reliance on results from testing at the VUEZ facility by Alion Science and Technology. As discussed with Mr. Jared S. Wermiel, Deputy Director of the Division for Engineering and Safety Systems, in a telephone call with SNC on September 11, 2008, the NRC identified several critical issues with the test protocol used in the testing at VUEZ. The NRC staff has stated that based on their review of information provided by Alion on the VUEZ testing, it is highly unlikely that SNC's

reliance on the VUEZ testing, performed to date to demonstrate strainer adequacy, will provide an adequate technical basis to resolve GL 2004-02.

SNC letter dated November 7, 2008, NL-08-1583 (Reference 102), requested an extension, in accordance with SECY-06-0078, for completion of chemical effects testing and closeout activities for GL 2004-02 as a result of NRC's concern that SNC's reliance on the VUEZ performed testing would not provide an adequate technical basis to resolve GL 2004-02 for VEGP. NL-08-1583 also noted that the RAIs issued by the staff on September 17, 2009 were from a partial review of SNC's responses to GL 2004-02 and did not represent a comprehensive set of RAIs. In addition, SNC submitted milestone dates supporting a closeout of GSI-191 and a final response to the staff by November 20, 2009, predicated on a reasonable submittal of SNC test-for-success protocol with NRC review and comment cycle, and resolution of any issues associated with pending revision of WCAP-16793-NP.

NRC letter dated December 2, 2008, NL-08-1829 (Reference 87), provided RAIs for prior SNC supplemental responses, letters dated February 28, 2008; May 21, 2008; July 31, 2008; and August 22, 2008 (References 95, 96, 98, and 100, respectively). The NRC requested RAI responses within 90 days.

SNC letter dated February 10, 2009, NL-09-0159 (Reference 103), notified the NRC that a single response to the RAIs issued by the NRC letter dated December 2, 2008, would be submitted once the new testing and analysis discussed in the November 7, 2008, letter was completed. SNC also notified the NRC that the planned completion date for the GL 2004-02 response was November 20, 2009, excluding issues related to WCAP-16793-NP.

SNC letter dated November 19, 2009, NL-09-1839 (Reference 104), stated that the schedule for completion of GSI-191 activities for VEGP is contingent upon resolution of generic issues and their impact to the remaining 26 RAIs for VEGP. Of the 29 RAIs issued December 2, 2008, SNC discussed 26 satisfactorily with the staff during public telecoms between SNC and the staff on August 13, 2009, and October 13, 2009. The three remaining RAIs concerned the following generic issues: Nukon ZOI, fibrous insulation erosion, and in-vessel downstream effects. During the telecom between SNC and the staff on October 13, 2009, the above generic issues were discussed as to how the outcome of said issues would impact the 26 RAIs for VEGP. It was agreed to by the staff in this telecom that a written response to the resolved RAIs was not required by November 20, 2009.

NEI letter to NRC dated May 4, 2012 (Reference 91), highlighted the current industry status and recommended actions for closure of GSI-191 based on licensees providing a docketed submittal to the NRC by December 31, 2012, outlining a GSI-191 resolution path and schedule pursuant to the Commission direction in Staff Requirements Memorandum (SRM) SECY-10-0113 (Reference 90).

An NEI letter to the NRC dated November 15, 2012 (Reference 92), and a subsequent NRC review of the schedule (Reference 93) recommended that licensees delay submittal of GSI-191 resolution path and schedule until January 31, 2013. The letter also recommended an alternative option to submit 30 days following placement of both the Commission's response to SECY-12-0093 (Reference 70) and the NRC SE on WCAP-16793-NP (Reference 94) into the public record.

The Commission approved the staff's recommendation in SRM-SECY-12-0093 (Reference 70) dated December 14, 2012, to allow licensees the flexibility to choose any of the three options discussed in the paper to resolve GSI-191. Further, the Commission encouraged the staff to remain open to staggering licensee submittals and the associated NRC reviews to accommodate the availability of staff and licensee resources.

The SE for WCAP-16793-NP (Reference 94) was made publicly available by the NRC on April 16, 2013.

SNC Letter dated May 16, 2013, NL-13-0953 (Reference 105), transmitted the VEGP Units 1 and 2 resolution path forward and schedule for resolution, summary of actions completed for GL 2004-02, and defense-in-depth and mitigation measures, using the industry template developed by NEI. VEGP provided a basis for continued operation in the interim while the industry and NRC collaborated on how to proceed towards resolution of the issue. VEGP is following the "STP Piloted Risk-Informed Approach for GSI-191," as submitted by the following South Texas Project Nuclear Operating Company (STPNOC) letters to the NRC (Reference 105).

STPNOC letter to the NRC dated November 13, 2013, NOC-AE-13003043 (Reference 44), submitted Supplement 1 to the STP pilot risk-informed approach for GSI-191.

STPNOC letter to the NRC dated August 20, 2015, NOC-AE-15003241 (Reference 45), submitted Supplement 2 to the STP pilot submittal.

STPNOC letter to the NRC dated October 20, 2016, NOC-AE-16003401 (Reference 109), submitted Supplement 3 to the Revised STP pilot submittal.

SNC Letter to the NRC dated April 21, 2017, NL-16-2002 (Reference 112), transmitted the VEGP Units 1 and 2 GL 2004-02 supplemental response, which superseded all previous responses.

SNC Letter to the NRC dated July 11, 2017, NL-17-1201 (Reference 113), responded to an NRC request for information regarding the seismic probabilistic risk assessment peer review. The letter also clarified the requested scope of the NRC's review for the April 21, 2017 submittal.

The SE for the STP risk-informed GSI-191 submittal (Reference 118) was made publicly available by the NRC on July 11, 2017.

SNC Letters NL-17-1848 (Reference 114, dated November 9, 2017), NL-17-2044 (Reference 115, dated January 2, 2018), NL-17-2045 (Reference 116, dated January 9, 2018), NL-18-0086 (Reference 117, dated February 12, 2018), and NL-18-0611 (Reference 119, dated May 23, 2018) responded to NRC's requests for additional information after the NRC's review of the April 21, 2017 submittal.

3.0 Specific Information Regarding Methodology for Demonstrating Compliance:

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.

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

Response to 3.a.1:

The VEGP debris generation calculation followed the methodology of NEI 04-07 and associated NRC SE (References 2 and 3, respectively), with the exception that it analyzed a full range of breaks, not just the worst-case breaks as suggested by NEI 04-07. The purpose of the calculation is to obtain debris quantities for the range of possible break scenarios. The calculation evaluated debris generation quantities for breaks on every inservice inspection (ISI) weld within the Class 1 pressure boundary. Both DEGBs and partial breaks were considered. All break sizes analyzed are assumed to fall into one of three high-level categories: small-break LOCA (SBLOCA) – a break smaller than 2 inches, medium-break LOCA (MBLOCA) – a break greater than or equal to 2 inches and less than 6 inches, or large-break LOCA (LBLOCA) – a break greater than or equal to 6 inches with the largest break being a DEGB of the 31-inch crossover leg.

In the debris generation calculation, a three-dimensional CAD model of VEGP Unit 1 containment building was updated to work with ENERCON's BADGER software. Note that the Unit 1 containment was used to represent both containments because the VEGP units are virtually identical. BADGER was used to place ZOIs representing possible breaks on every ISI weld identified in containment. See Figure 3.a.1-1 for weld locations and Table 3-8 in Enclosure 3 for a complete list of welds inside the first isolation valve. In Figure 3.a.1-1, welds labeled as "In" are on the RCS side of the first isolation valve and welds labeled as "Out" are downstream of the first isolation valve. As discussed in Enclosure 1 Section 3.2, non-pipe LOCAs were not explicitly analyzed.

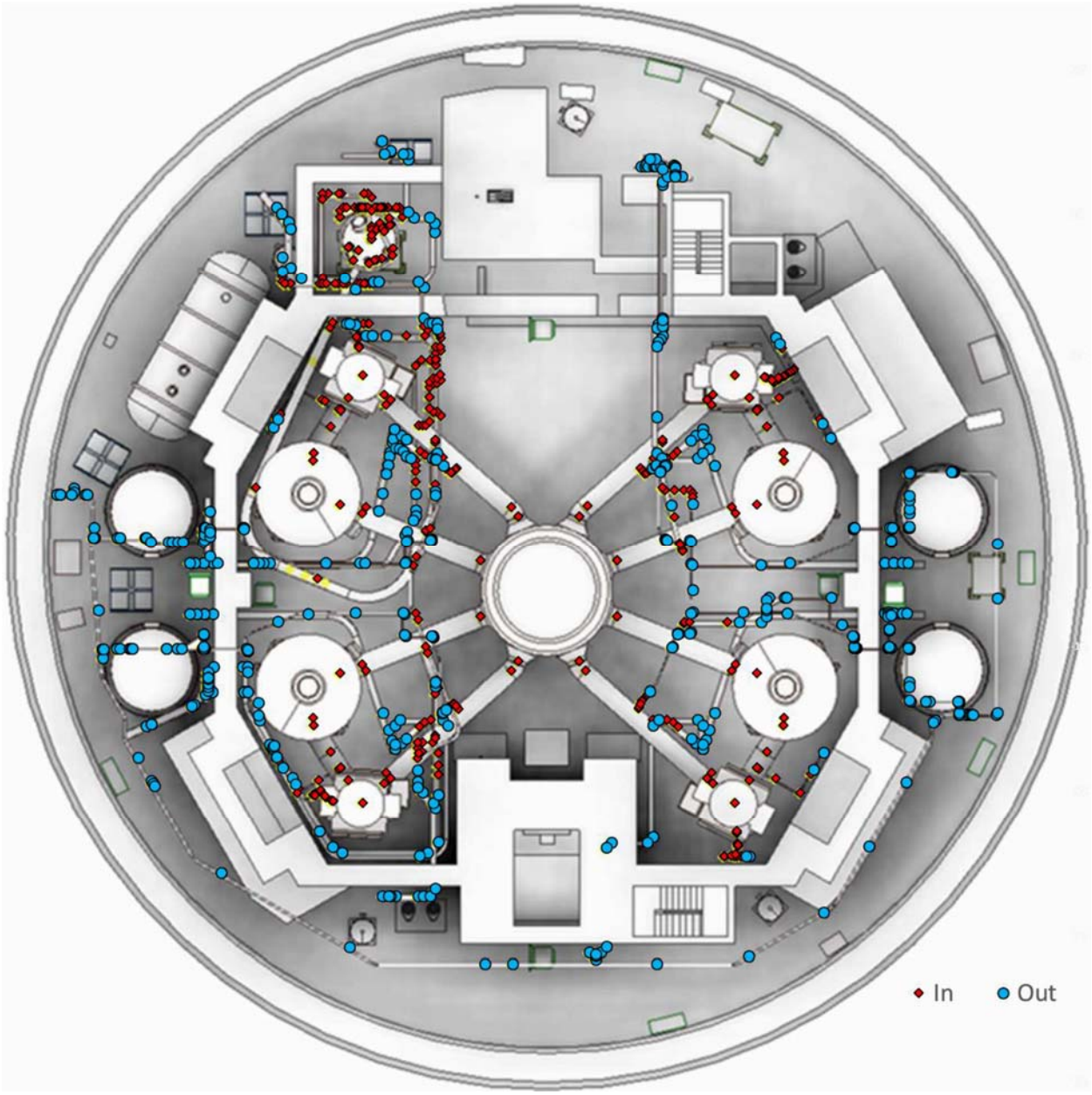

Figure 3.a.1-1 – Weld Locations where Postulated LOCAs Occur

It should be noted that, while DEGBs on main loop piping are typically bounding with regard to the volume of debris generated, small partial breaks are much more likely to occur. A partial break is any break smaller than a DEGB (i.e., sidewall break). Partial breaks are described by equivalent break size and are represented by a hemispherical ZOI with a radius proportional to the equivalent break size (see Figure 3.a.1-2).

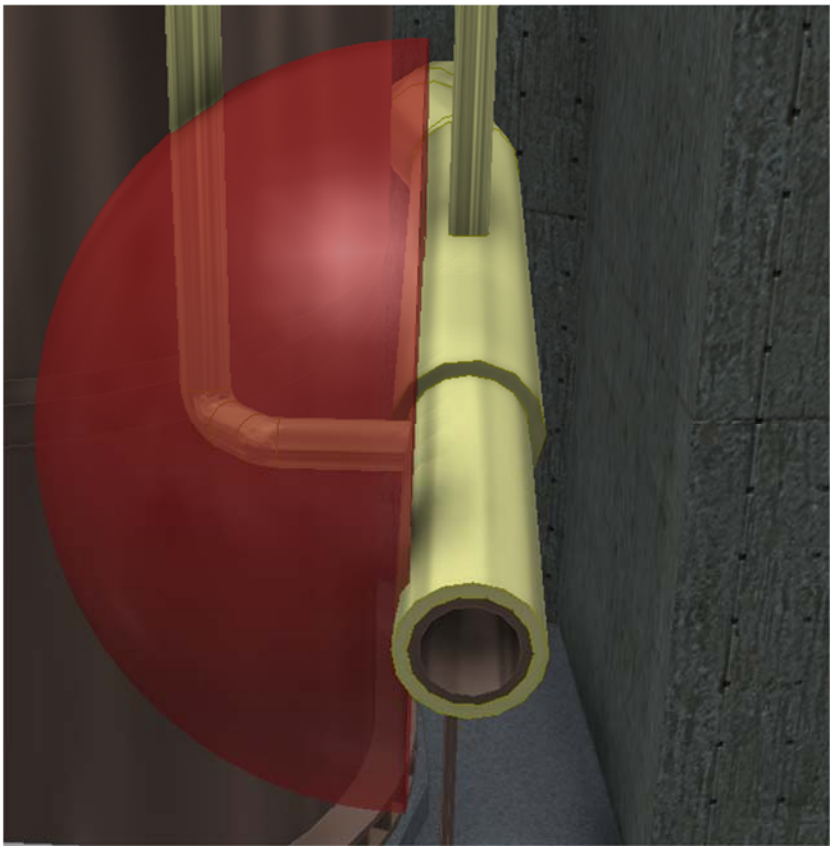

Figure 3.a.1-2 – Single Partial Break Zone of Influence

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 the probability is low, a secondary side break inside containment (SSBI) could require ECCS recirculation in a feed and bleed scenario. Therefore, secondary side breaks from the steam generator feedwater lines and main steam lines were analyzed.

Because secondary side breaks occur at a lower pressure and temperature than the primary side breaks, the ZOI size corresponding to the insulation destruction pressure would be smaller. The appropriate ZOI sizes were calculated based on the ANSI jet methodology described in Appendix I of NEI 04-07 Volume 2 (Reference 3). The main steam and feedwater pressures, temperatures, and calculated ZOI sizes are presented in Table 3.a.2-1.

Debris loads for the secondary side breaks were evaluated using BADGER and breaks were postulated approximately every 5 ft on each of the main steam and feedwater pipes. All secondary side breaks were assumed to be DEGBs. Only Nukon insulation was considered for the secondary-side breaks because there is no

fire barrier within the vicinity of the main steam and feedwater lines. Additionally, the coatings debris quantities for the secondary side breaks are bounded by the primary side breaks due to the larger pipe size and higher pressure on the primary side. Note that even the largest coatings debris loads of the primary side breaks (Table 3.e.6-15) are bounded by the debris limit (Table 3.f.5-1). Therefore, none of the secondary side breaks would fail due to exceeding the coatings debris limit.

Table 3.a.2-1 – Secondary-Side Line ZOI Summary

Main Steam Lines			**Feedwater Lines**		
P_0 (psia)	**T_0 (°F)**	**Pipe Inner Diameter (inch)**	**P_0 (psia)**	**T_0 (°F)**	**Pipe Inner Diameter (inch)**
985	545	24.0	1150	445	12.8
Mass Flux ($lb_m \cdot ft^{-2} \cdot s^{-1}$)	**C_T**		**Mass Flux ($lb_m \cdot ft^{-2} \cdot s^{-1}$)**	**C_T**	
1,978	1.25		19,238	1.65	
Insulation / Destruction Pressure (psig)	**ZOI Radius (D)**		**Insulation / Destruction Pressure (psig)**	**ZOI Radius (D)**	
Coatings / 40	3.0		Coatings / 40	2.8	
Interam / 10.2	7.9		Interam / 10.2	7.2	
Nukon / 6	10.5		Nukon / 6	11.3	

Note: C_T is the thrust coefficient

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 ISI weld within the Class 1 pressure boundary. Both DEGBs and partial breaks were considered. 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.

The total quantity of each type of debris generated within a particular ZOI is unique for every break scenario. Therefore, the bounding break-specific debris loads contained in the BADGER database were used on a break-specific basis for the analysis. The results of the debris generation calculation are presented below.

When reading the tables in this section it should be noted that the individual quantities for fines, small pieces, large pieces, and intact blankets do not necessarily add up to the total fiber quantity within the ZOI because the minimum, maximum, and average values for each size do not necessarily come from the same break. All average values are based on an equal probability of all breaks and do not consider differences in weld-specific break frequencies or the lower frequencies associated with larger break sizes.

These results reflect the following conservatisms:

- All debris sources within the reactor cavity were assumed to be available for destruction by all breaks within the reactor cavity, despite the likelihood that ZOIs would be restricted by structures and restraints.
- All qualified coatings on steel and concrete were analyzed as having the worst-case coating system for each surface.
- 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.

Debris Generated by DEGBs

Table 3.a.3-1 shows the location of the worst-case break for each debris type. Table 3.a.3-2, Table 3.a.3-3, and Figures 3.a.3-1 through 3.a.3-8 show the minimum, average, and maximum debris quantities by debris type for DEGBs and partial breaks upstream of the first isolation valve at VEGP Units 1 and 2. Note that, although most of the qualified coatings quantities are shown in mass in this section, volumes were used when comparing with the tested coatings debris loads in the NARWHAL analysis.

Table 3.a.3-1 – Location of Maximum Debris by Debris Type

Debris Type	Worst-Case Location	Amount
Nukon (ft^3)	SG Compartment 1/4 Side	2229.2
Interam E-50 Series (lbm)	SG Compartment 1/4 Side	59.8*
Qualified IOZ (lbm)	SG Compartment 2/3 Side	65.3[+]
Qualified Epoxy (lbm)	Reactor Cavity	220.4[^]

*The limiting quantity of Interam E-50 is generated by a partial break. This is possible because partial breaks are analyzed as being centered on the edge of the pipe, whereas DEGBs are centered on the axis of the pipe (see Figure 3.a.1-2). Because of this, partial breaks can extend further than DEGBs up to the outside radius of the pipe.

[+] This qualified IOZ debris load corresponds to a volume of 0.3 ft^3.

[^] This qualified epoxy debris load corresponds to a volume of 2.1 ft^3.

Table 3.a.3-2 – Debris Generated by DEGBs Upstream of First Isolation Valve

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 (ft^3)	Fines (Individual Fibers)	0.0	0.2	0.7	0.0	3.3	12.2	10.9	98.1	289.3
	Small Pieces (< 6" a side)	0.0	0.6	2.3	0.0	10.8	40.8	35.7	331.6	999.5
	Large Pieces (> 6" a side)	0.0	0.4	1.4	0.3	7.1	25.6	22.1	173.8	549.6
	Intact (Covered) Blankets	0.0	0.5	1.5	0.5	7.7	27.7	23.9	187.8	594.0
	All Debris Within ZOI	**0.0**	**1.7**	**5.9**	**1.9**	**28.9**	**104.1**	**92.7**	**791.6**	**2229.2**
Fire Barrier Debris (lbm)	Fiber	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.7	12.1
	Particulate	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.6	28.3
IOZ Qualified Coatings (lbm)		0.0	<0.1	0.1	0.0	<0.1	1.3	0.0	19.4	65.3[+]
Epoxy Qualified Coatings (lbm)		0.0	<0.1	0.3	0.0	0.2	4.9	0.0	86.2	220.4[^]

[+] This qualified IOZ debris load corresponds to a volume of 0.3 ft^3.

[^] This qualified epoxy debris load corresponds to a volume of 2.1 ft^3.

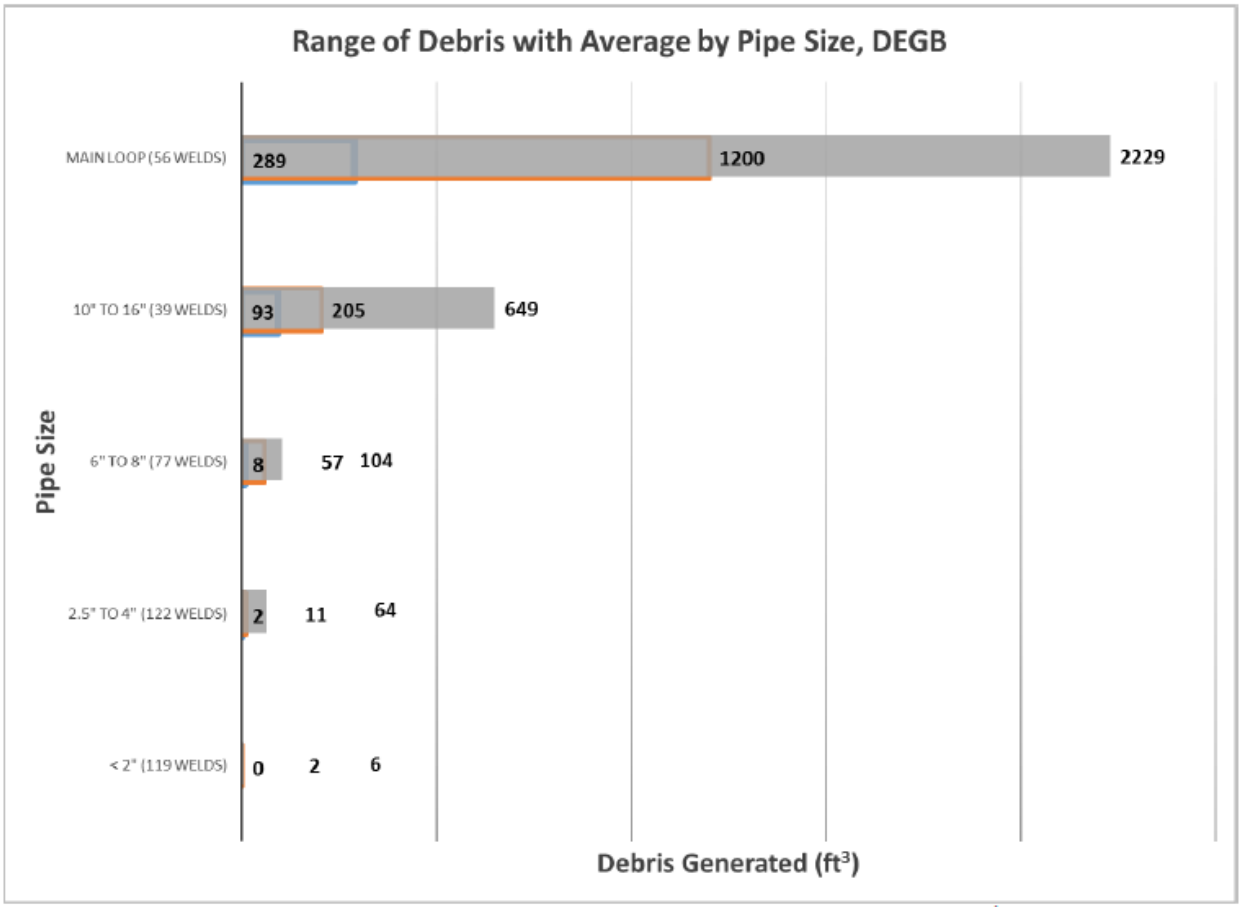

Figure 3.a.3-1 – Range of Nukon Debris Generated by DEGBs Upstream of First Isolation Valve

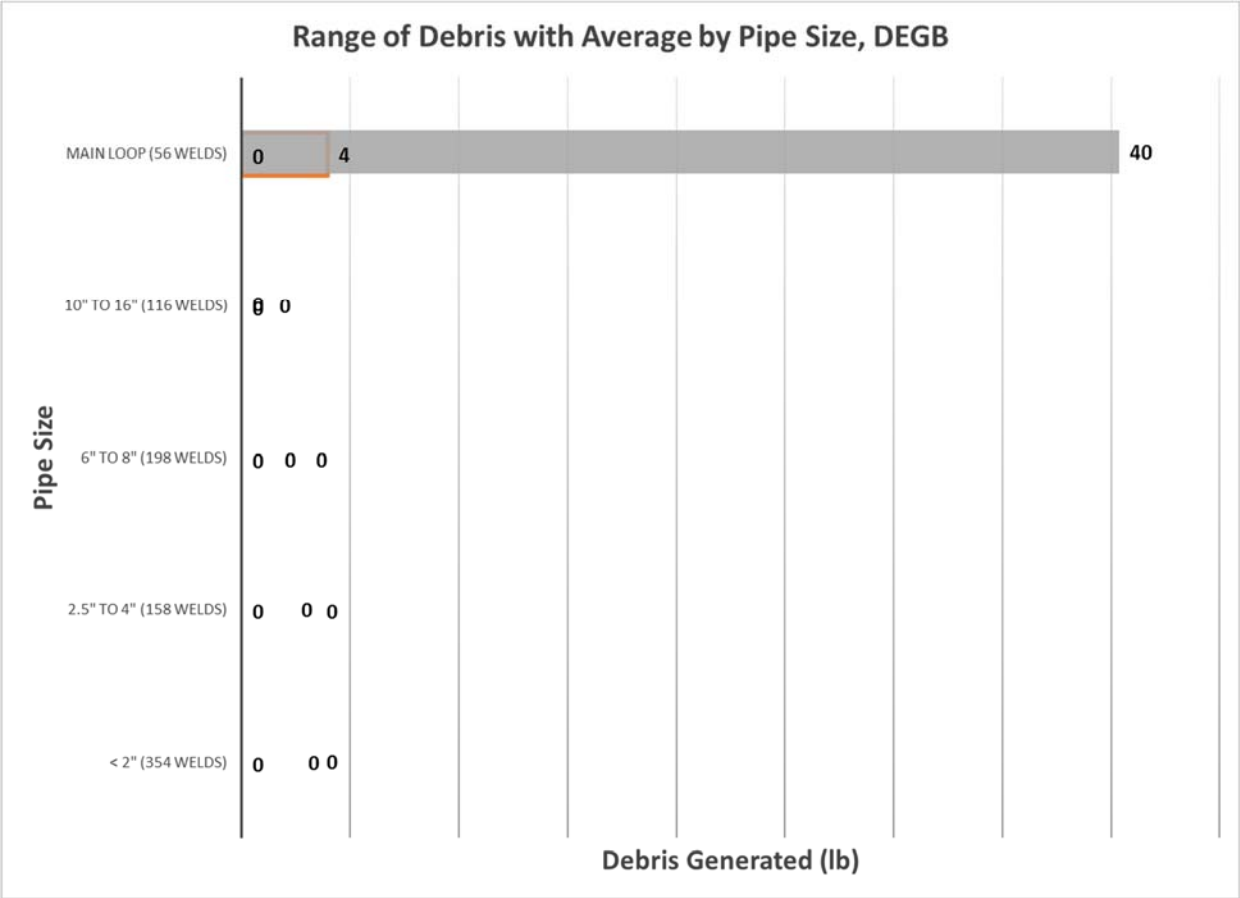

Figure 3.a.3-2 – Range of Fire Barrier Debris Generated by DEGBs Upstream of First Isolation Valve

Note that Figure 3.a.3-2 shows the total amount of fire barrier destroyed (fiber plus particulate).

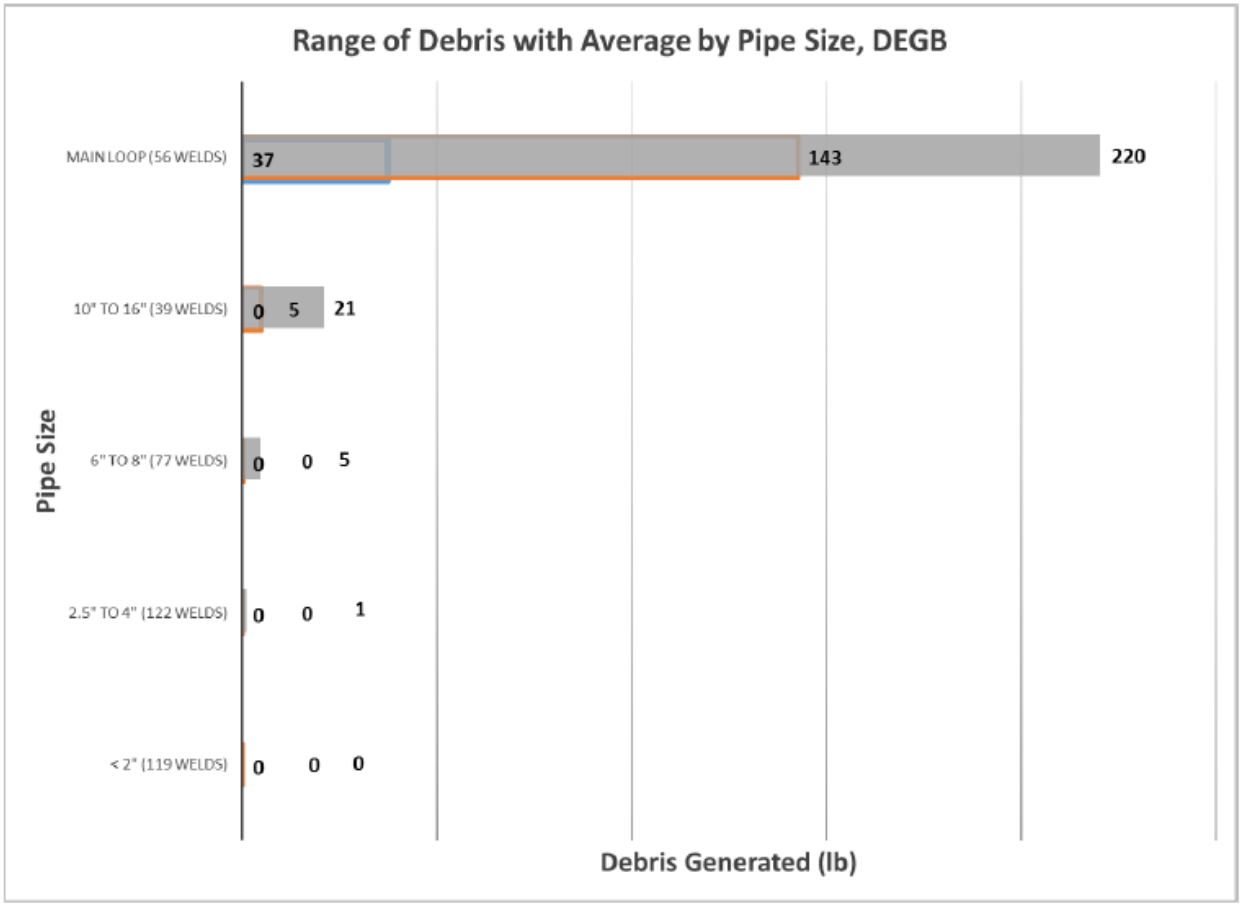

Figure 3.a.3-3 – Range of Epoxy Coatings Debris Generated by DEGBs Upstream of First Isolation Valve

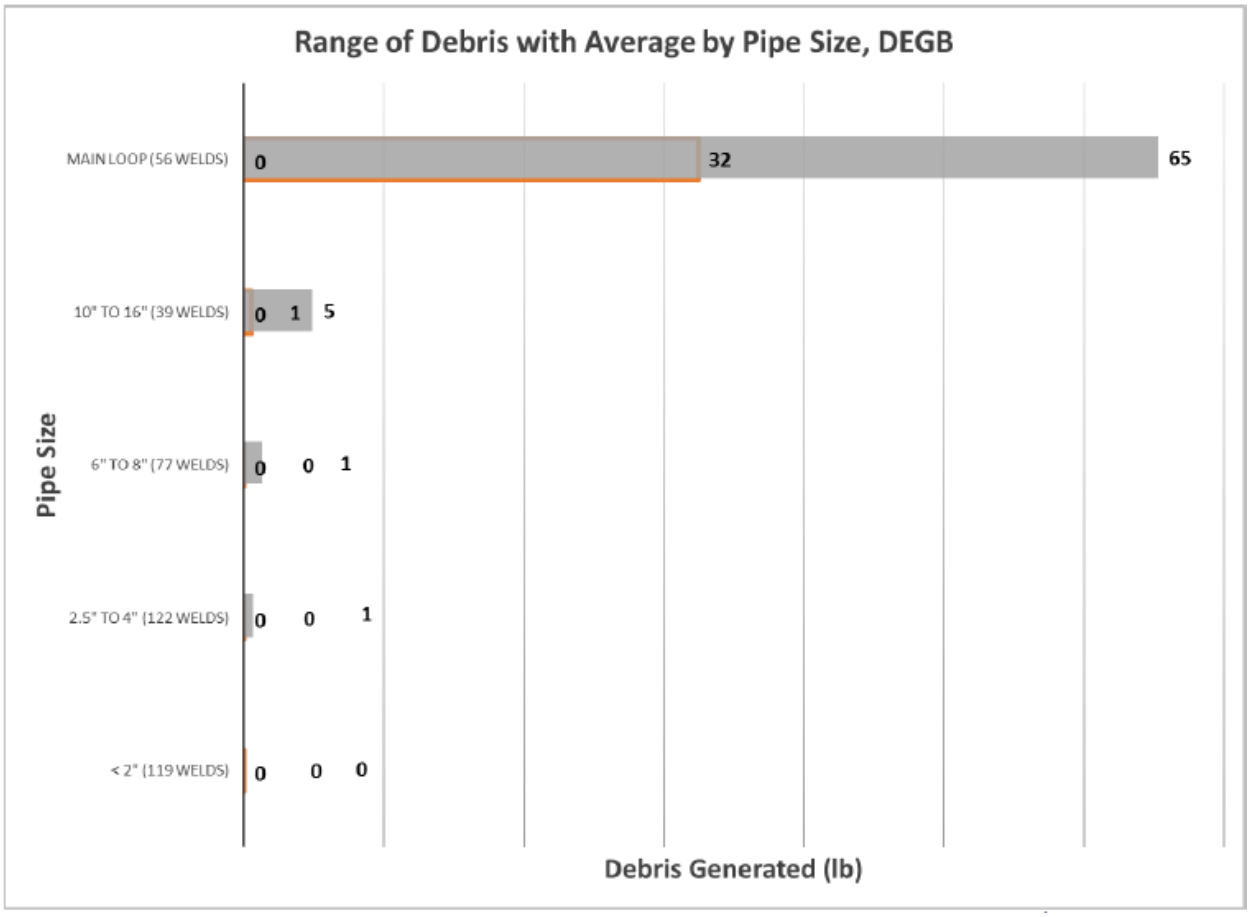

Figure 3.a.3-4 – Range of IOZ Coatings Debris Generated by DEGBs Upstream of First Isolation Valve

Table 3.a.3-3 – Debris Generated by Partial Breaks Upstream of First Isolation Valve

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 (ft^3)	Fines (Individual Fibers)	0.0	<0.1	0.2	0.0	1.1	8.9	0.6	40.6	223.7
	Small Pieces (< 6" a side)	0.0	<0.1	0.8	0.0	3.6	28.5	1.7	136.0	794.1
	Large Pieces (> 6" a side)	0.0	<0.1	0.3	0.0	2.6	22.7	0.0	75.7	438.4
	Intact (Covered) Blankets	0.0	<0.1	0.3	0.0	2.8	24.5	0.0	81.8	473.7
	All Debris Within ZOI	**0.0**	**0.2**	**1.6**	**0.0**	**10.0**	**79.5**	**4.5**	**334.3**	**1783.0**
Interam E-50 Series (lbm)	Fiber	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.2	17.9
	Particulate	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.4	41.9
IOZ Qualified Coatings (lbm)		0.0	<0.1	0.1	0.0	<0.1	2.1	0.0	7.8	40.9+
Epoxy Qualified Coatings (lbm)		0.0	<0.1	0.4	0.0	0.2	3.9	0.0	32.6	149.1^

+ This qualified IOZ debris load corresponds to a volume of 0.2 ft^3.

^ This qualified epoxy debris load corresponds to a volume of 1.5 ft^3.

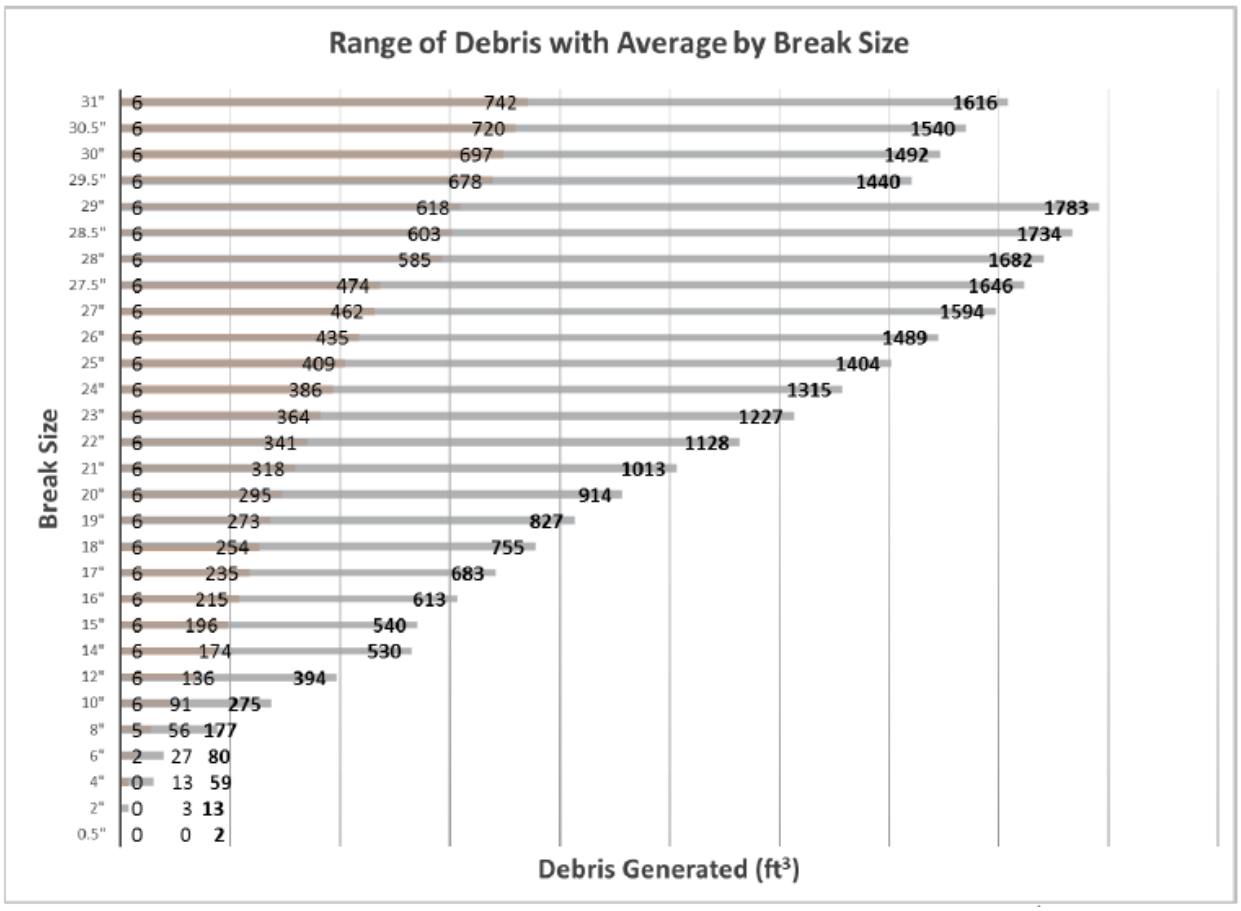

Figure 3.a.3-5 – Range of Nukon Debris Generated by Partial Breaks Upstream of First Isolation Valve

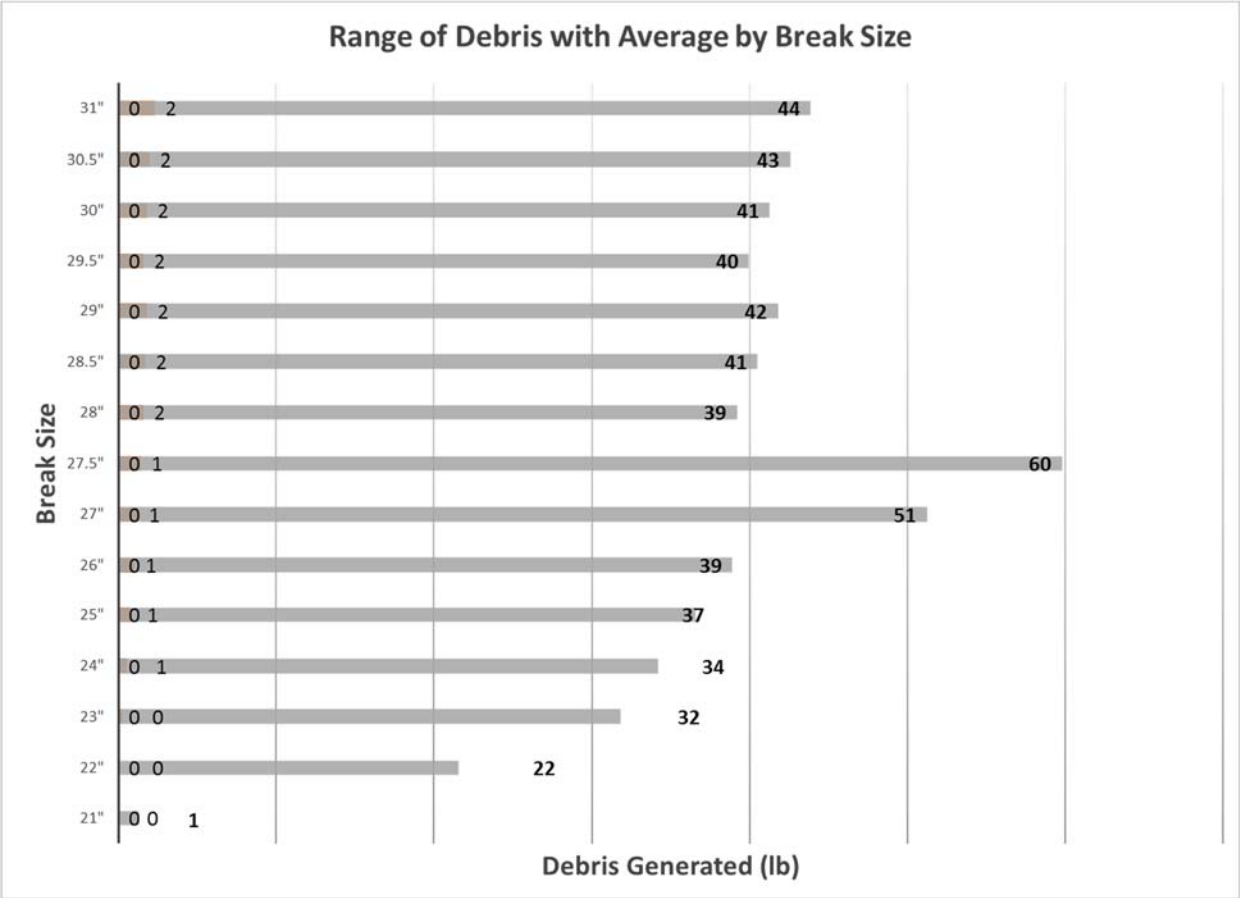

Figure 3.a.3-6 - Range of Fire Barrier Debris Generated by Partial Breaks Upstream of First Isolation Valve

Note that Figure 3.a.3-6 shows the total amount of fire barrier destroyed (fiber plus particulate). Note that no fire barrier material is destroyed by partial breaks smaller than 21 inches.

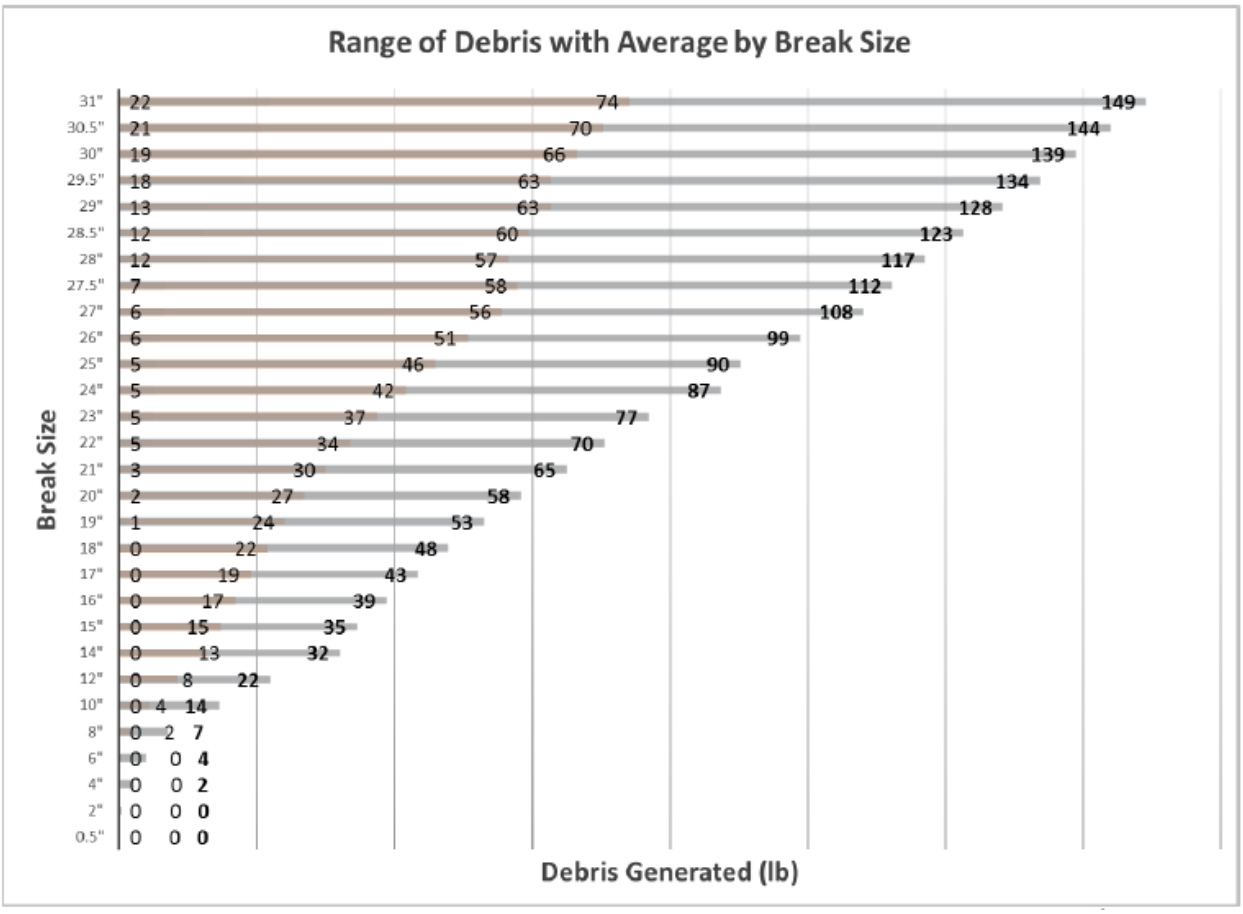

Figure 3.a.3-7 – Range of Epoxy Coatings Debris Generated by Partial Breaks Upstream of First Isolation Valve

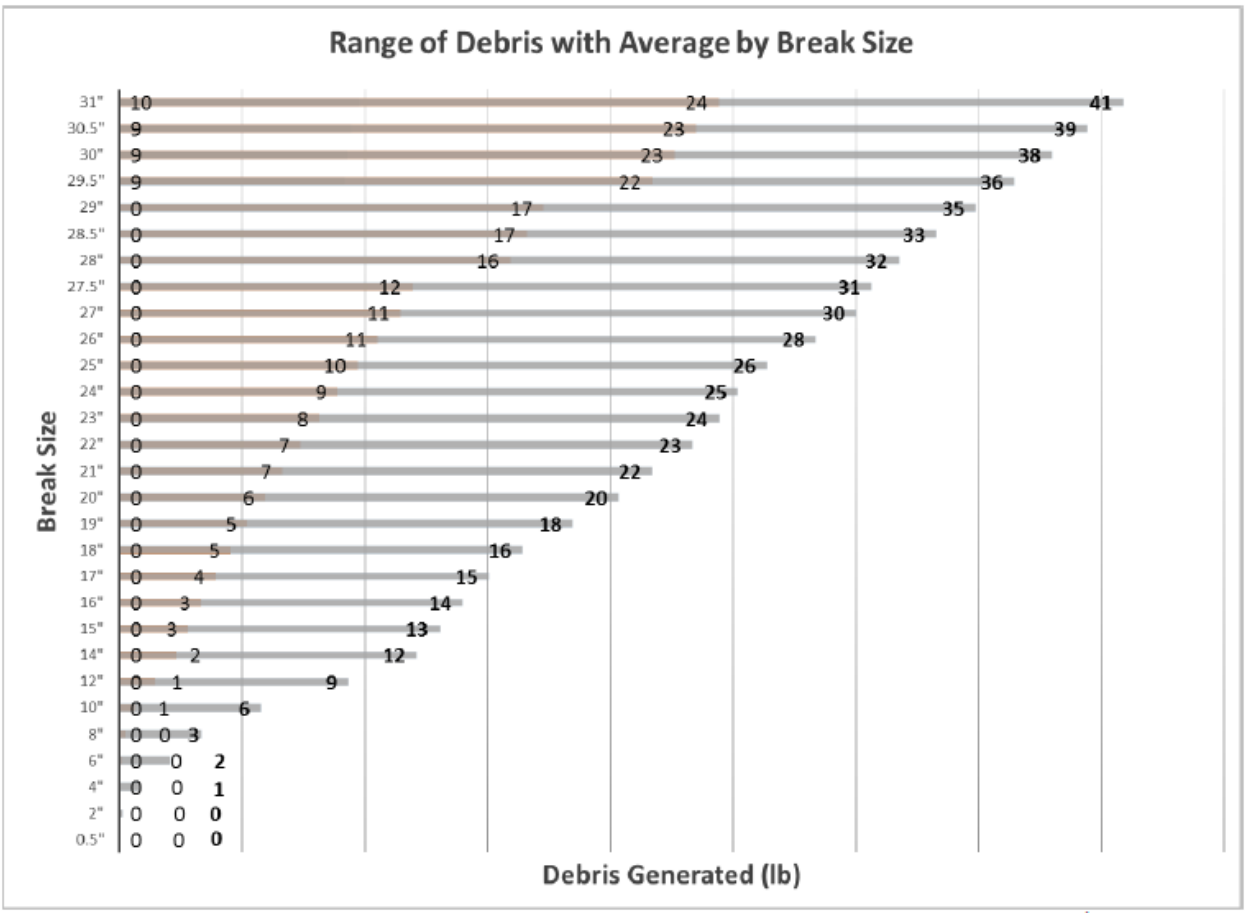

Figure 3.a.3-8 – Range of IOZ Coatings Debris Generated by Partial Breaks Upstream of First Isolation Valve

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; (2) the amount of debris generated by the break jet forces.

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:

   For DEGBs, the ZOI is defined as a spherical volume about 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 PWR reactor containment building, the worst-case pipe break would 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 2) 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), NEI 04-07, Volume 2 (Reference 3) suggests a hemispherical ZOI centered at the edge of the pipe. 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.

Since different insulation types have different destruction pressures, different ZOIs must be determined for each type of insulation. 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 VEGP Units 1 and 2 containment buildings. See Table 3.a.2-1 for ZOI sizes for SSBIs.

Table 3.b.1-1 – Primary Side Break ZOI Radii for VEGP Insulation Types

Insulation Type	Destruction Pressure (psi)	ZOI Radius/Break Diameter (L/D)
Unjacketed Nukon	6	17.0*
Qualified Coatings	Unknown	4.0**
Fire Barrier Material	Unknown	11.7***

* NRC SE for NEI 04-07 (Reference 3)

** "Revised Guidance Regarding Coatings Zone of Influence for Review of Final Licensee Responses to Generic Letter 2004-02" ADAMS # ML100960495

*** The destruction pressure of the Interam E-50 series fire barrier material at VEGP is unknown. However, its ZOI size was assumed to be 11.7D based on comparison to the robustness of Temp-Mat.

In some cases, if the ZOI for a particular material is very large (i.e., it has a low destruction pressure or is located on a large pipe); the radius of the sphere 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 3, Section 3.4.2.3). Due to the compartmentalization of containment in VEGP Units 1 and 2, the insulation on the opposite side of the compartment walls can be assumed to remain intact. However, the SG 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 and compartment openings per NEI 04-07 Volume 2 (Reference 3).

Volumetric debris quantities were determined by measuring the interference between a ZOI and its corresponding debris source. This was done within the CAD model.

No insulation debris would be generated outside of the ZOIs (Reference 2). This practice is considered acceptable by the NRC as stated in the SE for NEI 04-07 (Reference 3, Section 3.4.3.2).

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. 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:

VEGP applied the ZOI refinement discussed in NEI 04-07 Volume 2 (Reference 3, 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.

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 this section, the breaks selected in the Response to 3.a.1, and the size distribution provided in the Response to 3.c.1 of this enclosure, quantities of generated debris for each break case were calculated for each type of insulation. The quantities of debris generated for the four most limiting break cases are listed below in Table 3.b.4-1 as determined in the debris generation calculation. The quantities of debris generated for the four most limiting break cases that do not fail any of the acceptance criteria in the NARWHAL evaluation (see Enclosure 3 Section 13.0) are listed below in Table 3.b.4-2. See Table 3.h.5-1 in the Response to 3.h.5 for the quantity of qualified and unqualified coatings for the four most limiting break locations. See Response to 3.d.3 for the quantity of latent debris.

Table 3.b.4-1: The Four Overall Worst-Case Breaks

Break Location		11201-004-6-RB	11201-001-5-RB	11201-001-3-RB	11201-004-4-RB
Break Size		29"	29"	29"	29"
Break Type		DEGB	DEGB	DEGB	DEGB
Nukon (ft^3)	Fine	289.3	287.5	280.9	276.0
	Small	999.5	991.4	962.9	938.6
	Large	451.6	454.0	460.1	473.8
	Intact	487.9	490.5	497.1	511.9
Fire Barrier (ft^3)	Fine	0.0	2.4	2.4	0.0
	Small	0.0	2.4	2.4	0.0
Fire Barrier (lbm)	Particulate	0.0	26.9	26.8	0.0

Table 3.b.4-2: The Four Worst-Case Breaks that Do Not Fail Any Acceptance Criteria for the Single Train Failure Equipment Configuration

Break Location		11201-004-4-RB	11201-001-3-RB	11201-003-5-RB	11201-002-5-RB
Break Size		20"	23"	19"	16"
Break Type		Partial	Partial	Partial	Partial
Nukon (ft^3)	Fine	48.4	47.2	50.7	52.3
	Small	151.2	160.5	186.5	168.0
	Large	122.1	80.9	46.6	118.6
	Intact	132.0	87.4	50.3	128.1
Fire Barrier (ft^3)	Fine	0.0	0.0	0.0	0.0
	Small	0.0	0.0	0.0	0.0
Fire Barrier (lbm)	Particulate	0.0	0.0	0.0	0.0

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 or foreign materials were evaluated via walkdown. As with latent debris, a foreign material walkdown was only performed for Unit 1. However, based on the similarity between units discussed previously, Unit 1 data is considered applicable for Unit 2. The amount of foreign materials found during the walkdown was 2.0 ft^2. However, for conservatism, a total surface area of 4.0 ft^2 was assumed in the VEGP debris generation calculation, and 50 ft^2 was used in the NARWHAL conditional failure probability (CFP) calculation.

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.

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 below.

Table 3.c.1-1 – Debris Material Properties

Debris	Distribution	Density (lbm/ft³)	Characteristic Size (µm)
Nukon	See section below	2.4 (bulk) 159 (fiber)	7
Fire Barrier	Fiber (30% of total mass): 50% Fines 50% Smalls	54.3 (bulk) 175 (fiber)	1.5 (fiber)
	Particulate (70% of total mass): 100% Particulate	151 (particulate)	10 (particulate)
Qualified Coatings	100% Particulate	208 (IOZ)	10
		107 (Epoxy)	
Unqualified and Degraded Coatings	100% Particulate	208 (IOZ)	10
		109 (Epoxy)	
		122 (Alkyd)	

Nukon Low-Density Fiberglass Insulation

The bulk density of Nukon is 2.4 pounds mass per cubic foot (lbm/ft³), and the individual fiber density is 159 lbm/ft³, per NEI 04-07 (Reference 2). The characteristic diameter of the individual fibers is 7 micrometers (µm).

A baseline analysis of Nukon includes a size distribution with two categories: 60 percent small fines, and 40 percent large pieces, per NEI 04-07 (Reference 2). The debris generation calculation uses a four-category size distribution based on the guidance in NEI 04-07 Volume 2 (Reference 3). 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 17D 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 pipe diameters (C).

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

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

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

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

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. 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, specific surface areas were not calculated or used for the Vogtle head loss evaluation.

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

   Response to 3.c.4:

   The Interam E-50 Series fire barrier material was assumed to be comprised of 70 percent particulate and 30 percent fiber by weight. These values fall within the ranges given in the E-50 Material Safety Data Sheet (MSDS). It was assumed that

the fiber constituent of 3M E-50 would fail as 50 percent fines and 50 percent small pieces sized between ½ inch and 4 inches. This engineering judgment is based on observations from exploratory testing with a 1500 psi pressure washer (the same type used for NEI fiber preparation). This assumption is conservative because the observations indicate that it is a very robust material and is less likely to break up into fines than low density fiberglass (LDFG).

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.

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

   Response to 3.d.1:

   The evaluation for latent debris at VEGP was performed in a manner consistent with the NRC NEI 04-07 SE approved methodology (Reference 3, Section 3.5.2.3). The total source term was determined through the collection of debris samples from multiple locations throughout the containment. Conservatism was added by sampling those areas that exhibited unusually large concentrations of dirt and dust. In addition to dirt and dust, foreign materials and other debris sources were surveyed and documented including lint, paint chips, fibers, pieces of paper (shredded or intact), plastic, tape, adhesive labels, and fines or shards of thermal insulation, fireproof barrier, or other materials that are already present in the containment prior to a postulated break in a high-energy line inside containment.

   Vertical, horizontal, and equipment surfaces were sampled for dirt and dust by wiping with muslin cloth. Sample areas were chosen by cognizant engineering personnel with the intent to produce bounding results. The containment was divided into categories from which a minimum of three samples were taken. Prior to collecting samples, the containment was surveyed through a series of walkdowns to locate desirable sample locations.

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

   Response to 3.d.2:

   See Response to 3.d.3.

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:

Latent debris includes dirt, dust, lint, paint chips, fines, and shards of loose thermal insulation fibers that could potentially transport to the sump strainers during recirculation. Latent debris can be introduced into containment several ways, including by deterioration of items such as insulation and coatings and by personnel tracking in particulate and fibers from outside containment. The quantity of latent debris is calculated in the debris generation calculation. A walkdown of VEGP Unit 1 was performed to measure quantities of latent debris, and the total quantity was calculated based on those samples. The total amount of latent debris calculated based on walkdown data was 60 lbm, but 200 lbm is assumed in the debris generation calculation. This conservatively bounds the 60 lbm of actual latent debris with ample operating margin.

Table 3.d.3-1 lists the assumed latent fiber and particulate constituents and their material characteristics. Latent debris is assumed to consist of 15 percent fiber and 85 percent particulate by mass, per the NRC NEI 04-07 SE (Reference 3, page 50).

Based on NEI 04-07 Volume 2 (Reference 3, Sections 3.5.2.3, 3.7.2.3.2.3), the size and density of latent particulate were assumed to be 17.3 µm and 168.6 lbm/ft³, respectively. Additionally, the bulk density and microscopic density of latent fiber were assumed to be 2.4 lbm/ft³ and 93.6 lbm/ft³, respectively. Latent fiber is assumed to have a characteristic size of 5.5 µm. This is reasonably conservative, as it is the smallest fiber diameter listed in Table 3-2 of the general reference for low-density fiberglass found in NEI 04-07 (Reference 2).

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

	Latent Debris (lbm)	Bulk Density (lbm/ft³)	Microscopic Density (lbm/ft³)	Characteristic Size (µm)
Particulate (85%)	170*	-	168.6	17.3
Fiber (15%)	30	2.4	93.6	5.5
Total	200			

* This mass of latent particulate debris corresponds to a volume of 1.01 ft³ based on a density of 168.6 lbm/ft³.

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

Response to 3.d.4:

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

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.

1. Describe the methodology used to analyze debris transport during 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 guidance and the associated NRC SE (Reference 3), as well as the refined methodologies suggested by the SE in Appendices III, IV, and VI (Reference 3). The specific effect of each of four modes of transport was analyzed in the debris transport calculations for each type of debris generated. These modes of transport are:

- Blowdown Transport – the vertical and horizontal transport of debris to all areas of containment by the break jet
- Washdown Transport – the vertical (downward) transport of debris by the containment sprays, break flow, and condensation
- Pool Fill-Up Transport – the transport of debris by break and containment spray flows from the RWST to regions that may be active or inactive during recirculation
- Recirculation Transport – the horizontal transport of debris from the active portions of the recirculation pool to the sump screens by the flow through the ECCS

The logic tree approach was applied for each type of debris determined from the debris generation calculation. The logic tree shown in Figure 3.e.1-1 is slightly different from the baseline. This departure was made to account for certain non-conservative assumptions identified by the NRC SE (Reference 3), including the 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 screens during pool fill-up.

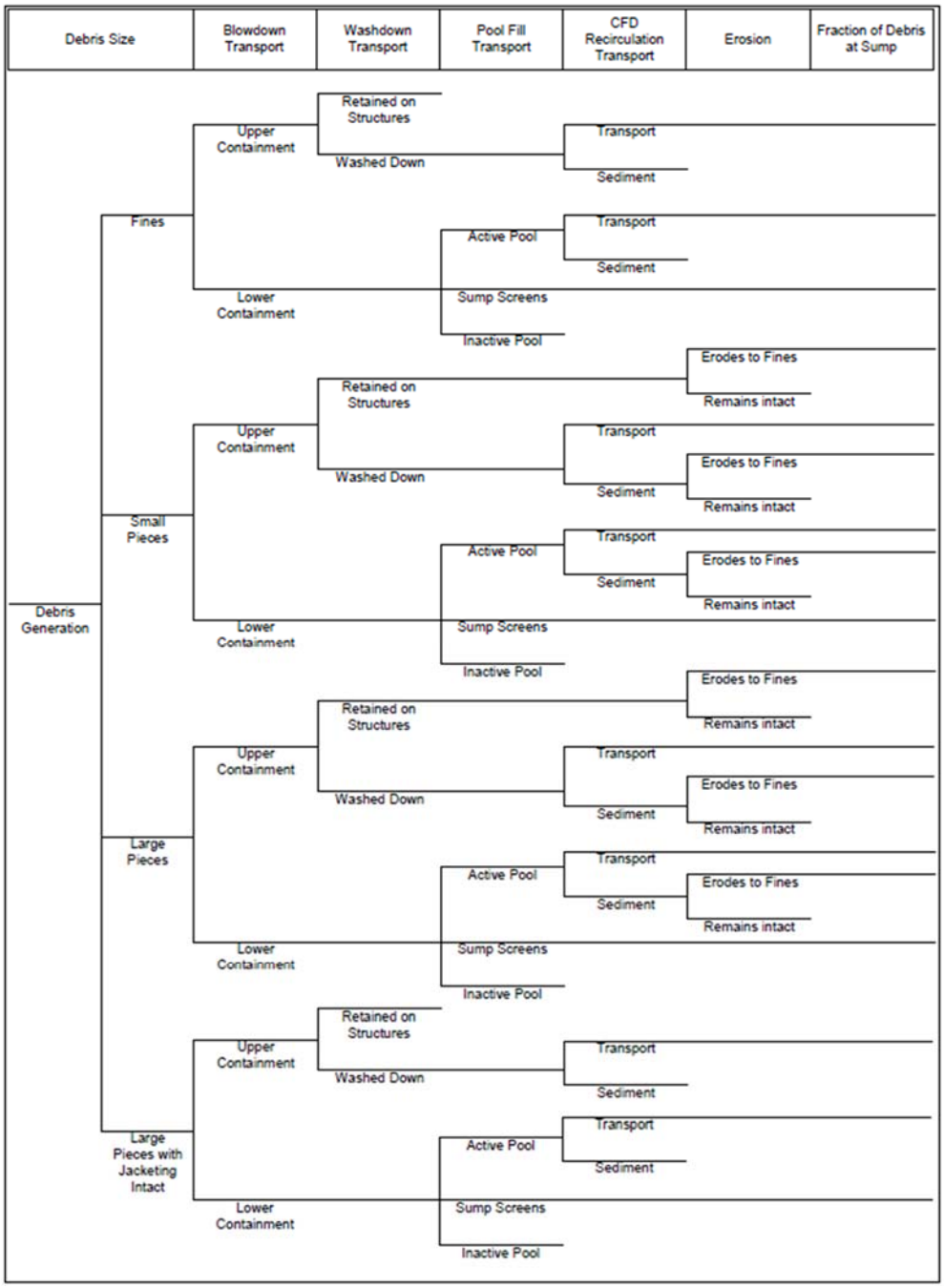

Figure 3.e.1-1: Generic Debris Transport Logic Tree

The basic methodology for the VEGP transport analysis is summarized below.

1. The CAD model was provided as input to determine break locations and sizes.
2. The debris generation calculation was provided as input into the calculation for debris types and sizes.
3. Potential upstream blockage points were qualitatively addressed.

4. The fraction of debris blown into upper containment and lower containment for each compartment was determined based on the volumes of upper and lower containment. Inertial capture was not credited for fine debris. Inertial capture (by miscellaneous structures, grating and 90° flow turns) was considered for small pieces of fiberglass, 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 spray.
5. The fraction of debris washed down by containment spray flow was determined along with the locations where the debris would be washed down.
6. The quantity of debris transported to inactive areas or directly to the sump strainers was calculated based on the volume of the inactive and sump cavities proportional to the water volume at the time these cavities are filled.
7. The location of each type/size of debris at the beginning of recirculation was determined based on the break location. All debris that transports to lower containment (e.g., following blowdown and/or washdown) is assumed to be in the pool.
8. A CFD model was developed in Flow-3D to simulate the flow patterns 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 initial recirculation transport fractions from the CFD analysis were gathered to determine the final recirculation transport fractions for 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/size of debris was determined by combining each of the previous steps into logic trees.

Potential Upstream Blockage Points

Potential upstream blockage points were qualitatively addressed in the debris transport calculation. It was determined that there are not any upstream blockage points in the VEGP containment building. Upstream effects are discussed in the Response to 3.l.

CFD Model of Containment Recirculation Pool

A diagram showing the significant parts of the CFD model is shown in Figure 3.e.1-2. The sump mass sink and the various direct and runoff spray regions are highlighted.

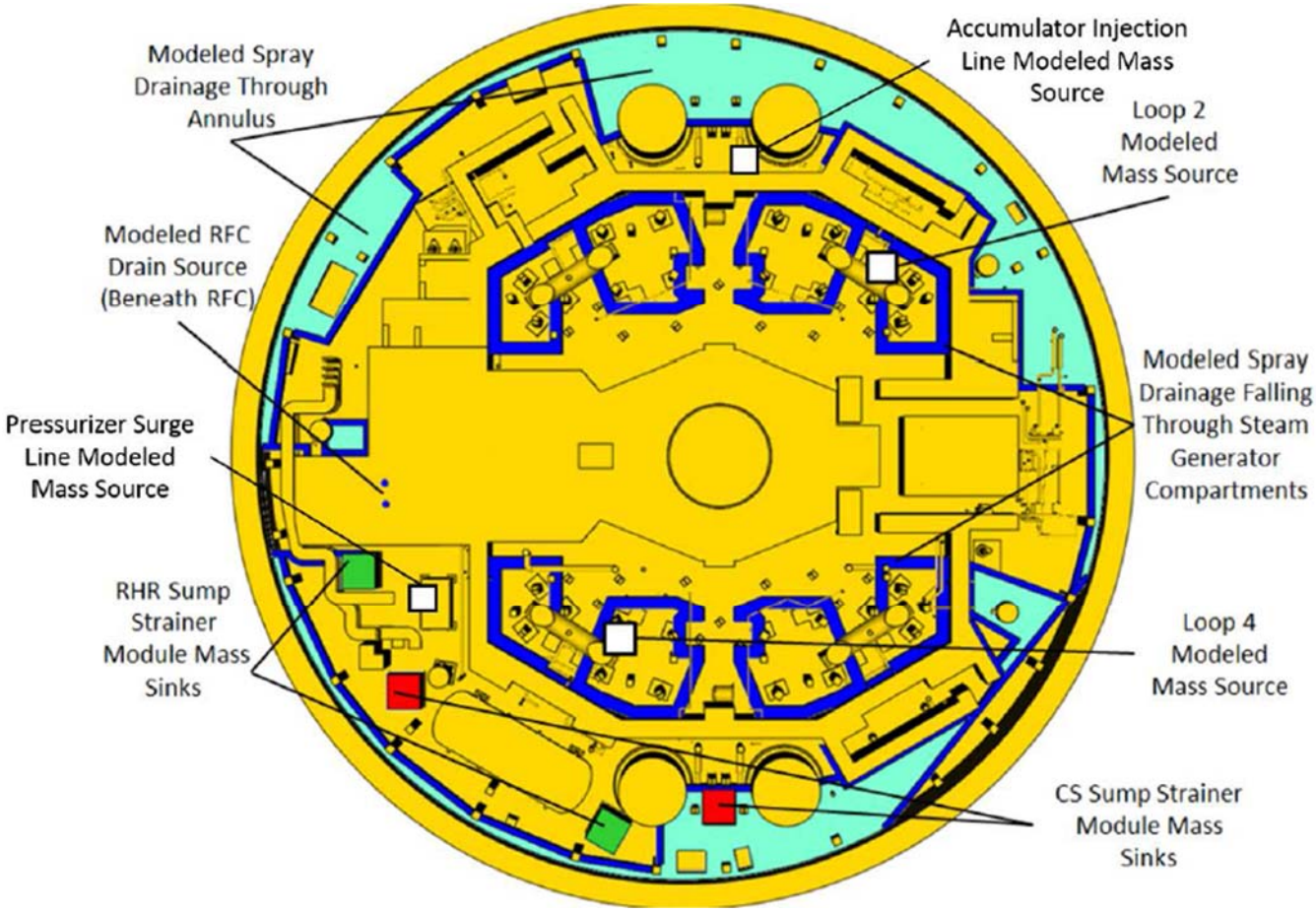

Figure 3.e.1-2: Significant Features in CFD Model

The key CFD modeling attributes/considerations included the following:

Computational Mesh

A rectangular mesh was defined in the CFD model 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 5 inch by 5 inch was used in the x and y directions and 3-inch to 4-inch mesh space was used in the z direction.

Modeling of Containment Spray Flows

For CFD cases with CS activated, various plan and section drawings, as well as the containment building CAD model, were considered. Spray water would drain to the pool through many pathways. Some of these pathways include the steam generator enclosures, the various openings in the operating deck, the annulus through the various open sections of grating, and the refueling canal drains. The sprays were 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 that a freefalling object would have if it fell the vertical distance from the location of the break to the surface of the pool.

Modeling of the Sump Strainers

Each sump strainer in VEGP consists of four columns of stacked disks with a solid plate on top. In the CFD model, each strainer was modeled with a plate above it to prevent flow from entering through the top of the strainers. The sump strainers were modeled as having flow across their surfaces proportional to the areas of the strainers. A negative flow rate was set for the sump mass sink, which tells the CFD model to draw the specified amount of water from the pool over the entire exposed surface area of the mass sink obstacle.

Turbulence Modeling

Several different turbulence-modeling approaches can be selected for a Flow-3D calculation. 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 this 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 TKE 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. The steady-state conditions were used to determine the recirculation debris transport fractions. A plot of mean kinetic energy was used to determine when steady-state conditions were reached. Checks were also made of the velocity and turbulent energy patterns in the pool to verify that steady-state conditions were reached.

CFD Simulation Cases

For the recirculation transport fractions, CFD simulations were performed for seven different LBLOCA cases in the debris transport calculation. The key input parameters for these cases are tabulated below:

Table 3.e.1-0a – CFD Simulation Cases for Sump Recirculation Transport

CFD Case No.	Break Location	Containment Spray Activation	No. of Active Trains	Pump Flow Rates (gpm)	Water Level (ft)
1	Loop 4 Crossover Leg	Off	2	3,734 gpm (RHR)	6.597
2	Pressurizer Surge Line in Annulus	Off	2	3,734 gpm (RHR)	6.597
3	Accumulator Injection Line in Annulus	Off	2	3,734 gpm (RHR)	6.597
4	Loop 4 Crossover Leg	Off	1	4,500 gpm (RHR)	6.597
5	Loop 4 Crossover Leg	Off	2	3,734 gpm (RHR)	7.534
6	Loop 2 Crossover Leg	On	2	4,500 gpm (RHR) 3,200 gpm (CS)	5.25
7	Loop 4 Crossover Leg	On	2	4,500 gpm (RHR) 3,200 gpm (CS)	5.25

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 VEGP transport analysis originate from the sources below.

- NUREG/CR-6772 Tables 3.1 and 3.2 (Reference 37)
- NUREG/CR-6808 Figure 5.2 and Tables 5-1 and 5-3 (Reference 39)

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. Containment pool debris distributions were defined for fine debris (including latent debris and unqualified coatings, as well as Nukon, fire barrier, and qualified coatings fines generated inside the ZOI). However, because very low turbulence is required to transport fine debris, the recirculation transport fraction for all fine debris is 100% regardless of the initial distribution in the containment pool.

For the breaks inside the secondary shield wall, small and large pieces of insulation debris blown to the containment pool were assumed to be uniformly distributed inside the secondary shield wall. This is reasonable since the blowdown and the majority of the pool fill phases are multidirectional flows that would tend to disburse debris around the area inside the secondary shield wall (including areas with lower transport potential). The small piece debris blown to upper containment was assumed to be distributed in the vicinity of the locations where it is washed down. For breaks in the annulus that are near or far from the sump strainers, small and large pieces of debris were assumed to be distributed near the break location.

Figure 3.e.1-2a through Figure 3.e.1-2d graphically depict the initial debris distribution areas for the various types/sizes of debris and different break locations. Figure 3.e.1-2a shows the distribution area of small piece debris washed down from upper containment to the steam generator compartments inside the secondary shield wall (SSW), through the refueling canal (RFC) drain, and to the annulus. Figure 3.e.1-2b shows the distribution area for small and large piece debris for breaks in the steam generator compartments. Figure 3.e.1-2c and Figure 3.e.1-2d show the distribution area for small and large piece debris for breaks in the annulus near and far from the sump strainers, respectively.

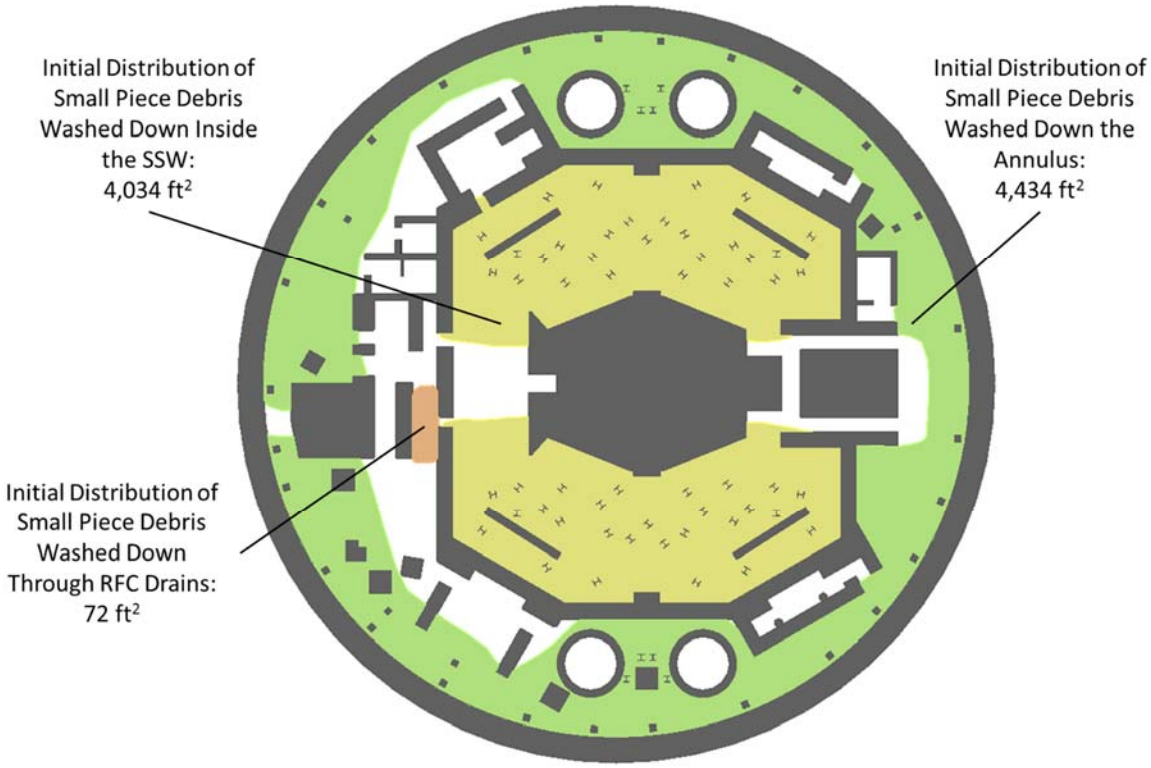

Figure 3.e.1-2a: Distribution of Small Piece Debris Washed Down from Upper Containment

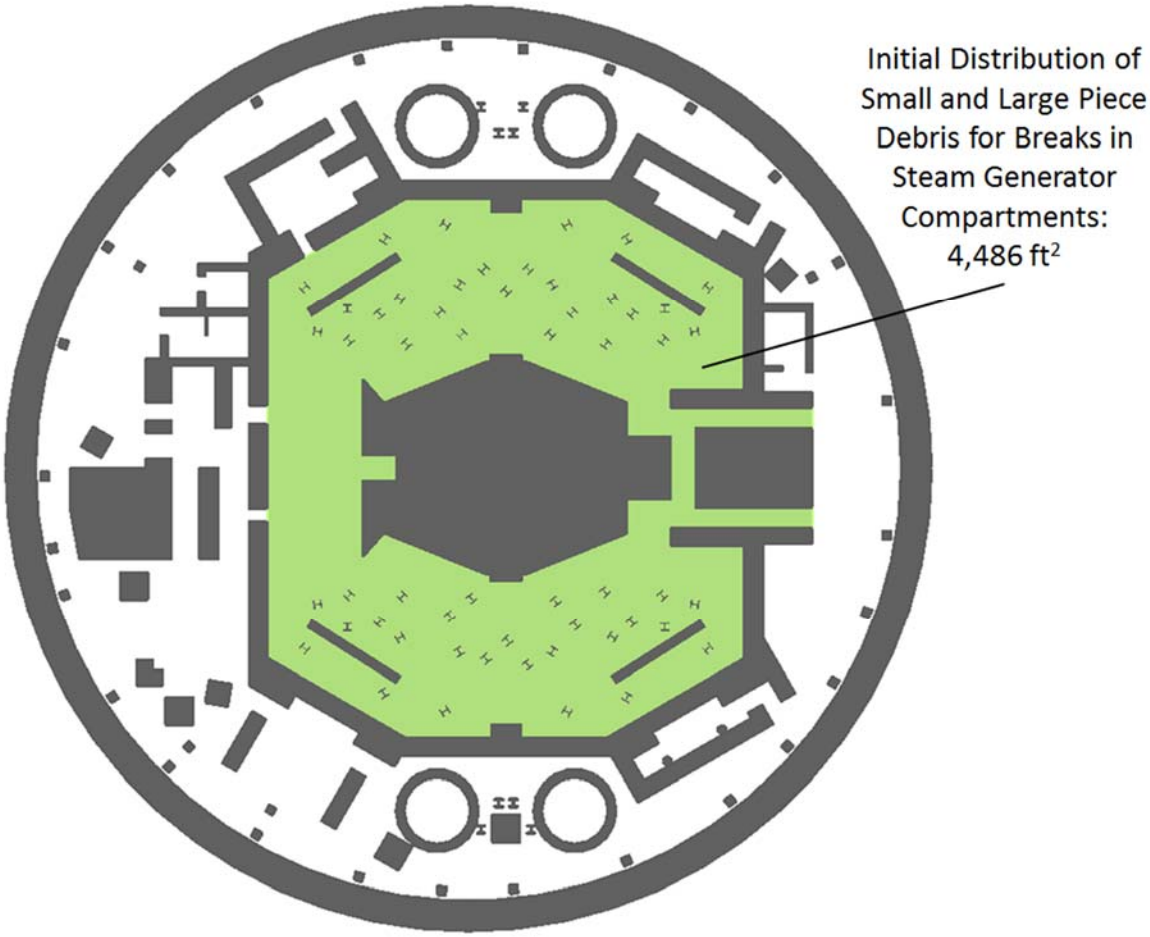

Figure 3.e.1-2b: Distribution of Small and Large Piece Debris in Lower Containment for Breaks in the Steam Generator Compartments

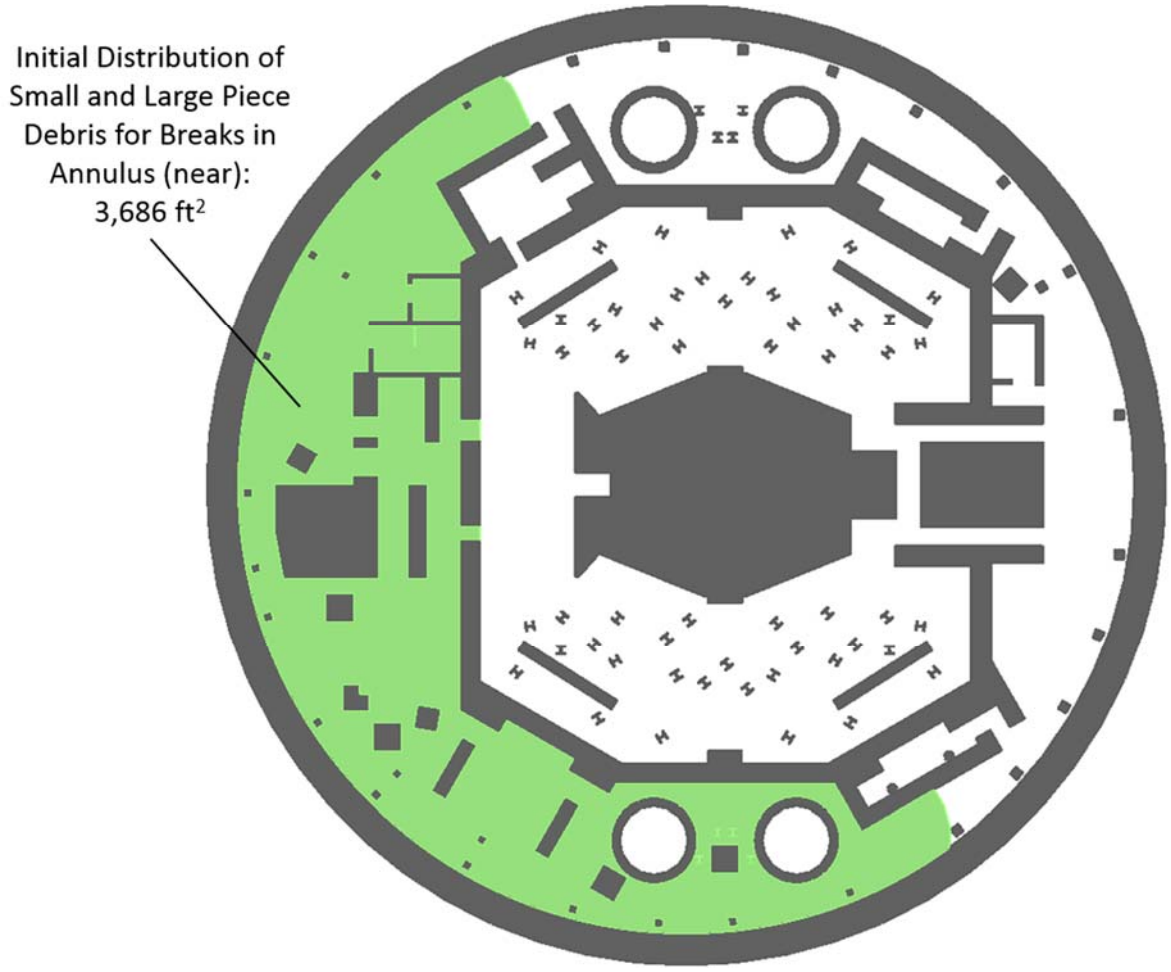

Figure 3.e.1-2c: Distribution of Small and Large Piece Debris in Lower Containment for Breaks in the Annulus (Near Sump Strainers)

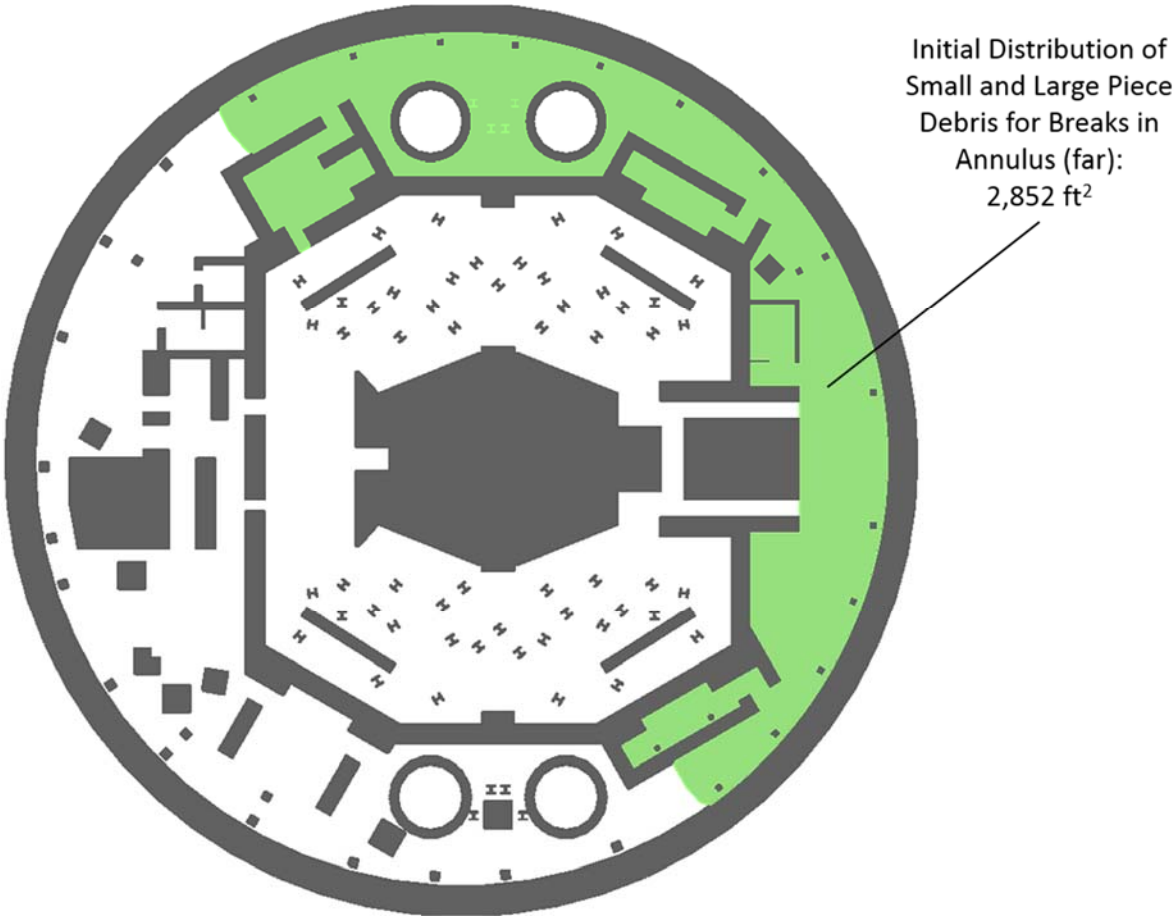

Figure 3.e.1-2d: Distribution of Small and Large Piece Debris in Lower Containment for Breaks in the Annulus (Far from Sump Strainers)

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 emergency sump screens. Detailed explanations of each bullet are provided in the paragraphs below.

- 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 an AutoCAD 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.

Plots showing the TKE and the velocity magnitude in the pool were generated for each case to determine areas where specific types of debris would be transported. The limits on the plots were set according to the minimum TKE or velocity metrics necessary to move each type of debris. The overlying yellow areas represent regions where the debris would be suspended, and the red areas represent regions where the debris would be tumbled along the floor (see Figure 3.e.1-4). The yellow TKE portion of the plots is a three-dimensional representation of the TKE. Since the TKE is a three-dimensional representation, the plots do not show the TKE at any specific elevation. Rather, any debris that is shown to be present in this yellow area will transport, regardless of the elevation of TKE in the pool. The velocity portion of the plots represents the velocity magnitude just above the floor level (1.5 inches), where tumbling of sunken debris could occur. Directional flow vectors were also included in the plots to determine whether debris in certain areas would be transported to the sump strainers or transported to less active regions of the pool where it could settle to the floor (blue regions).

The following figures and discussion are presented as an example of how the transport analysis was performed for a generic small debris type. This same approach was used for other debris types analyzed at VEGP.

As shown in Figure 3.e.1-3, the small debris (depicted by green shading) was initially assumed to be uniformly distributed between the break location and the sump strainers. The break location in this scenario is a break in the annulus on the pressurizer surge line.

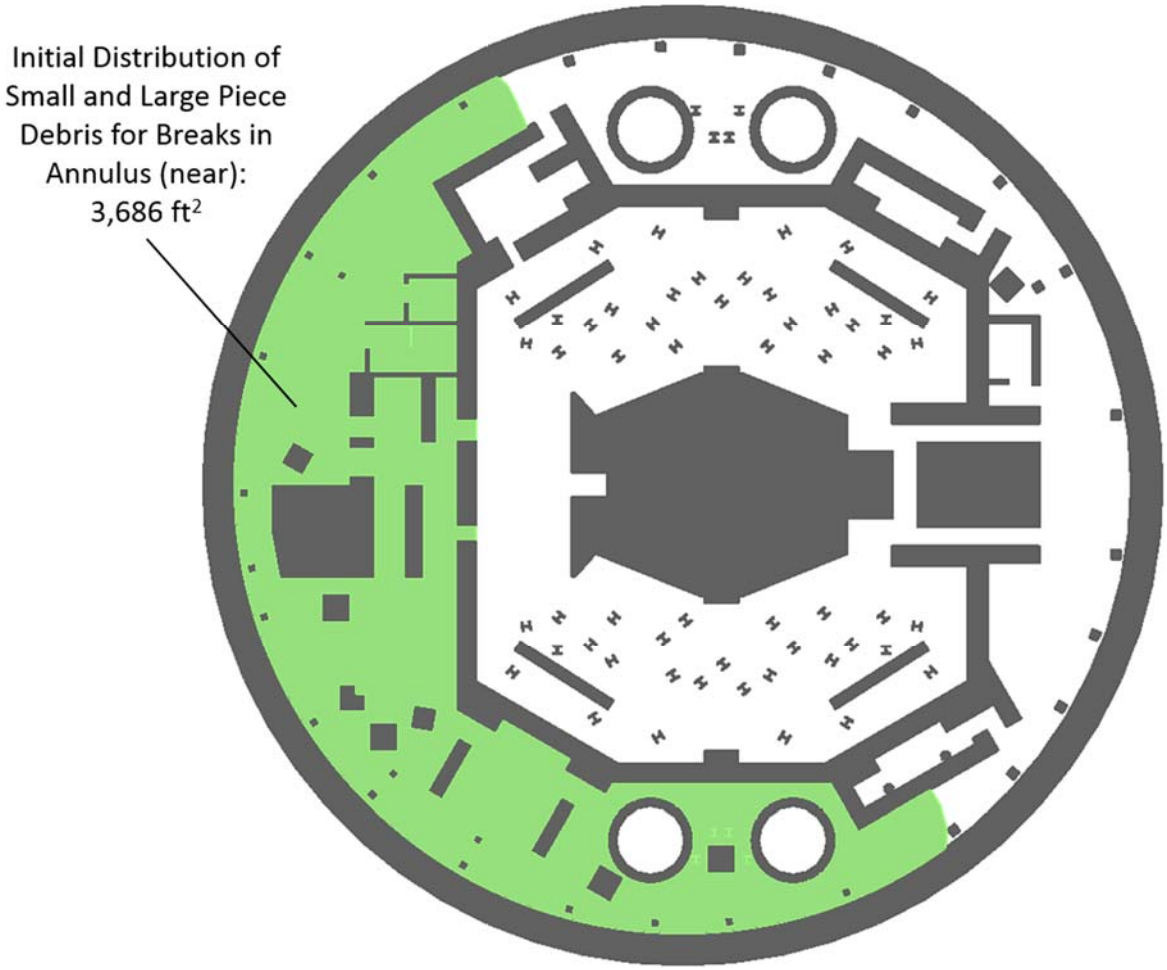

Figure 3.e.1-3: Distribution of Small Debris in Lower Containment

Figure 3.e.1-4 shows that the turbulence (yellow regions) and the velocity (red regions) in the pool (blue regions) are high enough to transport the generic small debris to the sump strainers during recirculation. The initial distribution area (Figure 3.e.1-3) was overlaid on top of the plot showing tumbling velocity, TKE, and flow vectors (Figure 3.e.1-4) to determine the recirculation transport fraction (Figure 3.e.1-5).

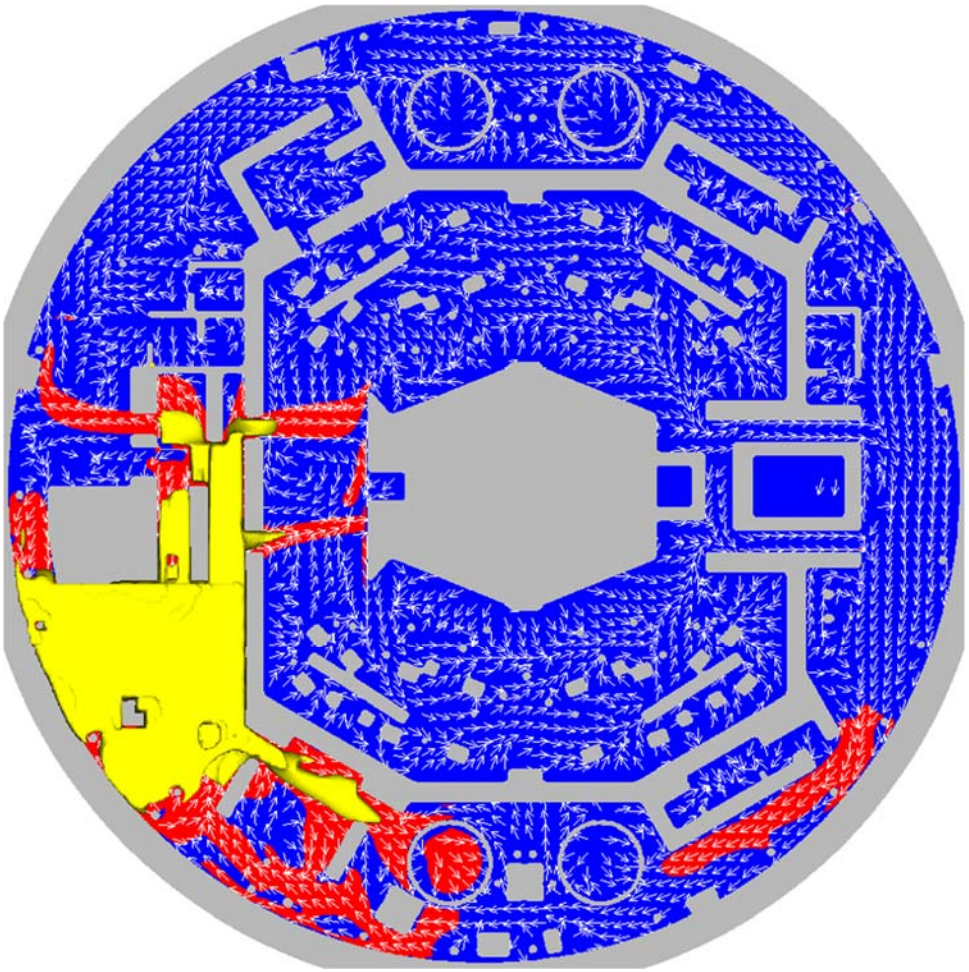

Figure 3.e.1-4: TKE and Velocity with Limits Set at Suspension/ Tumbling of Small Fiberglass Debris

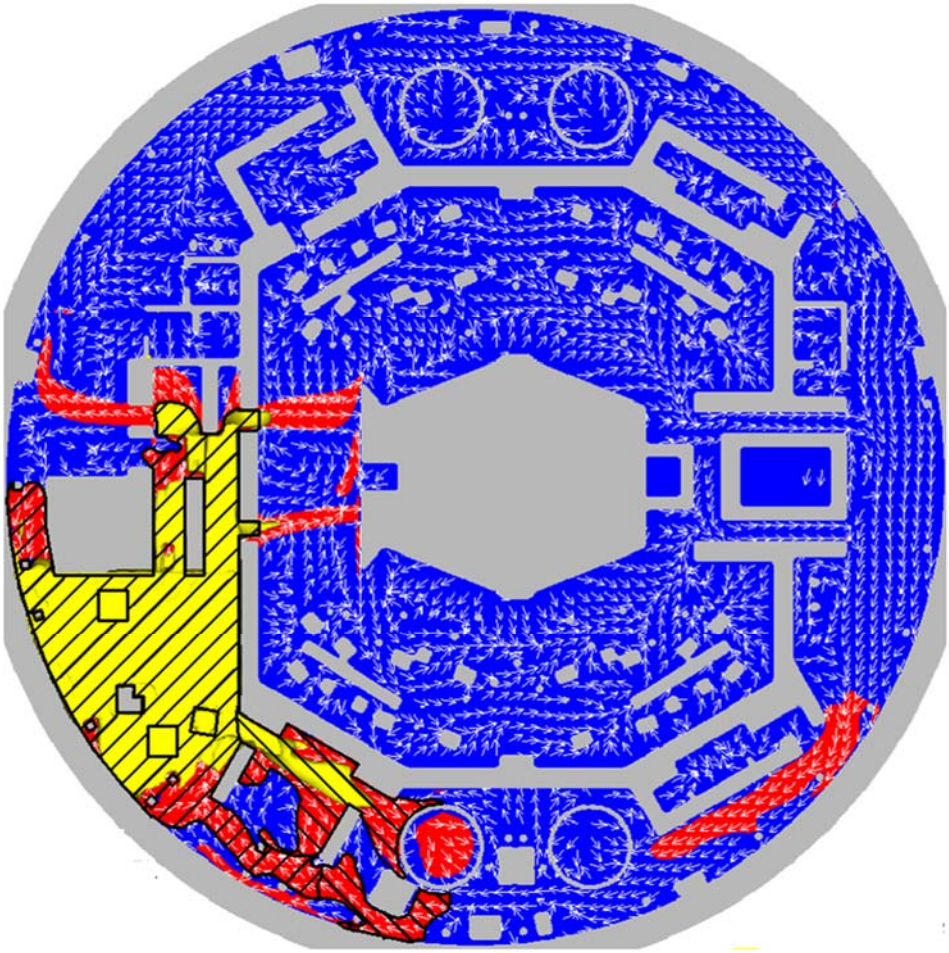

Figure 3.e.1-5: Floor Area where Small Fiberglass Debris Would Transport to the Sump Strainers (hatched area)

This same analysis was applied for each type of debris at VEGP. Recirculation-pool transport fractions were identified for each debris type associated with the location of its initial distribution. This includes a recirculation transport fraction for debris blown to lower containment, debris washed down inside the secondary shield wall, and debris washed down into the annulus.

Application of Recirculation Transport Fractions

The recirculation debris transport fractions derived from the seven CFD cases (see Table 3.e.1-0a) were applied to various break locations and equipment configurations, as summarized in the table below.

Table 3.e.1-0b: Application of CFD Cases to Different Break Scenarios

Break Location	Containment Spray Activation	Active Trains	CFD Case Applied
Steam Generator Compartments 1 & 4	Off	2	Case 1
Steam Generator Compartments 1 & 4	On	1 or 2	Case 7
Steam Generator Compartments 1 & 4	Off	1	Case 4
Steam Generator Compartments 2 & 3	Off	2	Case 1
Steam Generator Compartments 2 & 3	On	1 or 2	Case 6
Steam Generator Compartments 2 & 3	Off	1	Case 4
Reactor Cavity	Off	2	Case 1
Reactor Cavity	On	1 or 2	Case 7
Reactor Cavity	Off	1	Case 4
Pressurizer Compartment	Off	2	Case 1
Pressurizer Compartment	On	1 or 2	Case 7
Pressurizer Compartment	Off	1	Case 4
Annulus	Off	2	Case 2
Annulus	On	1 or 2	Case 7
Annulus	Off	1	Case 2

Erosion Discussion

Due to the turbulence in the recirculation pool and the force of break and spray flow, Nukon debris may erode into smaller pieces, making transport of this debris to the strainer more likely. Results of the Drywell Debris Transport Study indicate that debris exposed to containment sprays above the recirculation pool undergo an erosion fraction of less than 1%. Therefore, a 1% erosion fraction for debris held up on gratings and other miscellaneous structures at Vogtle was used.

Erosion Test Discussion

To estimate erosion that would occur in the recirculation pool at Vogtle, generic 30-day erosion testing was performed. [[

]][1]

Input Parameters

The flow conditions used for the testing were based on prototypical plant flow conditions. The target velocity selected for the erosion testing was [[
.]][1] based on the tumbling velocity required to transport small pieces of

[1] Alion trade secret

LDFG (Reference 37). Since small pieces of LDFG would transport at higher velocities, the non-transporting small pieces of LDFG on a containment pool floor would be exposed to a velocity less than [[ ]][1]. Typically, in regions where the velocity is lower than 0.12 ft/s, the pool is relatively quiescent, and the turbulence levels are very low. Based on a review of the average turbulence levels for various plants in the quiescent "non-transport" regions, a target turbulence of [[ ]][1] was selected for the erosion testing.

To prevent potential contamination of the samples from the minerals in tap water, deionized (DI) water was selected for the erosion testing. In prototypical plant conditions, the containment pool water is borated and buffered. Based on observations during chemical effects testing, chemical precipitates tend to accumulate on exposed fiberglass. This effect can mask the actual erosion. However, by using pure water, this phenomenon was eliminated in the erosion testing.

Erosion Test Durations

Table 3.e.1-1 shows the test matrix for the erosion testing. The length of the pre-test was selected based on the longest period of time that any one filter was installed during the primary test and post-test, or 5 days. The length of the primary test was based on a full 30-day mission time. The length of the post-test was based on the filter measurements during the primary test as well as the initial filter measurements during the post-test, or 5 days. Since the primary test measurements showed that the majority of erosion occurs within the first 10 days, a 10-day test was determined to be an adequate length of time to accomplish the purposes of the post-test. This test was run for a total of 13 days.

Table 3.e.1-1 Erosion Test Durations

Test	Duration	Description
Pre-Test	5 days	Quantify weight change of a filter under test conditions without any fiberglass in the flume.
Primary Test	30 days	Measure overall weight loss of fiber clumps after 30 days exposure to flow. Also, determine time dependent erosion curve by measuring weight change of filters throughout.
Post-Test	13 days	Determine repeatability of primary test, and confirm that there are no unknown long-term phenomena that resulted in a non-conservative weight gain of the fiber samples during the later stages of the primary test.

In summary, appropriate erosion fractions were applied for small and large pieces of fiberglass retained in upper containment and those in the pool (including both the transportable and non-transportable debris). In NARWHAL, the fines generated due to erosion are subtracted from the quantity of small and large pieces and added to

the quantity of fines prior to the start of recirculation. This is conservative since early arrival of fines (when the pool temperature is higher) is more detrimental for strainer failures.

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 and does not deviate from the NRC approved NEI 04-07 guidance and the associated NRC SE (Reference 3) for refined analyses, as well as the refined methodologies suggested by the SE in Appendices III, IV, and VI.

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, several CFD simulations were performed using Flow-3D, a commercially available software package. Seven breaks were investigated that included single- and two-train recirculation with sprays both on and off to ensure a conservative representation of the post-LOCA containment-sump flow velocities. Breaks were also evaluated inside and outside the secondary shield wall to determine which scenario(s) would maximize debris transport. The simulation results include a series of contour plots of velocity and TKE. These results have been combined with settling and tumbling velocities from the GSI-191 literature to determine the recirculation transport fractions for all debris types present in the VEGP containment building. See Response to 3.e.1 for additional discussion of the CFD results.

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

   Response to 3.e.4:

   No credit was taken for debris interceptors.

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 settling of fine debris.

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 listed in Table 3.e.6-1 through Table 3.e.6-14 are inputs to the NARWHAL CFP calculation. Note that these fractions result in the bounding quantity of debris transported to the strainer. The debris transport quantities are provided in Tables 3.e.6-15 and 3.e.6-16.

Blowdown Transport

Table 3.e.6-1 shows the bounding blowdown transport fractions as a function of break location and debris type.

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 (all)	80%	20%	0%
	Small Nukon & Fire Barrier	39%	61%	0%
	Large Nukon	0%	100%	0%
Reactor Cavity	Fines (all)	80%	20%	0%
	Small Nukon & Fire Barrier	39%	61%	0%
	Large Nukon	0%	100%	0%
Pressurizer Compartment	Fines (all)	80%	20%	0%
	Small Nukon & Fire Barrier	69%	9%	22%
	Large Nukon	0%	0%	100%
Annulus - Pressurizer Surge Line	Fines (all)	80%	20%	0%
	Small Nukon & Fire Barrier	35%	18%	47%
	Large Nukon	0%	25%	75%
Annulus - Accumulator Injection Line	Fines (all)	80%	20%	0%
	Small Nukon & Fire Barrier	17%	83%	0%
	Large Nukon	0%	100%	0%

Washdown Transport

Table 3.e.6-2 shows the bounding washdown transport fractions as a function of containment spray activation and debris type. Note that these transport fractions do not depend on the location of the break.

Table 3.e.6-2: Washdown Transport Fractions

Sprays Initiated?	Debris Type	Transport Fraction		
		Washed Down in Annulus	Washed Down Inside SSW	Washed Down RFC Drains
Yes	Fines (all)	53%	37%	10%
	Small Nukon & Fire Barrier	43%	37%	10%
	Large Nukon	0%	0%	10%
No	Fines (all)	10%		
	Small- and Large-Piece Debris	0%	0%	0%

Pool-Fill Transport

Table 3.e.6-3 shows the bounding pool fill transport fractions as a function of debris type.

Table 3.e.6-3: Pool fill Transport Fractions

Debris Type	Pool Fill Transport Fraction	
	Elevator Cavity	Each ECCS Sump
Fines (all)	2%	0.75%
Small Nukon & Fire Barrier	2%	0.75%
Large Nukon	2%	0.75%
Unqualified Coatings	0%	0%

Recirculation Transport

For the recirculation transport fractions, seven different cases were evaluated in the debris transport calculation, as discussed in the Response to 3.e.1. These cases are listed below:

- Case 1: LBLOCA in SG Compartment Loop 4, Sprays not Activated, Two Trains Operational
- Case 2: LBLOCA in Pressurizer Surge Line in Annulus, Sprays not Activated, Two Trains Operational
- Case 3: LBLOCA in Accumulator Injection Line in Annulus, Sprays not Activated, Two Trains Operational
- Case 4: LBLOCA in SG Compartment Loop 4, Sprays not Activated, One Train Operational

- Case 5: LBLOCA in SG Compartment Loop 4, Sprays not Activated, Two Trains Operational, High Water Level
- Case 6: LBLOCA in SG Compartment Loop 2, Sprays Activated, Two Trains Operational
- Case 7: LBLOCA in SG Compartment Loop 4, Sprays Activated, Two Trains Operational

The bounding recirculation transport fractions for fine debris are shown in Table 3.e.6-4. As discussed in the Response to 3.e.3, the recirculation transport fractions were determined using the results from CFD simulations. Because the strainers are in different physical locations, the bulk pool velocity varies significantly in the vicinity of each strainer, which ultimately affects the small and large fiberglass debris transport fractions to each strainer. This variation results in the differences in the recirculation transport fractions for small and large pieces to each of the strainers as shown in Tables 3.e.6-4 through 3.e.6-6, as well as the overall transport fractions shown in Tables 3.e.6-7 through 3.e.6-14.

Table 3.e.6-4: Recirculation Transport Fractions for Fine Debris

Case	Sump	Break Recirculation	Washed Inside Secondary Shield Wall	Washed In Annulus	Washed Down RFC
Case 1	RHR A	50%	NA	NA	NA
	RHR B	50%	NA	NA	NA
Case 2	RHR A	50%	NA	NA	NA
	RHR B	50%	NA	NA	NA
Case 3	RHR A	50%	NA	NA	NA
	RHR B	50%	NA	NA	NA
Case 4	RHR B	100%	NA	NA	NA
Case 5	RHR A	50%	NA	NA	NA
	RHR B	50%	NA	NA	NA
Case 6	CS A	21%	21%	21%	21%
	RHR A	29%	29%	29%	29%
	CS B	21%	21%	21%	21%
	RHR B	29%	29%	29%	29%
Case 7	CS A	21%	21%	21%	21%
	RHR A	29%	29%	29%	29%
	CS B	21%	21%	21%	21%
	RHR B	29%	29%	29%	29%

The bounding recirculation transport fractions for small fiber debris are shown in Table 3.e.6-5.

Table 3.e.6-5: Recirculation Transport Fractions for Small Fiber Debris

Case	Sump	Break Recirculation	Washed Inside Secondary Shield Wall	Washed In Annulus	Washed Down RFC
Case 1	RHR A	0%	NA	NA	NA
	RHR B	0%	NA	NA	NA
Case 2	RHR A	9%	NA	NA	NA
	RHR B	33%	NA	NA	NA
Case 3	RHR A	0%	NA	NA	NA
	RHR B	0%	NA	NA	NA
Case 4	RHRB	0%	NA	NA	NA
Case 5	RHR A	0%	NA	NA	NA
	RHR B	0%	NA	NA	NA
Case 6	CS A	24%	23%	28%	0%
	RHR A	0%	0%	9%	0%
	CS B	8%	8%	19%	100%
	RHR B	9%	10%	7%	0%
Case 7	CS A	20%	17%	25%	0%
	RHR A	0%	0%	10%	0%
	CS B	8%	9%	6%	100%
	RHR B	26%	25%	20%	0%

The bounding recirculation transport fractions for large fiber debris are shown in Table 3.e.6-6.

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

Case	Sump	Break Recirculation	Washed Inside Secondary Shield Wall	Washed In Annulus	Washed Down RFC
Case 1	RHR A	0%	NA	NA	NA
	RHR B	0%	NA	NA	NA
Case 2	RHR A	4%	NA	NA	NA
	RHR B	20%	NA	NA	NA
Case 3	RHR A	0%	NA	NA	NA
	RHR B	0%	NA	NA	NA
Case 4	RHR B	0%	NA	NA	NA
Case 5	RHR A	0%	NA	NA	NA
	RHR B	0%	NA	NA	NA
Case 6	CS A	0%	NA	NA	NA
	RHR A	0%	NA	NA	NA
	CS B	0%	NA	NA	NA
	RHR B	0%	NA	NA	NA
Case 7	CS A	0%	NA	NA	NA
	RHR A	0%	NA	NA	NA
	CS B	0%	NA	NA	NA
	RHR B	0%	NA	NA	NA

Overall Debris Transport

Transport logic trees, which include blowdown, washdown, pool fill, 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 bounding overall transport fractions are presented in Table 3.e.6-7 through Table 3.e.6-14. Note that the near annulus breaks represent breaks that are within close proximity to the strainers in the annulus, and that the far annulus breaks represent breaks that are far away from the strainer in the annulus. The values below are slightly different than what is calculated in NARWHAL. This is because the total transport fractions are entered in NARWHAL and the time-dependent fiber accumulation on the strainers is calculated based on the flow split. Additionally, some fiber penetrates the strainers and accumulates on the core, which also contributes to this slight difference.

Taking credit for the initial distribution of the unqualified coatings in the upper and lower containment results in different overall transport fractions depending on whether the containment sprays are initiated. The overall transport fractions for unqualified coatings are less than 100% for the cases with no containment spray, as shown in Tables 3.e.6-8 through 3.e.6-12 (see discussion in the Response to 3.h.2).

In the NARWHAL runs for VEGP, 100% of the latent debris was conservatively assumed to be in the containment pool at the start of the event. Given that 2% transports to the inactive elevator cavity and 3% transports to the four strainers during the pool fill phase (see Table 3.e.6-3), 95% of the latent debris would be in the active pool at the start of recirculation. The recirculation transport fraction for fine debris is 100% (see Table 3.e.6-4), resulting in an overall transport fraction of 98% (3% pool fill transport plus 95% recirculation transport) to the strainers for the latent debris.

Table 3.e.6-7: Overall Transport Fractions for an SG Compartment/ Reactor Cavity Break, Two Trains Operational, CS On

Debris Type	CS A	RHR A	CS B	RHR B	Total
Nukon Individual Fibers	21%	29%	21%	29%	100%
Nukon & Fire Barrier Small Pieces	20%	5%	13%	25%	63%
Nukon Large Pieces	3%	4%	3%	4%	14%
Nukon Intact Pieces	0%	0%	0%	0%	0%
Fire Barrier Fines & Particulate	21%	29%	21%	29%	100%
Qualified Coatings (IOZ, Epoxy)	21%	29%	21%	29%	100%
Unqualified Epoxy Coatings Particulate	21%	29%	21%	29%	100%
Unqualified IOZ Coatings Particulate	21%	29%	21%	29%	100%
Unqualified Alkyd Coatings Particulate	21%	29%	21%	29%	100%
Latent Dirt/Dust Particulate & Fiber	21%	28%	21%	28%	98%

Table 3.e.6-8: Overall Transport Fractions for an SG Compartment/ Reactor Cavity Break, One Train Operational, CS Off

Debris Type	CS A	RHR A	CS B	RHR B	Total
Nukon Individual Fibers	NA	NA	NA	27%	27%
Nukon & Fire Barrier Small Pieces	NA	NA	NA	6%	6%
Nukon Large Pieces	NA	NA	NA	10%	10%
Nukon Intact Pieces	NA	NA	NA	0%	0%
Fire Barrier Fines & Particulate	NA	NA	NA	27%	27%
Qualified Coatings (IOZ, Epoxy)	NA	NA	NA	27%	27%
Unqualified Epoxy Coatings Particulate	NA	NA	NA	47%	47%
Unqualified IOZ Coatings Particulate	NA	NA	NA	60%	60%
Unqualified Alkyd Coatings Particulate	NA	NA	NA	100%	100%
Latent Dirt/Dust Particulate & Fiber	NA	NA	NA	98%	98%

Table 3.e.6-9: Overall Transport Fractions for an SG Compartment/ Reactor Cavity Break, Two Trains Operational, CS Off

Debris Type	CS A	RHR A	CS B	RHR B	Total
Nukon Individual Fibers	NA	14%	NA	14%	28%
Nukon & Fire Barrier Small Pieces	NA	3%	NA	3%	6%
Nukon Large Pieces	NA	6%	NA	6%	12%
Nukon Intact Pieces	NA	0%	NA	0%	0%
Fire Barrier Fines & Particulate	NA	14%	NA	14%	28%
Qualified Coatings (IOZ, Epoxy)	NA	14%	NA	14%	28%
Unqualified Epoxy Coatings Particulate	NA	23%	NA	23%	46%
Unqualified IOZ Coatings Particulate	NA	30%	NA	30%	60%
Unqualified Alkyd Coatings Particulate	NA	50%	NA	50%	100%
Latent Dirt/Dust Particulate & Fiber	NA	49%	NA	49%	98%

Table 3.e.6-10: Overall Transport Fractions for a Pressurizer Compartment Break, Two Trains Operational, CS Off

Debris Type	CS A	RHR A	CS B	RHR B	Total
Nukon Individual Fibers	NA	14%	NA	14%	28%
Nukon & Fire Barrier Small Pieces	NA	1%	NA	3%	4%
Nukon Large Pieces	NA	0%	NA	0%	0%
Nukon Intact Pieces	NA	0%	NA	0%	0%
Fire Barrier Fines & Particulate	NA	14%	NA	14%	28%
Qualified Coatings (IOZ, Epoxy)	NA	14%	NA	14%	28%
Unqualified Epoxy Coatings Particulate	NA	23%	NA	23%	46%
Unqualified IOZ Coatings Particulate	NA	30%	NA	30%	60%
Unqualified Alkyd Coatings Particulate	NA	50%	NA	50%	100%
Latent Dirt/Dust Particulate & Fiber	NA	49%	NA	49%	98%

Table 3.e.6-11: Overall Transport Fractions for a Near Annulus Break, Two Trains Operational, CS Off

Debris Type	CS A	RHR A	CS B	RHR B	Total
Nukon Individual Fibers	NA	14%	NA	14%	28%
Nukon & Fire Barrier Small Pieces	NA	2%	NA	6%	8%
Nukon Large Pieces	NA	2%	NA	6%	8%
Nukon Intact Pieces	NA	0%	NA	0%	0%
Fire Barrier Fines & Particulate	NA	14%	NA	14%	28%
Qualified Coatings (IOZ, Epoxy)	NA	14%	NA	14%	28%
Unqualified Epoxy Coatings Particulate	NA	23%	NA	23%	46%
Unqualified IOZ Coatings Particulate	NA	30%	NA	30%	60%
Unqualified Alkyd Coatings Particulate	NA	50%	NA	50%	100%
Latent Dirt/Dust Particulate & Fiber	NA	49%	NA	49%	98%

Table 3.e.6-12: Overall Transport Fractions for a Far Annulus Break, Two Trains Operational, CS Off

Debris Type	CS A	RHR A	CS B	RHR B	Total
Nukon Individual Fibers	NA	14%	NA	14%	28%
Nukon & Fire Barrier Small Pieces	NA	5%	NA	5%	10%
Nukon Large Pieces	NA	6%	NA	6%	12%
Nukon Intact Pieces	NA	0%	NA	0%	0%
Fire Barrier Fines & Particulate	NA	14%	NA	14%	28%
Qualified Coatings (IOZ, Epoxy)	NA	14%	NA	14%	28%
Unqualified Epoxy Coatings Particulate	NA	23%	NA	23%	46%
Unqualified IOZ Coatings Particulate	NA	30%	NA	30%	60%
Unqualified Alkyd Coatings Particulate	NA	50%	NA	50%	100%
Latent Dirt/Dust Particulate & Fiber	NA	49%	NA	49%	98%

Table 3.e.6-13: Overall Transport Fractions for a Near Annulus Break, Two Trains Operational, CS On

Debris Type	CS A	RHR A	CS B	RHR B	Total
Nukon Individual Fibers	21%	29%	21%	29%	100%
Nukon & Fire Barrier Small Pieces	21%	3%	10%	27%	61%
Nukon Large Pieces	3%	4%	3%	4%	14%
Nukon Intact Pieces	0%	0%	0%	0%	0%
Fire Barrier Fines & Particulate	21%	29%	21%	29%	100%
Qualified Coatings (IOZ, Epoxy)	21%	29%	21%	29%	100%
Unqualified Epoxy Coatings Particulate	21%	29%	21%	29%	100%
Unqualified IOZ Coatings Particulate	21%	29%	21%	29%	100%
Unqualified Alkyd Coatings Particulate	21%	29%	21%	29%	100%
Latent Dirt/Dust Particulate & Fiber	21%	28%	21%	28%	98%

Table 3.e.6-14: Overall Transport Fractions for a Far Annulus Break, Two Trains Operational, CS On

Debris Type	CS A	RHR A	CS B	RHR B	Total
Nukon Individual Fibers	21%	29%	21%	29%	100%
Nukon & Fire Barrier Small Pieces	20%	5%	13%	25%	63%
Nukon Large Pieces	3%	4%	3%	4%	14%
Nukon Intact Pieces	0%	0%	0%	0%	0%
Fire Barrier Fines & Particulate	21%	29%	21%	29%	100%
Qualified Coatings (IOZ, Epoxy)	21%	29%	21%	29%	100%
Unqualified Epoxy Coatings Particulate	21%	29%	21%	29%	100%
Unqualified IOZ Coatings Particulate	21%	29%	21%	29%	100%
Unqualified Alkyd Coatings Particulate	21%	29%	21%	29%	100%
Latent Dirt/Dust Particulate & Fiber	21%	28%	21%	28%	98%

Transported Debris Quantities

The transported debris quantities for the most limiting break cases identified in Tables 3.b.4-1 and 3.b.4-2 are shown below and were derived using the debris transport fractions provided in this section for a single train failure case. The debris transport quantities for the four most limiting break cases are listed below in Table 3.e.6-15 as determined in the NARWHAL CFP calculation. The quantities of debris transported for the four most limiting break cases that do not fail any of the strainer or core acceptance criteria are listed in Table 3.e.6-16. Note that the fiber quantity includes fines, small pieces, large pieces, intact pieces, and latent fiber debris.

To calculate the transported quantities of debris presented in the following tables, the blowdown, washdown, pool-fill, and recirculation data (Table 3.e.6-1 through Table 3.e.6-6) are input into NARWHAL. However, the NARWHAL CFP calculation takes into account certain factors that the transport calculation does not consider in order to calculate the time-dependent arrival of debris on the strainer. For example, it takes into account various factors such as the RHR strainer switching over to recirculation before the CS strainer, and the flow split between the strainers. Therefore, the calculation of debris transported to the strainer in the NARWHAL CFP calculation is not a straightforward one. All breaks listed in the tables below occur on the hot leg and activate containment sprays since the break size for each break is greater than 15”. The recirculation transport fractions for Case 7 (LBLOCA in SG Compartment Loop 4, Sprays Activated, Two Trains Operational) from the transport calculation were conservatively input into NARWHAL for the single train failure with containment sprays activated. This was done since there was not a CFD case in the transport calculation that examined one train failure with containment sprays activated, and it is conservative since the turbulence and the velocities in the pool during recirculation for two train operation with containment sprays activated are maximized.

Table 3.e.6-15: Transported Debris for the Four Overall Worst-Case Breaks

Break Location		11201-004-6-RB	11201-001-5-RB	11201-001-3-RB	11201-004-4-RB
Break Size		29"	29"	29"	29"
Break Type		DEGB	DEGB	DEGB	DEGB
Fiber (ft^3)	RHR	639.8	635.4	617.9	605.8
	CS	279.5	277.6	272.7	267.4
Coatings and Latent Particulate (ft^3)	RHR	10.60	10.60	10.57	10.54
	CS	4.58	4.57	4.61	4.60
Calcium Phosphate (lbm)	RHR	73.5	73.5	73.5	73.5
	CS	42.3	42.2	42.1	42.1
Sodium Aluminum Silicate (lbm)	RHR	86.2	86.1	85.8	85.8
	CS	0.5	0.5	0.5	0.5
Fire Barrier Particulate (lbm)	RHR	0.0	18.7	18.5	0.0
	CS	0.0	8.1	8.1	0.0
Fire Barrier Fiber[1] (ft^3)	RHR	0.0	2.6	2.6	0.0
	CS	0.0	1.1	1.1	0.0

[1] The volumes of fire barrier fiber debris shown here were converted from their mass values using the density of low density fiber glass of 2.4 lbm/ft^3. For example, the 2.6 ft^3 of fire barrier fiber debris corresponds to a mass of 6.2 lbm.

Table 3.e.6-16: Transported Debris for the Four Worst-Case Breaks that Do Not Fail the Acceptance Criteria

Break Location		11201-004-4-RB	11201-001-3-RB	11201-003-5-RB	11201-002-5-RB
Break Size		20"	23"	19"	16"
Break Type		Partial	Partial	Partial	Partial
Fiber (ft^3)	RHR	107.2	107.1	107.6	107.5
	CS	46.9	46.7	46.4	46.6
Coatings and Latent Particulate (ft^3)	RHR	10.14	10.16	10.17	10.16
	CS	4.42	4.43	4.39	4.38
Calcium Phosphate (lbm)	RHR	46.4	38.3	34.0	47.8
	CS	16.4	14.1	12.8	16.8
Sodium Aluminum Silicate (lbm)	RHR	54.3	52.5	51.5	54.6
	CS	0.3	0.3	0.3	0.3
Fire Barrier Particulate (lbm)	RHR	0.0	0.0	0.0	0.0
	CS	0.0	0.0	0.0	0.0
Fire Barrier Fiber (ft^3)	RHR	0.0	0.0	0.0	0.0
	CS	0.0	0.0	0.0	0.0

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.

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

   Response to 3.f.1:

   See Figure 3.f.1-1 and Figure 3.f.1-2 for ECCS and CSS schematics, respectively.

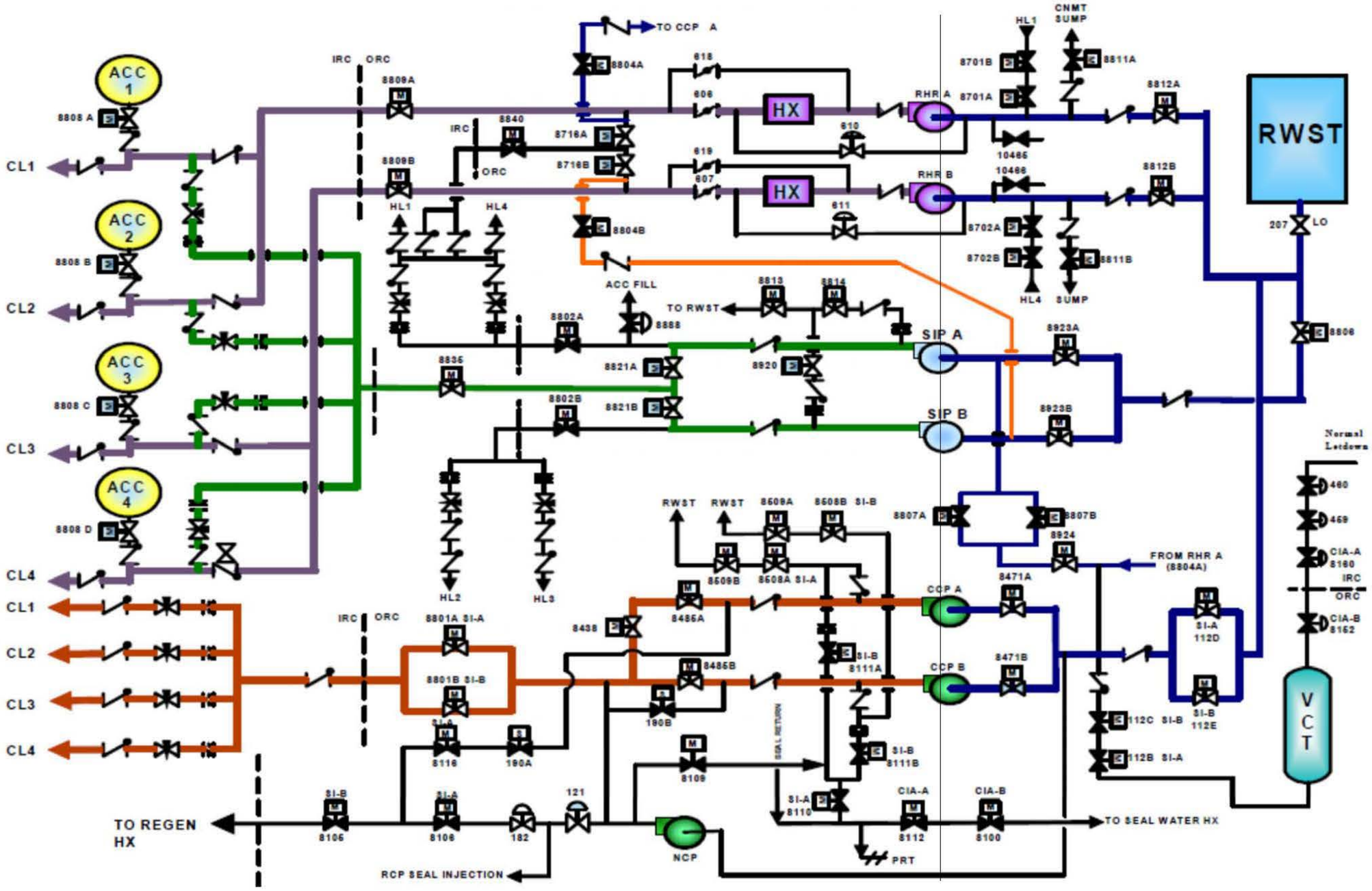

Figure 3.f.1-1 Emergency Core Cooling System Schematic

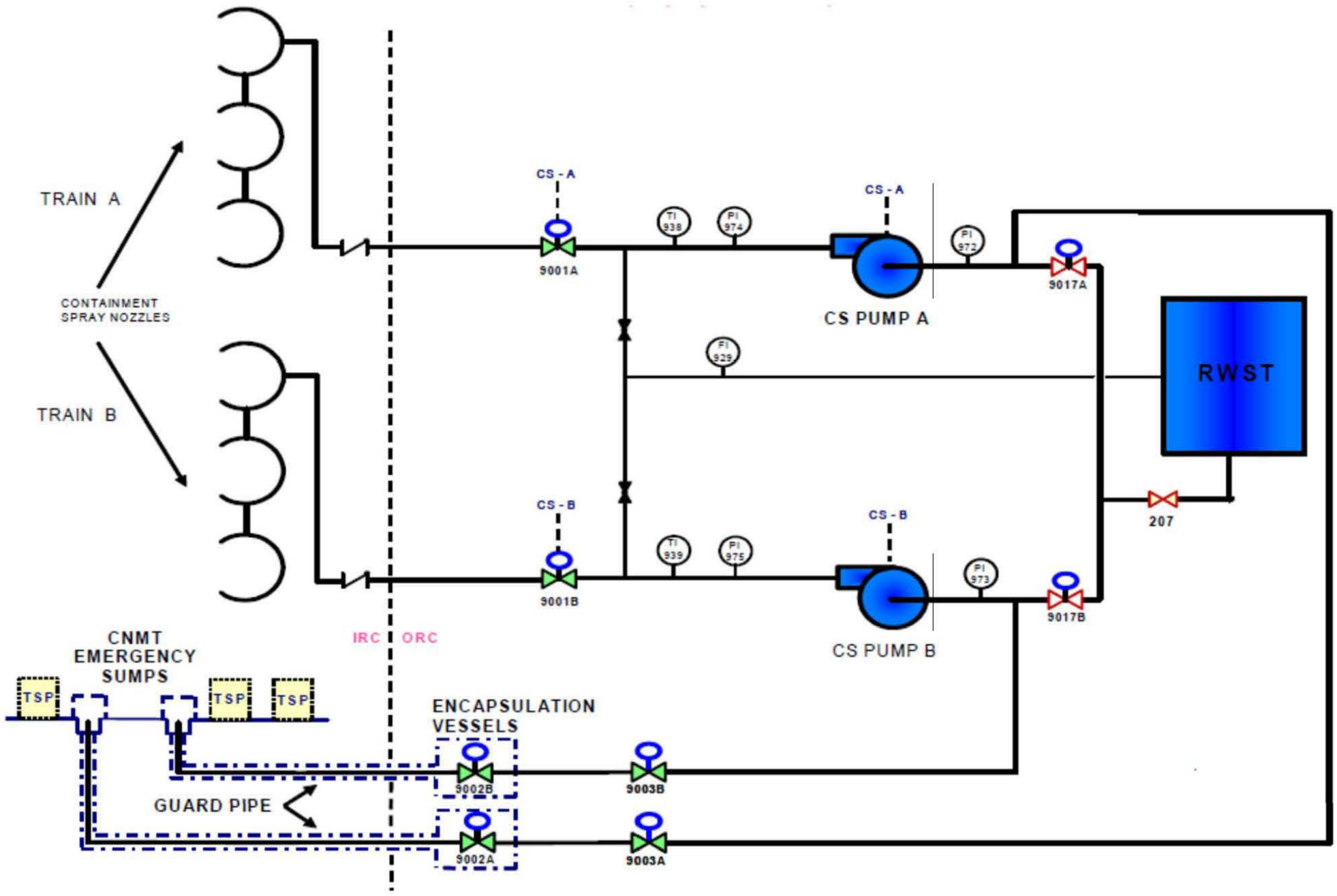

Figure 3.f.1-2 Containment Spray System Schematic

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

Response to 3.f.2:

The sump strainers are fully submerged during recirculation in all cases except reactor nozzle breaks (see Table 3.g.1-3). The highest elevation of the RHR strainer disk is 53-1/4 inches (or 4.438 ft) above the containment floor (see Figure 3.f.2-1). The submergence of the highest elevation point of the RHR strainer is conservatively taken to be its minimum submergence. The height of the 16-disk RHR strainer bounds (i.e., is greater than) the height of the 14-disk CSS strainer, and the minimum submergence of the RHR strainers is always less than the minimum submergence of the CS strainers.

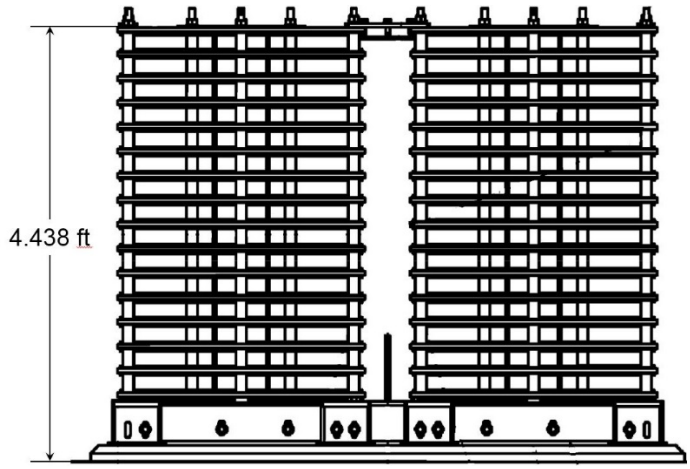

Figure 3.f.2-1: Side View of 16-Disk RHR Strainer

The RHR and CS sump strainers are fully submerged under an LBLOCA that is not a reactor nozzle break. As shown in the Response to 3.g.1, the minimum LBLOCA water level during recirculation for a break that is not a reactor nozzle break is 4.536 ft, and the minimum submergence of the RHR strainer is 0.098 ft.

The RHR sump strainers are also fully submerged for an LBLOCA at a reactor nozzle when CS is not activated. As shown in the Response to 3.g.1, the minimum water level during recirculation for this case is 4.977 ft, and the minimum submergence of the RHR strainer is 0.539 ft.

The RHR and CS sump strainers are fully submerged under an SBLOCA. As shown in the Response to 3.g.1, the minimum SBLOCA water level during recirculation is 5.186 ft. Therefore, the minimum submergence of the RHR strainer is 0.748 ft.

The RHR and CS sump strainers are not fully submerged for an LBLOCA caused by a reactor nozzle break that actuates CS. As shown in the Response to 3.g.1, the

minimum water level for this case (3.054 ft) occurs at the start of recirculation. This pool level is 1.384 ft below the top of the RHR strainer. It should be noted that the water level increases to 4.788 ft when the sump recirculation switchover is complete. This corresponds to a strainer submergence of 0.35 ft.

According to Regulatory Guide (RG) 1.82 (Reference 107), the total strainer head loss of a partially submerged strainer should be less than half the submerged height of the strainer. This ensures the average hydrostatic head of the submerged portion of the strainer will be greater than the head loss through the debris bed. This requirement was used when evaluating partially-submerged strainers for VEGP.

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

Response to 3.f.3:

Summary of Vortex Tests

In 2009, vortex testing was performed on a prototypical strainer module to observe the size, shape, and location of vortices that may develop as both the flow rate through the strainer and the submergence of the strainer module were varied. The vortex tests were performed during the head loss test described in the Response to 3.f.4. Both clean screen and debris laden vortex tests were performed. See Figure 3.f.4-1 for the layout of the test strainer and test tank.

Two vortex tests were conducted at clean strainer conditions, as summarized below.

- The first clean strainer vortex test was started at a submergence level of 3.625 inches and an average approach velocity of 0.0258 ft/s. No vortexing was observed. The average approach velocity was then increased to 0.0355 ft/s. No vortexing was observed. The water level was then reduced to 1.825 inches below the top of the strainer. Again, no vortexing was observed.
- A second clean strainer vortex test was started with a strainer submergence of 4.175 inches and an approach velocity of 0.0306 ft/s. This approach velocity was maintained throughout the duration of this test. The water level in the tank was reduced to just below the top of the strainer. Some pump noise was audible and a small surface swirl was visible in the front right corner of the tank but no persistent vortices were observed.

Debris laden vortex tests were performed at the end of the thin bed and full-load head loss tests after adding all conventional and chemical debris. The test loop setup was the same as that used for the clean screen vortex test and the approach velocity for all debris laden vortex tests was 0.0136 ft/s. It should be noted that, during the course of the thin bed and full-load tests, the tank water level was

maintained at 3.675 ± 0.5 inches above the strainer, and no appreciable vortices were visually observed.

- Water level was slowly reduced at the end of the thin bed test. Air ingestion was not observed until the water level was 0.25 inches below the top of the strainer.
- For the first full debris load test, when the water level was reduced to approximately 3 inches above the strainer, air-entrainment vortices were observed. The vortices became persistent when the water level reached 2.25 inches above the strainer.
- At the end of the second full debris load test, when the water level was reduced to approximately 3 inches above the strainer, air-entrainment vortices were observed. The vortices were not persistent until the water level reached 1.5 inches below the top of the strainer.

Vortexing of Plant Strainers

For reference in the discussion below, the average approach velocity of the 16-disk RHR strainers in the plant is 0.0122 ft/s, and the average approach velocity for the 14-disk CS strainers in the plant is 0.0098 ft/s. These approach velocities are calculated using the strainer flow rates and surface areas shown in Table 3.f.3-1. These approach velocities are well bounded by that used during the clean strainer (0.0355 ft/s) and debris-laden (0.0136 ft/s) vortex tests.

Table 3.f.3-1: Plant Strainer Average Approach Velocities

	RHR Strainer	CS Strainer
Flow Rate (gpm)	3,700	2,600
Surface Area (ft²)	677.6	590
Approach Velocity (ft/s)	0.0122	0.0098

Table 3.g.1-3 presents the minimum strainer submergences for different breaks at different times following the accident. As shown in the table, for the following four cases, the strainer submergence is greater than 3 inches after the start of sump recirculation. The debris-laden vortex test showed that, even with all debris loaded to the strainer, no vortices were observed for submergences greater than 3 inches. It is reasonable to conclude that vortexing will not occur for these four cases because at the start of recirculation[2], the strainer is expected to be clear of debris.

- SBLOCA without CS
- MBLOCA without CS

[2] The start of recirculation for the breaks that actuate containment spray is the time when the RWST level reaches the Low-Low level alarm and the sump suction valves for the RHR pumps open. For the breaks that do not actuate containment spray, start of recirculation is when the switchover of the RHR pump suctions from the RWST to sump is completed at the Empty level alarm.

- LBLOCA without CS
- Reactor nozzle break without CS

For an LBLOCA with CS, the minimum strainer submergence is 0.098 ft (or 1.2 inches) at the start of recirculation when the strainer is still clean. Since the clean strainer vortex showed no vortices even for a partially submerged strainer, it is concluded that vortexing is not a concern for this case at the start of recirculation. Table 3.g.1-3 shows that, for an LBLOCA with CS , the minimum strainer submergence increases to 1.803 ft when the switchover to sump recirculation is completed, which occurs approximately 20 minutes after start of recirculation. Since this submergence is much higher than the 3-inch limit identified by the debris-laden vortex test and only small amount of debris is expected to be transported to the strainer during the period of time it would take to reach a submergence of 3 inches, vortexing will not be an issue at debris-laden conditions for an LBLOCA with CS.

Lastly, for LBLOCA reactor nozzle breaks with CS, Table 3.g.1-3 shows that the strainer is partially submerged at the start of recirculation. After that, the minimum strainer submergence increases over time and is equal to 0.35 ft (or 4.2 inches) when switchover to sump recirculation is completed, which is approximately 20 minutes after the start of recirculation. Similar to the discussion presented above for the LBLOCA with CS, the reactor nozzle break with CS is also bounded by the debris-laden vortex test with respect to formation of vortices.

Based on the discussion above, vortexing is not a concern for any of the analyzed break scenarios with debris loads bounded by the head loss testing. Note that the breaks that have debris loads exceeding those tested are assumed to fail per the debris limit failure criteria, as stated in the Response to 3.f.5.

4. Provide a summary of 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:

Head loss tests were performed 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, debris quantities, and flow rates that were prototypical to VEGP. Different test cases were performed with the thin bed and full debris load protocols, following the 2008 NRC Staff Review Guidance (Reference 111).

The results of the head loss tests provided a matrix of head loss data for various combinations of conventional and chemical debris loads. This matrix was used in the NARWHAL CFP calculation to determine the debris head loss for the debris load associated with each postulated break.

Test Setup

The test strainer assembly consists of seven stacked disks that are duplicates of the disks in the plant strainer. The top surface of the top disk and bottom surface of the bottom disk are solid steel rather than perforated plate. This results in a total of six disks contributing to the effective surface area of the test strainer. The test strainer was placed in a corner of a 6 ft tall by 6 ft wide by 10 ft long test tank on top of a horizontal plenum that simulated the plenum configuration present in the plant. The gaps between the test strainer and the surrounding walls of the test tank simulated the configuration of the plant strainer. See Figure 3.f.4-1 for an illustration of the test strainer and tank.

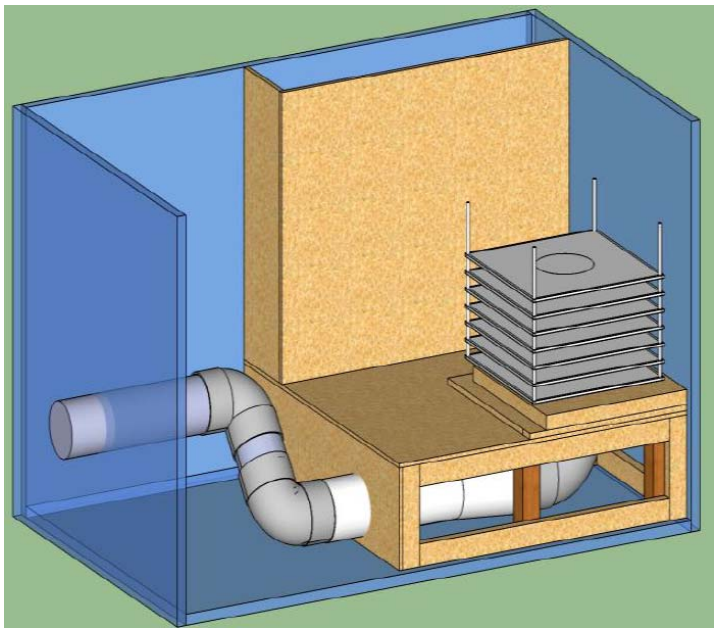

Figure 3.f.4-1: Isometric of Head Loss Test Strainer Assembly inside Test Tank

A schematic piping diagram of the test loop is provided in Figure 3.f.4-2 below. The test loop had a recirculation pump that took suction from the plenum underneath the test strainer and returned the water back into the test tank. The return flow exit into the tank was located such that the turbulence from the flow did not affect the debris bed on the test strainer. A flow element was used to measure the flow rate through the loop. Flow control valves and heating and cooling loops were used to control the test flow rate and water temperature. The test water was maintained at temperatures of at least 80 degrees F throughout the tests.

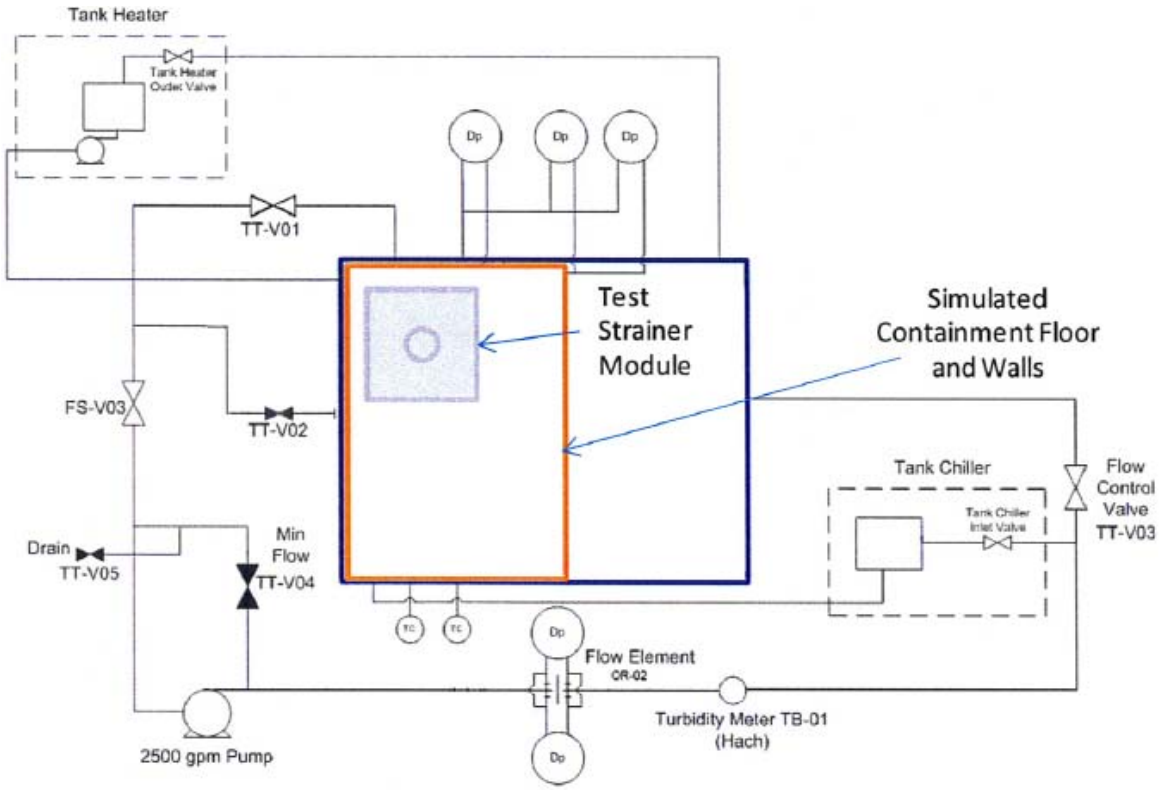

Figure 3.f.4-2: Piping Diagram of Head Loss Test Loop

Test Parameters and Scaling

The test strainer replicates all hydraulic dimensions of the plant strainer except for the number of strainer disks and the number of gaps between disks. The test debris quantities and test flow rate were scaled from plant values based on the ratio between the numbers of gaps between disks of the test strainer to that of the plant strainer. This is analogous to scaling the debris loads and flow rate based on the ratio of the test strainer surface area to the plant strainer surface area.

The surface area of the test strainer was calculated to be 65.57 ft². This surface area of the test strainer was scaled from the RHR strainer surface area based on the number of disks. The post-modification RHR strainer at the plant consists of four stacks of 15.5 disks (the top surface of the top disk for each stack is solid steel, so the top disk counted as ½ a disk). The test strainer has a similar configuration as the plant strainer except the test strainer has only seven disks with the top surface of the top disk and bottom surface of the bottom disk being solid steel. Therefore, the active surface area of the test strainer is equivalent to six disks. This simple scaling method is reasonable because the test strainer disks and spacer rings were fabricated to the same dimensions as the disks and spacer rings installed in the plant.

To scale debris loads from the test to the plant, the debris load is multiplied by the ratio of the plant strainer area to the test strainer area (shown above). The 16-disk RHR strainers have a surface area of 677.6 ft², and the 14-disk CS strainers have a surface area of 590 ft² (see Table 3.f.3-1). This scaling is used when determining the strainer debris limits at the plant scale in Table 3.f.5-1.

During VEGP head loss testing, a nominal test flow rate of 400 gpm was used. Therefore, using the test strainer surface area shown above, the average approach velocity of the test strainer was 0.0136 ft/s, which bounds the approach velocities of the current plant RHR and CS strainers, as shown in Table 3.f.3-1.

In the NARWHAL CFP calculation, the measured strainer head losses from the 2009 testing were corrected from the testing conditions (e.g., strainer approach velocity and water temperature) to plant conditions of interest using the flow sweep data collected during testing. See response to 3.f.10.

Debris Materials and Preparation

The following materials were used as conventional debris for head loss testing: Nukon, Interam E-54A, green silicon carbide powder, and silica sand. The method of preparation prior to introduction to the test tank for each material is discussed below.

Nukon fines were used as surrogate for latent fiber, as recommended in NEI 04-07 and associated NRC SE (References 2 and 3, respectively). Nukon was also used to represent fines and small pieces of LDFG insulation debris.

To prepare Nukon fines, Nukon fiberglass sheets were first shredded and inspected to ensure that the shredded Nukon met the size distribution requirements defined in NUREG/CR-6808 (Reference 39). Afterward, the required quantity was weighed out and boiled for 10 minutes to remove the binder. The boiled fiber was then placed in a bucket of water that was within +/-10 degrees F of the testing water temperature. The fiber was mixed thoroughly with a paint mixer attached to an electric drill until a homogeneous slurry was formed. Prepared fiber fines consisted of Class 1–3 fibers as defined in NUREG/CR-6224 (Reference 29). For small pieces of Nukon, the preparation was similar. However, the prepared small pieces consisted of interwoven strands of fiber, equivalent to or smaller than the Class 4 small fiber clusters, as defined in NUREG/CR-6224.

Interam fire blanket was processed (double-shredded) through a leaf shredder, similar to the manner in which the fiber debris was shredded. After shredding, the Interam was added to buckets with sufficient water to suspend the debris. The buckets were then stirred to wet and suspend the Interam debris in the bucket.

Silica sand was used as a surrogate for latent particulate debris and was prepared by Performance Contracting, Inc. (PCI). The size distribution of the silica sand was

prepared to be consistent with that of latent particulate debris provided in the NRC SE for NEI 04-07 (References 3).

Green silicon carbide powder with a density of 199.25 lbm/ft^3 was used as a surrogate for both qualified and unqualified coatings. Per NEI 04-07 and the associated NRC SE (References 2 and 3, respectively), the coatings particulate debris was assumed to be 10 µm diameter spheres. The majority of the silicon carbide surrogate had a size distribution range from 4 µm to 20 µm, a median size of 10.25 µm, and a mean size of 10.46 µm.

The required amount of silicon carbide and silica sand was weighed out and placed in a bucket of water with a temperature within +/-10 degrees F of the testing temperature. The particulate was mixed with water using an electric paint stirrer until no agglomeration or clumping was observed. Before introducing the particulate to the test tank, all particulate batches were mixed once again with an electric paint stirrer to create a thin slurry.

Two types of chemical debris surrogates were used for the head loss testing: sodium aluminum silicate (SAS) and calcium phosphate. The chemical debris was prepared in accordance with WCAP-16530-NP-A (Reference 73). The 1-hour settling volume for each batch of chemical precipitates was determined at the time the batch was produced. The chemical precipitate settling time was also measured within 24 hours from the time the surrogate was to be used. Chemical precipitates that did not meet the settling requirements were discarded and not used for testing.

Debris Introduction

Debris was added at the side of the tank adjacent to the return flow exit into the tank and away from the simulated sump floor and walls. This allowed for an even and representative debris accumulation on the test strainer. A sparger system was installed on the return flow exit in the tank away from the strainer to aid in suspension of debris. Two mechanical mixers were also installed at the tank corners opposite from the test strainer.

During the thin-bed test, all of the debris was added over the mixers while the recirculation pump was running. Additional manual agitation was applied to help the Interam arrive near the test strainer. During the later full-load tests, the Interam was introduced to the floor area immediately adjacent to the test strainer to aid the transport. The agitation from the sparger, mechanical mixers, and manual agitation helped keep the debris suspended in the water, and near-field settling was not credited. The debris bed formed on the strainer was not affected by debris addition or agitation in the tank.

After conventional debris introduction was completed for each test, chemical precipitate debris was added to the test tank. Calcium phosphate was first

introduced in batches, and the head loss was allowed to stabilize between batches. SAS batches were added last.

Head Loss Test Cases and Results

Three head loss tests were performed for VEGP: one thin bed test and two full debris load tests. The two full-load tests targeted the same flow conditions and debris loads.

For the thin bed test (VOG-1-TB), all of the particulate debris (including Interam, coatings surrogate, and latent particulate surrogate) was introduced to the test tank at the beginning of the test. Once all of the particulate debris was added and allowed to circulate through the test loop, fine fiber batches were incrementally added in batch sizes equivalent to a 1/8-inch theoretical uniform debris bed thickness. Only fiber fines were used for the thin bed test. Thin-bed formation was observed visually, via head loss and turbidity measurement. The head loss was allowed to stabilize (less than or equal to 1 percent change over a 1-hour period) after the final batch of fiber fines was added. Once the head loss stabilized, chemical precipitates were incrementally added. The debris batch composition and size for the thin bed test are summarized in Table 3.f.4-1.

Table 3.f.4-1: Debris Batches Added for the Thin Bed Head Loss Test

Batch	Test Nukon Quantity (lbm)		Test Interam Quantity (lbm)	Test Silicon Carbide Quantity (lbm)	Test Dirt/Dust Quantity (lbm)	Test Calcium Phosphate Quantity (L)	Test SAS Quantity (L)
	Fines	Smalls					
VOG-1.2-TB-P	0	0	29.15	358.42	5.3	0	0
VOG-1.3-TB-F1	1.49	0	0	0	0	0	0
VOG-1.4-TB-F2	1.49	0	0	0	0	0	0
VOG-1.5-TB-F3	1.49	0	0	0	0	0	0
VOG-1.6-TB-F4	1.49	0	0	0	0	0	0
VOG-1.7-TB-F5	1.49	0	0	0	0	0	0
VOG-1.10-TB-CP1	0	0	0	0	0	160.24	0
VOG-1.11-TB-CP2	0	0	0	0	0	160.24	0
VOG-1.12-TB-CP3	0	0	0	0	0	160.24	0
VOG-1.13-TB-NAS1	0	0	0	0	0	0	122.86
VOG-1.14-TB-NAS2	0	0	0	0	0	0	122.86
VOG-1.15-TB-NAS3	0	0	0	0	0	0	122.86
Total	7.45	0	29.15	358.42	5.3	480.72	368.58

Table 3.f.4-2 summarizes the stabilized head losses after adding each batch of debris during the thin bed test.

Table 3.f.4-2: Thin Bed Head Loss Test Results

Batch	Stabilized Head Loss ($ft\text{-}H_2O$)
VOG-1.2-TB-P	0.171
VOG-1.3-TB-F1	0.226
VOG-1.4-TB-F2	0.262
VOG-1.5-TB-F3	0.308
VOG-1.6-TB-F4	0.368
VOG-1.7-TB-F5	0.625
VOG-1.10-TB-CP1	1.02
VOG-1.11-TB-CP2	1.54
VOG-1.12-TB-CP3	1.65
VOG-1.13-TB-NAS1	2.12
VOG-1.14-TB-NAS2	2.27
VOG-1.15-TB-NAS3	2.56

For the full-load tests (VOG-2-FL-B and VOG-2-FL-B2), the particulate and fiber debris was introduced simultaneously in equal batches maintaining the same fiber to particulate ratio until the full conventional debris load was reached. This debris addition sequence resulted in a homogenous debris bed accumulation. Each debris batch consisted of Nukon fiber fines and small pieces, and particulate debris. After the final batch of conventional debris was introduced, the head loss was allowed to stabilize with a less than 1 percent change over a 1-hour period. Chemical precipitates were then incrementally added. The debris batch composition and size for the full-load tests are summarized in Table 3.f.4-3.

Table 3.f.4-3: Debris Batches Added for the Full-load Head Loss Tests

Batch	Test Nukon Quantity (lbm)		Test Interam Quantity (lbm)	Test Silicon Carbide Quantity (lbm)	Test Dirt/ Dust Quantity (lbm)	Test Calcium Phosphate Quantity (L)	Test SAS Quantity (L)
	Fines	Smalls					
VOG-2.2-FL-F1	4.58	2.04	7.29	89.61	1.32	0	0
VOG-2.3-FL-F2	4.58	2.04	7.29	89.61	1.32	0	0
VOG-2.4-FL-F3	4.58	2.04	7.29	89.61	1.32	0	0
VOG-2.5-FL-F4	4.58	2.04	7.29	89.61	1.32	0	0
VOG-2.6-FL-CP1	0	0	0	0	0	160.24	0
VOG-2.7-FL-CP2	0	0	0	0	0	160.24	0
VOG-2.8-FL-CP3	0	0	0	0	0	160.24	0
VOG-2.9-FL-NAS1	0	0	0	0	0	0	122.86
VOG-2.10-FL-NAS2	0	0	0	0	0	0	122.86
VOG-2.11-FL-NAS3	0	0	0	0	0	0	122.86
Total	18.32	8.16	29.16	358.44	5.28	480.72	368.58

As discussed above, two full-load head loss tests were performed. However, the first full-load test VOG-2-FL-B reported higher head losses. Table 3.f.4-4 summarizes the stabilized head losses after adding each debris batch during test VOG-2-FL-B. The measured head losses from the full-load test are much higher than the thin-bed test.

Table 3.f.4-4: Bounding Full-load Head Loss Test Results

Batch	Stabilized Head Loss (ft-H_2O)
VOG-2.2-FL-F1	0.276
VOG-2.3-FL-F2	1.06
VOG-2.4-FL-F3	2.42
VOG-2.5-FL-F4	5.46
VOG-2.6-FL-CP1	5.29
VOG-2.7-FL-CP2	6.22
VOG-2.8-FL-CP3	6.57
VOG-2.9-FL-NAS1	7.16
VOG-2.10-FL-NAS2	7.24
VOG-2.11-FL-NAS3	11.81

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:

The 2009 head loss test program evaluated debris loads based on the Nukon debris quantities calculated using a 7D ZOI from WCAP-16710-P, which was later rejected by the NRC. The current Nukon debris quantities were calculated with a 17D ZOI in BADGER, and this results in fiber debris loads greater than that tested. Therefore, the debris quantities used in the 2009 test program do not bound what is predicted for some breaks.

Debris limits were implemented in the NARWHAL analysis for each strainer in operation for a given scenario. The debris limits were applied to individual strainers, not to the total amount of transported debris. If the debris on the strainer exceeded any of the debris limits, a failure was recorded for that postulated break. The debris limits are based on the maximum quantity of conventional and chemical debris that was tested in 2009. Table 3.f.5-1 shows the debris limits for each of the debris types at the test scale and plant scale for one RHR strainer.

A combined limit was shown for the coatings and latent particulate, which was calculated by dividing the total tested mass (358.44 lbm for silicon carbide and 5.28 lbm for dirt/dust, see Table 3.f.4-3) by the density of silicon carbide (199.25 lbm/ft^3). This approach resulted in a conservatively low debris limit since the density of dirt/dust is lower (168.6 lbm/ft^3). The plant coatings debris loads were compared with the tested limits on a volume basis rather than a mass basis since the surrogate material used for coatings during testing has a different density from the actual coatings debris. The calcium phosphate debris and SAS debris loads were converted from volume to mass using concentrations of 5 g/L and 11 g/L, respectively. The debris limits at the plant scale shown in Table 3.f.5-1 were determined by multiplying the debris limits at the test scale by the ratio of the RHR strainer area (677.6 ft^2) to the test strainer area (65.57 ft^2), as discussed in the response to 3.f.4.

Table 3.f.5-1: Debris Limit Failure Criteria

Debris Type	Debris Limit at Test Scale	Debris Limit at Plant Scale for One RHR Strainer
Fiber	11.03 ft^3	113.98 ft^3
Coatings and Latent Particulate	1.825 ft^3	18.86 ft^3
Fire Barrier (particulate)[3]	20.41 lbm	210.92 lbm
Fire Barrier (fiber)	3.65 ft^3	37.72 ft^3
Calcium Phosphate	5.30 lbm	54.77 lbm
Sodium Aluminum Silicate	8.94 lbm	92.39 lbm

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 (or high particulate to fiber ratio) debris bed formed by a thin layer of fibrous debris that can effectively filter particulate debris. The 2009 VEGP head loss testing included a test for thin-bed effects. During this test, the full particulate load was added into the test tank first, followed by fiber fines in batches equivalent to a 1/8-inch theoretical uniform bed thickness. This batching schedule allowed the formation of a debris bed with a high particulate to fiber ratio. As a result, any thin-bed effects, should they occur, would be captured by the measured head losses.

   As discussed in Section 3.f.10, head loss testing results from the thin bed test are used by the NARWHAL CFP calculation for postulated debris loads less than or equal to 3.1 ft^3 of fiber at test scale (corresponding to a theoretical uniform bed thickness of 0.57 inches). See Section 3.f.10 for additional discussion of how the total head loss is determined.

7. Provide the basis for strainer design maximum head loss.

   Response to 3.f.7:

   There are several failure criteria evaluated by NARWHAL based on the head loss across the strainer: strainer structural margin, strainer debris limits, strainer partial submergence limits, void fraction limits, flashing, and pump NPSH. A postulated break that exceeds one or more of these criteria for the RHR strainers/pumps is considered to be a failure of the ECCS system. Each of the failure criteria was

[3] Fire barrier debris was treated as 70% particulate and 30% fiber.

evaluated at each time step within the NARWHAL model to determine if an ECCS system failure would occur.

Strainer Structural Margin Limits

The strainer structural margin for each strainer is 24.0 ft. The head loss across each of the RHR and CS strainers due to conventional and chemical debris loading is compared to this value to ensure that the structural margin was not exceeded. See Section 3.k.1 for additional information on how the structural margin was determined.

Debris Limits

See Section 3.f.5 for discussion of conventional and chemical debris limits used in NARWHAL.

Unsubmerged Strainer Limits

If the strainers are partially submerged, the NARWHAL CFP calculation assumed that the strainer would fail if the head loss across the debris bed and strainer is greater than half of the submerged strainer height per RG 1.82 (Reference 25). Note that the pump NPSH and strainer structural limits are also applicable for a partially submerged strainer. The NARWHAL CFP calculation tracks time-dependent accumulation of debris on the strainer. When the strainer is partially submerged, the evaluation only credits the active (i.e., submerged) portion of the strainers for flow and debris accumulation.

Void Fraction Limits

A pump failure due to degasification was recorded if the steady state gas void fraction at the pump is greater than 2 percent by volume. Note that bubble compression was not credited.

Flashing Failure Limits

A flashing failure was recorded for a postulated break if, at any time during sump recirculation, the pressure downstream of the strainer was lower than the vapor pressure at the sump temperature. The pressure downstream of the strainer was calculated by NARWHAL based on the strainer submergence, containment pressure and head loss across the strainer. Note that a small increase in containment pressure was credited in the flashing analysis, see Section 3.f.14 for additional information.

Pump NPSH Limits

A pump failure was recorded if the head loss across the strainer exceeded the clean strainer NPSH margin (i.e., margin that is available for debris laden head loss). It should be noted that because the safety injection (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 were calculated in NARWHAL. See Section 3.g.16 for details of the NPSH margin used in NARWHAL.

8. Describe significant margins and conservatisms used in head loss and vortexing calculations.

Response to 3.f.8:

Vortexing Testing

Testing was conducted to determine if vortexing is expected to occur. As discussed in the Response to 3.f.3, the vortex tests were performed at both clean strainer and debris-laden conditions.

All vortex tests used strainer approach velocities higher than those expected for the plant strainer (0.0122 ft/s and 0.0098 ft/s for the RHR and CS strainers, respectively, see Table 3.f.3-1). The clean strainer vortex tests used strainer approach velocities up to 0.0258 – 0.0355 ft/s. For the debris laden vortex tests, a strainer approach velocity of 0.0136 ft/s was used.

As shown in the response to 3.f.3, plant strainer minimum submergence at the start of the recirculation is compared with the submergence limit established by the debris-laden vortex tests. It should be noted that these tests were performed after all conventional and chemical debris has been added to the test tank. This is conservatively bounding because, at the start of recirculation, the strainer is expected to be clear of debris.

Strainer Head Loss

The quantity of latent debris used to determine the strainer head loss is 200 lbm, but the actual amount of latent debris documented for the plant is only 60 lbm. Similarly, the amount of miscellaneous debris used in the analysis is 50 ft^2, but, as stated in Response to 3.b.5, the amount of miscellaneous debris was conservatively assumed in the debris generation calculation to be 4 ft^2, which bounds the 2 ft^2 identified in containment during the walkdown.

When correcting the debris head loss from the test conditions (e.g., water temperature and strainer approach velocity) to plant conditions, head loss coefficients from both the full debris load test and thin bed test were applied in the analysis. The coefficients that result in the higher head loss are used to calculate

the debris head loss. Additionally, specific flow sweep data points were excluded from the analysis if the use of the points would result in lower corrected head loss values. See section 3.f.10 for additional discussion.

Finally, the rule-based approach described in Section 3.f.10 conservatively applies the maximum head loss result from the thin bed test for all postulated breaks with minimal fiber debris generation. Similarly, all postulated breaks that result in debris at the strainer greater than the thin bed test and less than or equal to the maximum amount of debris tested in the full load test have the maximum head loss from the full load test.

9. Provide a summary of 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 to be 4.40 in-H_2O at the RHR pump runout flow rate of 4,500 gpm. This flow rate conservatively bounds the flow rate of the RHR and CS strainers. Note that a value of 4.5 inches (0.375 ft) was used in the NARWHAL calculation.

The clean strainer head loss was calculated by modeling flow through one 18-disk strainer stack per the following steps:

- The cross-sectional areas of flow for various parts of the strainer stack were calculated from the physical dimensions of the strainer components.
- Loss coefficients were calculated for the flow paths through the strainer components based on flow path geometry. Loss coefficients were determined for the perforated plate, wire cloth, and converging cross flow from the flow through the disk with flow through the core tube.
- A system of mass balance and energy balance equations were iteratively solved to calculate the flow and resulting pressure drop for each disk in the stack.
- The difference between the initial pressure and the pressure of the fluid before entering the plenum was calculated and reported as the head loss through the strainer stack.
- The head loss inside the sump pit below the strainer stacks was also included. For each pit, the flow through the four strainer stacks combines inside the space below the strainer assembly and upstream of the ECCS or CS suction pipe openings. Head losses due to flow exiting the strainer stacks and turning inside the pit were accounted for based on conservative loss coefficients and velocities.

Several assumptions were used when applying the above methodology to determine the clean strainer head loss. The temperature of water was assumed to be 120 degrees F, the strainer was assumed to be fully submerged (i.e., flow is through all disks), and head loss along the outside face of the disks, elevation head, and

coupling effects were ignored. Additionally, friction loss between and within the strainer disks was ignored as it is negligible compared to the screen and perforated plate losses. The surface friction loss in the strainer core was also ignored because the radial influx of water from the strainers disks and spacers prevents significant flow and friction loss along the surface of the strainer core. The flow turn inside the pit after exiting a strainer stack was conservatively assumed to be confined in a 90° steel mitre bend.

It is acceptable to use the clean screen head loss calculated for an 18-disk strainer as the clean screen head loss for a 16-disk strainer for the following reasons:

- The flow distribution in the clean screen head loss calculation shows that the vast majority of flow is through the first six disks closest to the pump suction. In fact, only 0.1 percent of the total flow comes from the top two disks.
- A flow rate of 3,700 gpm through the RHR strainer is used in the NARWHAL analysis rather than the pump runout flow rate of 4,500 gpm used in the clean screen head loss calculation. The clean screen head loss at 3,700 gpm is less than the clean screen head loss at 4,500 gpm.

10. Provide a summary of methodology, assumptions, bases for the assumptions, and results for the debris head loss analysis.

Response to 3.f.10:

The total head loss across the strainer is the sum of the clean strainer head loss, the conventional debris head loss, and the chemical head loss. The conventional and chemical head loss values were based on VEGP-specific head loss test results that were corrected from the test conditions (i.e., strainer approach velocity and water temperature) to plant conditions. Additionally, the test results were extrapolated to the end of the 30-day strainer mission time.

Debris Head Loss Correction

A head loss correction factor (based on the strainer approach velocity and pool temperature) was implemented into NARWHAL to scale the measured head losses from test conditions to plant conditions. For each time step for which conventional and chemical head losses are evaluated, the head loss value is corrected based on the plant flow rate through the strainer and the pool temperature. The correction was performed based on the debris bed characteristics obtained through flow sweeps conducted during head loss tests. For the 2009 test program, flow sweeps were performed at the end of the thin bed and full-load tests. To account for the uncertainty in the flow sweeps for each test, the resulting correction parameters from both tests were applied at each time step, and the maximum resulting head loss was returned.

Table 3.f.10-1 and Table 3.f.10-2 show the flow sweep data for the thin bed and full load tests, respectively. Note that the thin bed test flow sweep was conducted at a water temperature of 86 degrees F, and that the full load test was conducted at a water temperature of approximately 93 degrees F. The flow sweep data was used to determine the correction parameters, which were then used to scale measured head losses. The volumetric flow rates documented in Tables 3.f.10-1 and 3.f.10-2 were converted to approach velocities using the test strainer area of 65.57 ft².

Table 3.f.10-1: Thin Bed Test Flow Sweep Data

Flow Rate (gpm)	Head Loss (ft-H_2O)	Approach Velocity (ft/s)
403	2.6	0.0137
371	2.37	0.0126
200	1.31	0.0068
436	2.87	0.0148
395	2.56	0.0134

Table 3.f.10-2: Full Load Test Flow Sweep Data

Flow Rate (gpm)	Head Loss (ft-H_2O)	Approach Velocity (ft/s)
393	11.81	0.0134
369	10.61	0.0125
200	3.47	0.0068
446	11.57	0.0152
395	8.81	0.0134

Figure 3.f.10-1 and Figure 3.f.10-2 show the debris head loss as a function of approach velocity for the thin bed and full load test flow sweeps, respectively. In addition, the data was fit with a second-order polynomial in the following form:

$$\text{Head Loss} = K_1 v^2 + K_2 v$$

Here, K_1 and K_2 are the fitting coefficients (as shown in the figures), and v is the strainer approach velocity in ft/s. Note that the polynomial was forced through the origin, because the head loss would be zero at an approach velocity of zero. Also, note that for the full load test, the test data points represented by the two orange points were excluded from the curve fit. This produces a conservative curve fit because it results in a higher predicted head loss for lower approach velocities.

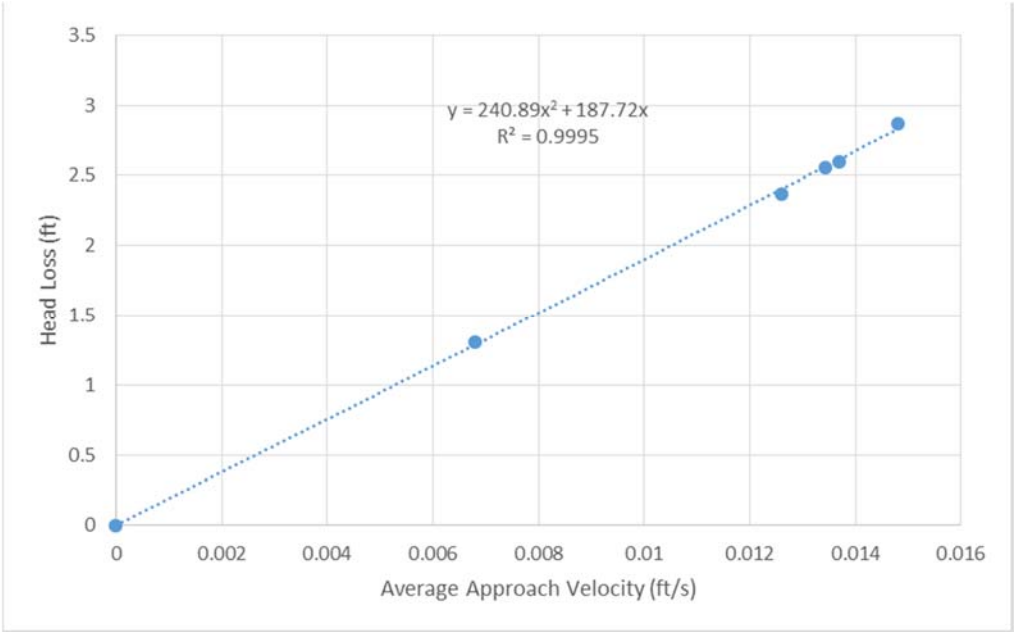

Figure 3.f.10-1: Head Loss Fitting Coefficients for Thin Bed Test

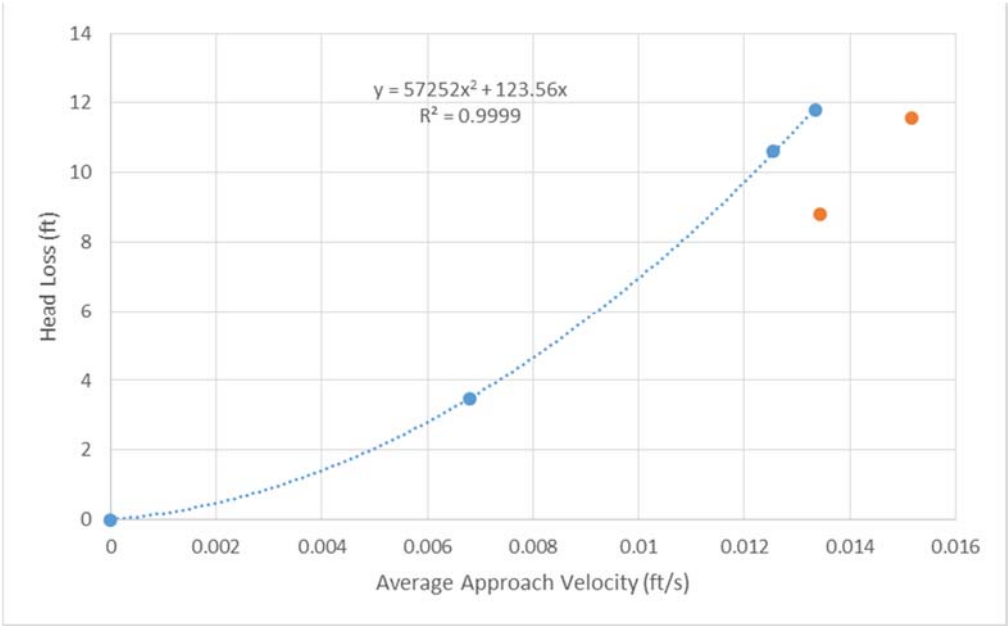

Figure 3.f.10-2: Head Loss Fitting Coefficients for Full Load Test

With these curves defined, the head loss fitting parameter can be calculated. NARWHAL accepts *a*, *b*, and flow sweep head loss as inputs into the following correction equation.

$$X_{HL} = \frac{a \times \mu \times v_{strainer} + b \times \rho \times v_{strainer}^2}{\Delta P_{HL}}$$

Nomenclature:

X_{HL}	=	Head loss correction factor
$v_{strainer}$	=	Approach velocity of the strainer at plant condition, ft/s
a	=	Coefficient determined from flow sweep curve fitting parameter K_2 and water viscosity at test temperature, K_2/μ
b	=	Coefficient determined from flow sweep curve fitting parameter K_1 and water density at test temperature, K_1/ρ
μ	=	Viscosity of water at plant condition, lbm/(ft-s)
ρ	=	Density of water at plant condition, lbm/ft³
ΔP_{HL}	=	Head loss at the test approach velocity and temperature, and the flow sweep debris load, ft-H_2O

The thin bed head loss test was conducted at a temperature of 86 degrees F, which corresponds to a water density of 62.16 lbm/ft³ and a water viscosity of 0.000536 lbm/(ft-s). The head loss correction coefficients were calculated as follows for the thin bed test flow sweep data:

$$a = \frac{K_2}{\mu} = \frac{187.72\ \mathrm{s}}{0.000536 \frac{\mathrm{lbm}}{\mathrm{ft} * \mathrm{s}}} = 350{,}223.9\ \frac{\mathrm{s}^2 \cdot \mathrm{ft}}{\mathrm{lbm}}$$

$$b = \frac{K_1}{\rho} = \frac{240.89 \frac{\mathrm{s}^2}{\mathrm{ft}}}{62.16 \frac{\mathrm{lbm}}{\mathrm{ft}^3}} = 3.875 \frac{\mathrm{s}^2 \cdot \mathrm{ft}^2}{\mathrm{lbm}}$$

$$\Delta P_{HL} = 2.6\ \mathrm{ft}$$

The full load head loss test was conducted at a temperature of 93 degrees F, which corresponds to a water density of 62.08 lbm/ft³ and a water viscosity of 0.000494 lbm/(ft-s). The head loss correction coefficients are calculated as follows for the full load test flow sweep data:

$$a = \frac{K_2}{\mu} = \frac{123.56\ \mathrm{s}}{0.000494 \frac{\mathrm{lbm}}{\mathrm{ft} * \mathrm{s}}} = 250{,}121.5 \frac{\mathrm{s}^2 \cdot \mathrm{ft}}{\mathrm{lbm}}$$

$$b = \frac{K_1}{\rho} = \frac{57{,}252\,\frac{s^2}{ft}}{62.08\,\frac{lbm}{ft^3}} = 922.229\,\frac{s^2 \cdot ft^2}{lbm}$$

$$\Delta P_{HL} = 11.81\ ft$$

By substituting the values of a, b, and ΔP_{HL} into the formula above, a head loss correction factor X_{HL} can be calculated for each set of flow sweep data. As stated earlier, the two correction factors calculated from the thin-bed and full debris load flow sweeps were multiplied by the total debris head loss at each time step and the higher resulting head loss was returned. Note that the total debris head loss is the sum of the conventional debris head loss, the chemical head loss, and the extrapolation constant (where applicable). The clean screen head loss is not corrected to different strainer approach velocities or pool temperatures.

Debris Head Loss Extrapolation

To address extrapolation of the head loss tests to the end of 30-day mission time, a head loss extrapolation constant was applied. The extrapolation constant was determined using the raw test data from the end of the head loss test. The raw test data was smoothed using a locally weighted least-squares method. The first order derivative of the smoothed data was reviewed to ensure that the slope of the data was trending towards zero, suggesting that the head loss profile was stabilizing.

A natural logarithmic function was fitted to the smoothed data and the function was shifted upwards, which bounded any peaks observed after the last chemical addition. The curve fit also had a similar slope (i.e., rate of increase in head loss over time) as the recorded head losses at the end of the test. This head loss extrapolation was performed for the thin-bed test and both full debris load tests. Figure 3.f.10-3 shows the recorded head losses and the logarithmic curve fit before and after the adjustment for the first full debris load test. Note that the test data used for the extrapolation analysis was recorded at least 12 hours after adding the last batch of chemical debris.

The extrapolation constant was calculated for all three tests. Since the constant of the first full debris load test (3.89 ft-H_2O) is larger than the other tests, this value was conservatively used for all NARWHAL analysis. Note that this extrapolation constant is at the testing condition and is corrected to plant conditions using the same approach as the debris head losses (see discussions earlier in this response). The extrapolation constant was applied at 7.5 hours after the accident. As shown in Table 3.g.16-1, this extrapolation constant was applied to the calcium phosphate debris head loss before the formation of aluminum precipitate (see head losses at sump temperatures of 153°F and 140°F).

Before 7.5 hours, no extrapolations are necessary because the full debris load test, from which the head loss results were obtained, lasted for over 48 hours from the start of introduction of conventional debris to the end of head loss stabilization after adding the calcium phosphate debris (see Figures 3.o.2.17-1 through 3.o.2.17-3). Even the test duration for the addition of calcium phosphate debris and head loss stabilization alone is close to 24 hours (see Figures 3.o.2.17-2 and 3.o .2.17-3). Therefore, any potential head loss increases before 7.5 hours following the accident were captured by the recorded head loss data from testing. As a result, no additional extrapolation is necessary.

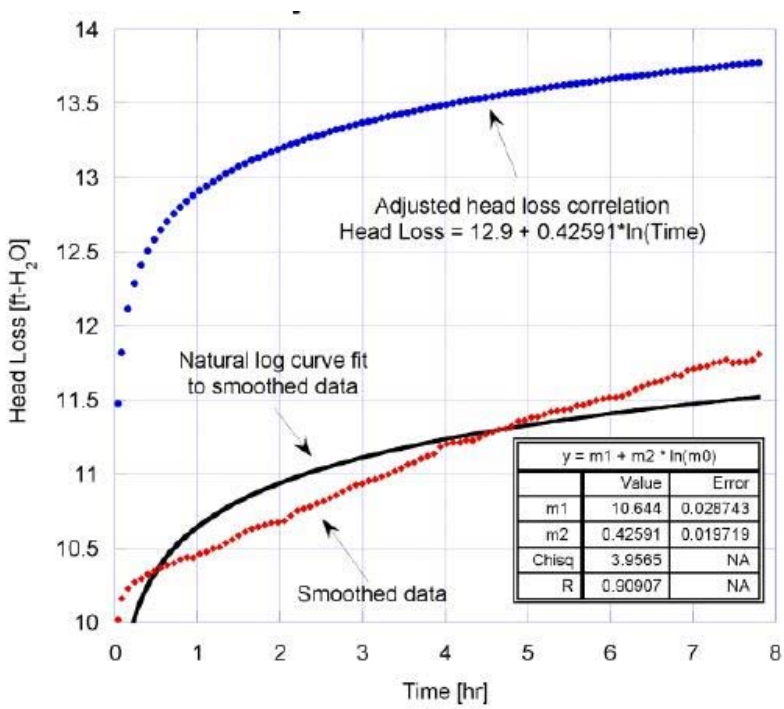

Figure 3.f.10-3: Logarithmic Curve Fit of Bounded Test Data Used for 30-Day Head Loss Extrapolation

Clean Strainer Head Loss

The clean strainer head loss varies as a function of strainer approach velocity. However, the bounding clean strainer head loss of 4.4 inches at 4,500 gpm was rounded up to 4.5 inches (0.375 ft) and was used for all cases in NARWHAL. No flow or temperature correction was applied to the clean strainer head loss.

Conventional Debris Head Loss

NARWHAL uses a rule-based approach to calculate head loss based on the results of head loss testing. As shown in Table 3.f.10-3, if the fiber debris load at the strainer is less than the tested quantity from the thin bed test (3.1 ft^3 at test scale, corresponding to a theoretical uniform bed thickness of 0.57 inches), the maximum thin bed conventional debris head loss was returned. If the quantity was greater

than what was tested in the thin bed test, the conventional head loss of the full-load test was returned (5.46 ft-H_2O) since this head loss is higher than that of the confirmatory test, as shown in Table 3.o.2-1. For a given time step, NARWHAL scaled the plant debris load to the test scale based on the active plant strainer surface area before determining the conventional debris head loss.

Table 3.f.10-3: Conventional Head Loss Values

Fiber Debris Load at Test Scale (ft^3)	Head Loss (ft-H_2O)
≤3.1*	0.625
>3.1*	5.46

* The 3.1 ft^3 of fiber at test scale corresponds to 32.04 ft^3 at the plant scale for one RHR strainer.

Chemical Head Loss

A conservative chemical head loss model was implemented in the NARWHAL CFP calculation. The head loss effects of calcium phosphate and SAS were each analyzed separately from the 2009 full-load head loss test. Note that the confirmatory head loss test resulted in greater head loss increases after adding the calcium phosphate and SAS than the full-load test. However, the overall head losses of the confirmatory test are much lower, as shown in Table 3.o.2-1 and, therefore, are not used. Table 3.f.10-4 shows the head loss applied to each strainer once precipitate starts to accumulate on the strainer. The variability in strainer head loss was shown to have little impact on ΔCDF, as shown in Tables 3-11 and 3-12 of Enclosure 3.

Table 3.f.10-4: Chemical Head Loss Values

Chemical Precipitate	Quantity (lbm)	Head Loss (ft-H_2O)
Calcium Phosphate	> 0	1.11
Sodium Aluminum Silicate	> 0	5.24

Note that chemical head loss is not applied until a 0.45-inch thick theoretical uniform fiber debris bed has formed on the strainer. This approach is reasonable because, for fiber quantities smaller than this, large areas of open screen are present on the strainer. This is supported by the 2009 thin bed head loss test data. During the thin bed test, particulate debris was added to the test tank before fiber debris was batched in. Figure 3.f.10-4 shows a negligible increase in debris head loss for fiber loads up to and including 5.96 lbm. This fiber load corresponds to a uniform fiber bed thickness of 0.45 inches.

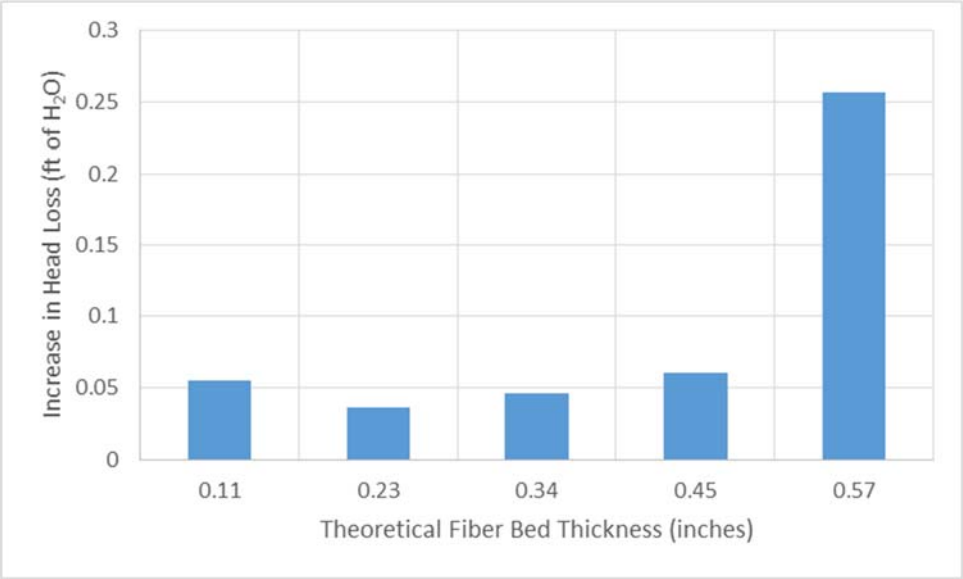

Figure 3.f.10-4: Increase in Head Loss as a Function of Fiber Bed Thickness

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 NPSH margin were applied to address potential inability to pass the required flow through the strainer.

Response to 3.f.11:

As shown in Table 3.g.1-3, for some of the postulated breaks (specifically, reactor cavity breaks with CS actuated), the strainers could be partially submerged at the start of recirculation for a short period but become fully submerged before the switchover to recirculation is completed. When strainers are not fully submerged, the unsubmerged strainer head loss failure criterion discussed in Section 3.f.7 was used.

The NARWHAL CFP calculation showed that the head loss during the time when the RHR strainer is partially submerged does not challenge the failure criterion which states that head loss cannot be greater than half of the submerged strainer height for any of the break scenarios. The calculation evaluated the most limiting break in terms of fiber at the RHR strainer during the time that the RHR strainer is partially submerged. The DEGB at Weld 11201-001-1-RB (located in the reactor cavity on the hot leg) with a single train failure configuration resulted in the most amount of fiber on the RHRA strainer for all breaks that have partially submerged strainers during recirculation. For this break, the strainer is partially submerged for a total of 11 minutes. The amount of debris accumulated during this period of time did not challenge the failure criterion for partially submerged strainers.

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 head loss testing. Sufficient turbulence was provided in the tank to ensure that all debris had an opportunity to collect on the surfaces of the test strainer, while not disturbing the debris bed formation. Additionally, manual stirs were applied as necessary to prevent debris from settling after introduction.

Two mechanical stirrers were required to suspend the debris due to the strainer configuration and flow rate, one in the pit below the strainer, and another within the area underneath the strainer, bounded by the simulated containment walls and floor. A sparger system was installed on the return line to aid in suspension of debris. Additionally, a sump pump and attendant tubing were used to provide flow from beneath the simulated containment floor to ensure that particulate debris did not accumulate there. Hand-stirring and manual adjustment of the mechanical stirrers was performed as necessary during the additions of the fibrous and particulate debris, with much care and consideration given to avoid disturbing the bed or otherwise artificially influencing the bed formation.

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:

Head loss values were scaled from test conditions to plant conditions using both temperature (i.e., viscosity and density as a function of temperature) as well as strainer approach velocity. As shown in the response to 3.f.10, these equations were derived from VEGP-specific flow sweep data.

During the 2009 head loss testing, flow sweeps were conducted at the end of each test to characterize the flow through a prototypical debris bed. Therefore, any boreholes and other differential-pressure induced effects on bed morphology were captured and properly accounted for when scaling the head loss. In addition, as stated in the NARWHAL CFP calculation, two sets of flow sweep data were collected following the thin-bed and full debris load tests. To account for the uncertainties in the flow sweeps, the resulting correction parameters from both tests were applied at each time step and the higher resulting head loss was used.

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:

The NARWHAL software was used to evaluate the potential for flashing due to the pressure drop across the strainer and debris bed. For a given break, a flashing failure was recorded if, at any time during sump recirculation, the pressure downstream of the strainer was lower than the vapor pressure at the sump temperature. The pressure downstream of the strainer was calculated by NARWHAL based on the strainer submergence, containment pressure, and head loss across the strainer. Note that, for flashing analysis, the strainer submergence is evaluated from the top of the strainer. As discussed in Enclosure 3, Section 6.7, 3.5 psi of accident pressure was credited for the first 2.5 hours of event in order to preclude flashing. This approach is reasonable, since, as shown below, even the smallest margin in the containment pressure for preventing flashing is higher than the 3.5 psi credited in the analysis.

The margin in containment pressure for preventing flashing immediately downstream of the strainer is evaluated for time-dependent post-accident containment and sump conditions. For each given set of conditions, the sump pool temperature is obtained from the design basis profile evaluated for a double-ended reactor coolant pump (RCP) suction break with minimum safeguards. The strainer head loss at each given pool temperature is taken from the NARWHAL CFP calculation. The post-accident containment pressure is from the design basis profile evaluated for a double-ended RCP suction break with maximum safeguards.

The minimum containment pressure that is required to prevent flashing is calculated by adding the strainer head loss to the water vapor pressure. Afterwards, this minimum required containment pressure is compared with the expected post-accident containment pressure to determine the margin.

The evaluation contains the following conservatisms:

- When calculating the minimum containment pressure required to prevent flashing, the submergence of the strainer is conservatively neglected. Including the submergence would reduce the minimum pressure required and increase the margin.
- For sump temperatures above 212 degrees F, the strainer head loss is conservatively assumed to be the same as that at 212 degrees F. In reality, the head loss adjusted to the actual temperature would be lower due to the lower water viscosities at higher temperatures.
- When determining the post-accident containment pressures from the Vogtle FSAR chart, the values are rounded down, which results in conservatively smaller margins.

As shown in Table 3.f.14-1, the minimum margin in the containment pressure to prevent flashing is over 6 psi at 3,000 seconds after the accident. Therefore, crediting 3.5 psi of accident pressure for the first 2.5 hours of the event in the NARWHAL CFP calculation for the flashing evaluation is reasonable and conservative. Figure 3.f.14-1 compares the design basis post-accident containment pressure with the minimum containment pressure required to prevent flashing. The vertical difference between the two curves represent the margin in containment pressure for preventing flashing.

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 (ft)	Min Cont. Pressure Req'd to Prevent Flashing (psia)	Accident Pressure (psia)	Margin in Containment Pressure (psi)
1,800	251	30.66	5.515	32.90	39.4	6.5
3,000	249	29.47	5.515	31.72	37.8	6.1
3,400	248	28.70	5.515	30.95	38.4	7.4
7,020	212	14.81	5.515	17.10	33.4	16.3
7,980	205	12.88	5.544	15.19	33.4	18.2
10,020	195	10.49	5.589	12.83	31.4	18.6
19,980	165	5.42	5.729	7.85	27.4	19.5
30,000	153	4.08	9.006	7.92	25.4	17.5
60,000	140	2.96	9.128	6.86	22.9	16.0
90,060	133	2.47	13.615	8.30	21.9	13.6
500,460	120	1.74	13.826	7.68	19.4	11.7

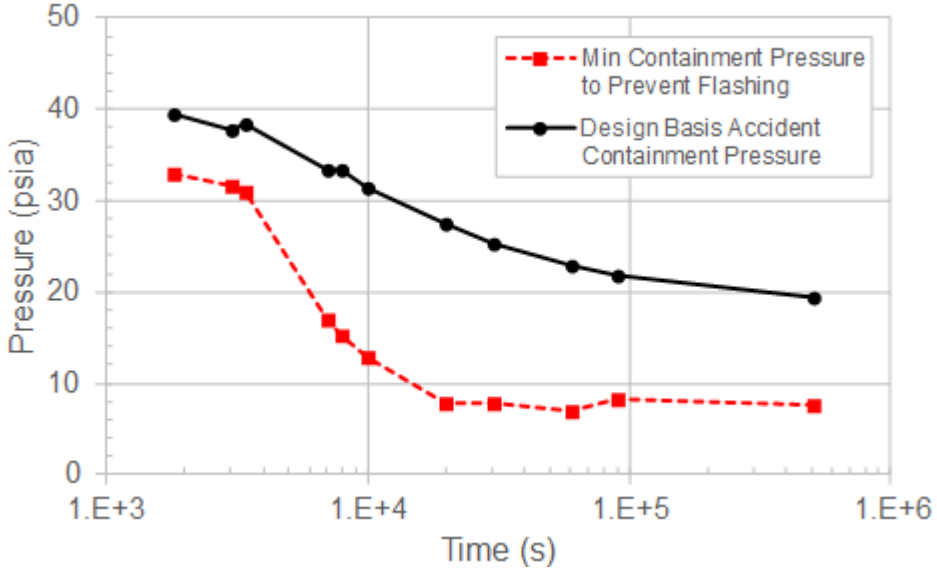

Figure 3.f.14-1: Margin in Containment Pressure for Preventing Flashing based on DBA Curves

To demonstrate that margin also exists for breaks that have a lower pressure than the DBA profile, the same approach was also applied to the best-estimate post-accident containment pressure and sump pool temperature profiles to derive the margins in containment pressure for preventing flashing. Note that the best-estimate curves have lower values than the design basis curves because the thermal hydraulic modeling for the best-estimate cases used less conservative inputs.

Figure 3.f.14-2 compares the best-estimate containment pressure curve for a double-ended guillotine cold leg break with the minimum pressure required for preventing flashing evaluated using the corresponding sump pool temperature profile. The vertical difference between the two curves is the margin for preventing flashing. The results show that the minimum margin is 9.3 psi at approximately 30,000 seconds after the accident. Note that, for the cold leg break, sump recirculation starts at 3,398 seconds after the accident.

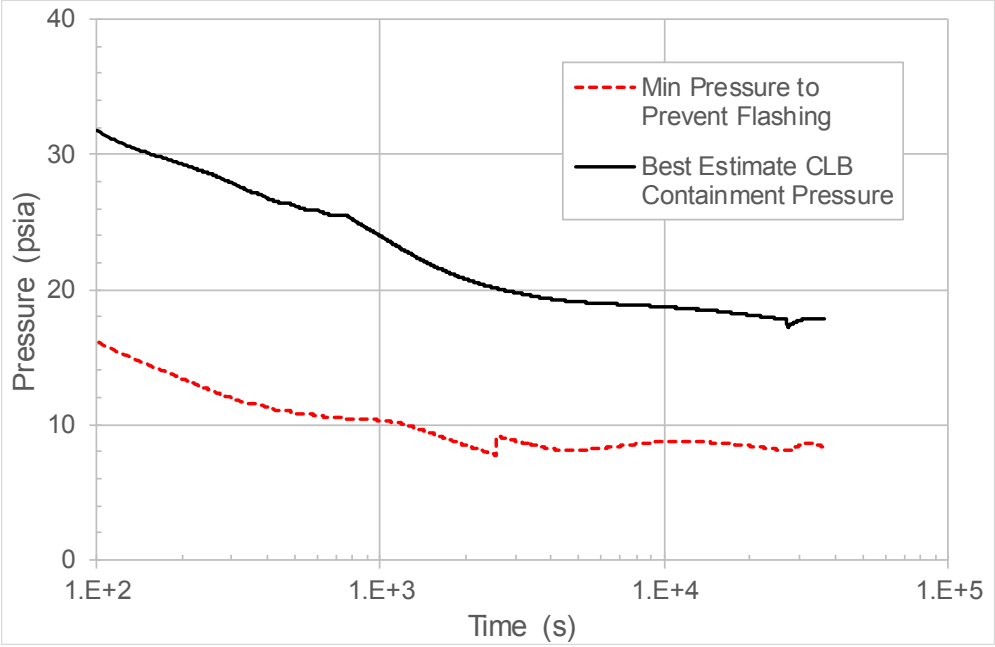

Figure 3.f.14-2: Margin in Containment Pressure for Preventing Flashing based on Best-Estimate Cold Leg Break Curves

Similar evaluations were also performed using the best-estimate hot leg break curves. The results are shown in Figure 3.f.14-3. The minimum margin is 8.7 psi at approximately 2,500 seconds after the accident. Note that, for the hot leg break, sump recirculation starts at 2,263 seconds after the accident.

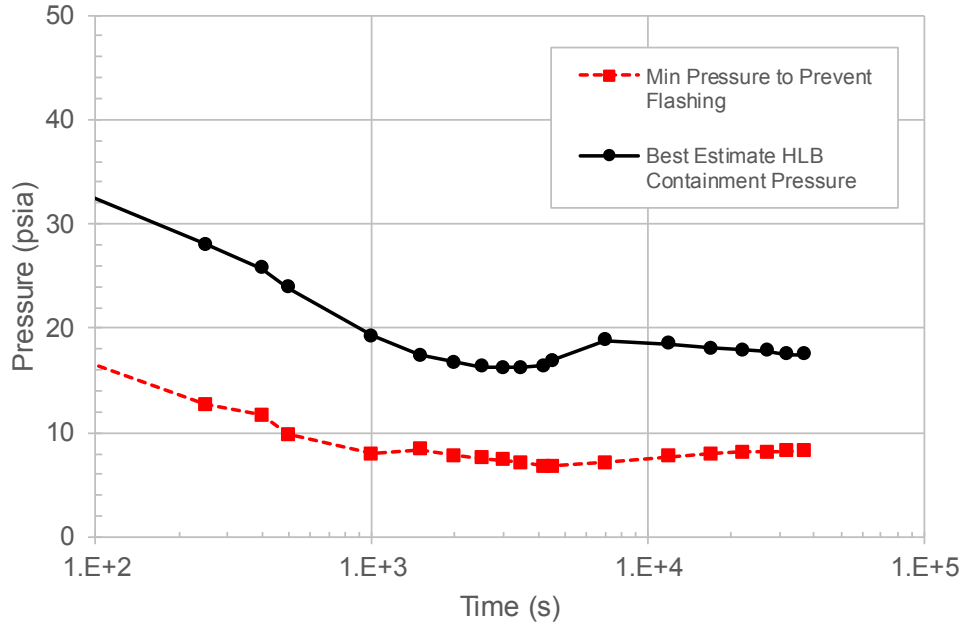

Figure 3.f.14-3: Margin in Containment Pressure for Preventing Flashing based on Best-Estimate Hot Leg Break Curves

In summary, for the best-estimate containment pressure and sump temperature curves, the minimum margin in containment pressure for preventing flashing is at least 8.7 psi. Therefore, crediting 3.5 psi of accident pressure for the first 2.5 hours of the event in the NARWHAL CFP calculation for the flashing evaluation is reasonable and conservative. Note that containment pressure and sump temperature are intrinsically related. While the best-estimate containment pressures are lower than the design basis case, the corresponding pool temperatures are also lower, which results in lower pressures required for preventing flashing. The resulting margins in containment pressure for the best-estimate curves are either comparable or actually greater than those derived based on the design basis curves.

g. Net Positive Suction Head

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

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:

Pump/ Sump Flow Rates

ECCS and CS pump design flow rates used in the NARWHAL model are presented in Table 3.g.1-1. The total recirculation sump flow rates are provided in Table 3.g.1-2. Note that the SI pumps and CCPs piggyback off of the RHR system during recirculation. Additionally, each of the RHR and CS pumps has its own dedicated sump and strainer.

Table 3.g.1-1: Applicable Pump Flow Rates

Pump	Design Flow Rate (gpm)
RHR	3,700
SI	425
CC	150
CS	2,600

Table 3.g.1-2: Total Recirculation Sump Flow Rates

Sump	Design Flow Rate (gpm)
RHR	3,700
CS	2,600

Minimum Water Level

Minimum sump pool levels were calculated in the NARWHAL CFP calculation and in a bounding hand calculation. The NARWHAL calculation performs comprehensive evaluation of GSI-191 phenomena in a self-consistent and time-dependent manner. For each accident evaluated, the entire duration of RWST injection and sump recirculation was divided into smaller time steps. The minimum sump pool volume was calculated for each time step by subtracting the transitory and geometric hold up volumes from the total quantity of water in containment. The NARWHAL CFP calculation evaluated the NPSH margin for each break scenario. Impact on the results due to variabilities in the inputs was evaluated by sensitivity analyses.

The minimum water level hand calculation evaluated bounding minimum sump pool volumes and levels which were used as inputs in the vortexing evaluation (see the Response to 3.f.3) and chemical precipitate debris hand calculation (see the Response to 3.o.1). Table 3.g.1-3 summarizes the results of the minimum water level hand calculation. The short-term water level values (prior to 60 hours post-LOCA) and long-term water level values (at 60 hours post-LOCA) differ due to the transient inputs, such as RWST injection, reactor cavity hold-up, and containment temperature (which influences vapor hold-up).

The submergence values in Table 3.g.1-3 were calculated by subtracting the RHR strainer height (4.438 ft, discussed in Response to 3.f.2) from the water level above the containment floor. A negative value for strainer submergence indicates the strainer is partially submerged. The submergence values in Table 3.g.1-3 bound the minimum submergence of the CS strainers because the CS strainers are shorter than the RHR strainer and switchover of CS pumps to recirculation occurs after the RHR pumps.

The VEGP sump recirculation switchover evaluation showed that, for breaks that do not actuate CS, the ECCS pumps continue drawing all of their flow from the RWST until the Empty level setpoint is reached. In other words, for these breaks, recirculation will not start until the time labeled as “completion of switchover” in Table 3.g.1-3. Therefore, for the breaks that do not actuate CS, no strainer submergence values are shown at the time when the sump suction valves open.

For those breaks that do actuate CS, the VEGP sump recirculation switchover evaluation showed that the ECCS pumps start to draw flow from the sump as soon as the sump suction valves open. Therefore, for these breaks, sump recirculation begins when the sump suction valves open at the RWST Low-Low level.

Table 3.g.1-3: Minimum Sump Pool Water Levels from Hand Calculation

Break Case Description	Time	Time of Occurrence (sec)	Pool Height (ft)	Strainer Submergence (ft)
LBLOCA with Containment Spray	Sump Suction Valves Open[1]	1,929	4.536	0.098
	Completion of Switchover	3,288	6.241	1.803
	5.5 Hours	19,800	6.058	1.620
	60 Hours	216,000	5.311	0.873
SBLOCA without Containment Spray	Sump Suction Valves Open	18,109	4.108	N/A
	Completion of Switchover[2]	24,137	5.739	1.301
	5.5 Hours	19,800	6.003	1.565
	60 Hours	216,000	5.186	0.748
MBLOCA without Containment Spray	Sump Suction Valves Open	6,038	4.648	N/A
	Completion of Switchover[2]	8,048	6.353	1.915
	5.5 Hours	19,800	6.318	1.880
	60 Hours	216,000	5.501	1.063
LBLOCA without Containment Spray	Sump Suction Valves Open	3,655	4.692	N/A
	Completion of Switchover[2]	5,104	6.407	1.969
	5.5 Hours	19,800	6.318	1.880
	60 Hours	216,000	5.501	1.063
Reactor Nozzle Break LBLOCA with Containment Spray	Sump Suction Valves Open[1]	1,929	3.054	-1.384
	Completion of Switchover	3,288	4.788	0.350
	5.5 Hours	19,800	4.971	0.533
	60 Hours	216,000	5.039	0.601
Reactor Nozzle Break LBLOCA without Containment Spray	Sump Suction Valves Open	3,655	3.235	N/A
	Completion of Switchover[2]	5,104	4.977	0.539
	5.5 Hours	19,800	5.161	0.723
	60 Hours	216,000	5.229	0.791

Notes: [1] Beginning of recirculation for the breaks that actuate CS is when the RWST level reaches Low-Low setpoint and the sump suction valves for the RHR pumps open.
[2] Beginning of recirculation for the breaks that do not actuate CS is when the switchover of the RHR pump suctions from the RWST to sump is completed.

When determining the minimum sump pool water levels, the NARWHAL model uses many inputs and assumptions that are consistent with the hand calculation. However, the hand calculation includes some additional conservatisms that are not included in the NARWHAL model. The following items are the most significant differences between the NARWHAL calculated pool level and the hand calculations:

- The hand calculation assumed a break at the highest elevation of the RCS (above the top of the pressurizer), which results in a significantly larger RCS hold-up volume compared to a break at a lower elevation. In NARWHAL, RCS hold-up is based on the break-specific elevation (i.e., the maximum RCS hold-up volume is applied for breaks at the highest elevations such as the pressurizer spray line piping, and smaller hold-up volumes are applied for breaks at lower elevations such as the pressurizer surge line or primary loop piping).
- The hand calculation assumed a pool temperature of 100°F to maximize the pool water density and minimize the pool water level. NARWHAL calculates the pool density as a function of the time-dependent pool temperature, which is significantly higher than 100°F early in the event.

The NARWHAL calculated pool level is shown in Table 3.g.1-4 for a 19-inch hot leg break at Weld 11201-003-5-RB. The hand-calculated water levels for an LBLOCA with containment sprays shown in Table 3.g.1-3 were adjusted to use an RCS hold-up volume consistent with the hot leg break elevation, as well as pool temperatures that are consistent with the design basis pool temperatures used in NARWHAL. As shown in the table, the NARWHAL calculation and hand calculation results are very similar, and only vary slightly due to simplifications made in the hand calculation for time-dependent input parameters.

Table 3.g.1-4: Water Level Comparison for an LBLOCA with Containment Spray Activated

Time	NARWHAL Calculation Pool Level (ft)	Hand Calculation Minimum Pool Level (ft)
Sump Suction Valves Open	5.365	5.271
Completion of Switchover	6.976	6.989
5.5 Hours	6.411	6.579
60 Hours	5.763	5.726

Sump Temperature

The VEGP NARWHAL CFP calculation used the design-basis sump temperature profile calculated for a double-ended pump suction LOCA with minimum safeguards. Note that the minimum safeguards temperature profile shown in Figure 3.g.1-1 is

conservatively higher than the temperature profile for the maximum safeguards case.

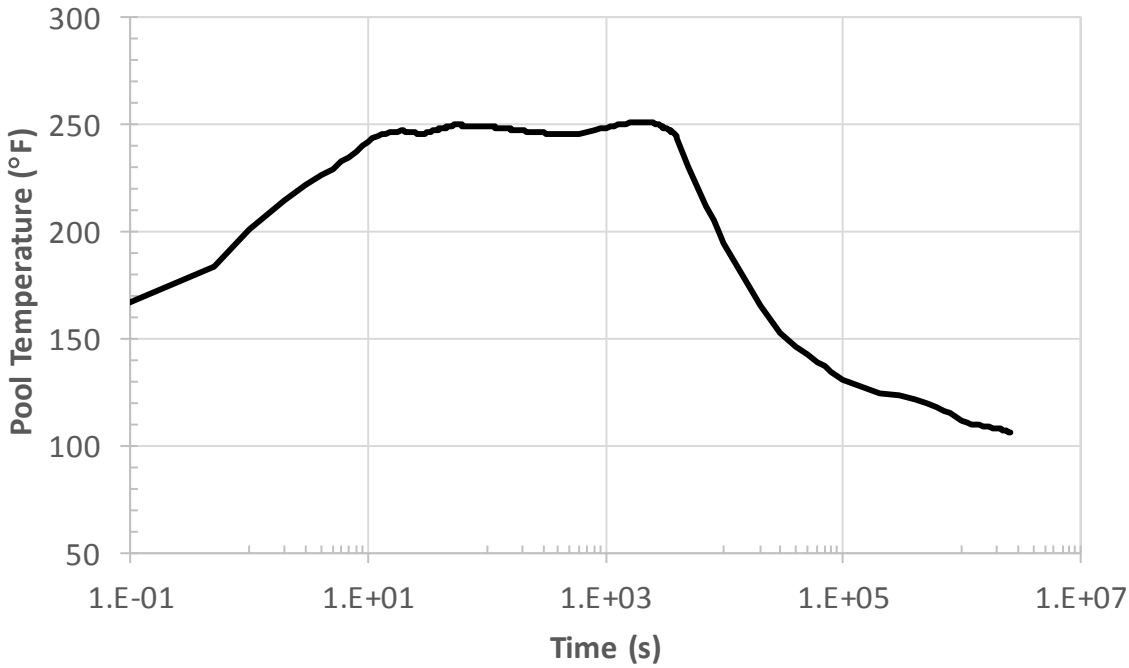

Figure 3.g.1-1: Sump Temperature for Double-Ended Pump Suction Break with Minimum Safeguards

As discussed above, the recirculation duration was divided into smaller time steps. When applying the sump temperature profile, the value that is closest to the current time step is used. Consider an example where the current time-step is 220 seconds and the profile has values corresponding to 219 and 229 seconds. NARWHAL would return the value at 219 seconds because it is closer to the current time step.

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 Rate

As discussed in the VEGP NARWHAL CFP calculation, the RHR flow rate was assumed to be 3,700 gpm. The design flow rate for the RHR pumps is 3,000 gpm. Using a higher flow rate is generally conservative in terms of recirculation timing, flashing calculations, and head loss correction as performed in the NARWHAL CFP calculation. This 3,700-gpm flow rate is also consistent with the value used in the design-basis NPSH calculation and the single train value used in the ECCS system head curve.

In the NARWHAL CFP calculation, the flow rates for the SI pumps and CCPs are their design flow rates based on the SI system description. Similarly, the CS pump flow rate is also the design flow rate from the CS system description. Using the design flow rates for these pumps is reasonable because they are more closely aligned to what would be expected in post-LOCA mitigation than run-out flow rates when taking into consideration system pipe losses.

In the NARWHAL CFP calculation, the same pump flow rates were used consistently for all breaks regardless of break size. Using higher flow rates for the smaller breaks is conservative in terms of NPSH and flashing failures because recirculation starts sooner when the pool temperature is higher.

Minimum Water Level

As stated in the response to 3.g.1, minimum sump pool water levels were calculated in both the NARWHAL CFP calculation and a hand calculation. The major assumptions used in these evaluations are listed as follows.

1. The density of the inventory of the RWST, the reactor coolant system (RCS), and the SI accumulators is assumed to be the same as pure water. This is a reasonable assumption because the concentration of boric acid in the water is extremely small, with a maximum of 1,900 ppm for the RCS; 2,600 ppm for the RWST; and 2,600 ppm for the accumulators.
2. It is assumed that SBLOCAs will not result in rapid, full depressurization of the RCS; therefore, the SI accumulators will not inject when evaluating the minimum water levels for SBLOCAs. This is a conservative assumption because this will minimize the pool volume.
3. It is assumed that MBLOCAs and LBLOCAs will result in full depressurization of the RCS; therefore, during recirculation, the RCS will retain water up to the elevation of the break. This is a reasonable assumption because these breaks result in rapid cooling from the SI accumulators, which are triggered through RCS depressurization.
4. The hand calculation reported minimum sump water levels for three break size categories, defined as follows. This definition matches that used in the Vogtle PRA model.
   a. An SBLOCA is defined as a break smaller than 2 inches.
   b. An MBLOCA is defined as a break greater than or equal to 2 inches, less than 6 inches.
   c. An LBLOCA is defined as a break greater than or equal to 6 inches with the largest break being a double-ended guillotine break of the crossover leg.
5. It is assumed that a reactor nozzle break will cause the entire reactor cavity (up to the seal ring) to fill with water before any water reaches the containment floor. This conservatively maximizes the transient reactor cavity hold-up, thereby minimizing the pool level.
6. By maximizing the vapor hold-up in the atmosphere of containment, water is withheld from the pool, thereby conservatively minimizing the pool level. To

maximize the vapor hold-up in the atmosphere of containment, three complementary assumptions were made.

a. The relative humidity within containment post-LOCA was maximized. A post-LOCA relative humidity of 100 percent was used to saturate the atmosphere in containment completely, thereby maximizing the change in water vapor in the air from pre-LOCA to post-LOCA conditions.
b. The relative humidity within containment pre-LOCA was minimized. A pre-LOCA relative humidity of 0 percent was used to obtain a pre-LOCA vapor hold-up of 0 gal, thereby maximizing the change in water vapor in the air from pre-LOCA to post-LOCA conditions.
c. The transient values used for containment temperature are maximum values, which result in maximum vapor pressures. These maximum vapor pressures maximize the vapor hold-up in the air.

7. The containment sprays were assumed to only be activated for hot leg breaks greater than 15 inches, which includes all partial breaks and DEGBs greater than 15 inches on the hot legs. However, no breaks on the cold or intermediate legs were assumed to actuate containment sprays. This assumption is consistent with the results of best-estimate thermal-hydraulic modeling for a range of potential break sizes on the hot and cold leg piping. This modeling showed that a hot leg DEGB resulted in containment pressures exceeding the CS actuation setpoint of 21.5 psig, while all other evaluated breaks (including a cold leg DEGB and partial 15 inch breaks on both the hot and cold legs) did not. Assuming that hot leg breaks greater than 15 inches activate CS is reasonable because it represents what was learned from the best-estimate thermal hydraulic modeling. It is recognized that there is some uncertainty in which breaks initiate CS. Sensitivity runs were therefore performed on actuation limits and spray duration using NARWHAL, as summarized in Enclosure 3.
8. The accumulators were assumed to not inject for any secondary side break. This is a reasonable assumption because secondary side breaks do not result in rapid depressurization of the RCS, which would trigger accumulator injection.

Sump Temperature

The sump temperature profile used for the GSI-191 analysis was from the design-basis containment analysis for a DEGB on the crossover leg with minimum safeguards and 11.12 percent fan cooler degradation. This analysis was performed for evaluating post-LOCA containment integrity to support the VEGP Units 1 and 2 measurement uncertainty recapture power uprate program. Note that, in addition to the crossover leg break, the containment analysis also modeled the containment response following a main steam line break (MSLB). The results showed that the crossover leg break with minimum safeguards resulted in higher sump temperatures than the MSLB and the crossover leg break with maximum safeguards.

3. Provide the basis for the required NPSH values, e.g., 3 percent head drop or other criterion.

   Response to 3.g.3:

   The NPSH required (NPSHR) values were taken from the bounding pump vendor curves. These curves were obtained by the pump manufacturer through testing in accordance with the Hydraulic Institute guidelines in effect at the time. Typically, the 3 percent head drop criterion was used in pump NPSH testing.

4. Describe how friction and other flow losses are accounted for.

   Response to 3.g.4:

   The verification of adequate NPSH margin to the RHR and CS pumps from the containment sump was performed using the NARWHAL model. For each time step, NARWHAL calculates the pump NPSH available (NPSHA), NPSHR, and strainer head loss using the inputs of that time step (e.g., sump water level, sump temperature, and pump flow rates). Note that the calculated NPSHA accounted for the piping head loss from the sump to pump suction but not the strainer head loss. If the NPSH margin, determined by subtracting NPSHR from NPSHA, is less than the total strainer head loss, a failure is recorded.

   The total strainer head loss was calculated by combining the clean strainer and debris bed head losses, and extrapolation constant as necessary. The head loss of the suction piping between the strainer exit and the pump suction was accounted for when calculating NPSHA. The piping frictional loss was calculated using the standard Darcy formula with the friction factor determined from an empirical equation. 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.

5. Describe the system response scenarios for LBLOCAs and SBLOCAs.

   Response to 3.g.5:

   In response to a LOCA, the RHR pumps, SI pumps, 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 high-pressure setpoint for CS actuation. The CS pumps also take suction from the RWST during the injection phase. When the RCS depressurizes to approximately 600 psia, all four accumulators begin to inject borated water into the RCS loops.

   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 ECCS pumps

open automatically when the RWST level reaches the Low-Low setpoint. The switchover for the CS pumps starts manually when the RWST level reaches the Empty setpoint. The switchover is complete when the suction valves from the RWST for all pumps are manually closed, which occurs between the RWST Empty and Dead Volume levels. For the breaks that do not actuate CS, the switchover to sump recirculation for the ECCS pumps follows the same logic.

Approximately 7.5 hours following an accident, the ECCS line-up is modified for simultaneous cold and hot leg recirculation. For this operating mode, the SI pumps 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 response sequence described above is typical for the ECCS and CSS following a LOCA. The differences between the responses 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.

6. Describe the operational status for each ECCS and CSS pump before and after the initiation of recirculation.

Response to 3.g.6:

Residual Heat Removal Pumps

In the event of a LOCA, both RHR pumps are started 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. When the RWST level reaches the Low-Low setpoint, the suction valves to the sump automatically open. The RHR pumps could take suction simultaneously from the RWST and the containment sumps. After the RWST level reaches the Empty setpoint, the suction valves to the RWST are manually closed. Afterwards, the RHR pumps take suction from the sumps only. The RHR pumps continue to supply flow to the RCS cold legs and to the SI pumps and CCPs.

Centrifugal Charging Pumps

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. The CCPs supply flow to the RCS cold legs. After switching to the sump recirculation phase, flow to the CCPs is provided from the RHR pump discharge. The CCPs continue supplying flow to the RCS cold legs during simultaneous cold leg and hot leg recirculation.

Safety Injection Pumps

In the event of a LOCA, both SI pumps 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 leg. Similar to the CCPs, flow to the SI pumps is supplied from the containment emergency sump via the RHR pumps during the recirculation phase.

Containment Spray System Pumps

The CS pumps can be actuated manually from the control room or automatically on receipt of two out of four containment pressure (high-3) signals. These signals start the CS pumps and open the discharge valves to the spray headers. During the injection phase, the CS pumps take suction from the RWST. As discussed in the Response to 3.g.5, the pump suction is manually switched to the containment recirculation sump when the RWST level reaches the Empty setpoint.

7. Describe the single failure assumptions relevant to pump operation and sump performance.

Response to 3.g.7:

As described in Enclosure 3, Sections 4.0, 6.3, and 14.1, the VEGP risk-informed evaluation considered many different equipment configurations and wasn't limited to the worst single failure. The high likelihood configuration calculation used the VEGP PRA model of record, which accounts for human reliability analysis (HRA), and identified the following twelve equipment failure combinations.

1. No Equipment failure
2. RHR Pump B failure
3. RHR Pump A failure
4. Charging Pump A failure
5. Charging Pump B failure
6. SI Pump B failure
7. SI Pump A failure
8. Train B failure
9. Train A failure
10. CS Pump B failure

11. CS Pump A failure
12. Both CS Pumps failure

Note that the VEGP inputs and NARWHAL methodology allow for this list to be reduced based on train symmetry. A pump failure for one train is analytically identical to a pump failure for the other train. Therefore, the following seven equipment configurations were analyzed in the NARWHAL CFP calculation.

1. No Equipment failure
2. RHR Pump B failure
3. Charging Pump B failure
4. SI Pump B failure
5. Train B failure
6. CS Pump B failure
7. Both CS Pumps failure

It was assumed that all random equipment failures evaluated occur at the beginning of recirculation. This is a conservative assumption because it results in a quicker switchover to recirculation when compared to failure at the beginning of the event. Additionally, for CS pump and/or RHR pump failure cases, it results in more debris accumulation on the remaining active strainers.

The CCP B and SI pump B failure cases are identical to the no equipment failure case. This is because the failure is applied at the start of recirculation. The flow rate through the RHR strainers is not affected by the charging pump failure because the RHR pump provides the same flow rate regardless of which piggybacked pump fails. The CS pump B failure case is similar to the no equipment failure case. It only affects hot leg breaks greater than 15 inches. Thus, this case only has a slight effect on the results even though there is one less active strainer during recirculation.

8. Describe how the containment sump water level is determined.

Response to 3.g.8:

As discussed in the response to 3.g.1, the post-LOCA minimum sump pool level was determined in both the NARWHAL CFP calculation and a hand calculation. The two calculations used the methodology described below:

1. A correlation was first developed for the relationship between the containment water level and the water volume using a 3-D CAD model.
2. The quantity of water added to containment from the RWST, RCS, and SI accumulators was calculated.
3. The quantity of water that is diverted from the containment sump by the following effects was evaluated:
   - Hold-up within the reactor cavity and in-core tunnel.
   - RCS hold-up volume required to fill the RCS steam space.

- Water volume required to fill the CS pump discharge piping that is empty pre-LOCA.
- Water in transit from the containment spray nozzles and the break to the containment sump.
- Steam hold-up in the containment atmosphere.
- Miscellaneous hold-up volumes throughout containment, such as containment sumps, the elevator pit, and containment floor drains.

4. Given the net mass of water added to the containment floor based on Items 2 and 3 listed above, the post-LOCA containment water level is calculated using the correlation developed in Item 1.

As discussed earlier, the NARWHAL CFP calculation used self-consistent inputs and evaluated time-dependent pool volumes and water levels for each postulated break. The hand calculation determined bounding minimum containment water levels for LBLOCA, MBLOCA, and SBLOCA and provided inputs for evaluating chemical precipitate debris quantities and vortexing. While the NARWHAL CFP calculation determines the water level at each time step within the simulation, the hand calculation only reported water levels at a few different times after the accident, as shown in Table 3.g.1-3.

9. Provide assumptions that are included in the analysis to ensure a minimum (conservative) water level 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 VEGP containment water volume calculation.

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 VEGP containment water volume calculation as hold-up volumes that remove water from the containment pool: CS discharge piping (initially empty spray piping), water in transit from both the containment spray nozzles and the break itself, and the water droplets on containment walls.

11. Provide assumptions (and their bases) as to what equipment will displace water resulting in higher pool level.

Response to 3.g.11:

The volumes occupied by structures, equipment, and equipment supports, etc. will displace water and result in a higher pool level. Examples of such equipment and structures include concrete walls, accumulator tanks, piping, and cable trays. These volumes were accounted for in the VEGP containment water volume calculation. The 3D CAD model of containment was used to determine the correlation between the containment pool volume and water level. Smaller equipment, cables, and instruments are excluded from the CAD model and therefore provide some conservatism in the resulting water levels.

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 following design inputs provided the basis for water sources and their volumes to determine the minimum containment water level for VEGP:

- The VEGP TS minimum initial RWST level was used for the initial RWST water level. As discussed in 3.g.1, when evaluating the minimum containment water level, the RWST level at the beginning of sump recirculation is either at the Low-Low level (minimum volume of water injected from the RWST at this level is 435,522 gal) or at the Empty level (the minimum volume of water injected from the RWST at this level is 580,497 gal).
- Four SI accumulators have a minimum volume of 6,555 gal/accumulator. The total minimum volume of the SI accumulators is therefore 26,220 gal. This volume is not credited in the SBLOCA cases because the RCS pressure is assumed to remain above their injection pressure as stated in Response to 3.g.2.
- The inventory of the RCS is assumed to remain relatively constant during normal operations. This is a reasonable assumption because during full power operation, the RCS remains at a fixed volume and remains at constant temperature and pressure. Due to the small volume of the RCS as compared to the RWST and its negligible variation in water volume, a best estimate value is representative. The best estimate RCS liquid volume is that associated with the total RCS liquid volume at hot full power conditions: 86,729 gal. The RCS represents both a source of water and a hold-up volume. The mass of water held up in the RCS may be more or less than the initial RCS mass depending on the elevation of the break (e.g., for a break at the top of the pressurizer, the vapor space of the pressurizer would be filled).

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 VEGP analysis for pump NPSH. Using the VEGP NARWHAL model, pump NPSH margin was calculated at each time step using inputs from that time step. The containment pressure is assumed to be equal to the saturation pressure at the sump temperature for sump temperatures greater than 210.96 degrees F. Note that the temperature of 210.96 degrees F corresponds to the saturation temperature at the VEGP TS minimum containment pressure of –0.3 psig. For sump temperatures below 210.96 degrees F, the minimum containment pressure of –0.3 psig (or 14.396 psia) was used as the containment pressure to calculate the pump NPSHA.

14. Provide assumptions made which minimize the containment accident pressure and maximize the sump water temperature.

Response to 3.g.14:

Containment Pressure

As discussed in the Response to 3.g.13, the VEGP TS minimum containment pressure is –0.3 psig, which corresponds to a saturation temperature of 210.96 degrees F. When calculating pump NPSH margin, the containment pressure was minimized by using the minimum containment pressure of –0.3 psig for sump temperatures below 210.96 degrees F. When the sump temperature is higher than 210.96 degrees F, the containment pressure was assumed to be equal to the saturation pressure at the sump temperature, which is necessary to maintain the sump in a liquid phase. No accident pressure was credited for NPSH calculations.

As stated in the NARWHAL CFP calculation, an accident pressure of 3.5 psi was credited for the first 2.5 hours of the event when evaluating degasification. The static head credited for degasification calculations is from the surface of the pool to the midpoint of the strainer.

Sump Temperature

As discussed in the Response to 3.g.1, the VEGP NARWHAL model used the design-basis sump temperature profile calculated for a double-ended pump suction LOCA with minimum safeguards. Note that the minimum safeguards temperature profile is conservatively higher than the temperature profile for the maximum safeguards case during recirculation through the sump strainers. The time-dependent sump temperature profile applied to all breaks is included as Figure 3.g.1-1.

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 Responses to 3.g.13 and 3.g.14.

16. Provide the NPSH margin results for pumps taking suction from the sump in recirculation mode.

Response to 3.g.16:

The RHR and CS pump NPSH margins were evaluated using the VEGP NARWHAL model. Table 3.g.16-1 provides a summary of the minimum NPSH margins for the RHR pumps in recirculation mode at various sump temperatures between 120 degrees F and 212 degrees F. The RHR pump NPSH margins shown in Table 3.g.16-1 are for a 19-inch partial break at weld location 11201-003-5-RB, as identified in Table 3.b.4-2. Note that the four breaks in Table 3.b.4-2 have the same NPSH margins due to the rule-based approach used in calculating head loss.

Table 3.g.16-1 Limiting NPSH Margin vs. Sump Temperature

Pool Temperature (°F)	NPSH Margin Before Subtracting Strainer Head Loss (ft-H_2O)	Strainer Head Loss (ft-H_2O)	Net NPSH Margin After Subtracting Strainer Head Loss (ft-H_2O)
212	23.119	5.515[a]	17.6
205	27.255	5.544[a]	21.7
195	32.441	5.589[a]	26.9
165	44.863	5.729[a]	39.1
153	45.657	9.006[b]	36.7
140	47.339	9.128[b]	38.2
133	44.076	14.359[c]	29.7
120	45.861	14.578[c]	31.3

[a] This includes clean strainer, conventional and chemical debris (calcium phosphate) head losses.
[b] This includes clean strainer, conventional and chemical debris (calcium phosphate) head losses, and extrapolation constant.
[c] This includes clean strainer, conventional and chemical debris (calcium phosphate and SAS) head losses and extrapolation constant.

Although the minimum net NPSH margins shown in Table 3.g.16-1 are for RHR Pump A, they are bounding for all of the RHR and CS pumps. As shown in the NARWHAL outputs, the NPSH margins for the CS pumps are consistently greater, compared with the RHR pumps. This is expected because of the lower head losses

of the CS strainers associated with the lower CS pump flow rate and smaller debris loads. An NPSH evaluation was not performed for the SI pumps and CCPs because these pumps take suction from the RHR pumps during recirculation.

Table 3.g.16-1 shows the NPSH margins before and after subtracting the total strainer head losses. The total strainer head losses include the clean strainer head loss, conventional debris (particulate and fiber) head loss, and chemical debris (calcium phosphate and SAS) head loss, as appropriate. Bounding head loss values, as shown in the Response to 3.f.10, were used in the evaluation. The head losses were also extrapolated to the end of the 30-day mission time as described in the Response to 3.f.10. As shown in the table, the minimum net NPSH margin for any given sump temperatures is over 17 ft. Therefore, adequate NPSH margin is available for Unit 1 and Unit 2 RHR and CS pumps to ensure their design functions.

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.

1. Provide a summary of type(s) of coating systems used in containment, e.g., Carboline CZ 11 Inorganic Zinc primer, Ameron 90 epoxy finish coat.

   Response to 3.h.1:

   The types of coating and systems used in containment are presented in Table 3.h.1-1.

Qualified Coatings

Table 3.h.1-1 – Coatings Systems Used in Analyses

Substrate	Layer	Type	DFT (mil)	Density (lbm/ft³)
Steel Surfaces	1st Coat	Carbozinc 11	5	208
	2nd Coat	Ameron 90	6	99.6
	3rd Coat	Ameron 90	6	99.6
	Total		17	
Concrete Surfaces	1st Coat	K&L 4129	1.5	69.0
	2nd Coat	K&L 4000	25	107.2
	3rd Coat	K&L D-Series	9	98.0
	Total		35.5	

Unqualified Coatings

Unqualified coatings could include coatings within containment that do not have a specified preparation, application, or inspection compliant with plant specifications; previously qualified coatings that have noticeably deteriorated; 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, including various types of epoxy, inorganic zinc, and alkyds.

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 NARWHAL model:

- The distribution of unqualified coatings in upper and lower containment was determined based on detailed logs maintained over the life of the plant, and this distribution was credited in the analysis. Unqualified coatings initially located in lower containment were assumed to fail directly into the sump at the beginning of the event. For the unqualified coatings initially located in upper containment, the washdown fraction is 100% when containment sprays are initiated, and 10% when containment sprays are not initiated (see Table 3.e.6-2).
- It was assumed that the unqualified and degraded qualified coatings in VEGP have a recirculation transport fraction of 100%. This is consistent with the

debris transport calculation, and is conservative since settling of this debris is not credited.

3. Discuss suction strainer head loss testing performed as it relates to both qualified and unqualified coatings. Identify surrogate material and what surrogate material was used to simulate coatings debris.

   Response to 3.h.3:

   Silicon carbide was used to simulate both qualified and unqualified coatings debris. See the Response to 3.f.4 for detailed information on coating surrogates and the amount added to the test.

4. Provide bases for the choice of surrogates.

   Response to 3.h.4:

   See the Response to 3.f.4.

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 assumptions related to coatings were made in the debris generation calculation:

   - Qualified coatings within the ZOIs were assumed to fail as 10μm diameter spheres; qualified coatings outside the ZOIs were assumed to remain intact. This is based on the guidance of NEI 04-07.
   - It was assumed that the FN-8 qualified coatings system was applied to steel structures, including columns, equipment supports and grating. The FN-14/19 system was applied to all concrete surfaces within containment. Using these two systems is conservative because they have the largest number of coats and the largest final dry film thickness of all field coating systems present within containment for their respective substrates. Both field coating systems, including the type, dry-film thickness, and density are presented in Table 3.h.1-1.

   The amount of coating debris generated at VEGP is shown in Tables 3.h.5-1 and 3.h.5-2. The unqualified coatings in upper and lower containment are quantified

based on detailed logs maintained over the life of the plant. The entire quantity of unqualified coatings, as shown in the tables, were assumed to fail for all breaks.

Table 3.h.5-1: Coatings Debris for the Four Overall Worst-Case Breaks

Break Location		11201-004-6-RB	11201-001-5-RB	11201-001-3-RB	11201-004-4-RB
Break Size		29"	29"	29"	29"
Break Type		DEGB	DEGB	DEGB	DEGB
Qualified Epoxy (ft^3)		0.51	0.50	0.50	0.48
Qualified IOZ (ft^3)		0.21	0.21	0.21	0.20
Unqualified Epoxy (ft^3)	**UC**	17.671	17.671	17.671	17.671
	LC	12.711	12.711	12.711	12.711
Unqualified Alkyd (ft^3)	**UC**	0.0	0.0	0.0	0.0
	LC	0.516	0.516	0.516	0.516
Unqualified IOZ (ft^3)	**UC**	0.117	0.117	0.117	0.117
	LC	0.265	0.265	0.265	0.265

Table 3.h.5-2: Coatings Debris for the Four Worst-Case Breaks that Do Not Fail the Strainer Acceptance Criteria

Break Location		11201-004-4-RB	11201-001-3-RB	11201-003-5-RB	11201-002-5-RB
Break Size		20"	23"	19"	16"
Break Type		Partial	Partial	Partial	Partial
Qualified Epoxy (ft^3)		0.068	0.089	0.067	0.055
Qualified IOZ (ft^3)		0.028	0.037	0.028	0.023
Unqualified Epoxy (ft^3)	**UC**	17.671	17.671	17.671	17.671
	LC	12.711	12.711	12.711	12.711
Unqualified Alkyd (ft^3)	**UC**	0.0	0.0	0.0	0.0
	LC	0.516	0.516	0.516	0.516
Unqualified IOZ (ft^3)	**UC**	0.117	0.117	0.117	0.117
	LC	0.265	0.265	0.265	0.265

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 2) and the associated NRC SE (Reference 3), all coating debris was treated as particulate and

therefore transported entirely to the sump strainer. See the Response to 3.h.1, 3.h.2, and 3.h.5 for additional description of debris characteristics.

7. Describe any ongoing containment coating conditions assessment program.

   Response to 3.h.7:

   SNC conducts condition assessments of coatings inside containment every outage under the site work control system. As localized areas of degraded coatings are identified, those areas are evaluated and scheduled for repair or replacement as necessary. 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.

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:

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:

   SNC procedure, "Containment Exit Inspection," provides detailed guidance for containment inspection to ensure no loose debris (e.g., rags, trash, clothing, etc.) is present in the containment that could be transported to the containment sump and

cause restriction of pump suctions during LOCA conditions. This procedure contains an extensive checklist detailing all areas of containment that must be inspected for cleanliness prior to plant startup after each outage.

SNC procedure, "Containment Entry," establishes guidance to inventory and control items carried into containment during non-outage entries. This procedure ensures that no loose debris (e.g., rags, trash, clothing, etc.) is present in the containment, which could be transported to the containment sump and cause restriction of pump suctions during LOCA conditions.

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:

SNC procedure, "Foreign Material Exclusion Program," establishes the administrative controls and personnel responsibilities for the foreign material exclusion (FME) program. The procedure describes methods for controlling and accounting for material, tools, parts, and other foreign material to preclude their uncontrolled introduction into an open or breached system during work activities. This procedure also provides guidance for establishing and maintaining system cleanliness, recovering from an intrusion of foreign material, and re-establishing system cleanliness requirements.

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:

An enhancement to the screening guidelines and considerations for the design input process, which is part of the design change procedure, has introduced a requirement to review the impact of a proposed change on the documentation that forms the design basis for the response to GL 2004-02. The specific areas that are addressed are:

- Insulation inside containment
- Fire barrier material inside containment
- Coatings inside containment
- Inactive volumes in containment
- Labels inside containment
- Buffer changes (iodine and pH control)
- Structural changes (i.e., choke points) in containment

- Downstream effects (piping components downstream of the ECCS sump strainers)

As shown in the VEGP containment sump pool chemical effects calculation, aluminum release into the sump pool is dominated by aluminum metal inside the containment. An SNC fleet engineering guideline directs the engineers to evaluate any impact on the amount of reactive metals (i.e., zinc and aluminum) due to plant modifications. For VEGP, the inventory of aluminum metal inside the containment is tracked in a calculation that addresses both chemical effects for GSI-191 and post-accident hydrogen generation.

The main source for calcium release is the insulation materials, for example, Calcium Silicate, Mineral Wool and E-Glass. As stated above, the VEGP design change procedure already incorporates requirements for reviewing the impact of proposed changes on insulation materials inside the containment.

Inclusion in the design input process ensures all design changes consider these attributes during the design process.

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:

Maintenance activities, including temporary changes, are subject to the provisions of 10 CFR 50.65(a)(4), as well as VEGP TSs. SNC fleet procedures also provide guidance. For instance, the 50.59 review process procedure provides details on maintenance activities and temporary modifications, while the on-line work management process procedure establishes administrative controls for performing on-line maintenance of structures, systems, components (SSCs) to enhance overall plant safety and reliability. Further guidance is also available in the temporary configuration change procedure.

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:

   All of the Min-K insulation located inside the steam generator compartments (original ZOI analyzed for GL 2004-02) was removed from VEGP Unit 1 and Unit

2 containments during refueling outage 1R13 (Fall 2006) and refueling outage 2R12 (Spring 2007).

There are no known quantities of Min-K in VEGP Unit 1 and Unit 2 containments outside of the secondary shield wall (outer wall of the steam generator compartments). However, Min-K was only used as insulation in penetrations, which are difficult to inspect. This leaves the containment wall as the only possible location of remaining Min-K. As shown in Figure 3.a.1-1, there is only one line outside of the steam generator compartments with analyzed breaks (i.e. welds inside the first isolation valve). If a worst case ZOI of 28.6D is assumed for this 2" line, the ZOI radius would be about 4.8 feet. Using the strainer as a reference dimension (square with each side approximately 5 feet), it is apparent that the ZOI could not reach the containment wall. Therefore, Min-K is not considered when analyzing 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.

Response to 3.i.5.b:

This suggested design and operational refinement was not used in the VEGP evaluation.

c. Modifications to equipment or systems conducted to reduce the debris burden at the sump strainers.

Response to 3.i.5.c:

This suggested design and operational refinement was not used in the VEGP evaluation.

d. Actions taken to modify or improve the containment coatings program.

Response to 3.i.5.d:

No specific actions were taken to modify or improve the containment coatings program; however, enhancements were made to the screening guidelines and considerations for the design input process to ensure that all design changes consider GL 2004-02 attributes during the design process. The specific areas that are addressed are listed in the Response to 3.i.3.

j. Screen Modification Package

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

1. Provide a description of the major features of the sump screen design modification.

Response to 3.j.1:

The currently installed strainers for RHR and CS consist of four parallel, vertically stacked, modular disk strainer assemblies that are connected to a plenum installed over each sump. Each RHR strainer assembly consists of 18 stacked disks that are 30 inches long by 30 inches wide, and the height of the disk portion of the strainer is 53.75 inches. The RHR strainer assemblies are 59.6 inches tall, measured from the containment floor. The four RHR strainer assemblies provide approximately 765 ft² of perforated plate surface area and 179 ft² of circumscribed surface area per sump. Each CS strainer assembly consists of 14 stacked disks that are 30 inches long by 30 inches wide, and the height of the disk portion of the strainer is 41.75 inches. The CS strainer assemblies are 47.6 inches tall measured from the containment floor. Figure 3.j.1-1 below shows a picture of one CS strainer. Each of the four CS strainer assemblies provides approximately 590 ft² of perforated plate surface area and 139 ft² of circumscribed surface area.

Subsequent risk-informed analysis has led to the proposed modification of the Unit 1 and Unit 2 RHR sump strainer assemblies. The RHR strainers will be modified to reduce the overall height approximately 6 inches by removing the two top disks per disk stack. The modified RHR strainer assembly will consist of 16 stacked disks, and the disk portion of the strainer is approximately 47.75 inches high (53.75 in. – 6 in. = 47.75 in.). As shown in the response to 3.f.2, the overall height of the modified RHR strainer is 53.25 inches, measured from the containment floor to the highest strainer disks. The four RHR modified strainer assemblies provide approximately 677.6 ft² of perforated plate surface area and 159 ft² of circumscribed surface area per sump as calculated below.

$$\left(\frac{4\ modules\left(4\frac{sides}{module}\right)(30\ in)(47.75\ in)(1\ ft^2)\left(\frac{circumscribed\ area}{side}\right)}{(144\ in^2)}\right) = 159\ ft^2$$

All of the analyses shown in this submittal were performed for the modified strainer configuration. Operating procedures are being revised, in addition to the planned physical modification, to ensure that the RHR strainers are completely submerged for an increased number of postulated LOCA scenarios.

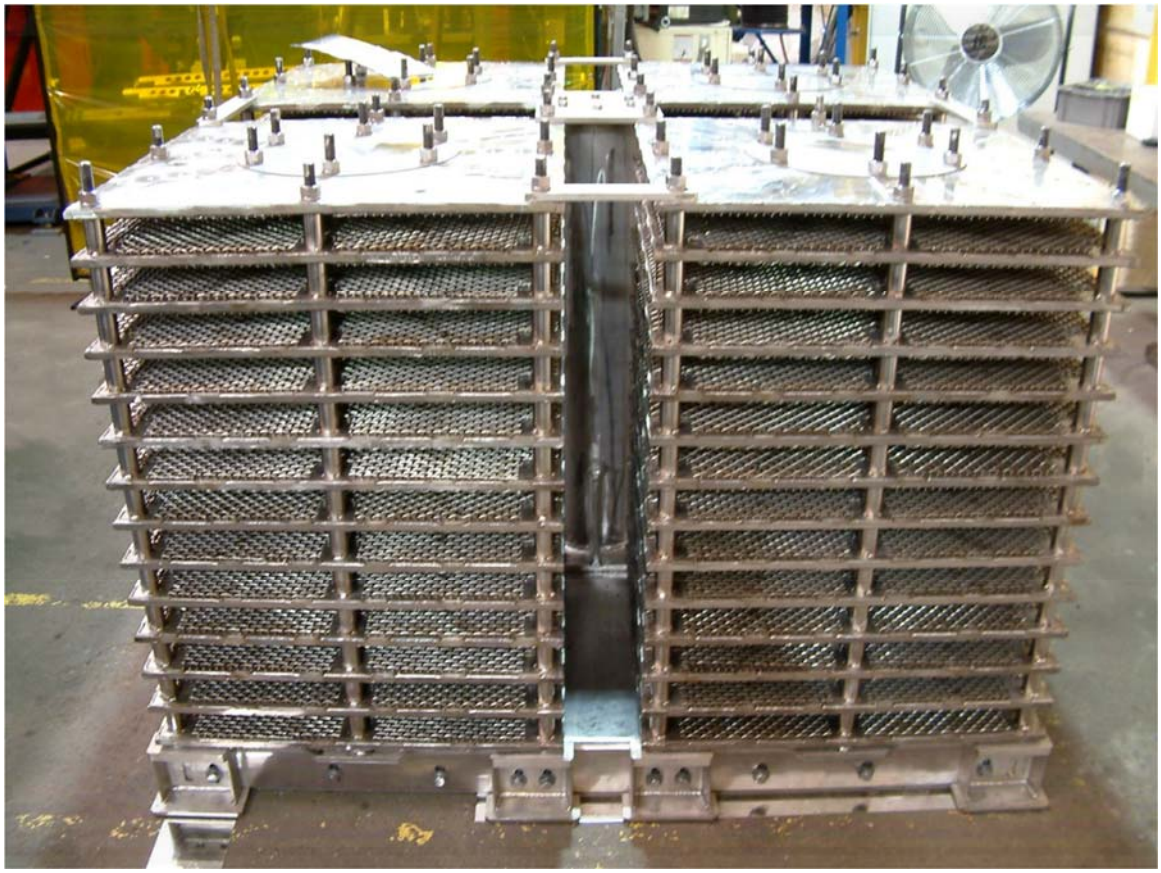

Figure 3.j.1-1 Containment Spray Strainer

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:

The following modifications were necessitated by those of the sump strainer:

- Installation of new and replacement of existing ECCS flow orifices to allow new ECCS throttle valve settings.
- Cage assembly vortex suppressors installed in the sumps removed.
- Temperature elements for the Units 1 and 2 RHR sumps replaced and relocated.
- Two conduit interferences at the Unit 2 RHR sump Train A screen rerouted through an area outside of the sump screen envelope.
- Three electrical interferences for the new Unit 2 CS sump Train A screen relocated/rerouted through an area outside of the sump screen envelope.
- The RHR strainers will be reduced in height by the removal of two disks from each stack to ensure full submergence for an increased number of postulated break scenarios as described in the Response to 3.j.1.

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.*

1. Summarize the design inputs, design codes, loads, and load combinations utilized for the sump strainer structural analysis.

   Response to 3.k.1:

   Design Codes

   (1) American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Subsection NC and ND, 1989 Edition.

   (2) American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Appendix I, 1989 Edition, Table I-6.0 for Modulus of Elasticity, Table I-5.0 for thermal expansion, and Table I-7.2 for allowable stress (S).

Material Properties

The Material properties come from the ASME Code and are tabulated in Table 3.k.1-1 below.

Table 3.k.1-1 Material Properties

Material / Property	@ Room Temperature (70°F)	@ Maximum Water Temperature (250°F)
SA-240 Type SS304 (Strainer):		
E, Elastic modulus, psi	28.3×10^{6}	27.45×10^{6}
Coefficient of thermal expansion, in/in/°F	8.6×10^{-6}	8.995×10^{-6}
Poisson's ratio	0.3	0.3
SA-479 Type SS410 (Tie Rod):		
E, Elastic modulus, psi	28.3×10^{6}	27.45×10^{6}
Coefficient of thermal expansion, in/in/°F	5.9×10^{-6}	6.1×10^{-6}

Load Combinations

Table 3.k.1-2 shows the load combinations specified for the VEGP passive suction strainer design.

Table 3.k.1-2 Load Combinations for VEGP Strainer Design

	Load Combination
Strainer Assembly	
Design	W + Po + OBE1
Level B	WD + Pd + OBE2 + TEmax + Pcr
Level D	WD + Pd +SSE2 + Pcr
Support Structure	
Design	W + Po + OBE1
Level B	WD + Pd + OBE2 + TEmax
Level D	WD + Pd +SSE2

Nomenclature:

W	=	Weight (Dry strainer Assembly Weight)
WD	=	Weight + Debris Weight + Hydrodynamic Mass (LOCA Event with Strainer in Water)
Pcr	=	Crush Pressure (During Suction Strainer Operation in Water Post LOCA)
Pd	=	Design Pressure (LOCA Event) + Water Head (Strainer Open System)
Po	=	Design Pressure (Strainer Open System)

OBE1 = Operating Basis Earthquake (Inertia Load in Air)
OBE2 = Operating Basis Earthquake (Inertia Load with Strainer in Water – Include Debris Weight + Hydrodynamic Mass)
TEmax = Thermal Expansion (Accident Condition)
SSE1 = Safe Shutdown Earthquake (Inertia Load with Strainer in Air)
SSE2 = Safe Shutdown Earthquake (Inertia Load with Strainer in Water – Include Debris Weight + Hydrodynamic Mass)

The seismic loads are based on the lateral and vertical accelerations of the response spectrum according to the first mode of frequency of the strainer assembly in water. The natural frequency checks in the original analysis show that the system is in the rigid range. The design pressure, Po or Pd, has no impact on the system because the strainer is an open system. The hydrodynamic mass and debris weight are distributed evenly and are added to the strainer finite element model by adjusting the density of the material.

A combined load table for the strainer component evaluation is summarized in Table 3.k.1-3.

For the design load case, the dry strainer weight (or 1G) is combined with the OBE vertical acceleration for a combined loading of 1.375G vertically. In addition, OBE horizontal acceleration of 0.27G is applied in both X and Y lateral directions.

For the Level B load case, the strainer weight in water including debris and hydrodynamic mass (or 1G) is combined with the OBE vertical acceleration for a combined loading of 1.375G vertically. In addition, OBE horizontal acceleration of 0.27G is applied in both X and Y lateral directions as well as crush pressure and thermal loading.

For the Level D load case, the strainer weight in water including debris and hydrodynamic mass (or 1G) is combined with SSE vertical acceleration for a combined loading of 1.6G vertically. In addition, SSE horizontal acceleration of 0.4125G is applied in both X and Y lateral directions as well as crush pressure.

Table 3.k.1-3 Load Table for the VEGP Strainer Design

Strainer Assembly	Load Combination	Inertia Z* (G)	Inertia X (G)	Inertia Y (G)	Pcr (psi)	Temp*** (°F)
Design	W + Po + OBE1	1.375	0.27	0.27		
Level B	WD + Pd + OBE2 + TEmax + Pcr	1.375	0.27	0.27	4.46**	180
Level D	WD + Pd +SSE2 + Pcr	1.6	0.4125	0.4125	4.46**	

* Axis orientation: Z Vertical, X and Y Lateral

** Equivalent to 10.1 ft of head loss, see Response to 3.k.2 for more information on an increase in crush pressure

*** Stress free temperature is assumed to be 70 °F, $\Delta T = (250 - 70)$ °F = 180 °F

Modal Analysis

Modal analyses were performed using the suction strainer finite element models. Modal results were obtained for the dry strainer and for the wet strainer with added debris weight and hydrodynamic mass during LOCA and post-LOCA events. The strainer structural mass and natural frequencies are calculated for the first four modes and are summarized in Table 3.k.1-4.

Table 3.k.1-4 Replacement Strainer Weight and Frequency

Typical RHR Strainer in Air	**W = 7,150 lbm**
Mode 1	37.311 Hz
Mode 2	37.722 Hz
Mode 3	39.240 Hz
Mode 4	85.327 Hz
Typical RHR Strainer in Water	**WD = 10,655 lbm**
Mode 1	30.566 Hz
Mode 2	30.902 Hz
Mode 3	32.146 Hz
Mode 4	69.901 Hz
RHR Train B Strainer in Water	**WD = 11,256 lbm**
Mode 1	31.421 Hz
Mode 2	32.037 Hz
Mode 3	33.426 Hz
Mode 4	68.966 Hz

Load Application

Loads used in the stress analysis of the strainer models include the weight of the strainer assembly, hydrodynamic mass and debris mass, the crush pressure due to suction strainer operation, and the lateral and vertical inertial accelerations of Response Spectrum (OBE & SSE) corresponding to the first mode frequency of strainer assembly in water.

The crush pressure is applied on the top and bottom surfaces of the disk sets accounting for debris blockage. The weight of the strainer assembly model in water (WD) is the sum of the weight of the strainer assembly in air (W), the debris weight, and the hydrodynamic mass. The debris and hydrodynamic mass are uniformly distributed over the strainer assembly and support for mode shape and stress analysis. The crush pressure is applied on the plenum for Level D load case.

The ASME code combination stress limits are summarized in Tables 3.k.1-5 and 3.k.1-6.

Table 3.k.1-5 Stress Limits for Strainer Components (250 degrees F)

Service Level	Stress Category	Stress Limit (ksi)	
Design	P_m $P_m + P_b$	S 1.5 S	17.15 25.725
Service Level B	P_m $P_m + P_b$ P_m* $P_m + P_b + Q$*	1.1 S 1.65 S S 3 Sm	18.865 28.3 16.35 69.9
Service Level D	P_m $P_m + P_b$	2.0 S 2.4 S	34.3 41.16

S: 17,150 psi for SS304
Sm: 23,300 psi for SS410

Table 3.k.1-6 Weld Stress Limits (250 degrees F)

Type	Service Level	Stress Category	Stress Limit (ksi)	
Fillet	ND-3929 & ND-5260*	Shear	0.85x0.7xS	10,200
Plug	ND-3929 & ND-5260*	Shear	0.65x0.8xS	8,918

S: 17,150 psi for SS304
* No specific weld inspection requirements. VT-visual test inspection will be performed.

2. Summarize the structural qualification results and design margins for the various components of the sump strainer structural assembly.

Response to 3.k.2:

Table 3.k.2-1 Stress Ratio Summary for Strainer Components Based on ASME Subsection NC

Component	Service Level	Stress Ratio*
Perforated Plates	Design – RHR model	18.55
Fingers	Design – RHR model	21.10
Finger Frames	Design – RHR model	40.70
Perforated Spacers	Design – RHR model	17.23
Center Post	Design – RHR model	39.27
Connecting Plates	Design – RHR model	41.69
Support Base	Design – RHR model	16.40
Base Frame	Design – RHR model	16.84
I-Beams	Design – RHR model	16.40
Perforated Plates	Level B – RHR model	2.30
Fingers	Level B – RHR model	3.65
Finger Frames	Level B – RHR model	13.78
Perforated Spacers	Level B – RHR model	10.49
Center Post	Level B – RHR model	26.13
Connecting Plates	Level B – RHR model	14.63
Support Base	Level B – RHR model	9.58
Base Frame	Level B – RHR model	9.58
I-Beams	Level B – RHR model	18.04
Tie Rods	Level B – RHR model	2.38
Perforated Plates	Level D – RHR model	3.33
Fingers	Level D – RHR model	5.30
Finger Frames	Level D – RHR model	19.83
Perforated Spacers	Level D – RHR model	12.44
Center Post	Level D – RHR model	30.11
Connecting Plates	Level D – RHR model	21.09
Support Base	Level D – RHR model	14.96
Base Frame	Level D – RHR model	14.96
I-Beams	Level D – RHR model	14.36
Perforated Plates	Level D – RHR Train B model	3.35
Fingers & Frames	Level D – RHR Train B model	5.21
Perforated Spacers	Level D – RHR Train B model	10.67
Center Post	Level D – RHR Train B model	26.78
Connecting Plates	Level D – RHR Train B model	20.47
Support Base	Level D – RHR Train B model	7.16
Base Frame	Level D – RHR Train B model	9.97
I-Beams	Level D – RHR Train B model	9.40

* Stress Ratio = ASME Code Stress Limit / Calculated Max Stress

Table 3.k.2-2 Stress Summary for Welds based on Service Level D Load

Weld Location (psi)	Weld Stress (psi)	Allowable Stress**	Stress Ratio*
Perforated Plate to Finger	3,708	8,918	2.4
Perforated Plate to Frame	9,016	10,200	1.13

* Stress Ratio = ASME Code Stress Limit / Calculated Max Stress
** Conservative Level A Stress Limits, ASME Code Section III, Subsection ND-3923 at 250 °F

The ASME Code combination stress limits are summarized in Tables 3.k.1-5 and 3.k.1-6.

Table 3.k.2-3 Typical RHR Strainer Stress Ratios for Service Level D

Component	Stress Category	Max Stress Intensity (psi)	Stress Limit (psi)	Stress Ratio*
Perforated Plates	P_m $P_m + P_b$	less than 12,347 12,347	34,300 41,160	2.78 minimum 3.33
Fingers	P_m $P_m + P_b$	less than 7,764 7,764	34,300 41,160	4.42 minimum 5.30
Finger Frames	P_m $P_m + P_b$	less than 2,076 2,076	34,300 41,160	16.52 minimum 19.83
Perforated Spacers	P_m $P_m + P_b$	less than 3,308 3,308	34,300 41,160	10.37 minimum 12.44
Center Post	P_m $P_m + P_b$	less than 1,367 1,367	34,300 41,160	25.09 minimum 30.11
Connecting Plates	P_m $P_m + P_b$	less than 1,952 1,952	34,300 41,160	17.57 minimum 21.09
Support Base	P_m $P_m + P_b$	less than 6,855 6,855	34,300 41,160	5.00 minimum 6.00
Base Frame	P_m $P_m + P_b$	less than 6,855 6,855	34,300 41,160	5.00 minimum 6.00
I-Beams	P_m $P_m + P_b$	less than 5,684 5,684	34,300 41,160	6.03 minimum 7.24

* Stress Ratio = ASME Code Stress Limit / Calculated Max Stress

Table 3.k.2-4 RHR Train B Strainer Stress Ratios for Service Level D

Component	Stress Category	Max Stress Intensity (psi)	Stress Limit (psi)	Stress Ratio*
Perforated Plates	P_m $P_m + P_b$	less than 12,289 12,289	34,300 41,160	2.79 minimum 3.35
Fingers & Frames	P_m $P_m + P_b$	less than 7,894 7,894	34,300 41,160	4.35 minimum 5.21
Perforated Spacers	P_m $P_m + P_b$	less than 3,856 3,856	34,300 41,160	8.90 minimum 10.67
Center Post	P_m $P_m + P_b$	less than 1,537 1,537	34,300 41,160	22.32 minimum 26.78
Connecting Plates	P_m $P_m + P_b$	less than 2,011 2,011	34,300 41,160	17.06 minimum 20.47
Support Base	P_m $P_m + P_b$	less than 5,750 5,750	34,300 41,160	5.97 minimum 7.16
Base Frame	P_m $P_m + P_b$	less than 4,130 4,130	34,300 41,160	8.31 minimum 9.97
I-Beams	P_m $P_m + P_b$	less than 4,380 4,380	34,300 41,160	7.83 minimum 9.40

* Stress Ratio = ASME Code Stress Limit / Calculated Max Stress

Weld Analysis

Since the finite element model with the typical RHR strainer configuration has slightly higher overall stress results, the ANSYS analysis results in this load case were used to calculate the load transfer through the welds. For a given weld location, the elements and corresponding nodes at the weld were selected on one side of the node, and the ANSYS post-processor was used to calculate the forces transferred across the weld section. These forces were then used to calculate the stresses based on the weld section properties. If the welds consisted of more than one weld, then the group section properties were used.

Weld stresses were calculated for simultaneous application of loads for Service Level D. These calculated stress values were compared with the ASME Code shear stress limits. The minimum weld stress ratios for all the weld locations are summarized in Tables 3.k.2-1 and 3.k.2-2.

For the welds between the fingers and perforated plate, robotic welding will be utilized to ensure a weld diameter of 3/16 inches. At the worst stress intensity finger location, a net shear Fx of 115.8 pounds force (lbf), a net shear Fy of 184 lbf, and a net tensile Fz of 184 lbf are obtained between two sides of the finger as seen in Figure 3.k.2-1.

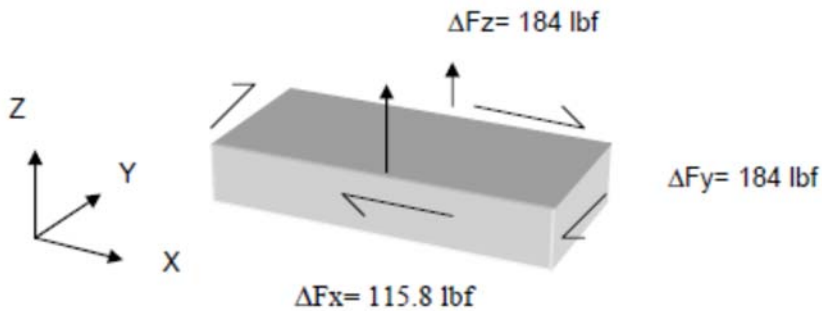

Figure 3.k.2-1 Worst Stress Intensity Finger Location Free Body Diagram

Considering a line of welds, net forces are reacted by the circular areas of the plug weld, Aw. The unbalanced Fz causes a moment of 30 inch-pounds force (in-lbf) and is reacted by the section modulus of the weld, Sw, when the weld is treated as a line. The unbalanced Fx and Fy cause torsion and are reacted by the twisting property of the weld, Jw. Jw is large because the line of weld is approximately 8 inches long. The stresses caused by torsion are therefore negligible.

Weld load treated as a line:

$$f = \sqrt{\left(\frac{\mathrm{M}}{\mathrm{Sw}} + \frac{\mathrm{Fz}}{\mathrm{Aw}}\right)^2 + \left(\frac{\mathrm{Fx}}{\mathrm{Aw}}\right)^2 + \left(\frac{\mathrm{Fy}}{\mathrm{Aw}}\right)^2}$$

$\mathrm{Sa} = \frac{\mathrm{f}}{\mathrm{Nt}}$ where Sa = 8,918 psi and t = 0.078 inches

Five welds along each finger will satisfy the stress allowable of 8,918 psi. The welds are to be distributed evenly along the finger.

Similarly, at the weld location between the finger frame and the perforated plate, a net shear Fx of 180 lbf., a net shear Fy of 325 lbf, and a net tensile Fz of 1,437 lbf are obtained between two sides of the frames. The moment from the unbalanced Fz is 566 in-lbf and is reacted by Sw of 1,200 in² for the square frame shape weld line. The fillet weld area, Aw, is 0.707 x 2t, where t equals 0.078 inches. In addition, an intermittent weld has a knock down factor of 0.66 for weld length of 3 inches and pitch distance of 5 inches. The weld calculation shows that 18 inches of weld length is recommended along each edge of the disk. The fillet welds should cover corners and at finger protrusion areas. Based on a width of 0.070 inches for the weld and stress allowable of 10,200 psi, the recommended intermittent welds should be 3 inches with 5-inch pitch.

Interface Load

The original Vogtle strainers were re-evaluated when the original 16 bolt anchor configuration was revised to a 32 bolt anchor configuration. The updated anchor bolt loads from the simplified GEH finite element model are documented in GEH letter no. JXDR7-2006-02, Rev. 1. The worst case anchor loads were for Level D load cases (Wd + SSE2). The largest tension load (z-direction) that an individual anchor bolt sees is 195 lbf. The largest lateral X-direction force is 237 lbf, and the largest Y-direction force is 279 lbf. The worst overall anchor bolt interaction ratio for each load case is provided below in Table 3.k.2-5.

Table 3.k.2-5 Worst Case Interaction Ratios (I.R.) for Anchor Bolts

	Fx (lbf)	Fy (lbf)	Fz (lbf)	I.R.*
WD + SSE2 (downward)	152.4	167.59	-195.4	0.423
WD + SSE2 (upward)	137.91	278.26	-38.734	0.350

*The Interaction Ratio (I.R.) is the inverse of a stress ratio in the tables above and equals the resultant force per anchor divided by the allowable force per anchor.

Justification for Increased Crush Pressure

The results and methodology presented above are based on the structural qualification performed by the hardware vendor, GEH, prior to the strainer installation. The crush pressure presented in Table 3.k.1-3 was an assumed value based on what was believed to be adequate at that time. As part of the change to a risk-informed methodology, GEH revisited the stress model and results to determine if a higher crush pressure could be justified. As shown in Table 3.k.2-2, the most limiting stress is in the welds between the perforated plate to finger and the perforated plate to frame for the Service Level D Load combination. Linear scaling was used conservatively assuming that total stress is scalable with crush pressure. The resulting max allowable crush pressure is 10.7 psi for the plate to finger weld condition, and 5.03 psi for the plate to frame weld. However, this approach is too conservative and not appropriate for the plate to frame weld. A single disk FEA model was run using ANSYS with the same mesh as the original analysis and determined that the impact of the crush pressure has very limited impacts to the stress of the plate to frame weld. Using this new model, the allowable crush pressure for the plate to frame weld was calculated to be 30.47 psi. This concluded that the limiting factor for the crush pressure was the plate to finger weld which has a max allowable crush pressure of 10.7psi.

Using ASME Code Section III Level D allowables, and the methodology above, the crush pressure was determined to be 10.39 psi for a 15-disk strainer, compared to 10.7 psi for the 18-disk strainer. The lower crush pressure for the 15-disk strainer is due to the larger debris weight for each disk (same total debris weight with fewer disks). Therefore, a crush pressure of 10.39 psi (24.0 ft) was conservatively used for the final configuration of 16 disks.

16-Disk ECCS Suction Strainer Summary

[[

{3}]]

Table 3.k.2-6: Service Level D Stress Summary for 16-Disk Strainer

[[				

{3}]]

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:

As shown in Figure 3.a.1-1, there is only one line outside of the steam generator compartments with analyzed breaks (i.e. welds inside the first isolation valve). However, this 2" line is at an elevation of 208 ft, which is 32 ft above the strainer at approximately 176 ft. Thus, pipe whip from this line impacting the strainer is not considered a credible scenario. The strainers are seismically qualified, robust structures designed with a crush pressure of 10.39 psi, which is approximately the impingement pressure at 11.5 pipe diameters from a break (Reference 3). Considering the distance from the analyzed break, jet impingement loads are not a credible concern. Finally, the line in question is on the side of the pressurizer cubicle wall, where no unsecured items would be located. Therefore, missiles generated by this analyzed break are not a credible concern.

There are no other high-energy lines in the area of the emergency sumps except for the RHR and HHSI lines that are used for accident mitigation and are not assumed to be the accident initiator.

The 12-inch RHR hot leg recirculation line is located more than 6 ft above the Train B CS strainer outside of the secondary shield wall. The RHR suction header has two isolation valves in series to isolate it from the RCS hot leg recirculation line. The valves are normally closed except when the RHR is operating. The inboard isolation valve is on the other side of the secondary shield wall inside the steam generator

compartment preventing high-energy RCS discharge outside of the secondary shield wall. Therefore, there is no possibility of pipe whip impacts or jet loads associated with this pipe. In addition, this line is only pressurized during shutdown, refueling, and accident mitigation. During normal operation, this line is not pressurized. Thus, this line is not considered for evaluation as a postulated location for a high-energy line break accident.

The 6-inch HHSI line is located approximately 69 inches above the Train B RHR strainer outside of the secondary shield wall. There is a check valve on the other side of the secondary shield wall inside the steam generator compartment preventing high-energy RCS discharge outside of the secondary shield wall. Therefore, there is no possibility of pipe whip impacts or jet loads associated with this pipe. In addition, this line is only pressurized during accident mitigation. During normal operation, this line is not pressurized. Thus, this line is not considered for evaluation as a postulated location for a high-energy line break accident.

The strainers are located outside of the steam generator compartments and inside the outer containment wall. Therefore, the strainers are adequately protected from the hazardous effects of missiles.

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:

Backflushing of the sump strainers, or any other active approach, is not credited in the VEGP analysis. The RHR strainer suction lines have check valves to prevent reverse flow through the strainers. If containment pressure is high enough to actuate containment spray, the pressure is high enough to prevent reverse flow through the CS strainers. Therefore, no structural analysis considering reverse flow is required.

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.*

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 following areas / items were considered as part of the evaluation to determine potential choke points for flow upstream of the sump:

   Refueling Cavity

   Evaluations of containment, along with review of the CFD model, indicated no significant areas would become blocked with debris and hold up water during the sump recirculation phase. The area of the refueling cavity, which is the area around the reactor head that is flooded prior to fuel movement, is the only significant area in containment that can retain water during an event that requires containment spray. However, this area is drained by a large clear flow path that cannot be easily blocked with debris. See the Response to 3.I.4 for additional information.

   Inside Secondary Shield Wall

   A postulated LOCA inside the secondary shield wall in the lower elevations of the containment was considered limiting with respect to flow restrictions upstream of the sump. The flow path from this break area to the sump strainers is primarily through two labyrinth-like walkways through the shield wall. There are also smaller openings through the shield wall for piping, but these are much smaller than the walkways. The walkways provide a large clear flow path from inside the shield wall to the screen area. In addition, any restriction of the smaller through-wall piping openings would have minimal effect on the overall flow path to the strainers, since water would simply flow through the open walkways.

Containment Spray Washdown

Containment spray washdown has a clear path to the containment sump area. Large sections of the floor on each level in containment are covered with grating that allows the water to pass.

A complete evaluation of containment, along with a review of the CFD model, indicated no significant areas would become blocked with debris and hold up water during the sump recirculation phase.

2. Summarize measures taken to mitigate potential choke points.

Response to 3.I.2:

Per the Response to 3.I.1, no measures were necessary to mitigate potential choke points.

3. Summarize the evaluation of water holdup at installed curbs and/or debris interceptors.

Response to 3.I.3:

There are no curbs or debris interceptors that provide water volume holdup in the VEGP containments.

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 cavity is drained by two 12-inch pipes. During refueling, these drains are secured by installing flanges. These flanges are removed prior to entry into Mode 4 and above. The VEGP limiting break with respect to upstream flow blockage occurs under the operating deck and inside the secondary shield wall. This break would result in a torturous path for large debris to travel above the operating deck and land in the refueling cavity. Large debris would have to travel through the RCP access ports or the Steam Generator and Pressurizer cubicles. The RCS access ports are covered with grating, and there are significant structural elements that would prevent any large pieces of debris from entering upper containment. The same is true for the path through the Steam Generator and Pressurizer cubicles. Each cubicle contains several levels of grating and significant structural elements that would make large pieces of debris entering upper containment via these paths highly unlikely. Note that a break in a pipe at the top of the pressurizer compartment could allow debris to transport to the refueling cavity without passing through grating. However, since the largest source of insulation in this area is on the pressurizer, which would be below the break, the debris would

tend to be blown downward with minimal transport of any large pieces out of the pressurizer compartment. Therefore, the clogging of the refueling cavity drains is minimized.

The drains into the area under the reactor (e.g., reactor cavity) could become blocked. For breaks outside of the reactor cavity, there is no detrimental impact of this blockage, as it would inhibit loss of water from the active ECCS sump to an inactive area beneath the vessel. The flooding analysis assumes this area floods during the event. For breaks inside the reactor cavity, there is also no detrimental impact of this blockage, since the majority of the flow from the break would travel to the ECCS sump through the hot leg and cold leg penetrations.

m. Downstream Effects – Components and Systems

The objective of the downstream effects, components and systems section is to evaluate the effect 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 2004-02 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.*

1. If NRC-approved methods were used (e.g., WCAP-16406-P-A 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:

   The following methodology was employed in the ex-vessel downstream effects evaluations. The evaluations did not use any unapproved methods or take any exceptions to NRC-approved methods.

Maximum Debris Ingestion Determination

Debris blockage of flow restrictions in the ECCS and CSS flowpaths downstream of the sump screen was addressed within the downstream effects evaluations for ECCS valves and equipment. Each unit has two sets of screens – RHR and CS emergency sump screens. The adequacy of the sump screens' mesh spacing or strainer hole size [nominal hole diameter of 0.09375 inches (3/32 inches)] is conservatively addressed by assuming that the maximum amount of particulate (coatings and latent debris) transported to the strainers passes through the strainers. Additionally, the evaluation used a quantity of fiber debris that passes through the strainers (100 g/FA), which is greater than the maximum total reactor vessel fiber load amount shown for a hot-leg break in Table 3.n.1-6. The ex-vessel downstream effects evaluations were based on this maximum amount of ingested debris (see Initial Debris Concentrations below).

The Unit 1 strainers were inspected after installation and found to conform to design specifications. No adverse gaps or breaches were found on the screen surface.

The Unit 2 strainers were inspected upon installation, and deficiencies in the fabrication of the Unit 2 CS sump screens were discovered; specifically, there were 124 holes greater than the nominal specified sump screen hole-diameter of 0.09375 inches (3/32 inches). No holes greater than a 0.25-inch diameter were found in the Unit 2 strainers; therefore, ingestion of debris will not cause plugging of downstream CS components because the smallest component diameter is 0.375 in.

Initial Debris Concentrations

Initial debris concentrations were developed using the assumptions and methodology described in Chapter 5 of WCAP-16406-P-A. Additionally, for conservatism, the maximum amount of particulate (coatings and latent debris) transported to the strainer were assumed to pass through the strainer. The total maximum initial debris concentration was determined to be 919.17 ppm, with fiber debris contributing 11.61 ppm, and particulate and coating debris contributing 907.56 ppm.

Flowpaths and Alignment Review

Both trains of the RHR system, SI system, component cooling system (CCS), and CSS were reviewed to ensure that all of the flowpaths and components impacted by the debris passing through the sump screens were considered. Documents used for this effort included piping and instrumentation diagrams (P&IDs), vendor manuals, equipment specifications, and other documents as applicable.

Component Blockage and Wear Evaluations Methodology

All component evaluations were performed based on WCAP-16406-P-A. Components addressed in the evaluations include pumps, heat exchangers, orifices, spray nozzles, instrumentation tubing, system piping, and valves required for the post-LOCA recirculation mode of operation of the ECCS and CSS. The evaluations included the following steps:

[

]a,c

2. Provide a summary and conclusions of downstream evaluations.

Response to 3.m.2:

Summary and Conclusions of Downstream Evaluations

The following is the summary of results and conclusions of the downstream effects evaluations:

ECCS/CSS Pumps

For pumps, the effects of debris ingestion through the sump screen on three aspects of operability (hydraulic performance, mechanical-shaft seal assembly performance,

and mechanical performance) were evaluated. The hydraulic and mechanical performances of the ECCS and CSS pumps were determined to be unaffected by the recirculating sump debris. The mechanical shaft seal assembly performance evaluation resulted in one action item with the suggested replacement of the RHR pumps' carbon/graphite backup seal bushings with a more wear-resistant material, such as bronze. However, because VEGP has an engineered safety feature (ESF) atmospheric filtration system in its auxiliary building, this action is not required per WCAP-16406-P-A.

ECCS/CSS Heat Exchangers, Orifices, Spray Nozzles, and System Piping

Heat exchangers, orifices, spray nozzles, and system piping were evaluated for the effects of erosive wear for an initial concentration of 919.17 ppm over the mission time of 30 days. The erosive wear on these components was determined to be insufficient to affect system performance.

The smallest clearance found for VEGP heat exchangers, orifices, spray nozzles, and system piping in the ECCS recirculation flow path is 0.188 inches, for the SI pressure breakdown orifices. Therefore, no blockage of the ECCS flow path is expected with the current ECCS sump screen hole size of 0.09375 inches. The smallest clearance found in the CSS recirculation flow path is 0.375 inches, for the CS nozzles. No blockage of the CSS flow path is expected because the maximum debris size able to bypass the CSS sump screens is 0.25 in.

System piping was evaluated for plugging based on system flow and material settling velocities. For all piping, the minimum flow velocity was found to be greater than 0.42 ft/s, the minimum velocity required to prevent debris sedimentation. All system piping passed the acceptable criteria for plugging due to sedimentation.

ECCS/CSS Valves

Valves were evaluated for plugging impact in the downstream effects evaluations. Valves that were determined to be "Not Critical" did not warrant further evaluation, but those valves identified as "Evaluation Required" received a more detailed evaluation. It was determined that all valves passed the acceptance criteria for the plugging evaluation.

Valves were evaluated for debris sedimentation. Valves identified as "No Evaluation" did not require additional analysis, but valves identified as "Evaluation Required" were analyzed further. The line velocity for all valves analyzed was found to be greater than [ ]^{a,c} thus, debris sedimentation was not an issue.

Valves were evaluated for wear impact. Valves determined to be "Not Critical" did not warrant further evaluation, but valves identified as "Evaluate" were analyzed further. It was found that four throttle valves did not pass the acceptance criteria of [ ]^{a,c} Because of this,

they were evaluated for additional opening. This additional opening is necessary for the valves to demonstrate acceptable wear. The turns open value (from a fully closed position) was calculated and rounded to the nearest half turn. With the new calculated turns open value, the valves passed the acceptance criteria of [ ]a,c

ECCS/CSS Instrumentation Lines

Instrumentation tubing was evaluated for debris settling. It was found that the largest settling velocity of debris source material found inside a PWR containment is [ ]a,c as provided by WCAP-16406-P-A. Therefore, as long as the recirculation flow velocity though the ECCS and CSS is greater than [ ]a,c failure of instrumentation due to debris settlement will not occur. The evaluation showed that all flow velocities are greater than [ ]a,c

3. Provide a summary of design or operational changes made because of downstream evaluations.

Response to 3.m.3:

As noted in the Response to 3.m.2, an adjustment to the minimum opening of four throttle valves was required in order to resolve erosion concerns. The minimum turns open value required to demonstrate acceptable valve wear was implemented into plant procedures.

The results of the VEGP downstream effects evaluations demonstrate that the evaluated components are acceptable for the expected mission time.

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 screens and into the reactor vessel has on core cooling.

1. Show that the in-vessel effects evaluation is consistent with, or bounded by, the industry generic guidance (WCAP-16793-NP), 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:

In-vessel downstream effects for VEGP were evaluated per the methodology in WCAP-16793-NP-A (Reference 22) and the associated NRC SE (Reference 94) using assumed values for in-vessel debris accumulation limits. The evaluation included the following:

1. Peak cladding temperature (PCT) due to deposition of debris on fuel rods (WCAP-16793-NP-A).
2. Deposition thickness (DT) due to collection of debris on fuel rods (WCAP-16793-NP-A).
3. Amount of fiber accumulation at the reactor core inlet and inside the reactor vessel (limits assumed for the purpose of exercising the risk-informed methodology).

These analyses concluded that post-accident long-term core cooling (LTCC) will not be challenged by deposition of debris on the fuel rods, accumulation of debris at the core inlet, or accumulation of debris in the heated region of the core for all postulated LOCAs inside containment. A brief summary of the relevant testing and analyses is provided below as it was used to inform the WCAP evaluations.

VEGP Fiber Penetration Testing

VEGP conducted fiber penetration testing in 2015. The purpose of the testing was to collect time-dependent fiber penetration data of a prototypical VEGP strainer under various conditions (e.g., approach velocity, water chemistry) and strainer configurations (e.g., number of strainer disks). The test results were used to derive a model that can be used to quantify fiber penetration for the RHR and CS strainers at plant conditions. Twelve penetration tests were conducted, nine of which are useful to inform the resolution of GL 2004-02. Within those nine tests, the approach velocity, the water chemistry, and the number of strainer disks were varied to investigate their effects on fiber penetration.

Test Loop Design

The test loop consisted of a metal test tank, which housed a test strainer at its downstream end. Water was circulated by a pump through the test strainer, a fiber filtering system, and various piping components. The test tank had a flume geometry, as shown in Figure 3.n.1-1. Debris was introduced in the high-agitation region. This region was equipped with two mechanical mixers to create adequate mixing and prevent the debris from settling. Mixing inside the low-agitation region was created by directing a portion of the returning flow through the perforated bottom plate of the region. This mixing motion kept fiber in suspension without disturbing the fiber bed on the strainer. The strainer region was designed such that the spacing between the test strainer and tank walls imitated the gaps between adjacent strainer stacks, or between the nearest object inside the containment and the strainer. The spacing between the strainer and tank walls was also designed to minimize settling of debris.

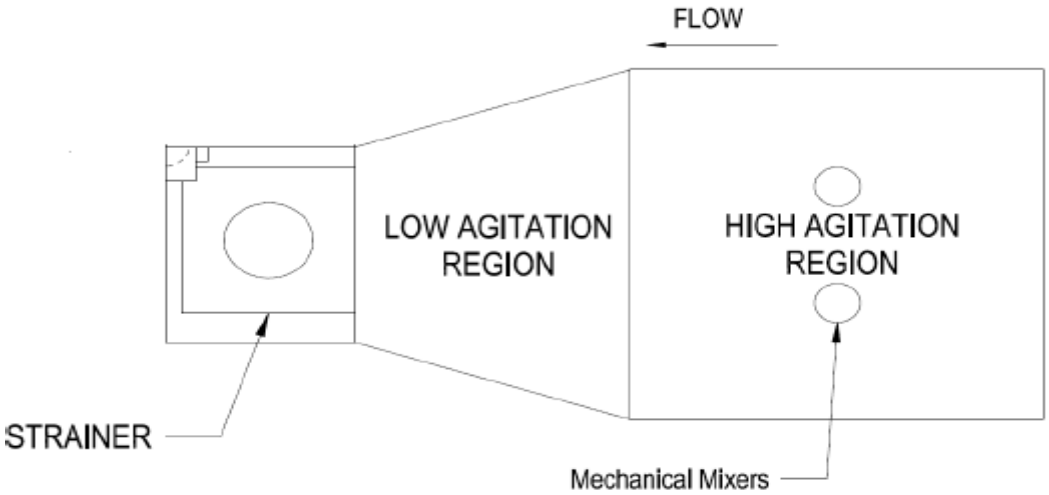

Figure 3.n.1-1 General Arrangement of Test Tank

The effectiveness of the agitation regions is displayed in Table 3.n.1-1, which documents the quantity of fiber that did not transport to the strainer and was collected from the high or low agitation regions after the conclusion of each test.

Table 3.n.1-1: Summary of Useful Fiber Transport Tests

Test #[1]	Gross Fiber Added (g)	Non-Transported Fiber (g)	Net Fiber Added (g)	% of Fiber Transport
1	17350.2	236.7	17113.5	98.6%
2	17350.2	397.9	16952.3	97.7%
3	11403.1	203.9	11199.2	98.2%
4	11401.4	205.2	11196.3	98.2%
5	14375.5	190.6	14184.9	98.7%
6	14466.6	171.9	14294.7	98.8%
7	17281.3	578.1	16703.2	96.7%
8	17281.3	278.3	17003.1	98.4%
10	14375.6	0.0	14375.6	100.0%

[1] The test numbers shown correspond to the number assigned to each test in the VEGP fiber penetration test report. Note that Tests 9, 11, and 12 are not shown because they were the three tests that were not applicable to this resolution.

Test Strainer

The test strainer was a prototypical strainer stack; the only difference was the number of disks installed on the test strainer. While the VEGP RHR and CS strainer stacks consist of 16 and 14 disks, respectively, the number of disks on the test strainer was varied among 12, 15, and 18 disks for different tests. The testing flow rate and debris load for each test were determined based on the area ratio of the test strainer to the prototypical strainer assembly, which was varied according to

the number of disks on the test strainer. As a result, the debris load per unit strainer area and water approach velocity were preserved among tests with varying number of strainer disks. This approach allowed the effect of varying the number of strainer disks on penetration quantities to be investigated separately. Though the number of disks for the tested strainer did not exactly match the plant strainers, the tested configurations bounded the plant configurations. As discussed later in this response, the results from those tests with different number of disks were interpolated to derive the penetration model for the plant strainer configurations.

Debris Types and Preparation

Nukon was the only debris type used in testing. This is appropriate because the only other type of fibrous debris in containment, fire barrier material, transports in negligible amounts of fine fiber. All Nukon used in testing was introduced as fines. Nukon fines were prepared according to the NEI protocol (Reference 46) and the prepared fiber was predominantly Class 2 fiber as defined in NUREG/CR-6224 (Reference 29). Nukon sheets, with an overall thickness of 2 inches, were baked single-sided into approximately half the thickness. The heat-treated sheets were then cut into cubes and weighed out according to batch size. Batches were then pressure-washed with test water following the NEI protocol (Reference 46).

Debris Introduction

Debris was introduced in eight separate batches of increasing batch size for each test. The first two batches corresponded to a theoretical uniform bed thickness of 1/16 inch. The third through seventh batches corresponded to a theoretical uniform bed thickness of 1/8 inch. The final batch corresponded to a theoretical uniform bed thickness of 1/4 inch. Although the batching size increased after the first two batches, this had little effect on the test results because the fiber concentration was maintained sufficiently low. Small-scale fiber penetration testing performed by VEGP demonstrated that fiber concentrations ranging between 0.00052 and 0.0177 ft^3 of fiber/ft^3 of water in the test tank had insignificant effects on fiber penetration. During the large-scale fiber penetration testing, the debris concentration was controlled by adjusting the addition time of each batch. The peak fiber concentrations were controlled below ~0.005 ft^3 of fiber/ft^3 of water for all batches. This concentration is enveloped by the range of concentrations studied in the small-scale testing. Therefore, it is reasonable to conclude that the batching size had little effect on the measured fiber penetration.

The total debris load used for each test was equivalent to a theoretical uniform bed thickness of 1 inch. This debris load was chosen because it was sufficient to circumscribe the test strainer. Subsequent debris addition after the development of a circumscribed debris bed would not result in an appreciable amount of penetration. Note that the debris introduction rate was controlled to maintain a prototypical debris concentration in the test tank.

Debris Capture

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 testing.

Fibers that passed through the strainer were collected by the fiber filtering system downstream of the test strainer. The filtering system had 5-micron filter bags installed in filter housings. All of the flow downstream of the strainer travelled through the filter bags before returning to the test tank. The capture efficiency of the filter bags was verified to be above 95 percent. The filtering system allowed the installation of two filter bags in parallel lines such that a filter bag could be left online at all times, even during periods in which filter bags were swapped.

Before and after each test, all of the filter bags required for the test were uniquely marked and dried, and their weights were recorded. The weight gain of the filter bags was used to quantify fiber penetration. After testing, the debris-laden filter bags were rinsed with DI water to remove residual chemicals before weighing. For each bag, the drying and weighing process was repeated until two consecutive bag weights (taken at least 1 hour apart) were within 0.05 g of each other.

A clean filter bag was placed online before a debris batch was introduced to the test tank and was left online for a minimum of five pool turnovers (PTOs) to capture the prompt fiber penetration. For Batches 1 and 3, two additional filter bags were used to capture the fiber penetration due to shedding. Before further debris addition, a visual confirmation was required to verify that all introduced debris had transported to the strainer. 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 fiber quantity on the strainer.

Test Parameters

The test water used for fiber penetration testing had a chemical composition prototypical to VEGP. The plant conditions selected for testing were those of minimum and maximum boron concentrations and the corresponding buffer (trisodium phosphate, TSP) concentrations. The low boron concentration was taken from an SBLOCA event, and the high boron concentration was taken from an LBLOCA with the CSS active. For the low boron concentration, a corresponding high plant TSP concentration was used, and the high boron concentration was coupled with a corresponding low plant TSP concentration. The chemical concentrations used in testing are shown in Table 3.n.1-2. Test water was prepared by adding chemicals to DI water until the prescribed concentrations were achieved.

Table 3.n.1-2: Summary of Penetration Test Water Chemistry

Chemical	High Level	Low Level
Boron (ppm)	2,522	2,169
TSP (ppm)	2,181	3,147

Several different strainer approach velocities, ranging between 0.0043 ft/s and 0.0130 ft/s, were determined from plant operating conditions and were used for the VEGP fiber penetration testing. As shown in Table 3.f.3-1, the 16-disk RHR strainers have a surface area of 677.6 ft², and the 14-disk CS strainers have a surface area of 590 ft². The design flow rates of the RHR and CS pumps are 3,700 gpm and 2,600 gpm, respectively. The average approach velocities of the RHR and CS strainers are therefore 0.0122 ft/s and 0.0098 ft/s, respectively. Both of these velocities were enveloped by the range tested.

In addition to water chemistry and approach velocity, the number of disks was also varied. A test matrix was designed to quantify fiber penetration for different combinations of test conditions. The test matrix is displayed in Table 3.n.1-3. Only the nine tests that are applicable to plant design conditions are shown.

Table 3.n.1-3: Large Scale Penetration Test Matrix

Test #	Approach Velocity (ft/s)	No. of Disks	Boron / TSP Concentration (ppm)
1	0.0130	18	2,522 / 2,181
2	0.0043	18	2,522 / 2,181
3	0.0130	12	2,522 / 2,181
4	0.0043	12	2,522 / 2,181
5	0.00314	15	2,522 / 2,181
6	0.0087	15	2,169 / 3,147
7	0.0043	18	2,169 / 3,147
8	0.0087	18	2,522 / 2,181
10	0.0087	15	2,522 / 2,181

Strainer Penetration Model Development

Data gathered from VEGP fiber penetration tests were used to develop a model for quantifying the RHR and CS strainer fiber penetration under prototypical plant conditions. Given the different characteristics and flow rates of the RHR and CS strainers, separate formulas were derived for the two strainers. 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 summation of the developed equations can be used to describe total fiber penetration. The equations contain coefficients whose values were determined separately for each test based on the test results.
- The results for each test were fit to the governing equations using various optimization techniques to refine the coefficient values. This produced a unique set of equations and thus a unique penetration model for each test. Figure 3.n.1-2 compares the curve fit with the test data for Test 8. As the figure shows, it is the summation of the prompt and shedding penetration curves that was fit to the test data. Since three parameters were varied during testing (approach velocity, water chemistry, and number of disks), the fitting coefficients are functions of these three parameters.

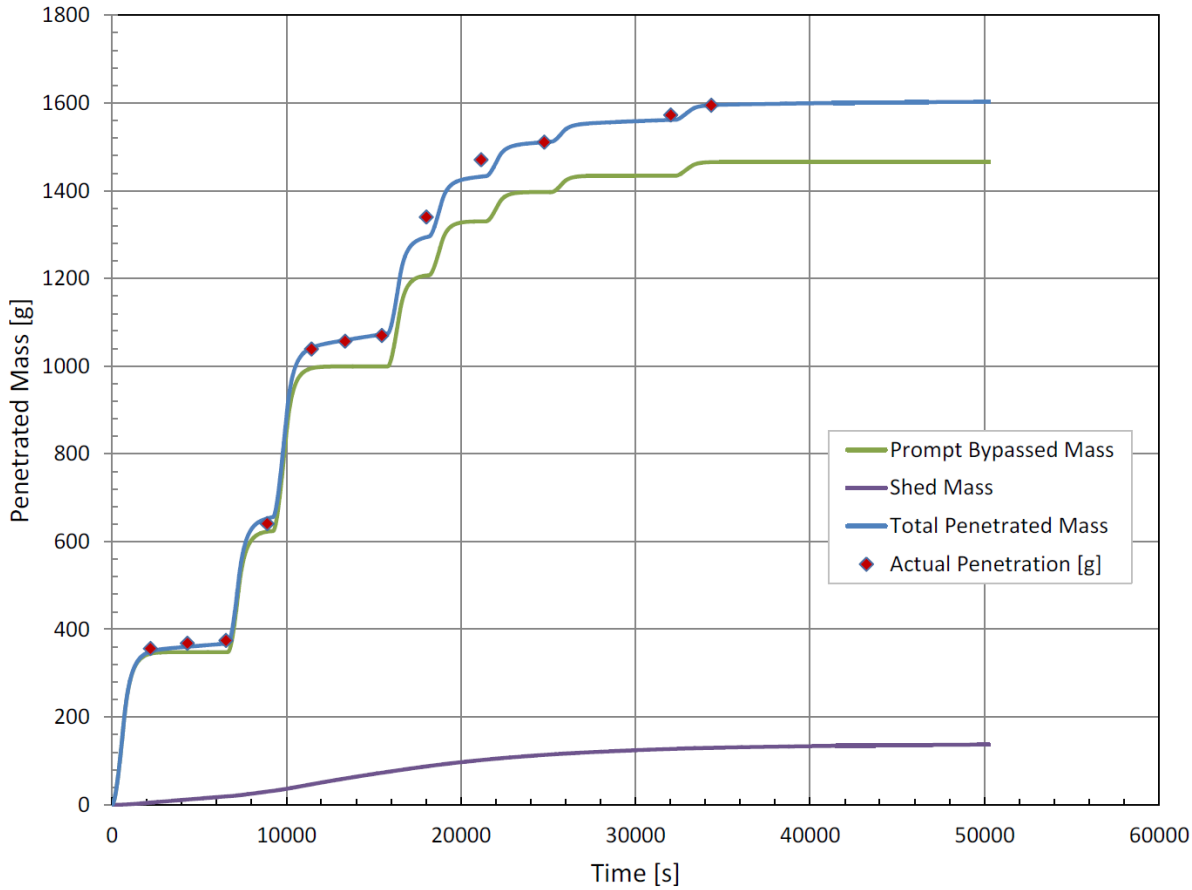

Figure 3.n.1-2: Test 8 Penetration Model Fit

- The test models from the previous step were then used to develop separate models for the RHR and CS strainers, respectively, by interpolating from the model coefficients of each test. During this process, different weighting factors were applied to each test model based on the similarity of the test conditions to the conditions of the actual RHR and CS strainer configurations (as shown in Table 3.n.1-4). Since the RHR and CS strainer conditions are different, each test was weighted separately for the RHR and CS model according to the similarity of its conditions to those of the RHR and CS strainers. Note that the RHR strainer configuration modeled below is based on the strainer modification described in the Response to 3.j.1. The effects of water chemistry on the penetration model were accounted for differently. Instead of interpolating between tests, the low-boron/high-TSP tests were used to represent the water chemistry of interest and were weighted more heavily when calculating the model parameters. This is conservative, firstly, because the large-scale fiber penetration test report showed that tests using low-boron and high-TSP concentrations resulted in higher fiber penetration under otherwise identical conditions, as shown in Figure 3.n.1-3. Secondly, the low-boron water chemistry conditions used in testing were determined with the intent of finding the

minimum sump boron concentration by using an unlikely combination of conservative inputs.

Table 3.n.1-4: RHR and CS Strainer Model Parameters

Strainer Stack	Disk #	Velocity (ft/s)	Boron (ppm)
RHR	16	0.0122	2,169
CS	14	0.0098	2,169

The resulting RHR fiber penetration model was applied to prototypical plant conditions to calculate the total fiber penetration as a function of time, which is shown as the dark solid line in Figure 3.n.1-3. The curve fit models for the actual test cases were also applied to the same plant conditions, and the results are shown in Figure 3.n.1-3 as lines of different colors. It should be noted that the bracketed values shown in the legend of the figure are the parameters used for each test, not the model conditions, which are common for each curve. As shown in the figure, the model developed for the RHR strainer provides a higher total penetrated fiber quantity than all of the test conditions. This is expected because the RHR strainer model was developed with high approach velocity and low boron concentration (referred to as “low chem” in the figure legend), which are the most conservative values for those parameters, and none of the tests were run with that combination. As shown in the figure, high approach velocity and low boron concentration, or “low chem”, increase fiber penetration under otherwise identical conditions.

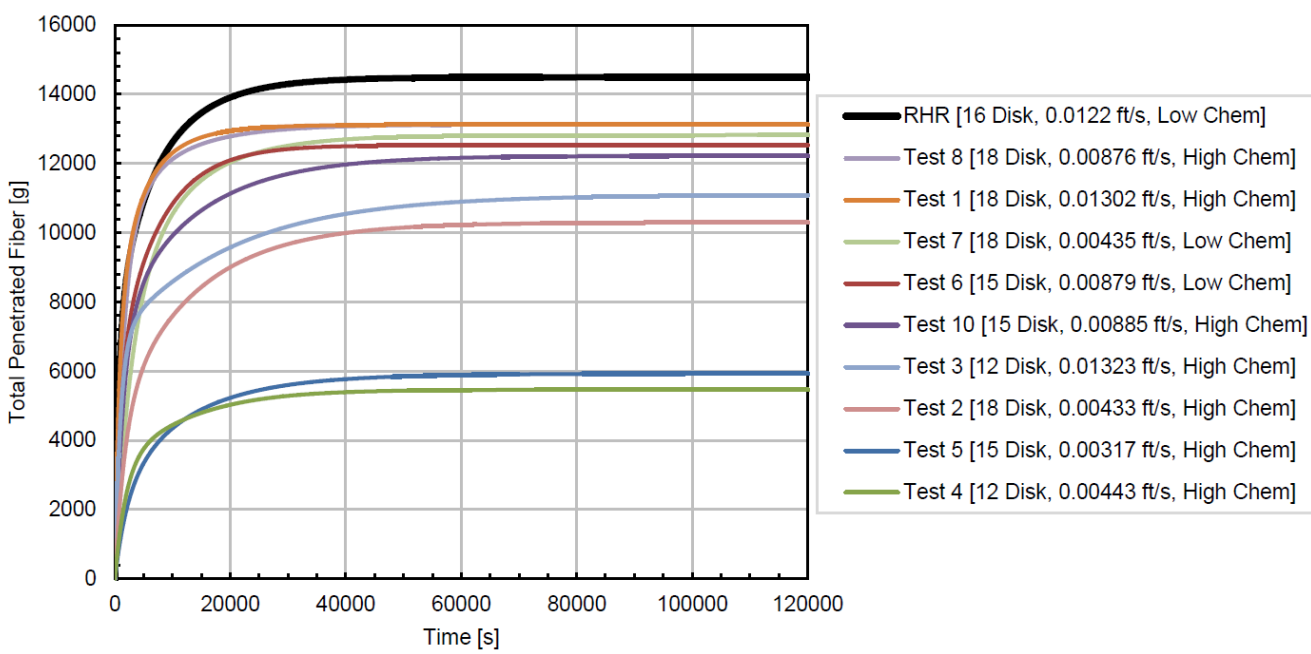

Figure 3.n.1-3: Comparison of RHR Model with Test Cases at Plant Scale

Figure 3.n.1-3a shows another comparison between the predicted fiber penetration quantities using the RHR strainer penetration model and the testing results. In the figure, the cumulative fiber penetration quantities are plotted as functions of the quantity of fiber added to the test tank. All debris quantities shown in the figure are normalized by the number of strainer disks. The RHR penetration model was applied to the plant RHR strainer approach velocity and disk number, along with the debris addition sequence of Test 1, Test 3, and Test 8 (e.g., debris introduction timing, duration and quantity). The results are shown in the figure as the thick solid lines. The measured fiber penetration quantities are shown for all 9 tests as the thin solid lines, identified by different markers. As shown in the figure, the predicted fiber penetration quantities for each test bound the measured data points of the corresponding test. Therefore, applying the fiber penetration model is conservative for quantifying fiber penetration for the evaluation of in-vessel downstream effects.

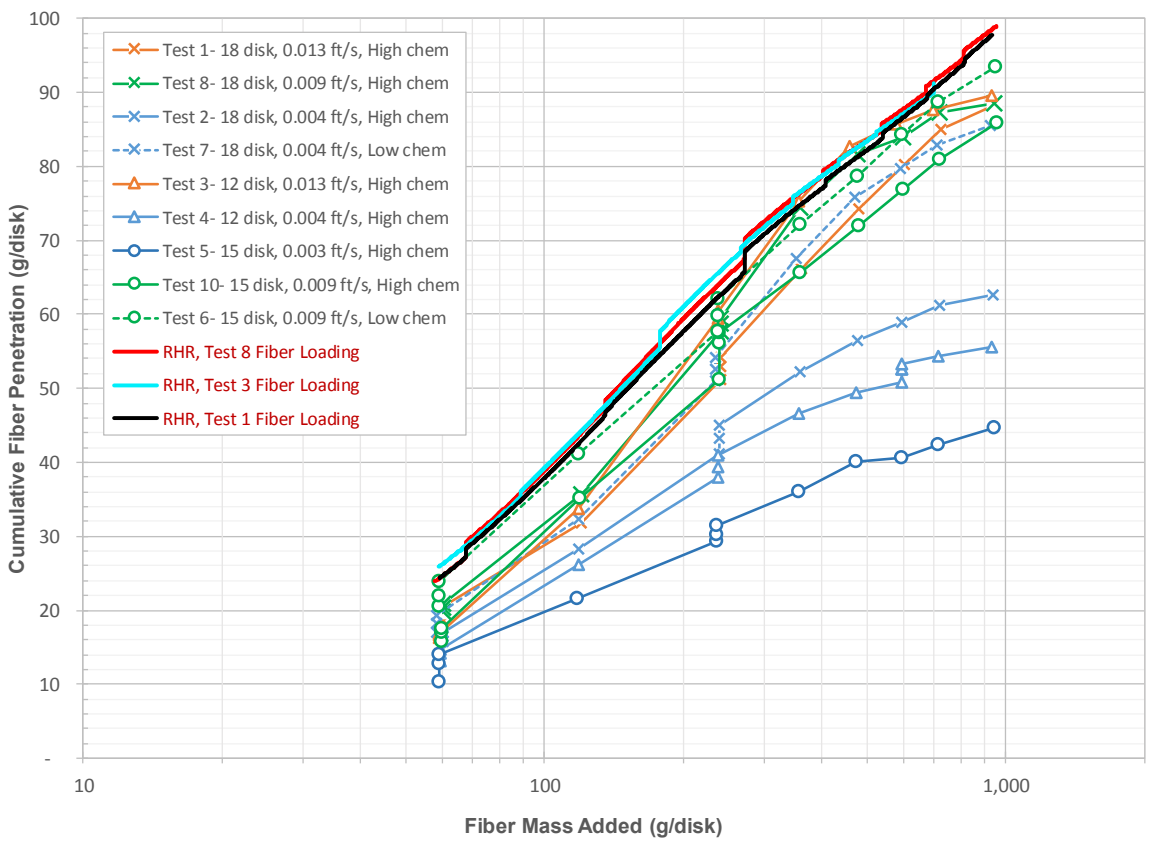

Figure 3.n.1-3a: Comparison between Predicted Fiber Penetration Quantities and Test Results

It should be noted that the 95% confidence interval uncertainty in the RHR model output is 137.7 g. This quantity is not included in the model output displayed in Figure 3.n.1-3, but it is added to the calculated in-vessel fiber accumulation quantity found in the in-vessel hand calculation discussed later in this section.

Figure 3.n.1-4 shows the prompt fiber penetration fraction as a function of fiber quantity on the strainer derived using the RHR strainer model. As expected, the prompt penetration fraction decreases as a fiber debris bed forms on the strainer. Because shedding penetration is a function of both fiber quantity on the strainer and time, it cannot be similarly shown.

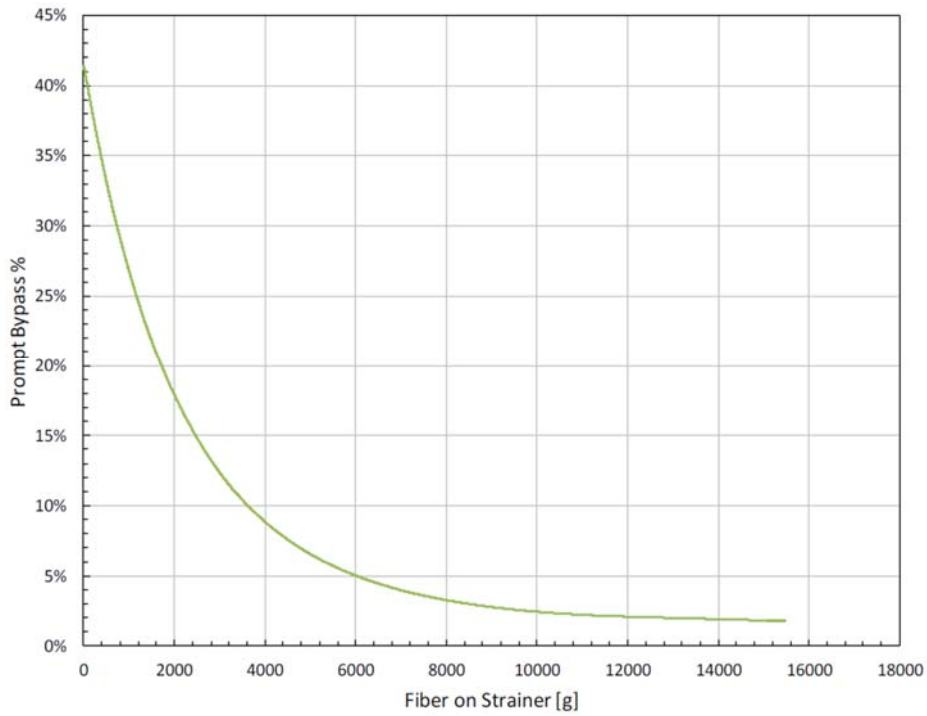

Figure 3.n.1-4: Prompt Fiber Penetration Fraction for RHR Strainer Model

In-Vessel Effects Evaluations

<u>Peak Cladding Temperature and Deposition Thickness</u>

The LOCA deposition model (LOCADM), which is contained as part of WCAP-16793-NP-A (Revision 22), was used to determine the scale thickness due to deposition of debris that passes through the strainer on the fuel rod surfaces and the resulting peak cladding temperature. The calculated scale thickness was then combined with the thickness of existing fuel cladding oxidation and crud build-up to determine the total deposition thickness. The calculated total deposition thickness and peak cladding temperature were compared with the acceptance criteria provided in WCAP-16793-NP. Note that the VEGP evaluation also considered the applicable requirements and recommendations from the following PWROG letters: OG-07-419, OG-07-534, OG-08-64, and OG-10-253.

Two different cases were considered in this evaluation per the WCAP: minimum containment sump pool volume (Case 1) and maximum containment sump pool volume (Case 2). Table 3.n.1-5 below summarizes the maximum PCT and DT for these two cases.

Table 3.n.1-5: Summary of PCT and DT for Cases 1 and 2

Case	PCT (°F)		DT (mils)	
	Results	Acceptance Criteria	Results	Acceptance Criteria
Case 1: Minimum Initial Sump Pool Volume	412	< 800	22.7	< 50
Case 2: Maximum Initial Sump Pool Volume	412		22.3	

For either case, the PCT is much lower than the acceptance criterion of 800 degrees F, and the DT is well within the acceptance criterion of 50 mils. Therefore, deposition of post-LOCA chemical precipitate on the fuel rods will not block the LTCC flow through the core, nor will it create unacceptable local hot spots on the fuel cladding surfaces.

The list below summarizes the key inputs and conservatisms used in the LOCADM analysis:

1. When calculating the “bump-up” factor to account for the fiber that passes through the strainer, a bounding value of 100 g/FA was used. As shown in Table 3.n.1-6, this quantity bounds the actual fiber loads for VEGP.
2. The surface area of aluminum coatings was conservatively calculated with operating margin.
3. The maximum sump pH, rather than the actual sump pH profile, was used for the entire 30-day mission time. This is conservative because higher sump pH values result in greater DT.
4. A combination of inputs was used to conservatively determine the PCT and/or DT in LOCADM.
   a. Spray was assumed to start immediately and continue for 30 days after the beginning of the LOCA.
   b. A conservatively high value for sump temperature was used to set the density of the sump water in order to minimize its mass for the given volume, thereby resulting in higher chemical concentrations within the sump.
   c. A conservatively high value for reactor vessel coolant temperature was used to set the density of the reactor coolant in order to minimize its mass for the given volume, thereby resulting in higher chemical concentrations within the reactor vessel.

The fiber limit at the reactor core inlet given in WCAP-16793-NP-A (15 g/FA) was not used. Instead, accumulation of fiber at the reactor core inlet and inside the reactor vessel was evaluated using the methodology described as discussed in the following section titled "Accumulation of Fiber inside Reactor Vessel."

The NRC Safety Evaluation of WCAP-16793-NP provided analysis and recommendations on the use of Westinghouse's WCAP-16793-NP, Revision 2 methodology and identified 14 limitations and conditions that must be addressed. VEGP's responses to these limitations and conditions are summarized below.

1. Assure the plant fuel type, inlet filter configuration, and ECCS flow rate are bounded by those used in the FA testing outlined in Appendix G of the WCAP. If the 15 g/FA acceptance criterion is used, determine the available driving head for a hot leg (HL) break and compare it to the debris head loss measured during the FA testing. Compare the fiber bypass amounts with the acceptance criterion given in the WCAP.

   VEGP Response:

   This limitation/condition is associated with the 15 g/FA limit established in WCAP-16793-NP-A, which does not apply to VEGP. In-vessel fiber accumulation is not calculated using WCAP-16793-NP-A.

2. Each licensee's GL 2004-02 submittal to the NRC should state the available driving head for an HL break, ECCS flow rates, LOCADM results, type of fuel and inlet filter, and amount of fiber bypass.

   VEGP Response:

   This limitation/condition is associated with the 15 g/FA limit established in WCAP-16793-NP-A, which does not apply to VEGP. In-vessel fiber accumulation is not calculated using WCAP-16793-NP-A.

3. If a licensee credits alternate flow paths in the reactor vessel in their LTCC evaluations, justification is required through testing or analysis.

   VEGP Response:

   This limitation/condition is associated with the 15 g/FA limit established in WCAP-16793-NP-A, which does not apply to VEGP. In-vessel fiber accumulation is not calculated using WCAP-16793-NP-A.

4. The numerical analyses discussed in Sections 3.2 and 3.3 of the WCAP should not be relied upon to demonstrate adequate LTCC.

   VEGP Response:

   VEGP does not use any of the conclusions drawn based on the fuel blockage modeling discussed in Sections 3.2 and 3.3 of the WCAP report. In-vessel fiber accumulation is not calculated using WCAP-16793-NP.

5. The SE requires that a plant must maintain its debris load within the limits defined by the testing (e.g., 15 g/FA), and any debris amounts greater than those

justified by generic testing in the WCAP must be justified on a plant-specific basis.

VEGP Response:

This limitation/condition is associated with the 15 g/FA limit established in WCAP-16793-NP-A, which does not apply to VEGP. In-vessel fiber accumulation is not calculated using WCAP-16793-NP-A.

6. The debris acceptance criterion can only be applied to fuel types and inlet filter configurations evaluated in the WCAP FA testing.

   VEGP Response:

   This limitation/condition is associated with the 15 g/FA limit established in WCAP-16793-NP-A, which does not apply to VEGP. In-vessel fiber accumulation is not calculated using WCAP-16793-NP-A.

7. Each licensee's GL 2004-02 submittal to the NRC should compare the PCT from LOCADM with the acceptance criterion of 800 degrees F.

   VEGP Response:

   As shown in Table 3.n.1-5 above, the calculated VEGP PCT is well within the acceptance criterion of 800 degrees F.

8. When utilizing LOCADM to determine PCT and DT, the aluminum release rate must be doubled to predict aluminum concentrations in the sump pool in the initial days following a LOCA more accurately.

   VEGP Response:

   The methodology outlined in PWROG Letter OG-08-64 was followed to double the aluminum release rate in the LOCADM analysis.

9. If refinements specific to the plant are made to the LOCADM to reduce conservatisms, the licensee should demonstrate that the results still adequately bound chemical product generation.

   VEGP Response:

   The VEGP LOCADM runs do not employ any conservatism-reducing refinements specific to the plant. Therefore, no additional justification is required.

10. The recommended value for scale thermal conductivity of 0.11 BTU/(h-ft-°F) should be used for LTCC evaluations.

    VEGP Response:

    As stated in Appendix E of WCAP-16793-NP, the recommended thermal conductivity of 0.11 BTU/(h-ft-°F) can be converted to 0.2 W/m-K, which is used in the LTCC calculation.

11. The licensee's submittals should include the means used to determine the amount of debris that bypasses the ECCS sump strainer and the fiber loading at the fuel inlet expected for the HL and cold leg (CL) break scenarios. Licensees

should provide the debris loads, calculated on a fuel assembly basis, for both the HL and CL break cases in their GL 2004-02 responses.

VEGP Response:

This limitation/condition is associated with the 15 g/FA limit established in WCAP-16793-NP-A, which does not apply to VEGP. In-vessel fiber accumulation is not calculated using WCAP-16793-NP-A.

12. Plants that can qualify a higher fiber load based on the absence of chemical deposits should ensure that tests for their conditions determine limiting head losses using particulate and fiber loads that maximize the head loss with no chemical precipitates included in the tests. In this case, licensees must also evaluate the other considerations discussed in the first limitation and condition.

    VEGP Response:

    This limitation/condition is associated with the 15 g/FA limit established in WCAP-16793-NP-A, which does not apply to VEGP. In-vessel fiber accumulation is not calculated using WCAP-16793-NP-A.

13. The size distribution of the debris used in the FA testing must represent the size distribution of fibrous debris expected to pass through the ECCS sump strainer at the plant.

    VEGP Response:

    This limitation/condition is associated with the 15 g/FA limit established in WCAP-16793-NP-A, which does not apply to VEGP. In-vessel fiber accumulation is not calculated using WCAP-16793-NP-A

14. Each licensee's GL 2004-02 submittal to the NRC should not utilize the "Margin Calculator" as it has not been reviewed by the NRC.

    VEGP Response:

    The VEGP evaluation does not use the "Margin Calculator".

In summary, the evaluation showed that the peak cladding temperature and deposition thickness due to accumulation of debris on the fuel rods met the acceptance criteria and did not cause any failures.

<u>Accumulation of Fiber 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 effect is evaluated for both hot leg break (HLB) and cold leg break (CLB) scenarios using the assumed acceptance criteria. The evaluation used time-dependent fiber penetration fractions obtained from VEGP 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 evaluation was performed in the NARWHAL CFP calculation as well as a bounding hand calculation. The NARWHAL model used assumed values and acceptance criteria to evaluate every break in a self-consistent and time-dependent manner. The recirculation phase was divided 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 are calculated using the model equations based on the quantity of fine fiber collected on the strainer at the beginning of the time step.
2. The amount of fine fiber that arrives at the strainer during the current time step is calculated by multiplying the fine fiber concentration in the pool by the strainer flow rate and time step.
3. The amount of prompt penetration is 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 is 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.

The model showed no failures for any of the postulated breaks due to accumulation of fiber at the core inlet or inside the reactor vessels. Impact on the results due to variabilities in the inputs was evaluated by sensitivity analyses.

The hand calculation served as a bounding evaluation in which the worst case combination of input parameters (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) were used. The steps shown above were implemented in an Excel spreadsheet. The uncertainty of the fiber penetration model was added to the calculated fiber quantities for conservatism. The results of the hand calculation are summarized in the table below and are compared with the acceptance criteria. The resulting fiber quantities for both the HLB and CLB are bounded by the assumed acceptance criteria. This conclusion is consistent with the results of the NARWHAL CFP calculation that showed no failures due to accumulation of fibers at the core inlet or inside the reactor vessel.

Table 3.n.1-6: Bounding Fiber Loads for HLB and CLB Scenarios

[ ]a,c

In summary, no failures were recorded for any of the postulated breaks due to accumulation of debris at the core inlet or inside the reactor vessel.

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.

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. Content guidance for chemical effects is provided in Enclosure 3 to a letter from the NRC to NEI dated March 2008 (Reference 106).

Response to 3.o.1:

The chemical effects strategy for VEGP includes:

- Quantification of chemical precipitates using the WCAP-16530-NP-A methodology with refinements for phosphate passivation of aluminum surfaces.
- Introduction of those pre-prepared precipitates in prototypical array testing.
- Application of an aluminum solubility correlation and integrated autoclave chemical test results to determine formation temperature/timing.
- Time-based determination of acceptable head losses.
- Extrapolation of the resulting head losses to 30 days.

As discussed in the Response to 3.a.1, VEGP has evaluated breaks at all Class 1 weld locations on the primary RCS piping, upstream of the first isolation valve. The amount of chemical precipitates was quantified individually for each of these breaks using the amount of LOCA generated debris for that respective break location.

Other plant-specific inputs such as pH, temperature, aluminum amount, and spray times were selected to maximize the generated amount of precipitates. These amounts were scaled by the ratio of the test strainer area to the plant strainer surface area and are compared with the chemical debris quantities used in the prototypical array tests to determine the resulting head loss across the strainers. Before the tests were conducted, the SAS and calcium phosphate were prepared according to the WCAP-16530-NP-A "recipes" and were verified to meet the 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. See the Response to 3.f.4 for further details on the head loss measured after introduction of chemical precipitates. See the Response to 3.f.10 for further details on how the chemical precipitate head loss was utilized in the NARWHAL CFP calculation.

See the in-vessel effects evaluations in the Response to 3.n.1 for the evaluation of chemical precipitate deposition on the fuel rod surfaces.

2. Content guidance for chemical effects is provided in Enclosure 3 to a letter from the NRC to NEI dated March 2008 (Reference 106).

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 106). VEGP's responses to the GL Supplement Content evaluation step for each debris characteristic are summarized below.

1. Sufficient 'Clean' Strainer Area: 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.

   VEGP Response:

   As discussed in the Response to 3.a.1, VEGP has evaluated breaks at all ISI weld locations on the primary RCS piping, upstream of the first isolation valve. Many of the breaks analyzed resulted in fiber loads sufficient to fully cover the sump strainer screens. Therefore, VEGP is not crediting clean strainer area to perform a simplified chemical effects analysis. See the Figure 1 flow chart in "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Plant-Specific Chemical Effect Evaluations" from March 2008 (Reference 106).

2. Debris Bed Formation: 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.

   VEGP Response:

   Three head loss tests were completed for VEGP: thin bed, full load, and confirmatory full load. The full load produced the highest head loss at each stage of the test, as shown in the table below. Therefore, the full load test was utilized to develop the contributions from conventional debris, calcium phosphate, aluminum precipitates, and the 30-day extrapolation. Table 3.o.2-1 lists the chemical head loss contributions.

Table 3.o.2-1: Chemical Head Loss Values

Test Point	Thin Bed (ft-H_2O)	Full Load (ft-H_2O)	Confirmatory Full Load (ft-H_2O)
Conventional Debris	0.625	5.46	3.50
Full Calcium Phosphate Load	1.65	6.57	5.75
Full Aluminum Precipitate Load	2.60	11.81	8.99
30-day Extrapolation	3.15	15.70	11.12

See the Response to 3.f.10 for additional chemical head loss information.

3. Plant-Specific Materials and Buffers: 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.

   VEGP Response:

   The VEGP 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.

   VEGP uses TSP to buffer the post-LOCA containment sump pool to a final pH between 7.12 and 7.78. In order to maximize chemical release, TSP is conservatively assumed to dissolve immediately, and the maximum pH of 7.8 was used for the containment sump pool for the entire 30-day event and for the containment spray while recirculating from the containment sump pool. A maximum pH of 5.72 was used for the containment spray during the post-LOCA RWST injection mode. To minimize aluminum solubility, the minimum containment sump pool pH of 7.0 was used, and precipitation was forced at 24 hours whether the solubility limit was reached or not. Different pH values for release and solubility were combined in a non-physical way, bounding the effects of all potential pH profile variations. The pH values are summarized in Table 3.o.3-1:

Table 3.o.3-1: VEGP pH Values

Sump and Recirculation Spray pH Used To Determine Chemical Release Rates	7.8
Maximum VEGP Long-Term Sump pH	7.78
Minimum VEGP Long-Term Sump pH	7.12
Sump pH Used To Determine Aluminum Solubility	7.0
Injection Spray pH Used To Determine Chemical Release Rates	5.72

The maximum temperature profiles of containment and the sump pool used for the analysis are from the double-ended pump suction LOCA with minimum safeguards case.

The total amount of unsubmerged aluminum exposed to containment sprays was assumed to be 926.6 ft^2. The total amount of submerged aluminum exposed to the containment sump fluid at VEGP Units 1 and 2 was assumed to be 348.4 ft^2.

The total amount of concrete assumed to be exposed and submerged in the containment sump pool is 10,000 ft^2. The quantity of chemical precipitates is 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 VEGP post-LOCA containment sump pool and is not tracked for this purpose.

The NARWHAL software (see Enclosure 3, Section 13.1 for general description of the software) analysis accounts for the change in water volume with respect to time and uses assumptions that minimize the water volume in containment. The water volume is dependent on break size, break location, and whether the containment sprays actuate, among other factors. The break size affects the volume in that the accumulators do not inject for breaks less than 2 inches. The RCS holdup volume is dependent on the break location/elevation. Finally, the containment spray activation affects the volume of water in transit.

It is acknowledged that water volume has competing effects with respect to chemical release versus solubility; therefore, water volume was included in the sensitivities evaluated in Enclosure 3. A spray duration of 24 hours is used in the analysis. Although there is some operational response flexibility in the spray duration, this is a reasonably conservative assumption because sprays are required to operate for at least 2 hours if they are initiated (assuming there are no containment radiation monitor alarms). Phosphate inhibition minimizes the effect of chemical release from extended spray durations. Enclosure 3, Section

14.3 includes sensitivity studies with both 2 hour and 30-day containment spray durations.

Variable Debris Amounts

Of the debris types described in WCAP-16530-NP-A, both E-Glass (Nukon) and Interam fire barrier are present in the VEGP containments. The quantities of these materials are specific to each break analyzed. Figure 3.a.3-1 through Figure 3.a.3-8 show the ranges of Nukon and Interam debris versus break size for DEGBs and partial breaks. The mass of latent fiber included as E-Glass for all breaks is 30 lbm.

Interam is a potential source of aluminum from its metal foil surface and is a source of leachable silicon (Reference 73). The Interam fire barrier at VEGP uses stainless steel foil and is, therefore, not a source of aluminum. Additionally, as discussed in the Response to 3.o.2.7.i, silicon release is not tracked because aluminum is assumed to precipitate only as SAS. Therefore, Interam does not impact the VEGP chemical model.

4. Approach to Determine Chemical Source Term (Decision Point): Licensees should identify the vendor who performed plant-specific chemical effects testing.

   VEGP Response:

   VEGP is using the separate chemical effects approach to determine the chemical source term. Alion Science and Technology Corporation performed the testing in their test lab in Warrenville, IL.

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).

   VEGP Response:

   VEGP is primarily using the WCAP-16530-NP-A chemical effects base model to determine the chemical source term. Refinements to this model for aluminum solubility and phosphate inhibition of aluminum release from metallic aluminum are discussed in the Response to 3.o.2.8 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.

      VEGP Response:

      This question is not applicable because VEGP is not using the AECL model. See the Figure 1 flow chart in "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Plant-Specific Chemical Effect Evaluations" from March 2008 (Reference 106).

   ii. Licensees should provide the chemical identities and amounts of predicted plant-specific precipitates.

      VEGP Response:

      This question is not applicable because VEGP is not using the AECL model. See the Figure 1 flow chart in "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Plant-Specific Chemical Effect Evaluations" from March 2008 (Reference 106).

7. WCAP Base Model:

   i. Licensees proceeding from block 7 to diamond 10 in the Figure 1 flow chart [in Enclosure 3 to a letter from the NRC to NEI dated March 2008 (Reference 106)] 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.

      VEGP Response:

      The VEGP chemical model includes quantification of chemical precipitates using the WCAP-16530-NP-A (Reference 73) methodology with refinements for phosphate passivation of aluminum surfaces and the application of an aluminum solubility correlation to determine formation temperature/timing. Silicon inhibition of aluminum release is not credited. Refinements to this model for aluminum solubility and phosphate inhibition of aluminum release from metallic aluminum are discussed in the Response to 3.o.2.9.i.

      The chemical precipitates assumed by the VEGP chemical model are calcium phosphate ($Ca_3(PO_4)_2$) and SAS ($NaAlSi_3O_8$). Although the WCAP-16530-NP-A model typically includes aluminum oxyhydroxide

(AlOOH), the VEGP model assumes that all precipitated aluminum forms SAS, independent of the quantity of silicon available. Per the WCAP-16530-NP-A SE, both aluminum precipitates are acceptable surrogates for aluminum precipitate in head loss testing.

As described in Enclosure 3, Section 8.0, the chemical quantity methodology is integrated into the NARWHAL software. The quantity of chemical precipitate is determined for each break as a function of thermal hydraulic inputs and debris generation quantities. The temperature profiles, pH profiles, aluminum metal surface area, and concrete surface area are constant for each break and were selected to maximize aluminum and calcium release and minimize aluminum solubility. The sump fluid volume, spray duration, and debris quantities are break-dependent variables in the NARWHAL calculations.

There are two parts to the determination of the chemical precipitate quantity: the elemental chemical release from substrates in containment and chemical product formation.

Elemental Chemical Release

The two classifications of substrates for which chemical release is analyzed are debris and exposed surfaces. Fiber debris and Interam debris contribute to chemical release, which is quantified using the WCAP-16530-NP-A release equations. Note that the quantity of each of these debris types is break-specific; therefore, the quantity of elemental chemical release will vary for each break analyzed. Also, note that the Interam only releases silicon, which does not contribute to the chemical precipitates being tracked for VEGP (see chemical product formation discussion below). This is because SAS is the only aluminum precipitate that is being tracked in the VEGP NARWHAL model, and NARWHAL conservatively assumes an infinite source of silicon when SAS is the only aluminum precipitate tracked.

The amount of elemental chemical release from a given debris source is limited by the quantity. Table 3.o.2.7-1 shows the chemical limits of fiberglass used in the chemical release model.

Table 3.o.2.7-1: Chemical Mass Limits

E-Glass	
Aluminum Mass Available per Material Mass	1.95%
Calcium Mass Available per Material Mass	2.16%

The exposed surfaces include aluminum metal and concrete surfaces that either are submerged in the containment pool or are exposed to containment

sprays. The same surface areas were analyzed for each break. Table 3.o.2.7-2 shows these areas. Note that the chemical release from exposed concrete was evaluated using the WCAP-16530-NP-A release equations, and the chemical release from aluminum was evaluated using the University of New Mexico (UNM) release equations.

Table 3.o.2.7-2: Other Source Release Values

Substrate	Surface Area (ft^2)
Aluminum Metal: Submerged	348.4
Aluminum Metal: Unsubmerged	926.6
Exposed Concrete: Submerged	10,000
Exposed Concrete: Unsubmerged	0

Chemical Product Formation

The chemical precipitates analyzed for the VEGP NARWHAL CFP calculation are SAS and calcium phosphate. The calcium phosphate is conservatively assumed to precipitate immediately. The SAS precipitates when the concentration of aluminum in the pool exceeds the aluminum solubility limit as calculated with the Argonne National Laboratory (ANL) solubility equation. Note that if precipitation of SAS is not predicted before 24 hours, then precipitation is forced at that time. Also, note that aluminum does not remain dissolved in the pool after precipitation occurs. Forcing precipitation at 24 hours and not taking credit for aluminum remaining dissolved in the pool are conservative factors in the chemical product formation model.

ii. Licensees should list the type (e.g., AlOOH) and amount of predicted plant-specific precipitates.

VEGP Response:

Chemical precipitate quantities were calculated in the NARWHAL CFP calculation and in a bounding hand calculation. The NARWHAL calculation performs comprehensive evaluation of GSI-191 phenomena in a self-consistent and time-dependent manner. It should be noted that the chemical debris quantities used for quantifying head loss were directly calculated in NARWHAL, not from the hand calculation. The NARWHAL calculation uses the plant-scale precipitate loads from the 2009 head loss testing as the maximum debris limit acceptance criteria (see Response to 3.f.5). The results from the hand calculation are provided below as bounding numbers.

The bounding precipitate surrogate masses that would be generated at VEGP are 40.2 kg (88.6 lbm) SAS and 63.2 kg (139.3 lbm) calcium phosphate.

These precipitate masses represent the bounding quantity of aluminum and calcium that could precipitate in the VEGP Unit 1 and Unit 2 containment sump pool. Both AlOOH and SAS chemical surrogates are considered equivalent generators of head loss across a debris bed. Therefore, AlOOH and SAS surrogates may be substituted for each other stoichiometrically relative to aluminum.

The maximum temperature where aluminum precipitation could occur in the containment sump pool was calculated to be 136.8 degrees F. Furthermore, the use of the aluminum solubility model for a TSP buffered solution is supported by the integrated autoclave experiments, as shown in Table 3.o.2.9-2. However, as the duration of these experiments was only 24 hours, aluminum precipitation is assumed to occur at 24 hours for calculating strainer head loss unless the solubility model predicts precipitation at an earlier time. Calcium phosphate is assumed to precipitate at all temperatures.

These results are bounding for both Unit 1 and Unit 2.

8. <u>WCAP Refinements</u>: State whether refinements to WCAP-16530-NP-A were utilized in the chemical effects analysis.

   VEGP Response:

   The chemical effects strategy for VEGP includes quantification of chemical precipitates using the WCAP-16530-NP-A (Reference 73) methodology with refinements for phosphate passivation of aluminum surfaces and the application of an aluminum solubility correlation to determine formation temperature/timing. Silicon inhibition of aluminum release was not credited. Refinements to the model for aluminum solubility and phosphate inhibition of aluminum release from metallic aluminum are discussed in the Response to 3.o.2.9.i.

9. <u>Solubility of Phosphates, Silicates and Al Alloys</u>:

   i. Licensees should clearly identify any refinements (plant-specific inputs) to the base WCAP-16530-NP-A model and justify why the plant-specific refinement is valid.

      VEGP Response:

      The VEGP chemical model includes quantification of chemical precipitates using the WCAP-16530-NP-A (Reference 73) methodology with refinements for phosphate passivation of aluminum surfaces and the application of an aluminum solubility correlation to determine formation temperature/timing. Silicon inhibition of aluminum release was not credited.

Phosphate Inhibition of Aluminum Surfaces

The release of aluminum from metallic aluminum material into the TSP sump solution was modeled using the equations developed by Howe et al. Equations 3.o.2.9-1 through 3.o.2.9-3 (UNM aluminum release equations) calculate the release rate for aluminum from sprayed and submerged aluminum metal in containment.

$$R_{Al,m} = R_{max}P \qquad \text{(Equation 3.o.2.9-1)}$$

$$R_{max} = 10^{6.2181-4.6454\frac{1000}{T_K}+1.7716(pH)-1.9550\frac{(pH)T_K}{1000}} \qquad \text{(Equation 3.o.2.9-2)}$$

$$P = e^{-\left(0.85586-4.9561\times10^{-3}T_{KP}+7.1978\times10^{-6}{T_{KP}}^2\right)t_{TSP}} \qquad \text{(Equation 3.o.2.9-3)}$$

Nomenclature:

$R_{Al,m}$ = release rate of aluminum from aluminum metal, mg/(m^{2}min)
R_{max} = non-passivated aluminum release rate, mg/(m^{2}min)
P = phosphate passivation term, unitless
T_K = temperature, K
T_{KP} = temperature utilized in the phosphate passivation term, K
pH = pH at 25 degrees C
t_{TSP} = time elapsed with phosphate present in solution, min

The above equations were developed in testing that was performed at temperatures from 55 degrees C to 85 degrees C (131 degrees F to 185 degrees F, 328.15 K to 358.15 K) and at pH values from 6.84 to 7.84. The following two constraints were used to extend the applicability of these equations:

1. The passivation term, P, is an exponential decay function that approaches zero as t_{TSP} increases. This term models the decrease in aluminum surface area available for release as the passivation layer forms. Since testing was not performed below a pH of 6.84, it is not known if this term is applicable at very low TSP concentrations. Therefore, phosphate is not considered "present in solution" unless the pH is above 6.84, and t_{TSP} is held at 0 minutes. When the pH rises above 6.84, as TSP dissolves into solution, the t_{TSP} "clock" starts.

   In practice, the VEGP analysis assumes that TSP is present in the containment sump pool at the start of the LOCA by assuming that the initial sump pH is at its maximum value of 7.8 (see Response to 3.o.2.3). This assumption conservatively increases the aluminum release rate by non-physically combining the highest sump pH with the high initial sump

temperatures. Similarly, the containment sprays are assumed to be at a pH of 7.8 (TSP is present) immediately upon the switch to recirculation.

2. The phosphate passivation function is not assumed to extrapolate beyond the test temperature range. Therefore, T_{KP} equals T_K unless above or below the temperatures described here. For temperatures below 328.15K, T_{KP} equals 328.15K. For temperatures above 358.15 K, T_{KP} equals 358.15 K. The passivation equation predicts faster passivation both above and below this range, which is not justifiable without additional testing.

A comparison of the VEGP chemical model with the WCAP-16530-NP-A model is shown in Figure 3.o.2.9-10 and Figure 3.o.2.9-11. Additionally, the aluminum release rate equations with the above constraints were verified for use by VEGP by modeling several integrated autoclave tests. Tests 40-01 and IBOB 40-01 used VEGP specific materials in testing to replicate the post-LOCA containment debris amounts reported for VEGP Units 1 and 2. Given that the relevant VEGP specific design inputs are a maximum sump pH of 7.78 and a post-LOCA debris type of E-Glass (i.e., no Calcium Silicate, Silica, or Mineral Wool insulation), tests 39-01, 42 01, 44-01, IBOB 39-01, IBOB 42-01, and IBOB 44-01 also use test inputs similar to that of VEGP [ ]a,c

These tests were run at a range of temperatures from [.

]a,c which bounds the VEGP maximum post-LOCA containment sump pool and containment temperature profiles. Therefore, tests 39-01, 40-01, 42-01,44-01, IBOB 39-01, IBOB 40-01, IBOB 42-01, and IBOB 44-01 were simulated using the WCAP-16530-NP-A methodology with the refined aluminum release equation. The critical parameters for the integrated autoclave tests 39-01, 40-01, 42-01, 44-01, IBOB 39-01, IBOB 40-01, IBOB 42-01, and IBOB 44-01 are summarized in Table 3.o.2.9-1 and Figure 3.o.2.9-1. Note that tests with the same four-digit identification were run at identical conditions with the exception of the debris placement out-of-bag for the IBOB tests. Tests without the IBOB designation were run with the debris contained within a mesh bag.

[ ]a,c

Figure 3.o.2.9-1: Integrated Autoclave Test Temperature Profiles

Table 3.o.2.9-1: Critical Integrated Autoclave Test Parameters

[ ]a,c

Vogtle's maximum pH (7.78) and the pH of tests 44-01 and IBOB 44-01 [ ]a,c are within the pH range for the UNM aluminum release equations of 6.84 through 7.84. However, IBOB 44-01 test results are all [ ]a,c and are considered non-conservative outliers. Therefore, test 44-01 serves as the primary validation of the UNM aluminum release equations in an integrated chemical environment. The aluminum concentration results for test 40-01, test IBOB 40-01, and the simulation are provided in Figure 3.o.2.9-2.

[ ]a,c

Figure 3.o.2.9-2: Test (IBOB) 44-01 (pH = 7.52) Aluminum Concentrations

Because the maximum pH of tests 44-01 and IBOB 44-01 [ ]a,c is below the VEGP maximum pH (7.78), integrated autoclave tests 39-01 [ ]a,c, 40-01 [ ]a,c, 42-01 [ ]a,c, IBOB 39-01 [ ]a,c, IBOB 40-01 [ ]a,c, and IBOB 42-01 [ ]a,c were also simulated to validate the UNM aluminum release equations at bounding pH values. However, these tests are above the pH range for the single effects tests (6.84 through 7.84) used to develop the UNM aluminum release equations. The aluminum concentration results for these tests and their simulations are provided in Figures 3.o.2.9-3 through 3.o.2.9-5.

a,c

[ ]

Figure 3.o.2.9-3: Test (IBOB) 39-01 (pH = 8.00) Aluminum Concentrations

a,c

[ ]

Figure 3.o.2.9-4: Test (IBOB) 40-01 (pH = 8.03) Aluminum Concentrations

a,c

[ ]

Figure 3.o.2.9-5: Test (IBOB) 42-01 (pH = 8.03) Aluminum Concentrations

In addition to the aluminum results, the integrated autoclave test calcium release results were also compared with the simulation in Figures 3.o.2.9-6 through 3.o.2.9-9 to demonstrate the overall conservatism of the VEGP chemical model. Calcium concentration results are not available for tests IBOB 40-01, IBOB 42-01, and IBOB 44-01.

a,c

[ ]

Figure 3.o.2.9-6: Test (IBOB) 39-01 (pH = 8) Calcium Concentrations

a,c

[ ]

Figure 3.o.2.9-7: Test 40-01 (pH = 8.03) Calcium Concentrations

a,c

[ ]

Figure 3.o.2.9-8: Test 42-01 (pH = 8.03) Calcium Concentrations

a,c

[ ]

Figure 3.o.2.9-9: Test 44-01 (pH = 7.52) Calcium Concentrations

The aluminum release results indicate that the aluminum release rate as a function of time in a TSP buffered solution decreases to approximately zero within 24 hours, as is predicted by the aluminum release equations. For integrated autoclave tests 40-01, 42-01, 44-01, and IBOB 40-01, the measured aluminum concentrations either follow or are bounded by the trends predicted by the UNM aluminum release equations, which justifies the UNM methodology with the extended pH (5 to 7.84) and temperature (104.8 degrees F to 266.5 degrees F) constraints as acceptable. Note that the aluminum concentrations for tests 39-01, IBOB 39-01, and IBOB 42-01 are under-predicted. The presence of zinc in solution can reduce the rate of aluminum corrosion. A reduction of approximately 2/3 of the released aluminum concentration was observed in the Howe et al. bench tests when zinc coupons were present. The Howe et al. equations do not credit zinc inhibition, which was shown by bench testing to result in an over-prediction of aluminum release under the maximum pH tested of 7.84. However, integrated autoclave tests 39-01, 40-01, 42-01, IBOB 39-01, IBOB 40-01, and IBOB 42-01 are above the pH range for the single effects tests. Of these six integrated autoclave tests, tests 39-01 and IBOB 39-01 contained the least amount of galvanized steel and contained less than 38% (0.117 ft^2/0.308 ft^2) of the galvanized steel surface area as Test 40-01, which used VEGP-specific material quantities. Furthermore, the aluminum release result for Test 40-01 is accurately predicted, and Test 42-01 is slightly over-predicted with 238% of the VEGP-specific galvanized steel quantity (0.732 ft^2/0.308 ft^2). Although these results demonstrate that the aluminum release equations accurately predict aluminum concentrations at elevated pH when VEGP-specific or greater zinc quantities are present, the maximum acceptable pH is not extended above the value of 7.84 as set by Howe, et al.

Additionally, as shown in Figures 3.o.2.9-6 through 3.o.2.9-9 calcium concentrations are significantly over-predicted by the WCAP-16530-NP-A model. As discussed in Section 3.o.2.7.i, calcium phosphate is conservatively assumed to form immediately as calcium is released. Finally, as discussed in Section 3.f.10, the full load calcium phosphate head loss is assumed as soon as calcium phosphate starts to accumulate on the strainer.

Because the VEGP chemical model results in over-predicted quantities of aluminum and calcium precipitates at VEGP-specific conditions, the overall methodology is conservative for use to determine the precipitate loading for strainer head loss.

In addition to the justification provided above, both quantified and qualitative conservatisms may be used to further justify the use of the Howe et al. equations for aluminum release from aluminum metals, despite the under-predictions of autoclave tests 39-01, IBOB 39-01, and IBOB 42-01. The following list of conservatisms is provided:

1. The total test head loss is applied when precipitate is present at the strainers, and high NPSH and structural margins remain available.

2. WCAP-16530-NP-A chemical surrogates were used in testing, which generate a significant head loss across a debris bed per the WCAP-16530-NP-A Safety Evaluation.

3. By using the same equations as submerged aluminum, unsubmerged aluminum is treated as fully wetted, or submerged, in the containment spray solution.

4. The autoclave tests demonstrate that little calcium would be released and calcium phosphate is unlikely to form and impact strainer head loss.

5. No credit is taken for aluminum that remains soluble after precipitation occurs.

6. The aluminum solubility equation predicted precipitation to occur in the filtration samples of the modeled autoclave tests although no precipitation was observed.

7. The Vogtle double-ended pump suction LOCA with minimum safeguards temperature profiles are used to determine chemical release rates.

8. The maximum design basis recirculation pH of 7.8 is used to determine chemical release rates. A recirculation pH of 7.0, which is less than the design basis minimum of 7.12, is used for aluminum solubility. Different pH values for release and solubility were combined in a non-physical way, bounding the effects of all potential pH profile variations.

9. The maximum design basis recirculation pH of 7.8 is used to determine chemical release rates. The best estimate maximum final containment sump pool pH was determined to be 7.42. The use of the design basis maximum pH for the Containment Sump Pool and the Containment Sprays (after the switchover to recirculation) conservatively overestimates the release of aluminum from aluminum metals by 230% versus the use of the best estimate maximum pH at the Vogtle minimum safeguards temperature profile.

In conclusion, the under-prediction of aluminum release during some of the autoclave tests is being considered in the VEGP chemical effects methodology through the use of both qualitative and quantified conservatisms. The conservatism inherent in the use of a pH of 7.8 for the containment sump pool and the containment sprays (after the switchover to recirculation) bounds the uncertainty in the use of the Howe et al. equations outside of the original range of pH values and temperatures.

Solubility of Aluminum

The aluminum solubility limit is determined using Equation 3.o.2.9-4, developed by ANL.

$$C_{Al,sol} = \begin{cases} 26980 \cdot 10^{(pH+\Delta pH)-14.4+0.0243T}, & \text{if } T \leq 175\ °F \\ 26980 \cdot 10^{(pH+\Delta pH)-10.41+0.00148T}, & \text{if } T > 175\ °F \end{cases} \quad \text{(Equation 3.o.2.9-4)}$$

Nomenclature:

ΔpH = pH change due to radiolysis acids
T = solution temperature, degrees F

The aluminum solubility limit equation was used to determine the temperature and timing of aluminum precipitation and to determine the aluminum concentration in solution for use in the aluminum release equations for concrete and insulation. When precipitation is predicted by this equation, the full amount of aluminum released is assumed to precipitate as SAS. The aluminum solubility limit equation was not used to reduce the predicted quantity of precipitate by crediting the amount remaining in solution.

Aluminum solubility testing developed by ANL was completed in boric acid/NaOH buffered solutions. As shown in Table 3.o.2.9-2, the method that Vogtle utilizes to predict aluminum precipitation temperature yields much higher temperatures than the filtration tests that did not detect precipitation. This demonstrates the applicability of the ANL equation in a boric acid/TSP buffered solution. Since the tests were performed for a 24-hour duration, the maximum amount of time allowed in the VEGP chemical model for aluminum precipitation to occur was capped at 24 hours.

Table 3.o.2.9-2: Aluminum Precipitation Test Results

[ ]a,c

Comparison of VEGP Chemical Model with WCAP-16530-NP-A

Enclosure 3 Section 14.4 includes a sensitivity study comparing the VEGP release model with phosphate inhibition credited (UNM Aluminum Metal Release Equation) to the WCAP-16530-NP-A chemical model without this refinement (WCAP-16530 Equation). Additionally, the bounding VEGP case for maximum precipitate generation (based on the hand calculation) was conducted using both the VEGP release model and the WCAP-16530-NP-A model without the refined equations for phosphate inhibition. Note that to address NRC concerns on the use of the WCAP-16530-NP-A for aluminum metal release rates early in the event, the aluminum metal release rate was doubled for the initial 15 days.

The maximum precipitation case uses the pH, temperature, aluminum metal, and concrete inputs described in the VEGP response to 3.o.2.3. The maximum sump pool mass of 6,675,858 lbm was used to maximize the dissolution of aluminum and calcium from concrete and insulation materials. The containment sprays were assumed to remain active for 30 days with a minimum RWST injection duration of 27.9 minutes to maximize aluminum release from sprayed aluminum surfaces. Lastly, the maximum generated quantity of Nukon insulation, 2,229.2 ft^3, was used. Note that these inputs apply to this comparison completed in the hand calculation, which were selected to yield bounding results for all break scenarios as opposed to the break-specific and time dependent inputs used in the NARWHAL CFP calculation.

Figure 3.o.2.9-10 shows a comparison of the aluminum concentration results for the two models over the full 30-day window, and Figure 3.o.2.9-11 shows the results for the first 24 hours. The UNM curve shows the results with aluminum passivation and the 2xWCAP-16530-NP-A curve shows the results without this refinement.

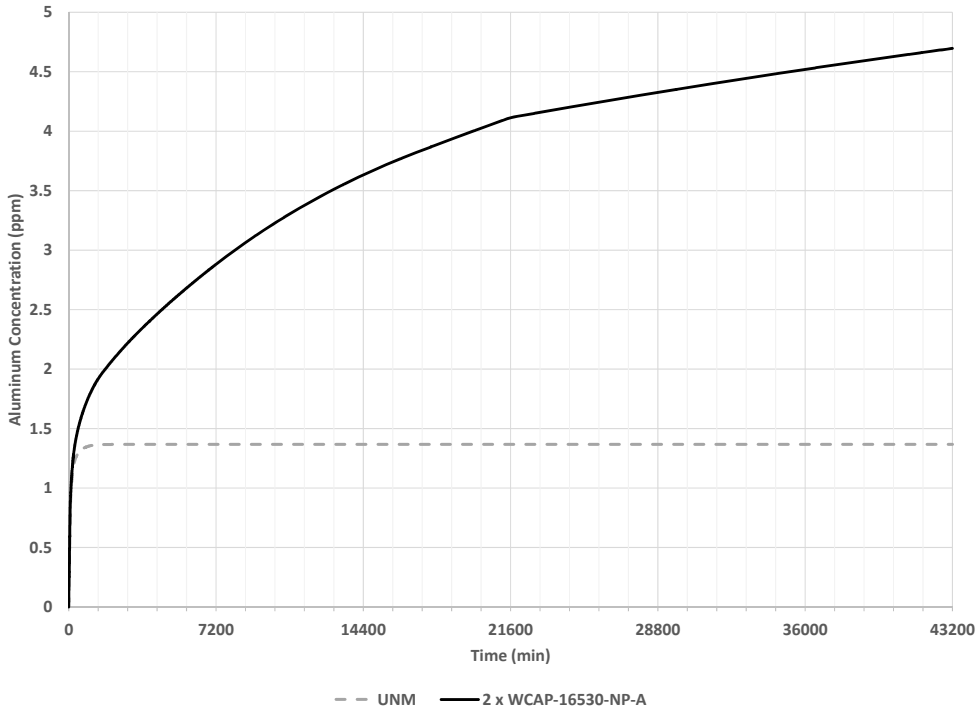

Figure 3.o.2.9-10: Maximum Aluminum Release Cases (30 days)

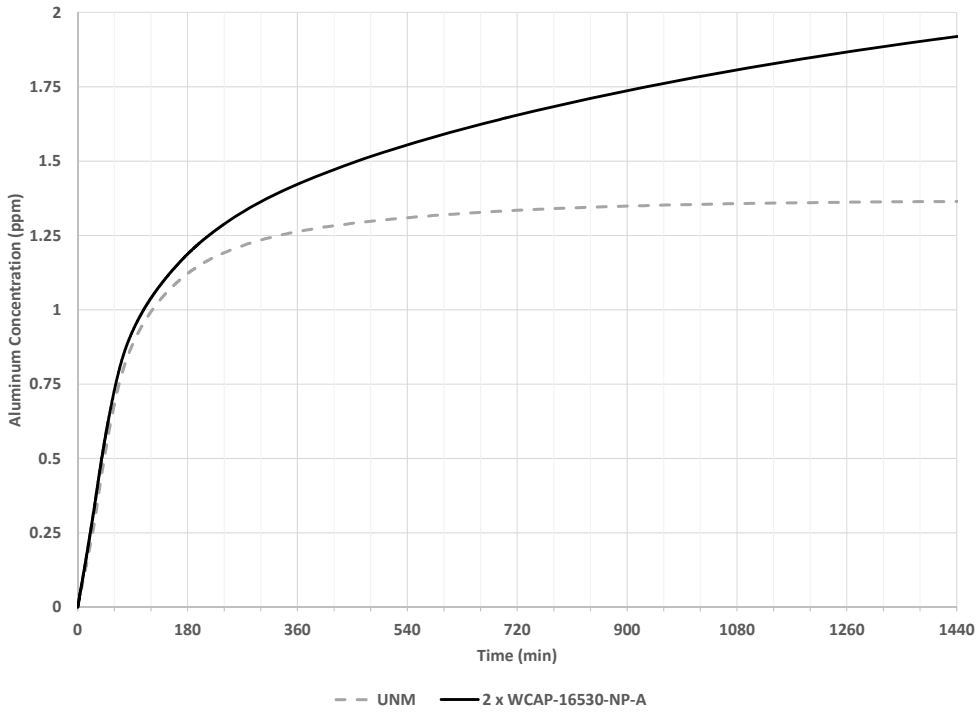

Figure 3.o.2.9-11: Maximum Aluminum Release Cases (24 hours)

The use of the aluminum passivation equations clearly decreases the release of aluminum predicted over 30 days. However, as would be expected, the initial release rate with passivation credited (UNM) follows the model without passivation credited (2xWCAP-16530-NP-A) closely over the initial two hours before diverging as the aluminum metal surface area available for release passivates.

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.

VEGP Response:

See the Response to 3.o.2.9.i.

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.

VEGP Response:

See the Response to 3.o.2.9.i.

iv. Licensees should list the type (e.g., AlOOH) and amount of predicted plant-specific precipitates.

VEGP Response:

See the Response to 3.o.2.7.ii.

10. Precipitate Generation (Decision Point): State whether precipitates are formed by chemical injection into a flowing test loop or whether the precipitates are formed in a separate mixing tank.

VEGP Response:

As discussed in the Response to 3.o.2.12, VEGP 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.

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.

    VEGP Response:

    As discussed in the response to item 3.o.2.12, VEGP 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. See the Figure 1 flow chart in "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Plant-Specific Chemical Effect Evaluations" from March 2008 (Reference 106).

    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.

    VEGP Response:

    See the Response to 3.o.11.i.

    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).

    VEGP Response:

    See the Response to 3.o.11.i.

12. Pre-Mix in Tank: Licensees should discuss any exceptions taken to the procedure recommended for surrogate precipitate formation in WCAP-16530-NP-A.

    VEGP Response:

    The chemical head loss tests employed the pre-mixed chemical surrogate methodology. The WCAP-16530-NP-A precipitate formation methodology for SAS and calcium phosphate was followed with no exceptions. The 1-hour settling volume for each batch of chemical precipitates was determined at the time that the batch was produced and was required to be 6 ml or greater. The chemical precipitate settling was also required to be measured within 24 hours of the time the surrogate was to be used, and the 1-hour settled volume was required to be 6 ml (SAS and AlOOH) or greater and within 1.5 ml of the freshly prepared surrogate (Reference 73). Chemical precipitates that failed the criteria

of being 6 ml or greater (initial test or re-test) and within 1.5 ml of the freshly prepared surrogate criteria were not used in testing.

13. Technical Approach to Debris Transport (Decision Point): State whether near-field settlement is credited or not.

    VEGP Response:

    VEGP chemical effects testing used agitation and turbulence in the test tank to ensure that essentially all debris analyzed to reach the strainer in the plant reached the strainer in head loss testing. VEGP did not credit any near field settlement in head loss testing.

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.

       VEGP Response:

       VEGP is not crediting near field settlement of chemical precipitate in chemical head loss testing. See the Figure 1 flow chart in "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Plant-Specific Chemical Effect Evaluations" from March 2008 (Reference 106).

    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.

       VEGP Response:

       See the Response to 3.o.2.14.i.

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.

       VEGP Response:

       Even though all debris had an opportunity to collect on the surfaces of the test strainer, a portion of the debris added to the test settled on the floor. Post-test photographs show that while nearly all of the Nukon had reached the strainers, approximately 10–15 percent of the (larger/heavier) dirt/dust surrogate and nearly all of the Interam debris settled on the floor in front of

the strainer array (see Attachment A). Additionally, a minor amount (approximately 10–20 lb.) of silicon carbide settled underneath the simulated containment floor, and less than 1.5 lbm of particulate debris was removed along with the water drained from the tank to ensure sufficient volume in the tank for chemical additions. Finally, a small amount of calcium phosphate (less than 0.25 L of the 480.72 L total) was spilled outside the tank such that it was unrecoverable. Because of the measures taken during the test, as described in Response to 3.f.12, to keep debris suspended and transportable to the test strainer, the amount of settled debris described above is considered acceptable.

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).

VEGP Response:

See the Response to 3.o.2.12.

16. Test Termination Criteria: Licensees should provide the test termination criteria.

VEGP Response:

The head loss was considered stable when the differential pressure across the debris bed changed by less than or equal to 1 percent over a 1-hr period. In addition, the rate of head loss increase was required to be significantly decreasing, or the head loss was required to be consistently steady at termination of the test.

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.

    VEGP Response:

    The pressure drop curves for the full load test are provided as Figures 3.o.2.17-1 through 3.o.2.17-4 below.

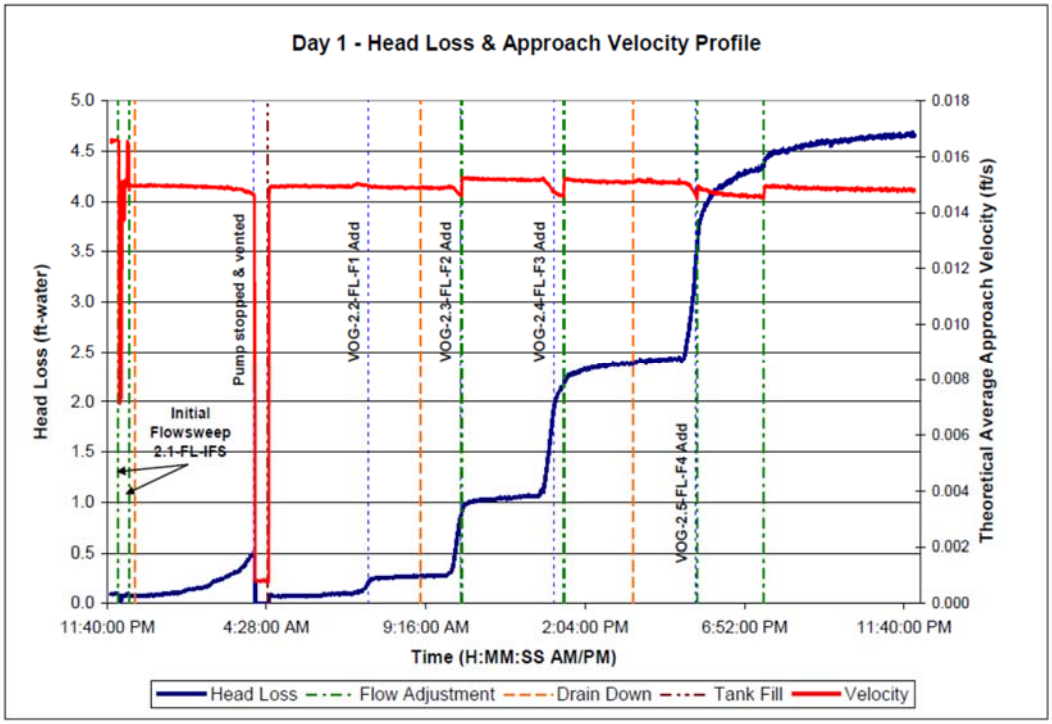

Figure 3.o.2.17-1: Full Load Test Differential Pressure and Velocity vs. Time – Day 1

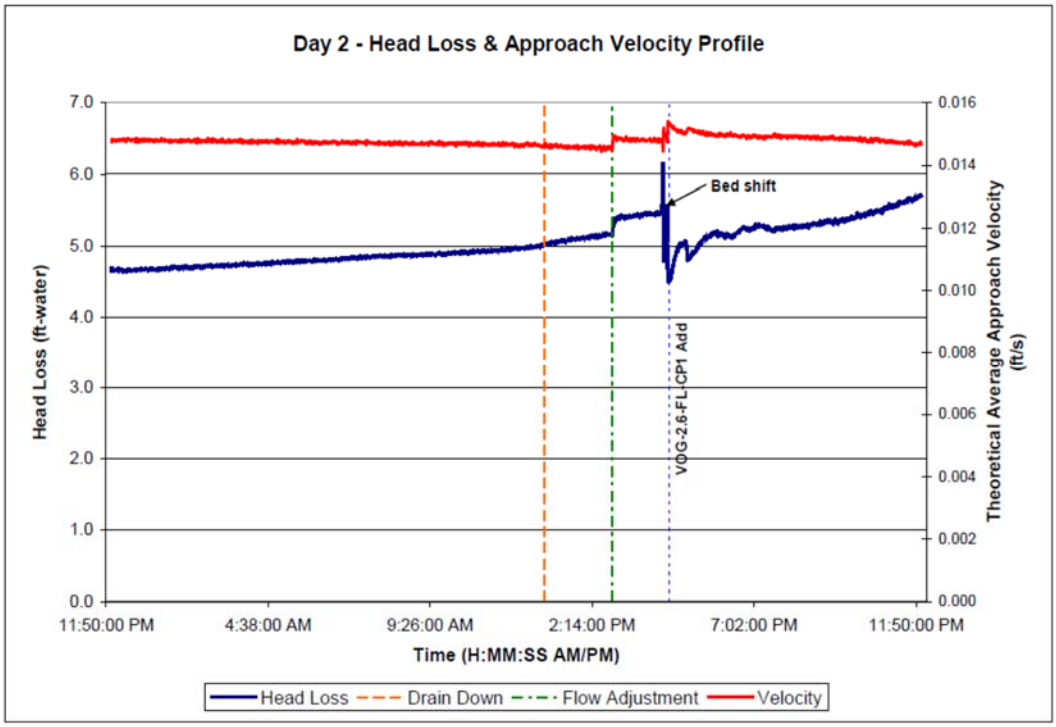

Figure 3.o.2.17-2: Full Load Test Differential Pressure and Velocity vs. Time – Day 2

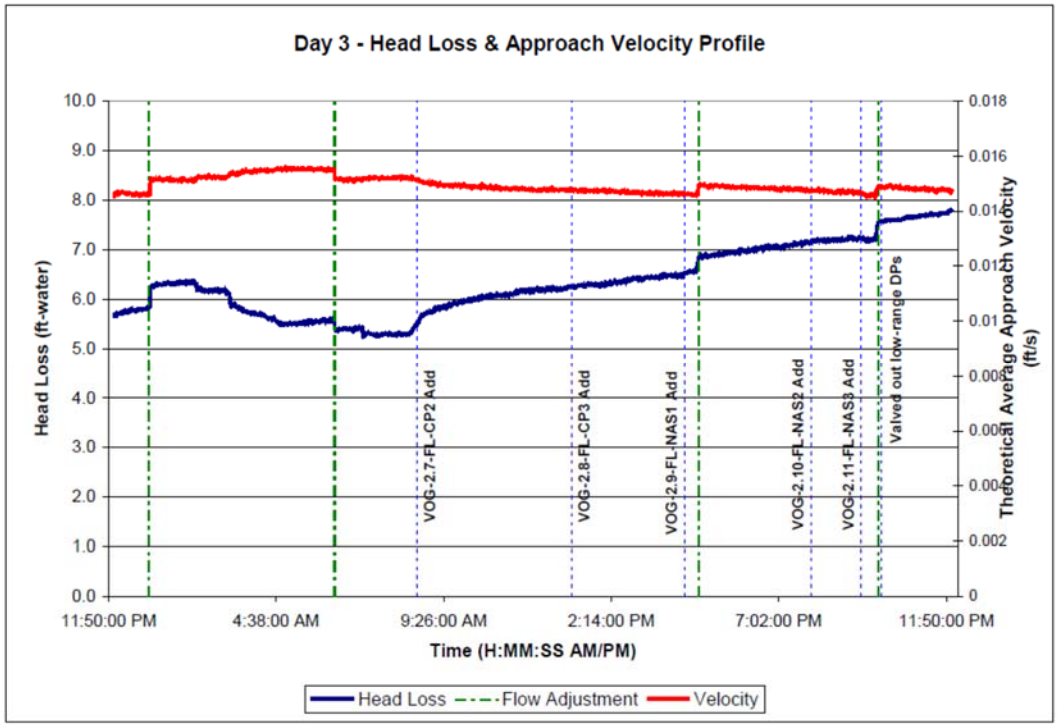

Figure 3.o.2.17-3: Full Load Test Differential Pressure and Velocity vs. Time – Day 3

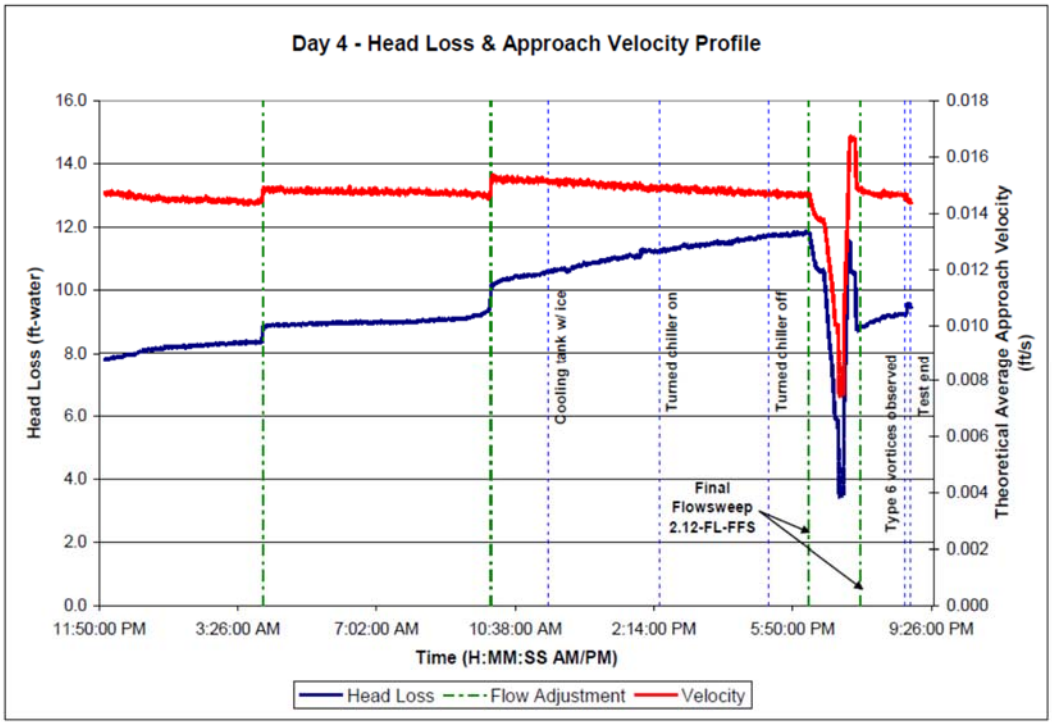

Figure 3.o.2.17-4: Full Load Test Differential Pressure and Velocity vs. Time – Day 4

ii. Licensees should explain any extrapolation methods used for data analysis.

VEGP Response:

See the Response to 3.f.10.

18. Integral Generation (Alion): Licensees should explain why the test parameters (e.g., temperature, pH) provide for a conservative chemical effects test.

VEGP Response:

VEGP is using the separate chemical effects approach to determine the chemical source term. This section is not applicable to the VEGP chemical effects analysis.

19. Tank Scaling / Bed Formation:

    i. Explain how scaling factors for the test facilities are representative or conservative relative to plant-specific values.

       VEGP Response:

       VEGP is using the separate chemical effects approach to determine the chemical source term. This section is not applicable to the VEGP chemical effects analysis.

    ii. Explain how bed formation is representative of that expected for the size of materials and debris that is formed in the plant specific evaluation.

       VEGP Response:

       VEGP is using the separate chemical effects approach to determine the chemical source term. This section is not applicable to the VEGP chemical effects analysis.

20. Tank Transport: Explain how the transport of chemicals and debris in the testing facility is representative or conservative with regard to the expected flow and transport in the plant-specific conditions.

    VEGP Response:

    VEGP is using the separate chemical effects approach to determine the chemical source term. This section is not applicable to the VEGP chemical effects analysis.

21. 30-Day Integrated Head Loss Test: 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.

    VEGP Response:

    VEGP is using the separate chemical effects approach to determine the chemical source term. This section is not applicable to the VEGP chemical effects analysis.

22. Data Analysis Bump Up Factor: 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.

    VEGP Response:

    VEGP is using the separate chemical effects approach to determine the chemical source term. This section is not applicable to the VEGP chemical effects analysis.

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.

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:

   VEGP is following the "STP Piloted Risk-Informed Approach for GSI-191," (References 44 and 45). The proposed change replacing the current deterministic methodology with a risk-informed methodology requires changes to the descriptions of how VEGP meets 10 CFR 50.46(a)(1), GDC 35, GDC 38, and GDC 41. Those changes require exemptions to certain requirements of 10 CFR 50.46(a)(1), GDC 35, GDC 38, and GDC 41, and the requests for the exemptions are provided in the future LAR.

   VEGP's risk-informed approach to assess the effects of LOCA debris replaces the existing deterministic approach described in the VEGP licensing basis. This, in turn, requires an amendment to the VEGP Unit 1 and Unit 2 operating licenses to incorporate the revised methodology per the requirements of 10 CFR 50.59. This proposed amendment to the operating license is included in the future LAR.

4.0 NRC Request for Additional Information:

VEGP received two requests for additional information from the NRC: NL-08-1497 and NL-08-1829 (References 101 and 87, respectively). The first RAI, issued September 17, 2008, addressed critical issues with test protocols used in chemical effects testing performed at the VUEZ facility by ALION Science and Technology. SNC responded to the VUEZ-specific RAIs with the issuance of NL-08-1583 (Reference 102), November 7, 2008. SNC determined the need to consider an alternate approach to demonstrating adequate performance of the emergency containment sump strainers after careful consideration of the NRC's concerns with the VUEZ test protocol. Therefore, the VUEZ-specific RAIs are no longer applicable and do not require a response, as recorded in NL-08-1583.

The second RAI, NL-08-1829, was issued December 2, 2008, containing 29 requests based on the review of all four VEGP Supplemental Responses to GL 2004-02: NL-07-1777, NL-08-0670, NL-08-1155, and NL-08-1228 (References 95, 96, 98, and 100, respectively). The final SNC responses to RAIs 1 through 29 are referenced in the table below.

Additionally, SNC provided its intended path forward for the resolution of GSI-191 letter to the NRC, NL-13-0953 (Reference 105). By letter dated November 14, 2013, the NRC sent SNC an RAI. The RAI and SNC's response are included below as provided in SNC letter to the NRC, NL-13-2544 (Reference 110). SNC has since revised the ERGs as discussed in the response below.

<u>NRC RAI</u>

SNC's May 16, 2013, letter did not identify that SNC had implemented, or identified for future implementation, any mitigative measures to deal with the potential for in-vessel blockage. SNC stated that it was evaluating Westinghouse recommendations for mitigative measures for in-vessel blockage and that appropriate procedure changes and operator training would be completed, as deemed necessary, following the evaluation. Please provide the mitigative measures chosen for the VEGP to deal with in-vessel blockage, should it occur.

<u>SNC Response to NRC RAI</u>

SNC will make changes to the Vogtle Units 1 and 2 Emergency Response Guidelines (ERGs). Specifically, after transferring to cold leg recirculation, Vogtle will monitor core exit temperatures, injection flow, and reactor vessel level indicating system (RVLIS) indications to identify any abnormal indications. Should abnormal indications exist, realignment to hot leg recirculation and back flowing through the reactor vessel will be necessary.

In April 2017, SNC submitted a GL 2004-02 supplemental response that superseded all previous responses (Reference 112). In 2017 and 2018, SNC has responded to NRC's requests for additional information after their review of the April 2017 submittal. The responses were documented in References 113 through 117.

RAI No.	Complete RAIs as Provided in NL-08-1829 (Reference 87)	Response
1	Please provide a description of the jacketing/banding systems used to encapsulate Nukon insulation at Vogtle (e.g., on piping, steam generators, reactor coolant pumps) and during jacketing/banding system qualification testing. The information should include the jacket materials used in the testing, geometries and sizes of the targets and jet nozzle, and materials used for jackets installed in the plant. Please provide information that compares the mechanical configuration and sizes of the test targets and jets, and the potential targets and two-phase jets in the plant. Pleases evaluate how any differences in jet/target sizing and jet impingement angle affect the ability of the insulation system to resist damage from jet impingement. In doing so, please provide a justification for applying debris generation test data obtained for the Nukon jacketing systems employed at the Wolf Creek and Callaway plants to the jacketing systems used at Vogtle and demonstrating that the Vogtle jacketing systems are as resistant to destruction as the jacketing systems tested. In responding to this question, please address the potential varied jacketing systems for various components of the reactor coolant system, which are within the LOCA ZOIs (e.g., piping, coolant pumps, and steam generators).	17.0D ZOI is used for all Nukon. WCAP-16710-P is no longer credited. Because we are no longer crediting the jacketing in the ZOI, a response to this portion of the question is not provided. See the Response to 3.c.1.
2	Please specify the ZOI radius used to calculate the quantity of Interam fire barrier debris that could be generated. Please provide the characteristics of the Interam fire barrier material including the type of Interam installed and its anticipated debris characteristics. Please provide information on how the material was prepared for inclusion in head loss testing or provide information on the surrogate material used and its properties. Please provide assumptions made regarding the physical properties of LOCA damaged Interam fire barrier material and the bases for how any surrogate material used in testing conservatively model these properties.	See the Response to 3.b.1, 3.c.1, and 3.f.4.

RAI No.	Complete RAIs as Provided in NL-08-1829 (Reference 87)	Response
3	Please provide the following additional information needed to support the assumption of 15% erosion of fibrous debris pieces in the containment pool: a. 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 expected plant conditions, and b. The durations of the erosion tests and how the test results were extrapolated to the sump performance mission time.	See the Response to 3.e.1. Note that a smaller fraction for erosion of fibrous debris pieces in the containment pool is used based on testing.
4	On pages E1-19 and E1-20 of the supplemental response dated February 28, 2008, it is indicated that, based on the fact that less than 25% of the strainer perimeter area is in excess of the curb lift velocity metric, 25% of small debris pieces are assumed to surmount the curb/plenum on which the strainer modules rest. However, based on the diagram of containment provided on page E1-9 of the same supplemental response, the staff expects that it is likely that flow and debris will preferentially approach the sump from openings in the shield wall. As a result, the fraction of debris approaching the sump in the higher velocity flow channel could significantly exceed 25%. In light of the considerations such as this, please provide a technical justification for the assumption that only 25% of small pieces of fibrous debris can surmount the curb/plenum and reach the sump strainers.	No longer applicable because the curb is not being credited as a debris interceptor. Any debris that transported during recirculation was assumed to reach the strainers. See the Response to 3.e.4 and 3.e.6.
5	The supplemental response states that the head loss test results were scaled to the full-sized strainer system based on temperature, velocity, and bed thickness differences. Without additional information on the methodology used to make these extrapolations, it is not possible to determine whether they were performed conservatively or prototypically. It appears that the head loss test result of 6.84 ft was extrapolated to 8.126 ft. Please provide the details of all extrapolations performed for the head loss test data. Please include the raw test data and conditions, and the final head loss value and the conditions to which it was extrapolated. Please include any differences in temperature, velocity, bed thickness between the head loss testing and anticipated plant conditions.	See the Response to 3.f.10.

RAI No.	Complete RAIs as Provided in NL-08-1829 (Reference 87)	Response
6	The supplemental response stated that the submergence value for the SBLOCA was not calculated. The stated [reason] for this was that an SBLOCA would create less debris and therefore result in a less challenging head loss. This is true for some portions of the evaluation. However, the vortexing evaluation is often most limiting when there is no debris on the strainer. In addition, if the strainer is not fully submerged, the acceptance criteria for maximum head loss may be limited by the strainer height (50% of the strainer height per RG 1.82, Rev. 3) instead of the pump NPSH margin. This would be a reduction to about 25% of the tested strainer head loss. Also, if the strainer is not submerged, air ingestion would have to be evaluated more rigorously. Un-submerged strainer area cannot be credited to accumulate debris, so other areas of the strainer would have to absorb the debris that cannot be collected on the uncovered portion of the strainer. Due to break location, the SBLOCA level may not include some RCS inventory and also may not include all or part of the accumulator volume. Please provide the minimum submergence for an SBLOCA. If the strainer is not fully submerged for this event, please provide appropriate evaluations for air ingestion and strainer head loss, including acceptability based on the guidance in RG 1.82 (or other appropriate methodology).	See the Response to 3.f.2 and 3.f.3.
7	Related to the RAI above on the response of the plant to cases where the strainers may not be fully submerged, have various scenarios such as an SBLOCA with the failure of one train of ECCS and no CS actuation been considered? For this case all debris would transport to a single ECCS strainer that may not be fully submerged. A thin bed with the bulk of the particulate debris could form on the operating strainer surface. Please provide an evaluation that demonstrates that adequate NPSH margin will be provided to the ECCS pumps (reference Regulatory Guide 1.82, Revision 3, Section 1.3.4.4).	See the Response to 3.g.6
8	It was implied that the debris was added to the testing prior to starting the recirculation pump. Please provide justification that this test sequence provides prototypical or conservative test conditions.	Debris was added after starting the recirculation pump. See the Response to 3.f.4.

RAI No.	Complete RAIs as Provided in NL-08-1829 (Reference 87)	Response
9	Scaling was based on the circumscribed area for the test and plant strainers. Normally scaling is based on the screen area. For the module testing the scaling factors based on circumscribed and screen area appear to be the same. The scaling factors for the sector testing could not be determined by the staff. Please provide information that justifies the use of the circumscribed area of the test and plant strainers for scaling of the sector tests.	Scaling was based on screen area for test and plant strainers. See the Response to 3.f.4 and 3.f.10
10	Without information on how debris was prepared and introduced into the thin bed tests it cannot be concluded that the thin bed testing was valid. It is possible that a properly conducted thin bed test would result in higher head loss than the full load test that was stated to be the limiting head loss condition for Vogtle. Please provide information that justifies that the sector testing conducted to determine the strainer's ability to deal with a thin bed was conducted under conditions that would conservatively model the debris bed. Please reference the staff Head Loss and Vortexing Guidance for thin bed testing considerations (ADAMS Accession No. ML080230038).	See the Response to 3.f.4 and 3.f.6.
11	The supplemental response stated that air ingestion was evaluated at the top of the module. The results of the vortexing evaluation were not provided. Please provide the results of the air ingestion/vortexting evaluation including the plant conditions assumed.	See the Response to 3.f.3.
12	No documentation of fiber size distribution used for testing compared to the fiber size distribution predicted to arrive at the strainer was provided. The supplemental response stated that only fine and small pieces of fiber would be created by the break. The size distribution of the fibrous debris used in testing was not provided. In general shredded fiber does not imply that all fine fibers are created. For thin bed testing, only fine fibrous debris should be added to the test flume until all fibrous fines predicted to be created are added. Please provide information regarding the size distribution of fibrous debris used in various tests and how these size distributions compare to the transport evaluation predictions.	See the Response to 3.f.4.

RAI No.	Complete RAIs as Provided in NL-08-1829 (Reference 87)	Response
13	The documentation of fibrous concentration during addition and methods of addition to the flume were unclear. Documentation should be provided showing that the concentration of debris during addition was controlled so that non-prototypical agglomeration of the debris would not occur. Please provide information that justifies that the debris introduction methods used during testing did not result in non-prototypical settling or agglomeration of debris. Also, please include the amounts of debris added during each addition, the actual size distribution of the debris, and the debris types.	Tank test was utilized which did not allow settling of debris. See the Response to 3.f.4.
14	Documentation of the amount of debris that settled in the agitated and non-agitated areas of the test tanks was not provided. Please provide the amount of debris that settled in the agitated and non-agitated areas of the test tank for each test.	See the Response to 3.f.1 and 3.f.4.
15	There is no discussion of extrapolation of head loss test results to ECCS mission times, nor discussion of test termination criteria and subsequent extrapolation. Please provide information that shows that the head loss testing was run to a maximum value, or that an extrapolation was performed to obtain the head loss at the end of the strainer mission time. Please provide sufficient time dependent test data so that the termination criteria and any extrapolations conducted can be verified. Please provide a graph of the head loss over time for the limiting module and sector tests. Please specify the sector test that created the limiting head loss.	See the Response to 3.f.10.
16	The flashing evaluation did not describe the margin to flashing through the strainer. The supplemental response stated that overpressure is credited, but the amount of overpressure required was not provided, nor was the available margin. The total head loss (without chemicals) is about 8 ft with a submergence of about 3 inches (LBLOCA). Please provide the minimum margin to flashing across the strainer throughout the strainer mission time. Please provide the assumptions used to determine this value.	See the Response to 3.f.14.

RAI No.	Complete RAIs as Provided in NL-08-1829 (Reference 87)	Response
17	The supplemental response stated that the vortex testing was conducted at a submergence that may have been slightly greater (non-conservative) than that expected under LBLOCA conditions (3.4 in vs. 3.675 +/- 0.5 in). Another section of the supplemental response stated that the testing was conducted with a representative or conservatively lowered water level, but no other details were provided. No vortex testing appears to have been conducted for SBLOCA conditions. No details on the test flow rates for vortex evaluations were provided. Vortex testing should be conducted with the minimum potential submergence and the maximum potential scaled flow rate through the strainer. Please provide an evaluation of vortex formation for the minimum level at which the strainer is required to operate (likely an SBLOCA condition). Please verify that the flow rates used in the vortex testing were conservative including the potential for higher flow rates in some sections of the strainer (generally those hydraulically closer to the pump suction). Please verify that the level that was tested for the LBLOCA case was in fact conservatively low. Please provide the submergence value for LBLOCA testing. Alternatively, please provide an updated evaluation considering all of these considerations.	See the Response to 3.f.3.
18	The clean strainer head loss (CSHL) calculation methodology was not provided. It was not clear how the CSHL was divided between strainer module head loss and piping head loss. Please provide the methodology used to determine CSHL. Please provide information indicating that each section of the strainer, plenum, or piping was included in the calculation, the head loss value for each section, and the method used to determine the head loss for each strainer section. Please include any assumptions made for each portion of the calculation.	See the Response to 3.f.9.
19	The supplemental response stated that for the sector tests debris was maintained in suspension using stirring. No information was provided to show that the stirring did not drive non-prototypical debris onto the bed nor prevent debris from collecting naturally on the strainer during these tests. For the module tests, from the provided diagrams it appeared that the stirrer was far enough from the strainer to prevent non-prototypical bed formation. Please provide information that justifies that the debris beds were not disrupted by the stirring and that the stirring did not result in non-prototypical debris accumulation on the strainer (accumulation of larger sizes of debris than would be expected).	Sector test is no longer utilized. See the Response to 3.f.4.

RAI No.	Complete RAIs as Provided in NL-08-1829 (Reference 87)	Response
20	During module tests stirring was used outside the area of the strainer. The supplemental response stated that the flow in the test flume conservatively represented the plant flow patterns in the area of the strainer to ensure that non-prototypical settling would not occur. No details were provided on how the plant and flume flow rates were modeled to assure that flow and turbulence would be prototypical. In general flow patterns in the plant are affected by, for example, upstream conditions, drainage into the sump pool, flow rates from various locations, upstream obstructions, and obstructions near the strainer. The boundary conditions in the models for determining typical plant flow patterns should be prototypical or conservative. Please provide information that justifies that the flow rates and patterns in the test flume for the module tests were prototypical or conservative with respect to plant conditions.	See the Response to 3.f.4.
21	Please provide the basis for the statement in the supplemental response that the debris head loss for the RHR strainers bounds the head loss for the containment spray strainers.	RHR and CS strainer head losses are evaluated independently. See the Response to 3.f.10.
22	Because of the large volume of debris and the relatively low submergence of the strainer it is possible for debris to collect on top of the strainer and provide a pathway for air ingestion. This was not discussed in the supplemental response. Air ingestion could result from a damming effect, or, if head loss exceeds submergence and holes form in the debris bed, these holes could allow air to be ingested through the debris bed. Please provide an evaluation of the potential for debris to collect on top of the strainer and provide a pathway for air ingestion into the strainer.	See the Response to 3.f.3.
23	It was unclear how varying the debris loading affected the results in all of the head loss testing. Please provide the debris amounts added to each test, the resulting theoretical bed thicknesses, and the maximum head loss determined for each test.	See the Response to 3.f.4, 3.f.6, 3.f.7, and 3.f.10.

RAI No.	Complete RAIs as Provided in NL-08-1829 (Reference 87)	Response
24	Please provide the containment sump/pool level both soon after the realignment to containment spray recirculation as well as at the time post-accident when all assumed water contributions and diversions/hold-ups have completely taken effect (except for subsequent sump/pool water thermal contraction). The differences in the assumptions and results for these two cases should be clearly explained, as should the times when the short-term and long-term results are applicable. The strainer submergence should be provided for both cases.	See the Response to 3.f.2 and 3.g.1. See the Response to 3.g.2 and 3.g.9.
25	The second portion of item 3.k of the revised content guide for the GL2004-02 supplemental responses requests that the licensee "summarize the structural qualification results and design margins for the various components of the sump strainer structural assembly." Please provide the actual and allowable stresses and show the design margins for the 16 bolt locations of the strainer base frame (in addition to the reaction forces already provided in Table 3.k.2-8 of the supplemental response).	See the Response to 3.k.2.
26	Item 3.k.3 of the revised content guide for the GL2004-02 supplemental responses requests that the licensee "summarize the evaluations performed for dynamic effects such as pipe whip, jet impingement, and missile impacts associated with high-energy line breaks (as applicable)." In addition to the information provided in your September 2005 and February 2008 responses, please submit a detailed summary along with any additional supporting information regarding your assessment that the strainers are not subject to the aforementioned dynamic effects.	See the Response to 3.k.3.
27	Please provide additional basis for concluding that the refueling cavity drains would not become blocked with debris. Please identify the potential types and characteristics of debris that could reach these drains. In particular, could large pieces of debris be blown into the upper containment by pipe breaks occurring in the lower containment, and subsequently fall into the cavity? In the case that partial/total blockage of the drains might occur, what would be the impact to minimum sump water level and ECCS and CS pump NPSH? Are there any potential flow restrictions in the two 12-inch refueling cavity drain lines (e.g., valves, meshing or gratings), and if so, how are these potential restrictions addressed so as ensure that these lines are not blocked during a LOCA?	See the Response to 3.l.4.

RAI No.	Complete RAIs as Provided in NL-08-1829 (Reference 87)	Response
28	The NRC staff considers in-vessel downstream effects to not be fully addressed at Vogtle Units 1 and 2, as well as at other PWRs. The supplemental response 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 safety evaluation (SE) for WCAP-16793-NP. The licensee may demonstrate that in-vessel downstream effects issues are resolved for Vogtle Units 1 and 2 by showing that the Vogtle 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 alternatively resolve this item by demonstrating, without reference to WCAP-16793-NP or the staff SE, that in-vessel downstream effects have been addressed at Vogtle Units 1 and 2. In any event, the licensee should report how it has addressed the in-vessel downstream effects issue within 90 days of issuance of the final NRC staff SE on WCAP-16793-NP. The NRC staff is developing a Regulatory Issue Summary to inform the industry of the staff's expectations and plans regarding resolution of this remaining aspect of GSI-191.	See the Response to 3.n.1.
29	The NRC Staff understands that SNC has changed its test approach to evaluate chemical effects. Please submit the revised chemical effects test results and analyses to the NRC when they become available.	See the Response to 3.o.1.

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," dated September 13, 2004
2. NEI Guidance Report NEI 04-07, Revision 0, "Pressurized Water Reactor Sump Performance Evaluation Methodology 'Volume 1 – Pressurized Water Reactor Sump Performance Evaluation Methodology'," December 2004
3. NEI Guidance Report 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, Revision 0, December 6, 2004'," December 2004
4. Not Used
5. NRC Bulletin 2003-01, "Requests for Additional Information, Bulletin 2003-01, Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors" for VEGP Electric Generating Plant, Units 1 and 2, Docket Nos. 50-424 and 50-425
6. 60 Day Response to NRC Bulletin 2003-01
   SNC-to-NRC NL-03-1514 dated 8/07/2003

Combined SNC response for Joseph M. Farley Nuclear Plant (FNP) and Vogtle Electric Generating Plant (VEGP) as required by NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors" (ML032240030)

7. Response to a Request for Additional Information on NRC Bulletin 2003-01
   NRC-to-SNC (NL-04-2013) dated 10/29/2004
   Combined SNC response for Joseph M. Farley Nuclear Plant (FNP) and Vogtle Electric Generating Plant (VEGP) as required by NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors"
8. Revised Response to a Request for Additional Information on NRC Bulletin 2003-01
   NRC-to-SNC (NL-05-1207) dated 7/22/2005
   Combined SNC response for Joseph M. Farley Nuclear Plant (FNP) and Vogtle Electric Generating Plant (VEGP) as required by NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors - Revision 1"
9. NRC-to-SNC (NL-05-1633) dated 8/26/2005
   Vogtle Electric Generating Plant, Units 1 and 2 - Response to NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors" (TAC Nos. MB9625 and MB9626)
10. 90 day response to GL 2004-02
    SNC-to-NRC NL-05-0290 dated 2/25/2005 (ML052430746)
    Joseph M. Farley Nuclear Plant, Vogtle Electric Generating Plant, Response to NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors"
11. Response (VEGP and FNP) to GL 2004-02
    SNC-to-NRC NL-05-1264 dated 8/31/2005
    Combined SNC response for Joseph M. Farley Nuclear Plant (FNP) and Vogtle Electric Generating Plant (VEGP) as required by NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors"
12. NRC Request for Additional Information
    NRC-to-SNC (NL-06-0279) dated 2/9/2006
    Vogtle Electric Generating Plant, Units 1 And 2, 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 Nos. MC4727 and MC4728)
13. NRC-to-SNC (NL-06-0753) dated 3/28/2006 (ML060870274)
    Alternative Approach for Responding to the Nuclear Regulatory Commission Request for Additional Information Letter Re: Generic Letter 2004-02,
14. VEGP 1st extension request to complete CAs (Unit 1 downstream effects) for GL 2004-02
    SNC-to-NRC (NL-06-1275) dated 6/22/06 (ML061730462)
    Vogtle Electric Generating Plant - Units 1 and 2 Request for Extension for Completing Corrective Actions for Generic Letter 2004-02, "Potential Impact of

Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors"

15. SNC-to-NRC (NL-06-1483) dated 7/28/2006
    Response to NRC RAI (6/30/06 phone call)on SNC Request for Extension for Completing Corrective Actions for Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors"
16. NRC-to-SNC (NL-06-2055) dated 9/7/2006 (ML062500269)
    Vogtle Electric Generating Plant, Unit 1, Approval of Generic Letter 2004-02 Extension Request (SNC request dated 6/22/2006)
17. NRC-to-NEI (NL-06-2686) dated 11/14/2006
    Nuclear Regulatory Communication Request for Additional Information to Pressurized Water Reactor Licensees Regarding Reponses to Generic Letter 2004-02
18. NRC-to-All Licenses (NL-07-0090) dated 1/4/2007
    Alternative Approach for Responding to the NRC request for Additional Information Letter Regarding GL 2004-02
19. SNC-to-NRC (NL-07-1969) dated 12/7/2007
    Vogtle Electric Generating Plant Units 1 and 2 Generic Letter 2004-02 Response Extension Request for completion of Chemical Effects testing and analysis, Downstream Effects analysis for Components - Systems, and Fuel - Vessel
20. NRC-to-SNC (NL-07-2367) dated 12/19/2007
    Vogtle Electric Generating Plant, Units 1 and 2 -Generic Letter 2004-02 "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," Extension Request Approval (to May 31, 2008)
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22. WCAP-16793-NP-A Revision 2.0, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," July 2013
23. WCAP-16568-P Revision 0.0, "Jet Impingement Testing to Determine the Zone of Influence (ZOI) for DBA-Qualified / Acceptable Coatings"
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26. NUREG/CR-0800, Revision 1, "U.S. Nuclear Regulatory Commission Standard Review Plan," Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," July 1981
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28. NUREG/CR-3616, "Transport and Screen Blockage Characteristics of Reflective Metallic Insulation Materials," January 1984
29. NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris, Final Report," Issued October 1995

30. NUREG/CR-6369, “Drywell Debris Transport Study, Final Report,” Volume 1, Issued September 1999
31. NUREG/CR-6369, “Drywell Debris Transport Study: Experimental Work, Final Report,” Volume 2, Issued September 1999
32. NUREG/CR-6369, “Drywell Debris Transport Study: Computational Work, Final Report,” Volume 3, Issued September 1999
33. NUREG/CR-6762, Volume 1, “GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance,” Issued August 2002
34. NUREG/CR-6762, Volume 2, “GSI-191 Technical Assessment: Summary and Analysis of U.S. Pressurized Water Reactor Industry Survey Responses and Responses to GL 97-04,” Issued August 2002
35. NUREG/CR-6762, Volume 3, “GSI-191 Technical Assessment: Development of Debris Generation Quantities in Support of the Parametric Evaluation,” Issued August 2002
36. NUREG/CR-6762, Volume 4, “GSI-191 Technical Assessment: Development of Debris Transport Fractions in Support of the Parametric Evaluation,” Issued August 2002
37. NUREG/CR-6772, “GSI-191: Separate Effects Characterization of Debris Transport in Water,” Issued August 2002
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40. NUREG/CR-6916, “Hydraulic Transport of Coating Debris, A Subtask of GSI-191,” Issued December 2006
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42. Westinghouse Technical Bulletin, TB-06-15, “Unqualified Service Level 1 Coatings on Equipment in Containment,” Dated September 28, 2006
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44. STPNOC Letter NOC-AE-13003043 to NRC, “Supplement 1 to Revised STP Pilot Submittal and Requests for Exemptions and Licensing Amendment for a Risk-Informed Approach to Resolving Generic Safety Issue (GSI)-191,” November 13, 2013 (ML13323A183)
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56. NUREG/CR-6877, Characterization and Head-Loss Testing of Latent Debris from Pressurized-Water-Reactor Containment Buildings, July 2005
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