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

April 15, 2014

Mr. Dennis L. Koehl
President and CEO/CNO
STP Nuclear Operating Company
South Texas Project
P.O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 – REQUEST FOR ADDITIONAL INFORMATION RELATED TO REQUEST FOR EXEMPTIONS AND LICENSE AMENDMENT FOR USE OF A RISK-INFORMED APPROACH TO RESOLVE THE ISSUE OF POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING DESIGN-BASIS ACCIDENTS AT PRESSURIZED-WATER REACTORS (TAC NOS. MF2400 AND MF2401)

Dear Mr. Koehl:

By letter dated June 19, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML131750250), STP Nuclear Operating Company (STPNOC, the licensee) submitted exemption requests accompanied by a license amendment request for a risk-informed approach to resolve the issue of potential impact of debris blockage on emergency recirculation during design-basis accidents Generic Safety Issue (GSI)- 191) for South Texas Project, Units 1 and 2. This initial submittal was superseded in its entirety by a revised version by letter dated November 13, 2013 (ADAMS Accession No. ML13323A128). In addition, the licensee provided additional information by letters dated November 21, 2013, two letters dated December 23, 2013, and letters dated January 9, February 13, February 27, March 17, and March 18, 2014 (ADAMS Accession Nos. ML13338A165, ML14015A312, ML14015A311, ML14029A533, ML14052A053, ML14072A076, ML14086A383; and ML14087A126, respectively). The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the information provided in your application and determined that additional information, as described in the enclosure to this letter, is required to complete review of your application.

A draft copy of the enclosed request for additional information (RAI) was provided to Mr. Wayne Harrison of your staff via e-mail on February 26, 2014. A clarification call was held on March 10, 2014 to discuss the RAI request. A separate call was held on March 26, 2014, to discuss the schedule for RAI response. Mr. Wayne Harrison agreed that STPNOC will provide response to the requested information by June 13, 2014.

If you have any questions, please contact me at 301-415-3016 or via e-mail at Balwant.Singal@nrc.gov.

Sincerely,

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosure:
Request for Additional Information

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

EXEMPTION REQUESTS AND LICENSE AMENDMENT REQUEST

RISK-INFORMED APPROACH TO RESOLVE THE ISSUE OF POTENTIAL

IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING

DESIGN-BASIS ACCIDENTS AT PRESSURIZED-WATER REACTORS

STP NUCLEAR OPERATING COMPANY

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

By letter dated June 19, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML131750250), STP Nuclear Operating Company (STPNOC, the licensee) submitted exemption requests accompanied by a license amendment request (LAR) for a risk-informed approach to resolve the issue of potential impact of debris blockage on emergency recirculation during design basis accidents (Generic Safety Issue (GSI)- 191) for South Texas Project (STP), Units 1 and 2. This initial submittal was superseded in its entirety by a revised version by letter dated November 13, 2013 (ADAMS Package Accession No. ML13323A128). In addition, the licensee provided additional information by letter dated November 21, 2013, two letters dated December 23, 2013, and letters dated January 9, February 13, February 27, March 17, and March 18, 2014 (ADAMS Accession Nos. ML13338A165, ML14015A312, ML14015A311, ML14029A533, ML14052A053, ML14072A076, ML14086A383; and ML14087A126, respectively. The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the information provided in the application and determined that the following additional information is required to complete its review.

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Please note that some of the RAIs may have already been transmitted formally and the licensee may have already provided responses to some of them. However, these RAIs are included below for completeness and are identified in *italics.* The licensee may respond to these RAIs by way of providing a reference to the previous letter transmitting the response.

Enclosure

Enclosure 4 to the licensee's letter dated November 13, 2013, contained Enclosures 4-1, 4-2, and 4-3 (defined as Volumes 1, 2, and 3, respectively). Also, Enclosure 5 was defined as Volume 6.2. The RAIs make multiple references to Volumes 1, 2, 3, and 6.2. Please note the following, as it relates to these references:

- "Volume 1" refers to Enclosure 4-1 of the LAR dated November 13, 2013, Project Summary
- "Volume 2" refers to Enclosure 4-2 of the LAR dated November 13, 2013, Probabilistic Risk Assessment (PRA)
- "Volume 3" refers to Enclosure 4-3 of the LAR dated November 13, 2013, Engineering (CASA [Containment Accident Stochastic Analysis] Grande) Analysis
- "Volume 6.2" refers to Enclosure 5 of the LAR dated November 13, 2013, Responses to NRC Request for Supplemental Information on the 2013 Submittal

REQUEST FOR ADDITIONAL INFORMATION

Probabilistic Risk Assessment Licensing Branch (APLA)

CASA GRANDE

General

1. Regulatory Guide (RG) 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 (ADAMS Accession No. ML100910006), Section 2.3.3, "Probabilistic Risk Assessment Technical Adequacy," states that the PRA model should be technically adequate for the application. Volume 6.2 contains a list of input variables and describes whether each variable was modeled as a point estimate or a distribution in CASA Grande. According to item 5.d (page 158 of 179), the decision as to whether to use a point estimate or a distribution was based on the availability of data for uncertainty analysis and the available consensus on the values assigned to specific factors (e.g., for some values, there is a high level of confidence by industry and NRC). Please describe the process used to assign point estimates or distributions in more detail. For each input parameter, please provide:

    a. The basis for using a point estimate or a distribution.

    b. The source of the parameter value (e.g., licensing basis calculation).

    c. Whether the parameter is based on an NRC-accepted value (e.g., as documented in the safety evaluation (SE) for NEI-04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology" (Package ADAMS Accession No. ML043280641).

Plant Configuration

1. RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2 (ADAMS Accession No. ML090410014), Section 1.4, "PRA Development, Maintenance, and Upgrade," states that plant information used in the PRA (e.g., expected thermal-hydraulic plant response to different states of equipment) should be as realistic as possible. Thermal-hydraulic simulations described in Volume 6.2 show that pool temperature is affected by parameters such as loss-of-coolant accident (LOCA) break size, component cooling water (CCW) temperature, and the status of containment spray, residual heat removal (RHR), and containment fan coolers. The simplified curves used by CASA Grande (Volume 3, Figure 2.2.13) assume "nominal" values for these parameters and an intact containment (Volume 6.2, page 6) yet are assumed to be bounding based on qualitative arguments stated by Volume 3, Assumption 1k, page 72.

    a. Please state if varying the aforementioned (or other) parameters from their nominal values produce time-temperature curves that would yield higher conditional probabilities of sump or core blockage for any duration of time during the event. Please include consideration of all 15 pump state-LOCA size combinations (i.e., five pump cases 1, 9, 22, 26, and 43, and three LOCA categories (S/M/L [small/medium/large]) for both scenarios where containment is intact and not intact.

    b. Please provide a technical justification for using only nominal values or calculate core damage frequency (CDF), large early release frequency (LERF), delta-CDF (ΔCDF), and delta-LERF (ΔLERF) using time-temperature curves that maximize the probability of sump and core blockage for the entire assumed duration of the event.

2. RG 1.174, Section 2.3.2, "Level of Detail Required to Support an Application," states that the level of detail of the PRA model must be sufficient to model the impact of the proposed change. Section 2.2.8 of Volume 3 states that Table 2.2.14 safety injection (SI) flow rates are based on simulations using nominal operating conditions (i.e., all emergency core cooling system (ECCS) trains operating, all fan coolers operating, and nominal CCW heat exchanger temperatures). Furthermore, Volume 2, page 38 states that "to evaluate the potential for generic safety issue (GSI)-191 phenomena, the total pump flow from the sump is the most important consideration."

    a. Please state whether varying the aforementioned (or other) operating conditions from their nominal values could produce flow rates or other thermal-hydraulic conditions that would yield higher conditional probabilities of sump or core blockage for any duration of time during the event. Please include consideration of all 15 pump state-LOCA size combinations (i.e., five pump cases 1, 9, 22, 26, and 43, and three LOCA categories (S/M/L) for both scenarios where containment is intact and not intact.

    b. Please provide a technical justification for assuming only nominal operating conditions or calculate CDF, LERF, ΔCDF, and ΔLERF using flow rates or other

thermal-hydraulic conditions that maximize the probability of sump and core blockage for the entire assumed duration of the event.

3. RG 1.174, Section 2.3.2 states that the level of detail of the PRA model must be sufficient to model the impact of the proposed change. Volume 3 (page 71 of 248), Assumption 2b provides a qualitative argument for why a combination of pumps failing in the same train is "worse" than the same set of pumps failing in different trains. This qualitative argument includes a set of examples captured in Volume 3, Tables 3.1, 3.2, and 3.3.

   a. Please justify this assumption and clarify if an engineering analysis was performed in support of this assumption.

   b. Please state if this assumption always increases the conditional probability of strainer failure (i.e., is this a conservative assumption?). In other words, please explain if there are any combinations of pumps failing in separate trains that would produce an equal or higher approach velocity and an equal or higher debris accumulation on any one strainer than the same combination of pumps failing in a single train. If so, please justify excluding them from the analysis.

   c. Please state if this assumption always increases the conditional probability of in-vessel effects. Assumption 2b acknowledges that some combinations of pumps failing in separate trains may produce an equal or higher amount of debris accumulation in the core when compared to the same combination of pumps failing in a single train. Please provide a list of these combinations and justify excluding them from the analysis.

<u>LOCA Frequencies</u>

1. RG 1.174, Section 2.3.3 states that the PRA model should be technically adequate for the application. Volume 3, Section 5.3.1 (page 124 of 248) states, in part, that

   > ...the relative weight[s] of breaks in various weld locations are based on specific degradation mechanisms for categories of welds. These frequencies were determined from an analysis of DM [degradation mechanism]-dependent weld failure rates based on service data, a Bayes method for uncertainty treatment developed in the [Electric Power Research Institute (EPRI)] risk-informed in-service inspection (RI-ISI) program, and estimates of conditional probability versus break size using information developed in NUREG-1829 ["Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," April 2008 (Volumes 1 and 2: ADAMS Accession Nos. ML082250436 and ML081060300)].

   a. Although not explicitly quantified, factors other than break size were considered by the NUREG-1829 panelists when developing LOCA frequencies. For example, NUREG-1829, Section 6.3.2, "Important Aging Mechanisms," describes the panelists' consideration of factors such as thermal fatigue, flow-accelerated

corrosion, inter-granular stress corrosion cracking, and mechanical fatigue. Please describe how these factors were used to quantify the break frequency for various pipe sizes. For any factors used in the STP pilot analysis that were already considered by the NUREG-1829 panelists, please explain why the proposed approach does not amount to "double counting."

b. The NUREG-1829 "total" LOCA frequencies referenced by the STP pilot application include contributions from both piping and non-piping (e.g., nozzles, component bodies, pressurizer heater sleeves, man ways, and control rod drive mechanism penetrations) weld failures. As shown in NUREG-1829, Figure 7.7, the contribution from non-piping LOCAs can be significant and exceeds the contribution from piping LOCAs for several categories. The approach described in Volume 3, Section 5.3, "LOCA Frequency," of the STPNOC submittal distributes the total LOCA frequencies onto pipe welds only. While this preserves the overall initiating event frequency, it does not explicitly consider the debris generation, transport, etc. of LOCAs caused by these contributors. Please explain how the debris-related risk from non-piping contributors was estimated in this study. Please provide a justification for any non-piping contributors that were excluded from the analysis.

2. RG 1.174, Section 2.3.4, "Plant Representation," states that PRA results should be derived from a model that realistically represents the risk associated with the plant. NUREG-1829 states that, in general, a complete rupture of a pipe is more likely than a partial rupture. It appears, however, that STP's methodology leads to the opposite result (i.e., a rupture of a given size is more likely to be caused by a partial rupture of a large pipe than a complete rupture of a smaller pipe). Please illustrate the results of your method by comparing the frequency of partial versus complete breaks for a set of representative pipe sizes. Please describe whether the methodology described in the STP pilot is consistent with the assumption of NUREG-1829 or provide justification for an alternate approach.

3. RG 1.174, Section 2.3.2 states that the level of detail of the PRA model must be sufficient to model the impact of the proposed change. Volume 3, Section 5.3 describes the process used to define non-uniform "sample bins" for each weld case. Although this section describes the process used to determine the number of bins for a given weld, the process used to define the bin sizes is not discussed. Please provide a description of this process.

4. RG 1.174, Section 2.3.3 states that the PRA model should be technically adequate for the application. Volume 3 describes the process used to assign break frequencies to welds in containment and cites the following two documents listed as References 7 and 8:

- Reference 7: KNF Consulting Services LLC and Scandpower Risk Management Inc. Development of LOCA Initiating Event Frequencies for South Texas Project GSI-191 Final Report for 2011 Work Scope. September 2011.

- Reference 8: University of Texas at Austin. Modeling and Sampling LOCA Frequency and Break Size for STP GSI-191 Resolution. September 2012.

Please provide Reference 7 on the docket and clarify exactly which aspects of the aforementioned references (e.g., by providing specific section or equation numbers) are used in the STP pilot.

CASA GRANDE TO PRA INTERFACE

General

1. Page 3 of Enclosure 1 of letter dated November 13, 2013, states: "Failure modes leading to core damage are explicitly modeled, excluding those that were previously addressed for the plant using deterministic evaluations." Also, based on information on page 20 of Volume 3 CASA Grande does not analyze failure mechanisms 4 and 6 (ex-vessel effects and crud on clad, respectively) because they have already been addressed deterministically. This is inconsistent with the risk-informed approach as set forth in RG 1.174, Section 1, "Element 1: Define the Proposed Change," in that the licensee should identify those aspects of the plant's licensing basis (LB) that may be affected by the proposed change.

    a. Please provide a basis for excluding the two failure mechanisms (4 and 6) from the risk assessment.

    b. Please identify other failure mechanisms or assumptions related to GSI-191 phenomena that rely on deterministic acceptance criteria (including deterministic criteria acceptable to the NRC) that were not included in the risk assessment.

2. RG 1.174, Section 2.3.2 states that the level of detail of the PRA model must be sufficient to model the impact of the proposed change. PRA models typically classify LOCA break sizes according to scenario differences and corresponding differences in structures, systems, and components (SSCs) available to mitigate the event and corresponding success criteria. For determining the effect of debris on the seven failure mechanisms defined in the submittal, a different set of break sizes might be more appropriate. For example, if there is a minimum size LOCA necessary to result in failure of recirculation due to debris, then including LOCAs below that size when determining failure probabilities may mask the true risk impact. Please provide the following information:

    a. The largest break size below which no failures were recorded during the CASA runs.

    b. In addition, for the chosen LOCA sizes, please describe scenario timing differences for the debris model compared to the base PRA, changes in success criteria as a result of debris, and changes in operator response.

3. RG 1.174 Section 2.3.2 states that the level of detail of the PRA model must be sufficient to model the impact of the proposed change. Table 2.2.11 (Volume 3, page 43 of 248) provides the "frequency of success pump combination states." Please explain what this term means, how the frequencies in the column titled, "Pump State Frequency" were derived, and how they were used in the analysis (both CASA Grande and the PRA models).

4. RG 1.174, Section 2.3.3 states that the PRA model should be technically adequate for the application. Assumption 4 on page 6 of Volume 2 states, in part, that

   > The CASA GRANDE models assume containment systems are successful (containment purge isolation, isolation of small containment penetrations, that at least two of six fan coolers operate, and that CCW is available to the RHR heat exchangers) for purposes of evaluating sump failure probabilities.

   High level requirement LE-E of American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-Sa 2009 ("The ASME PRA Standard") states: "The frequency of different containment failure modes leading to a large early release shall be quantified and aggregated." Please provide the following information for accident scenarios where the containment is not successfully isolated or where some containment systems do not operate as assumed:

   a. Please explain whether the probabilities of the various debris-related failure mechanisms are different for such scenarios.

   b. Please explain how any differences in those probabilities are accounted for in the PRA model.

   c. Please explain how the above assumption for CASA Grande meets high level requirement LE-E of the PRA Standard.

5. RG 1.174, Section 2.3.2 states that the level of detail of the PRA model must be sufficient to model the impact of the proposed change. Volume 2, page 25 states, in part, that

   > Early on in the assessment of GSI-191 phenomena it was determined that the only sequence classes requiring sump recirculation that would be affected are medium LOCAs (2"–6" diameter breaks) and large LOCAs (>6" diameter breaks).

   Also, Volume 1, page 24 states, in part, that

   > No failures were recorded for small- or medium-break events, and it transpired that only the higher range of large-break events contributed to failure. In addition to the composite PRA failure modes, total failure probability conditioned on the LOCA category is provided.

PRA models often include high and low pressure recirculation in event trees for small LOCAs. Please explain how it was determined that only medium and large LOCAs would require sump recirculation. Also, please explain why there were no failures for small or medium LOCAs, including an explanation of the physical phenomena that led to this result. Please include in this explanation a statement as to whether this result was due to insufficient debris generation or Volume 3, page 81, Assumption 11.

6. RG 1.174 Section 2.2 states that it is essential the uncertainties be recognized when assessing whether the principles of risk-informed decision-making are met. Page 84, Volume 2 states, in part, that

> The failure probabilities for [Top Event SUMP] are provided directly from CASA GRANDE output in Volume 3. The uncertainty in these failure probabilities [is] reported as discrete probability distributions with 5 points each.

This appears to conflict with Volume 3, Section 6 (page 233), which states that 15 point estimates of conditional failure probability are "averaged for use by the PRA."

a. Please explain how many point estimates (i.e., distinct conditional failure probabilities) associated with a single frequency vs. break size curve (e.g., one curve from figure 6.1 in Volume 3) were computed by CASA Grande?

b. Were these parameters passed to the PRA as a probability mass function, probability density function, or as a single mean value?

c. Please explain how were these point estimates used in the PRA parameter uncertainty evaluation?

STP PRA MODEL

<u>General</u>

1. RG 1.200, Section 2.2, "Industry Peer Review Program," states that when the NRC staff's regulatory positions contained in its appendices are taken into account, use of a peer review can be used to demonstrate that the PRA is adequate to support a risk-informed application. It appears that the latest peer review of the STP PRA model was performed to support STP's application for risk-managed technical specifications and the results of this peer review were captured by STP in its letter dated February 28, 2007 (ADAMS Accession No. ML070670369). Please describe any significant changes to the plant or PRA model that have been made since that time. Please state if any of these changes represent "model upgrades" as discussed in the ASME/ANS PRA Standard. Please provide the results of any focused-scope or full peer reviews conducted since letter dated February 28, 2007.

2. RG 1.174, Section 2.3.3 states that the PRA model should be technically adequate for the application. Please explain the basis of the statement on page 11 of LAR Enclosure 3 "since the STPNOC's PRA is compliant with RG 1.200, Revision 1 for

internal events, it is compliant with RG 1.200, Revision 2 for assessing the risk associated with GSI-191." Please explain how this conclusion reached.

3. RG 1.200, Section 2.1, "Consensus PRA Standards," states that the capability category (CC) needed for each technical requirement is dependent on the specific application, although CC-II is generally acceptable. Please provide a list of the supporting requirements and corresponding CCs that were determined to be applicable to the risk-informed resolution of GSI-191. For any cases where the necessary CC was not found to have been met by the peer review process, please provide a technical justification.

<u>Success Criteria</u>

1. RG 1.174, Section 2.3.3 states that the PRA model should be technically adequate for the application. Volume 2, page 7, item 14, states:

   > One out of three each from HHSI [high head safety injection] and LHSI [low head safety injection] pumps is assumed required for mitigation of medium LOCAs.

   Please provide the basis for this assumption including how debris effects on success criteria were considered.

2. RG 1.174, Section 2.3.3 states that the PRA model should be technically adequate for the application. ASME/ANS Ra-Sa-2009 SC-B3 states:

   > When defining success criteria, USE thermal/hydraulic, structural, or other analyses/evaluations appropriate to the event being analyzed, and accounting for a level of detail consistent with the initiating event grouping (HLR-IE-B) and accident sequence modeling (HLR-AS-A and HLR-AS-B).

   Volume 3, Section 5.10, "In-Vessel Downstream Effects" (page 222), and Volume 6.2, Item 5.a.14 describe a series of RELAP5 simulations used to assess small, medium, and large LOCAs on both the hot- and cold-leg side of the reactor coolant system (RCS) under full core blockage conditions. According to these sections, "only the medium and large cold leg breaks proceeded to core damage..."

   a. Please discuss how these thermal-hydraulic analyses contained a level of detail consistent with the initiating event grouping and accident sequence modeling. In other words, explain whether the break sizes and locations assumed in the RELAP5 simulations were consistent with those used in the PRA and elsewhere in CASA Grande.

   b. Please explain whether the plant conditions assumed in the RELAP5 simulations (e.g., number of available ECCS trains) were consistent with each accident sequence in which the results from the RELAP5 simulations were used. For example, a sequence involving plant state 43 (single ECCS train) and a medium-break LOCA on the hot-leg side would appear to have assumed conditional probability of core blockage due to boron of zero. Please explain if

the RELAP5 calculation showing adequate core cooling under this scenario accounts for the most limiting medium-break LOCA (break size/location) and if it is consistent with accident sequence modeling (i.e., models only one train of ECCS). The response should address all GSI-191 accident sequences that used RELAP5 simulations.

c. RELAP5 simulations are used to support the conclusion that adequate core cooling is achieved for some LOCAs even under the assumption of complete core blockage. Please describe how these simulations accounted for the reduction in cladding-to-coolant heat transfer that was calculated by the LOCADM computer code, as described in Volume 3, Section 5.10.1.

3. RG 1.174, Section 2.3.3 states that the PRA model should be technically adequate for the application. Volume 3, Section 4.2, "Structured Information Process Flow," describes the quantification of net positive suction head (NPSH) margin and core blockage as a function of time. According to this section, time-dependent values are compared against acceptance criteria from t=0 (i.e., time that the LOCA occurs) and t=30 days. This appears to conflict with Volume 3, assumption m, which states that "it was assumed that a 36-hr run time for the CASA Grande simulations is sufficient to predict the scenarios that would proceed to failure."

a. Please clarify the accident duration used by CASA Grande to calculate the conditional probability of sump and core blockage. Also, identify all other analyses performed outside of CASA Grande (e.g., time-dependent coatings failure) and their assumed accident duration.

b. Many of the simulations performed by CASA Grande concluded that none of the seven performance thresholds identified in Volume 1, Section 2.3.5, "LOCA Models," were exceeded. For these cases, please state if an analysis was performed to confirm that at the end of each simulation, the plant was in a safe, stable end state. For example, please explain if any cases were trending toward a performance threshold when the simulation was terminated. If so, please provide justification that the selected accident duration was appropriate to capture all physical phenomena (e.g., long-term chemical effects). If failures due to GSI-191 phenomena could occur after the CASA Grande analysis duration, please explain how was this was considered in the PRA model (note Supporting Requirement SC-A5 in the ASME/ANS PRA Standard).

c. Please state what plant conditions and configuration is assumed for the "safe, stable end state" in the PRA model. Please describe.

d. Please explain whether the conditional probabilities calculated by CASA Grande were adjusted in any way to match the PRA mission time of 24 hours.

Human Reliability Analysis

1. RG 1.174, Sections 2.3.1, "Scope," and 2.3.2, "Level of Detail Required to Support an Application," state that the scope and level of detail of the PRA model must be sufficient

to model the impact of the proposed change. Assumption 3.c in Volume 3 states that isolable breaks can be excluded from the evaluation since isolable breaks would not lead to recirculation. Please explain the basis for this assumption. Please describe what human error probability was used for the failure to isolate an isolable break. Please state whether there are any isolable breaks that could, if not isolated, result in the need to enter the ECCS recirculation mode.

2. RG 1.174, Sections 2.3.1 and 2.3.2 state that the scope and level of detail of the PRA model must be sufficient to model the impact of the proposed change. Under certain conditions, operator actions required to start or secure the pumps may depend on the effects of the debris generated by the specific pipe break. Please describe if CASA Grande considers the potential for the number of running pumps to change during a scenario because of operator actions taken in response to cues that debris is building up on the sump.

3. RG 1.174, Sections 2.3.1 and 2.3.2 state that the scope and level of detail of the PRA model must be sufficient to model the impact of the proposed change. Volume 2 describes two top events that model operator actions used to secure containment spray. Top event "OSI" (page 77) represents a short-term action to secure one train of containment spray prior to recirculation, given that all three trains are available. Top event "OFFS" (page 83) represents a longer-term action to secure all trains of containment spray once containment pressure and iodine levels are suitably low, these are conditions that may occur after recirculation is established. According to their descriptions, these top events are "always assumed successful when determining the failure probabilities introduced by the GSI-191 phenomena." Please provide the following information:

    a. Please state if the CASA Grande models the plant conditions (e.g., sump flow rates, washdown rates, refueling water storage tank (RWST) drain-down times, etc.) that would occur if three containment spray trains were running (i.e., if the manual actions modeled by top event OSI are unsuccessful.)

    b. Please state if the CASA Grande models the plant conditions (e.g., sump flow rates, washdown rates, RWST drain-down times, etc.) that would occur if the operators fail to secure containment spray long term once containment pressure and iodine levels are suitably low (i.e., the manual actions associated with OFFS are unsuccessful).

    c. If the answer to either (a) or (b) is "no", please provide a technical basis and explain how the PRA meets the ASME HLR-HR-G requirement to perform an assessment of post-initiator human failure events using a well-defined and self-consistent process that addresses scenario-specific influences on human performance.

4. RG 1.174, Section 2.3.2 states that the level of detail of the PRA model must be sufficient to model the impact of the proposed change. Volume 2, page 85 states that the probability of excess boron precipitation (top event BORON) depends on three factors: (1) whether the break is in the cold leg; (2) the extent of core flow blockage prior

to hot-leg switchover; and, (3) whether a low head safety injection (LHSI) train is realigned for hot-leg recirculation.

a. Please provide the probability assigned to BORON for each combination of these three factors used in the PRA model. A table or graphic may be a useful way to provide this information.

b. Assumption 1.j in Volume 3 (page 72 of 248) states that switchover to hot leg injection (factor 3) is assumed to occur between 5.75 and 6 hours after the start of the event. Please describe how the human error probabilities (HEPs) associated with this top event ("HLEG") were developed and how they account for LOCA size and plant configuration (e.g., number of pumps available, impact of debris, etc.), as well as factors 1 and 2 defined above.

5. RG 1.174, Section 2.3.2 states that the level of detail of the PRA model must be sufficient to model the impact of the proposed change. On page 32 of Volume 3, other important longer-term actions are listed, which include: (1) Securing one containment spray pump if all three containment spray pumps are successfully initiated; (2) Securing all containment spray pumps later in the event; (3) Switchover to ECCS sump recirculation after the RWST [refueling water storage tank] has been drained; and (4) Switchover to hot leg injection. Please explain how the CASA Grande results were developed to address the various combinations of success and failure of these operator actions. Please also explain how the consistency between the actual PRA scenario and the GSI-191 basic event failure probabilities developed in CASA Grande was assured.

6. RG 1.174, Section 2.5.5, "Comparison with Acceptance Guidelines," states that care should be taken to ensure that there are no unquantified detrimental impacts to proposed changes, such as an increase in operator burden. Section C.5.8, "Mitigation of Inadequate Reactor Core Flow," of Appendix C to Volume 1 lists a number of operator actions associated with debris. For operator actions that apply in both the GSI-191 PRA base case model and the GSI-191 PRA debris model, please explain how each operator action's HEP was modified as a result of debris consistent with ASME HLR-HR-G, which states that scenario-specific influences on human performance should be addressed

<u>PRA Scope</u>

1. A seismic event may potentially dislodge and transport insulation to the containment sump. Any subsequent sequence of events that leads to recirculation from the ECCS sump could be adversely impacted by debris. RG 1.174, Section 2.3.1 states that the scope of the PRA model must be sufficient to model the impact of the proposed change. Please identify such accident sequences and estimate the increase in core damage and large early release frequencies due to a seismic event as a result of having debris sources in the containment.

2. Volume 2 (page 47 of 257) states, in part, that

> ...a medium LOCA on one primary loop would be assumed to be accompanied by medium LOCA on all other loops. The result is that

seismically induced medium and large LOCAs are modeled as being excessive LOCAs-which have no success sequences by definition.

RG 1.174, Sections 2.3.1 and 2.3.2 state that the scope and level of detail of the PRA model must be sufficient to model the impact of the proposed change. While it is a common assumption that seismic failures among like components are 100 percent correlated, for this analysis this assumption leads to a lower calculated value of delta risk. That is, for cases where a single large pipe fails due to a seismic initiating event, or where multiple small bore pipes fail, it might be possible to mitigate the scenario to a greater extent if debris sources do not exist inside containment. Please justify not analyzing the additional risk of GSI-191 phenomena for seismic events without reliance on the conservative assumption of 100 percent correlation.

RESULTS INTERPRETATION

Quantification

1. Volume 2, page 3 states, in part, that

> The change in core damage frequency and large early release frequency is determined by comparing the results of two models: one with no source material in the containment capable of producing any GSI-191 effects and one representing the current plant conditions that includes both fibrous insulation that might be liberated following a LOCA and latent material found in the containment.

Also, elsewhere in the submittal, it says "failure branches" are not included since they lead to core damage with or without debris. This would imply that the analysis produced the delta risk directly by considering "success branches" that would be impacted by debris, without the need to subtract a base-case risk. It is important that the NRC staff understand how the risk was calculated. RG 1.174 Section 2.2 states that the licensee should assess the expected change in CDF and LERF. Please provide the following:

a. Please clarify whether the ΔCDF and ΔLERF were calculated by summing only the former success states that go to core damage due to GSI-191 phenomena or by requantifying the entire debris model and subtracting the base case.

b. Please explain if the same LOCA initiating event frequencies and parameter uncertainty distributions were used for both the baseline and debris models.

2. Please provide a list of the top 100 accident sequences that result in core damage due to one of the seven failure mechanisms identified in Volume 1; that is, include only sequences that include failure of recirculation core damage resulting from as a result of GSI-191 phenomena.

Uncertainty Analysis

1. RG 1.200 defines model uncertainty as an issue where no consensus approach or model exists and where the choice of approach or model is known to have an effect on the PRA (e.g., introduction of a new basic event, changes to basic event probabilities, changes in success criteria, introduction of a new initiating event.). Volume 1 Section 2.5.3, "Model Uncertainty," contains a general discussion of model uncertainty (using chemical effects and debris filtration as examples) and broadly states that conservatism was used to address model uncertainty. This is inconsistent with RG 1.200, Section 3.3.2, "Assessment of Assumptions and Approximations" (and ASME HLR-QU-E) guidance that states that all sources of model uncertainty should be identified and their effects on the PRA (as identified above) should be determined as discussed in RG 1.174, Section 2.5.5. Uncertainty must be evaluated for this analysis, including the PRA model, CASA Grande, and any supporting analyses. Please provide the following information:

    a. Please identify all sources of key model uncertainty as defined by RG 1.200.

    b. Please identify the key assumptions as defined by RG 1.200.

    c. Please describe the potential effect of the key assumptions on the results of this study; that is, on the CDF and LERF attributable to GSI-191 phenomena. Describe the results of any related sensitivity analyses that were performed.

2. Volume 3, Assumption 3.a (page 76 of 248) states that the geometric-mean aggregation of LOCA frequencies in NUREG-1829 is the most appropriate set of results to use for this evaluation. The basis provided is that geometric-mean aggregation produces frequency estimates that are approximately the same as the median estimates of the panelists. There is no justification about why the median estimate is preferred and emphasis on the median conflicts with the RG 1.174 guidance that the mean values be used for decision-making. Furthermore, information in NUREG-1829, Section 7.6.4, "Aggregation," shows that the use of the arithmetic mean instead of the geometric mean would increase the LOCA frequency by an order of magnitude or more for some LOCA categories and may therefore substantially increase the risk estimates. Consequently, selection of the geometric mean is a key assumption and selection of the arithmetic mean represents an alternative reasonable assumption as defined by RG 1.200. This is supported by RG 1.174, Section 2.5, "Comparison of Probabilistic Risk Assessment Results with the Acceptance Guidelines," which states, in part, that "the licensee should [identify] key assumptions in the PRA that impact the application." Sensitivity studies provide important information about how some of the key assumptions affect the final results as discussed in RG 1.174 Section 2.5.3. Please provide CDF, LERF, ΔCDF, and ΔLERF using the arithmetic mean aggregation of LOCA frequencies in NUREG-1829.

3. Volume 1, Section 1.3.1, "LOCA Frequency," states that LOCA frequencies were obtained from Table 7.19 of NUREG-1829 for 25 years' operation. Furthermore, assumption 3.b in Volume 3 (page 76) states that current-day LOCA frequencies are more appropriate to use for this evaluation than the end-of-plant-license frequencies. RG 1.174, Section 2.5.5 states that it is incumbent on the licensee to demonstrate that

the choice of reasonable alternative hypotheses, adjustment factors, or modeling approximations or methods to those adopted in the PRA model would not significantly change the assessment. Also, it is assumed that the STP plants will continue to operate for more than 25 years; RG 1.174 Section 3, "Element 3: Define Implementation and Monitoring Program," states that the licensee should define an implementation and monitoring program to ensure that no unexpected adverse safety degradation occurs do to the change. Please justify the use of the 25-year frequency estimates rather than the 40-year estimates provided by NUREG-1829. Please provide CDF, LERF, ΔCDF, and ΔLERF using the 40-year estimates.

4. The acceptance guidelines of RG 1.174, Section 2.5.5 "are defined such that the appropriate measure for comparison is the mean value of the uncertainty distribution on the corresponding metric." Typically, statistical sampling simulations will develop random variables that preserve the mean of the distribution from which the variables are sampled. STP has chosen to fit a Johnson bounded distribution that matches the expert-provided 5th, 50th, and 95th percentiles in NUREG-1829, but does not match the mean values. The properties of the distribution are such that, as fit, the mean of the fitted distribution is always less than the experts' means from the distributions in NUREG-1829.

    a. Please explain why the STP evaluation departs from the regulatory position in RG 1.174 regarding the use of mean values.

    b. The Johnson fit to 5th, 50th, and 95th percentiles is not unique. Alternative accurate fits can be constructed with arbitrary values of the scale parameter λ. The scale parameter λ defines a bound on the maximal frequencies sampled in the Monte Carlo model. By increasing the value of λ, the relative proportion of large to medium to small breaks can be altered, especially in the extrapolation range beyond the 95th percentile. Please provide a technical justification for the selection of the scale parameter λ (other selections appear possible that could change the outputs by CASA Grande).

    c. Please provide the maximum expected difference between the CDF, LERF, ΔCDF, and ΔLERF developed from bounded Johnson distributions that consider alternative values of the scale parameter λ, and other distributions that would preserve mean values reported in NUREG-1829. Note, in particular, that alternative bounded Johnson distributions with large values of the scale parameter λ can be built to accurately fit the NUREG-1829 5th, 50th, and 95th percentiles, and produce mean estimates closer to the NUREG-1829 values than current fits used by STP.

    d. In Table B.4-1 in Volume 2, LOCA exceedance frequencies are tabulated for different break sizes that are consistent with modified bounded Johnson distributions (the modified Johnson is a scaled bounded Johnson plus a uniform distribution) designed to match NUREG-1829 mean values. The fit to the 5th, 50th, and 95th NUREG-1829 percentiles of these modified Johnson distributions does not appear to be accurate. Therefore, these modified Johnson distributions inferred from Table B.4-1 appear to be different than the bounded Johnson

distributions summarized in Table 2.2.2 in Volume 3. Please justify the apparent use of different bounded Johnson distributions in the PRA and CASA Grande.

5. Volume 2, page 6 states, in part, that the "LOCA frequency uncertainties sampled in the PRA uncertainty analysis are assumed independent of the probabilities of failure from the uncertainty analysis of CASA GRANDE." This assumption does not account for the state of knowledge correlation because the PRA and CASA Grande rely on the same parameter for their quantification (LOCA frequency derived from NUREG-1829). RG 1.174, Section 2.5.2, "Parameter Uncertainty," states that the state of knowledge correlation should be accounted for unless it can be shown to be unimportant. Therefore, you are requested to either calculate CDF, LERF, ΔCDF, and ΔLERF accounting for the state-of-knowledge correlation or demonstrate that it is unimportant to this application.

6. RG 1.174 Section 2.5.5 states that it is incumbent on the licensee to demonstrate that the choice of reasonable alternative hypotheses, adjustment factors, or modeling approximations or methods to those adopted in the PRA model would not significantly change the assessment. This demonstration can take the form of well-formulated sensitivity studies or qualitative arguments. In general, the results of the sensitivity studies should confirm that the risk acceptance guidelines are still met even under alternative assumptions.

   Please provide the results of an aggregate analysis that quantifies the integrated impact on CDF, LERF, ΔCDF, and ΔLERF from all sensitivity studies that were performed. In this aggregate analysis, for those cases where individual assumptions have a synergistic effect on the results, a simultaneous analysis should be performed. For those cases where no synergy exists, a one-at-a-time analysis may be sufficient.

Radiation Protection & Consequence Branch (ARCB)

1. *In an effort to ensure a complete and accurate review of the dose consequence analyses, please provide additional information in tabular form describing, for each design basis accident affected by the proposed Risk Informed GSI-191 submittal, all the basic parameters used in the dose consequence analyses. For each parameter, please indicate the current licensing basis (CLB) value, the revised GSI-191 value where applicable, as well as the basis for any changes to the CLB. An example of the input/assumptions needed is provided in Table 4.3-11 "Dose Analysis Inputs for LOCA" provided in STP's alternate source term (AST) submittal dated March 22, 2007 (ADAMS Accession No. ML070890474). The NRC staff requests that the information include all of the basic parameters whether or not the individual parameter is being changed for the GSI-191 amendment. The NRC staff also requests that the information be presented in separate tables for each affected accident (i.e., LOCA, the fuel handing accident (FHA), the main steam line break accident (MSLB), the steam generator tube rupture accident (SGTR), the control rod ejection accident (CREA), and the locked rotor accident (LRA)).*

2. *STP identified the following condition related to the AST submittal:*

   Westinghouse Electric Company Nuclear Safety Advisory Letter (NSAL)-06-15, dated December 13, 2006, advised operators of Westinghouse plants that the single-failure scenario for the SGTR analysis that licensees used in their accident analysis may not be limiting. As stated in the STP AST NRC Safety Evaluation dated March 6, 2008 (ADAMS Accession No. ML080160013), "The licensee has evaluated the applicability of NSAL-06-15 against the accident analysis assumptions and has determined that the current single-failure assumption for the STP SGTR analysis is not limiting. Therefore, the licensee is operating under compensatory measures to meet regulatory dose guidelines. The licensee plans to resolve this condition at the earliest opportunity so that the assumptions, including the limiting single failure, for the SGTR accident analysis described herein are consistent with the plant response to this event. To support the limiting single-failure assumptions in the SGTR analysis, STP will maintain an administrative limit for reactor coolant system (RCS) dose equivalent iodine 131 (DEI) so that the radiological dose reference values for the SGTR analysis remain bounding, and the licensee will continue to comply with GDC [General Design Criteria] 19."

   Please state if this condition been resolved? If so, please describe how? Also, please provide justification that GDC 19 continues to be met.

3. *The LOCA analysis assumes that iodine will be removed from the containment atmosphere by containment spray and natural diffusion to the containment walls. As a result of these removal mechanisms a large fraction of the released activity will be deposited in the containment sump. The sump water will retain soluble gases and soluble fission products such as iodines and cesium, but not noble gases. The guidance from RG 1.183, "Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792), specifies that the iodine deposited in the sump water can be assumed to remain in solution as long as the containment sump pH is maintained at or above 7.*

   The AST application indicates:

   "After the first day, the containment sump pH will begin to decrease, reaching 6.8 by the end of the 30-day duration of the radiological consequence analysis for the Design Basis Accident (DBA) LOCA, and the impact of that decrease has been reflected in the Control Room and offsite doses."

   It is noted that the AST application further indicates:

   "The design inputs for calculating the containment sump pool pH were conservatively established by the licensee to maximize the acidic contribution to sump pH and minimize the basic contribution."

   The GSI-191 application indicates the possibility that debris generated during a LOCA could clog the containment sump strainers in pressurized-water reactors (PWRs) and

result in loss of NPSH for the ECCS and CSS [containment spray system] pumps, impeding the flow of water from the sump.

Please discuss the exemption justification as they relate to the effects on sump water pH, radiological consequences, and loss of the containment spray system (CSS).

Mechanical and Civil Engineering Branch (EMCB)

1 *In the application, the licensee provided a qualitative response regarding the structural analysis without any supporting quantitative data. Without actual and allowable stresses and design margins for the various components of the sump strainer structural assembly, the NRC staff is unable to make a determination about the inherent level of conservatism employed in the design. This information was not provided in the licensee's recent submittals concerning Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," (ADAMS Accession No. ML042360586).*

Please summarize the structural qualification results, including the actual and allowable stresses, and design margins for the various components of the sump strainer structural assembly.

Component Performance, NDE and Testing Branch (EPNB)

The NRC staff has reviewed the LOCA frequency estimates in Sections 2.2.3, "LOCA Frequencies," and 5.3, "LOCA Frequency," of Volume 3 and requests the following additional information.

1. Volume 3, Section 5.3.1, "Relative Weight of Breaks in Specific Weld Categories," page 125, specifies the degradation mechanisms that were considered in the LOCA frequency estimates. The risk-informed inservice inspection (RI-ISI) program at STP was based on EPRI TR-112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure (PWRMRP-05)," Revision B-A, Final Report, December 1999 (ADAMS Accession No. ML013470102). The NRC staff notes a discrepancy in the degradation mechanisms used between the RI-ISI program and the GSI-191 submittal. Several of the degradation mechanisms in Table 2-2 of EPRI TR-112657 report that are used in the RI-ISI program are not listed as the degradation mechanisms in the GSI-191 calculations. For example, erosion cavitation, corrosion fatigue, corrosion attack, and water hammer identified in Table 2-2 of the EPRI report are not considered in Section 5.3.1. Please discuss the discrepancy.

2. Volume 3, Section 5.3.1, page 125, states that Table 5.3.1, Category 6B, contains two weld sizes (nominal 0.75-inch and 1-inch pipes). However, Table 2.2.6, Category 6B shows only 1-inch weld size. Please discuss how the LOCA frequency calculations handle this discrepancy. That is, please discuss whether the initiating frequency calculation includes the frequencies from the 0.75-inch and 1-inch pipe sizes or only the 1-inch pipe size is used. This discrepancy also applies to Categories 6A and 8C which also contain two weld sizes.

3. Please discuss whether welds evaluated in the GSI-191 analysis contain flaws while in service. If yes, please discuss whether the LOCA frequencies for these welds are increased from that of NUREG-1829 estimates. If the pipe failure probabilities for these welds are not increased, please provide justifications. Volume 3, Table 5.3.1, page 126 shows that four welds at the pressurizer were weld overlaid. Please discuss whether the pipe failure probability LOCA frequencies for these mitigated welds were reduced from the frequency estimates of NUREG-1829. If not, please provide justifications.

4. Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, 50.55a(g)(6)(ii)(F) incorporates by reference American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Case N-770-1, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities, Section XI, Division 1," with conditions. ASME Code Case N-770-1 requires licensees to perform augmented inspections beyond those that are required by the ASME Code, Section XI, for piping with Alloy 82/182 dissimilar butt welds. Please discuss whether periodic inspections per ASME Code Case N-770-1 would reduce the LOCA frequency estimation for the Alloy 82/182 dissimilar metal welds. If yes, please discuss how much LOCA frequency estimates were reduced for these welds.

5. Volume 3, Section 5.3.5, "Sample Break Sizes at Each Weld Location," page 149, discusses how the break sizes are selected to derive the LOCA frequency estimates. Figure 5.3.4, page 151, provides the selected break sizes for weld case 1B as an example. (a) Figure 5.3.4 presents a total of 13 breaks being simulated at each weld belonging to Weld Case 1B (10 large breaks, two medium breaks and one small break). However, in Table 2.2.3, under weld case 1B, the NRC staff finds only 12 breaks. Please explain how 13 breaks were identified. (b) Please discuss how the break size distribution scheme in Figure 5.3.4 provides confidence and assurance that the break selection will result in appropriate debris generation as there are many possible scenarios for the break size distribution. The break sizes could be evenly distributed such that there are four small breaks, four medium breaks and four large breaks (assuming the total breaks are 12). The break sizes could be distributed skewed to the small size, such as 10 small breaks, one medium break and one large break. The break size could also be skewed toward medium breaks such as one small break, 10 medium breaks, and one large break. Please discuss how it was determined that the break size distribution in the CASA Grande analysis is appropriate (i.e., neither too conservative nor too non-conservative in terms of the debris generation) when examining the final probability result (the core melt frequency).

6. By letter dated September 10, 2012, the NRC approved the risk-informed inservice inspection (RI-ISI) program for the third 10-year inservice inspection interval at STP, Units 1 and 2 (ADAMS Accession No. ML12243A343). Please discuss the following:

    a. Please state if the LOCA frequency estimates used for welds in the GSI-191 submittal are consistent with the LOCA frequency estimates used in the RI-ISI program. If the comparison is appropriate, please provide numerical examples of

the comparison. If the comparison is not appropriate, please provide explanation.

b. If the LOCA frequencies for welds are not consistent between the two analyses, (1) please identify the differences and explain why there are differences, and (2) please discuss why the LOCA frequencies proposed in the GSI-191 submittal are acceptable if they are not consistent with that of the RI-ISI program.

Steam Generator Tube Integrity and Chemical Engineering Branch (ESGB)

Chemical Effects

1. The exponential probability density functions (PDF) for small-, medium-, and large-break LOCAs (e.g., Figure 5.6.4, Volume 3) are shown in terms of a chemical bump-up factor. The NRC staff has questions related to the conventional (non-chemical) head loss correlation. The review of a chemical bump-up factor is complicated by the fact that it is essentially a multiplier on a parameter that is concurrently being reviewed for adequacy. Please provide the following additional information.

    a. CASA Grande calculates the conventional head loss value for a given break and then applies a chemical bump-up factor independent of the conventional head loss. Please justify not correlating the chemical bump up factor to the conventional head loss since the same debris bed affects both values. Based on the NRC staff's experience observing testing, head loss for a given quantity of chemical precipitates should be related to both the type of precipitate and the filtering characteristics of the debris bed.

    b. In order to help the NRC staff judge the magnitude of the chemical head loss bump-up factor, please provide, by performing realizations for the existing CASA Grande model, a relative frequency plot of chemical effects for STP in terms of absolute units (e.g., feet of water) for the small-break LOCA (SBLOCA), medium-break LOCA (MBLOCA), and large-break LOCA (LBLOCA). For example, a histogram showing chemical head loss (feet) on the x-axis and number of occurrences on the y-axis would be very useful to the staff.

    c. Please provide additional details on how the results from the Chemical Head Loss Experiment (CHLE) testing, WCAP-16530-NP-A, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," March 2008 (ADAMS Accession No. ML081150379), calculations, and reasonable engineering judgment were used in the development of the exponential PDF. In addition, please supply the basis for choosing the exponential form of the PDF over others (e.g., Weibull).

    d. Please provide a detailed technical basis for the mean bump up factors shown for the SBLOCA, MBLOCA, and LBLOCA. The NRC staff has observed head loss testing where the greatest chemical bump-up factors are associated with thinner beds. Please discuss why the mean bump up factor would be higher for a LBLOCA. Please explain if it is more probable that a debris bed for smaller

and medium breaks (assuming the bed coverage criterion is met) would consist primarily of fiber fines that are the most readily transportable to the strainer. In general, finer fiber beds tend to lead to greater head loss.

2. Please explain how the highest head loss values (e.g., 90th, 95th percentile) obtained using CASA Grande compare with the head loss values obtained from the STP plant-specific strainer tests at Alden Laboratory. Please discuss both in terms of conventional head loss (i.e., before addition of chemical precipitates) and after chemical precipitates were added. Please discuss all strainer test results including the test that was terminated prior to addition of precipitates.

3. Please provide the technical basis for the 1E-05 probability for the maximum chemical effect for each break size. The engineering judgment used to determine that value appears to be arbitrary and other expert assessors could easily reach different conclusions concerning a tail probability.

4. Please provide the results of a sensitivity study related to the assumption of the tail probability that shows how different assumptions (e.g., 1E-02, 1E-03, 1E-04) for the maximum chemical effects bump-up tail would affect the probability of exceeding the ECCS pumps NPSH criteria and the impact to the core damage frequency and large early release frequency. Please evaluate these tail probabilities assuming the conventional head loss is constant and equal to (a) 0.5 feet, (b) 1.0 feet, and (c) 2 feet.

5. Since the STP chemical effects evaluation is heavily dependent on engineering judgment, the NRC staff needs clarification regarding the sensitivity of various assumptions to the overall risk informed evaluation. In addition to the sensitivity study requests contained in other chemical effects RAIs (e.g., RAI#4), please evaluate the key chemical effects area assumptions (e.g., bump up factors, type of distribution, etc.) and provide sensitivity studies that will help the staff to evaluate how changes in those assumptions will alter the probability of failures. Please identify any assumptions that are correlated and explain how the correlation was considered in the analysis.

6. Please describe the relative chemical contributions from the sprayed materials compared to the submerged materials. Please state if the chemical model input into CASA assumes a fixed, 6.5-hour spray duration. If so, please discuss the probability of containment spray operating at a time beyond 6.5 hours following a LOCA and how the chemical effects analysis would be changed. Please describe if any sensitivity study was performed on the spray time with respect to how it may affect the chemical source term, the probability of precipitation, and ultimately the GSI-191 failure modes.

7. CHLE Tank Tests 3 and 4 were performed with excessive quantities of aluminum relative to the plant and with a temperature profile intended to induce chemical precipitation. These tests resulted in chemical precipitation and provided useful information related to head loss loop response to chemical precipitates. The existing tests do not appear to address the extent of deviation from the best estimate plant conditions that could result in chemical precipitation. One potential method to inform engineering judgment with respect to chemical effects probabilities could be a series of smaller scale tests designed to evaluate the threshold concentrations of species that

could result in precipitation. For example, tests could be designed to evaluate how much aluminum or calcium in solution would cause precipitation that may result in significant increases in head loss. These types of tests were included in the original chemical effects test plans but were apparently cancelled. Please discuss any plans for smaller scale testing to investigate threshold values for precipitation and whether that information would provide greater confidence in determining the probability that a post-LOCA plant condition would result in chemical precipitate formation. If there are no plans for additional tests, please provide justification for this engineering judgment.

8. The STP chemical effects analysis assumes no precipitation prior to the sump pool cooling to 140 degrees Fahrenheit (°F). It is possible, however, to precipitate a calcium phosphate precipitate at higher temperatures if sufficient dissolved calcium is present. Please explain if the CASA Grande model includes calcium sources such as concrete dust, concrete ablated by the jet, and other plant materials such as insulation. Please state if there is a potential for some pipe breaks to produce enough calcium such that formation of a precipitate at a greater than 140 °F temperature should be included. Since some precipitates can form at temperatures greater than 140 °F, please explain how increasing the temperature threshold in Casa Grande affects the outcomes.

9. Please describe the key sources of uncertainty (aleatory and epistemic) associated with the dissolution model and the solubility limits and how is this uncertainty factored into the probability density functions and the chemical bump up factors?

10. The following parameters do not appear to be considered in the simplified approach used to quantify chemical effects for STP: pool chemistry, pool pH, and the amounts of aluminum, calcium, and zinc. If these items are not considered, please provide justification for acceptability of the analysis without their consideration.

11. The conclusions contained in document CHLE-014, "T2 LBLOCA Test Report," (letter dated October 13, 2013, available in ADAMS Accession No. ML13323A673) state, in part, "Chemical products did form under the simulated STP LBLOCA conditions but primarily were adhered to the galvanized coupons." In addition, CHLE-020, "Test Results for a 10-day chemical effects test simulating LBLOCA conditions (T5)," states on page 10, "The high turbidity at the beginning of Tests T5 and T2 shown in Figure 3b might be caused by detachment of zinc particles from the zinc coupons and galvanized steel coupons due to the high temperature during the first 80 minutes of the test." Page 75 of Volume 6.2 states, "Although a zinc (Zn) product was observed to form under STP LOCA test conditions, it was not included in this analysis since the product was determined to be crystalline and mainly adhere to structures within containment as opposed to readily travel with solution." Based on international experience, Framatome ANP, Inc. report titled, "Influence of Corrosion Processes on the Protected Sump Intake after Coolant Loss Accidents," December 2006 (ADAMS Accession No. ML083510156), zinc corrosion product dislodged by falling water caused a significant increase in head loss, please discuss:

    a. If following a LOCA, water either falling from a pipe break or from other locations in the containment building could dislodge zinc corrosion product from galvanized steel surfaces that could transport to the strainer.

b. Whether chemical effects contributions from zinc should be considered as part of the STP chemical effects analysis.

12. The CHLE test facility included three parallel vertical head loss loops that were intended to allow multiple bed evaluations with each test. The test results suggest there was potentially some bias in head loss results between the three loops. Please describe any evaluations that were performed and lessons learned. Also, please describe any significant modifications that were made to the facility loop during the course of testing and how those modifications may have affected the results.

13. The aluminum source for CHLE tests was aluminum scaffolding removed from the plant. The scaffolding was described in the test documents (e.g., CHLE-012) as a non-homogenous sample with unknown constituents from years of use which remained after cleaning. During the NRC staff's visit to observe CHLE testing, the staff observed what appeared to be a grout-like material covering a relatively small portion of a test sample. The submerged scaffolding samples were taken from the side of the scaffolding and had a different texture and appearance than the samples cut for the vapor space. Analysis of unused scaffolding indicated the presence of aluminum phosphate and aluminum oxide/hydroxide scales. The pre-existing scale may have reduced the aluminum released by corrosion. Since the corrosion of aluminum can have a significant impact on whether chemical precipitates form;

    a. Please describe the steps taken to ensure that the surface condition of the scaffolding used in the CHLE tests is representative of the remaining aluminum in the plant.

    b. Please explain if the corrosion behavior of the different parts of the scaffolding (i.e., the part used for submerged samples and the part used for vapor samples) was ever compared by placing them in the exact same test conditions, such as in a bench test.

    c. Please explain if the scaffolding surface condition was evaluated to determine if a LOCA jet or the thermal transient from a LOCA would cause the oxide to spall, potentially resulting in a greater corrosion rate than was observed during the CHLE testing.

14. Please discuss how uncertainties from the following items are considered in the STP chemical effects analysis:

    a. Radiation effects on precipitate formation.

    b. Radiation effects on debris bed degradation. In addition, for this item, the submittal states (Volume 6.2, page 84) that breakdown of the fiber bed is not considered to be a significant issue due to similar materials being used for filtration media for high activity particulate. Please discuss how the filter service life in the referenced application compares to the ECCS mission time following a LOCA.

c. Effects of unqualified coatings degradation. For example, please explain if leached chemicals from the coatings at STP contribute to potential chemical effects. Please explain if the coatings themselves become a debris source that is more problematic than particulates (e.g., gelatinous). As part of the response, please compare the coatings tested in the report referenced in the LAR to the STP plant-specific unqualified coatings.

15. The CHLE tests simulated a 15-inch LBLOCA. Please describe how the CASA Grande chemical model determines the chemical source term for different size breaks, such as a smaller than 15-inch LBLOCA or a larger than 15-inch LBLOCA. In addition, please explain if CASA Grande considers how a smaller but potentially more focused jet that takes longer to blow down may affect the calcium and aluminum concentrations.

16. Volume 1, Section 1.2.6, "Chemical Release and Precipitation Model," states that several scenarios were investigated using the WCAP-16530-NP-A formula for chemical release. The scenarios used different combinations of liquid temperature, pH, water volume, and fiber quantity for several different break sizes up to a double ended guillotine break. Please clarify if Tables 2.5.34 and 2.5.35 in Volume 6.2 summarize the results of these investigations. Please provide the minimum and maximum values for the pH, fiber quantity, and water volume in the tables. Also please discuss if post-LOCA values could reasonably exceed the minimum and maximum values used in the evaluations. For example, please explain if it is plausible for the pH to be greater than was assumed to be the maximum pH.

17. Page 187 of Volume 3 states "the chemical effects bump-up factor should never be less than one, and there is a practical maximum above which all events will lead to sump failure." Please discuss in more detail the approximate value of a bump up factor that will lead to sump failure. Please provide the values for conventional head loss that are assumed.

18. Volume 6.2, "Item 5.a.6: Corrosion and Dissolution Model," on page 72 states the following:

> The determination of whether a chemical product would form was based on a combination of engineering judgment and limited thermodynamic modeling. The total quantity of material released was not assumed to fully precipitate into chemical products. Instead, solubility limits of chemical products expected to form ... were calculated as a function of temperature and pH using Visual MINTEQ to determine the lowest concentration of metal required for product formation from the range of selected conditions. Sodium aluminum silicate and aluminum oxyhydroxide are the aluminum products described as possible precipitates in WCAP-16530-NP-A; however only the aluminum hydroxide solubility limit (Log K of 10.8 ...) was considered in this analysis since it was determined as a suitable substitute for sodium aluminum silicate in head loss testing.... Calcium phosphate (Log K of -28.25) solubility limits were also evaluated.

> The lowest concentration of metals required to form these chemical products were determined by identifying the lowest solubility over the pH range of 7.0 to 7.3 at a defined temperature.

Using this approach, the concentration of aluminum expected to result in formation of a chemical product is approximately 4.9 milligram per liter (mg/L). The calcium concentration expected to result in the formation of a chemical product was 0.8 mg/L. These values were used to assess the presence of chemical product formation from the calculated material release.

a. Please describe why the NRC staff's judgment that aluminum oxyhydroxide and sodium aluminum silicate precipitates prepared using the WCAP-16530-NP-A method can be substituted for each other in head loss testing is relevant to solubility when evaluating whether an aluminum based precipitate will form in a post-LOCA fluid containing dissolved aluminum.

b. Using Equation 4 in the Argonne National Laboratory Technical Letter Report "Aluminum Solubility in Boron Containing Solutions as a function of pH and Temperature," dated September 19, 2008 (ADAMS Accession No. ML091610696), the lowest aluminum solubility in the pH range 7.0 to 7.3 is approximately 2.7 parts per million (ppm). Please discuss how the analysis results would be affected by assuming the aluminum solubility was 2.7 mg/L instead of 4.9 mg/L.

c. The discussion states that calcium phosphate solubility limits were evaluated. Figure 2.5.34 (Volume 6.2) shows the calcium hydroxide solubility in borated-TSP [trisodium phosphate] solution. Please discuss the rationale for the 0.8 mg/L solubility for calcium and whether it was based on a calcium hydroxide solubility or calcium phosphate solubility.

19. The caption for Figure 5.6.6 (Volume 3) states "Typical sample of sump-strainer head loss histories generated under the assumption of exponential chemical effects factor and artificial head-loss inflation." Please clarify if the "artificial" head loss inflation refers to the NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," October 1995 (ADAMS Accession No. ML083290498), correlation multiplied by 5 or some other value.

20. Please discuss what benchmarking was performed with a) STP specific strainer tests and b) industry test data with similar conditions for the baseline head loss and chemical effects bump up factor.

21. The amount of crud released following a LOCA is estimated to be 5-24 pound-mass (lbm) (Volume 6.2, page 85). For the Casa Grande analysis, please discuss the quantity of crud or other activated debris that is assumed to reach the strainer and how it affects head loss. Please compare the total crud quantity estimated with the amount of crud collected during a controlled crud burst performed at the beginning of refueling outages.

22. A total of five CHLE tank tests were performed to evaluate STP plant-specific chemical effects tests. CHLE Tests 1 and 2 were intended to evaluate an MBLOCA and an LBLOCA, respectively. Please address the following questions related to Tests 1 and 2:

    a. Please discuss why the test screen debris bed is an acceptable method for detection of chemical precipitates given the earlier test "CHLE-010, CHLE Tank Test Results for Blended and NEI [Nuclear Energy Institute] Fiber Beds with Aluminum Addition," that showed no head loss response even in the presence of large quantities of aluminum oxyhydroxide precipitate generated according to the WCAP-16530-NP-A protocol. Note: Additional details are available a September 6, 2012, meeting summary dated October 4, 2012 (ADAMS Accession No. ML12270A055).

    b. Please describe why the use of only aluminum and fiberglass in the MBLOCA test adequately represents the plant specific environment.

Coatings

1. Please provide the basis for the unqualified epoxy size distribution reported in Table 2.2.18 in Volume 3. The NRC staff has previously allowed licensees to assume that degraded qualified epoxy coatings fail in pieces larger than fines. This allowance was limited to epoxy coatings that were originally qualified and have become degraded. The same treatment has not been accepted for epoxy coatings that were unqualified, since these are typically less robust coating systems that would disintegrate into fines. Please specify the epoxy coating in question and provide a basis (i.e., testing) for assuming it fails in pieces larger than fines.

2. Table 2.2.16 in Volume 3 provides the quantity of qualified coatings generated within the Zone of Influence (ZOI) for various pipe diameters. The ZOI used to calculate these quantities is not provided. Please provide the ZOI used for both epoxy and inorganic zinc (IOZ) qualified coatings (e.g., epoxy = 4D, IOZ = 10D).

3. Section 5.4.5, "Unqualified Coating Debris," in Volume 3 states that the total failure fraction is assumed to be 100% for all unqualified coatings. Given this statement, please describe the significance of the failure fraction analysis provided on pages 11 through 17 in Volume 6.2. Please clarify if the failure fraction analysis used in any manner in the CASA model or if the unqualified coatings are always assumed to have a 100 percent failure fraction.

4. For unqualified coatings that are not located in the upper containment, the NRC staff's understanding is that 100 percent of the coatings are assumed to fail and are available for transport to the strainer. The staff also understands that 100 percent of the coatings that are calculated to transport to the strainer are assumed to arrive at the strainer uniformly over the first 36 hours. Please confirm that the staff's understanding is correct. Also, please provide details related to the unqualified coating failure assumptions in terms of percentages, timing and quantities that arrive at the sump strainer.

5. Equations 27 and 28 (page 157, Volume 3) refer to F(t) as the "fraction of coatings that fail during a specific time period." Please provide the value of F(t) and describe the analysis performed to arrive at that value for the timeframe during which upper containment spray is active and capable of transporting coatings (the initial 24 hours). Please state if F(t) is the same for all unqualified coating types. In addition, please provide the cumulative mass of unqualified coatings that fail in the upper containment in the first 24 hours in the current analysis.

6. The Volume 6.2 responses to request for supplemental information, indicate that the failure timing analysis relies heavily on filter data from the EPRI DBA testing of original equipment manufacturer (OEM) coatings. Please address the following questions regarding STP's use of this test data:

   a. Please describe what STP has done in terms of documentation review or testing of plant materials in order to ensure that the plant-specific unqualified coatings at STP are the same as the coatings used in the EPRI testing.

   b. The final proprietary EPRI report on OEM Coatings (EPRI 1011753, "Design Basis Accident Testing of Pressurized Water Reactor Unqualified Original Equipment Manufacturer Coatings. Final Report," September 2005), states that, "Due to the prohibitive nature of the task, there was no attempt to quantify the amount of debris captured in the filters." This testing included many different coating types with varying color, density, and constituent particle size. The autoclave was not opened and the tested components were not examined until the entire test was complete. The NRC staff could not determine if lighter coatings which would be less visible on a filter (and certainly less visible on a photo of a filter) failed at the same rate as darker coatings or if they may have been present on one or more of the filters removed early in the test. Given this information and the fact that the testers stated that they made no attempt to quantify debris on the filters, please provide additional justification for using this test data to assign a failure time to unqualified coatings.

   c. The EPRI OEM report also states that, "With regard to timing of the coating failures, the filters do not demonstrate a definitive time of failure, however in subjective terms, it appears that much of the failure occurred in the 24- to 48-hour timeframe." STP seeks to reduce the transported unqualified coating debris from upper containment by 94 percent compared to a deterministic approach (100 percent failure in a deterministic evaluation, 6 percent for STP). The NRC staff is not persuaded that a subjective review of photographs from a test performed in 2005 is adequate justification for the proposed failure timing. Please provide additional justification for the current analysis or provide a revised value for the failure timing.

7. Please describe any ongoing containment coating condition assessment program. Please include the frequency and scope of the inspections, acceptance criteria, and the qualification of personnel who perform containment coatings condition assessment inspections.

Containment and Ventilation Branch (SCVB)

1. In support of Enclosure 2-3, "Request for Exemption from Certain Requirements of General Design Criterion [GDC] 38," please provide the following:

   a. Please list the specific STP plant systems that will not meet the requirements of GDC-38.

   b. Please describe the specific requirements of GDC-38 that will not be met by each of the plant systems listed in response to item (a) above.

2. In support of Enclosure 2-4, "Request for Exemption from Certain Requirements of General Design Criterion 41," please provide the following:

   a. Please list the specific STP plant systems that will not meet the requirements of GDC-41.

   b. Please describe the specific requirements of GDC-41 that will not be met by each of the plant systems listed in response to item (a) above.

3. Enclosure 3, page 4, paragraph "Use of a Risk-Informed Approach to Resolving GSI-191", states:

   > The design and licensing basis descriptions of accidents requiring ECCS operation, Including analysis methods, assumptions, and results provided in UFSAR [Updated Final Safety Analysis Report] Chapters 6 and 15 remain unchanged. This is based on the functionality of the ECCS and CSS during design basis accidents being confirmed by demonstrating that the calculated risk associated with GSI-191 for STP Units 1 and 2 is "Very Small" and less than the Region III acceptance guidelines defined by RG 1.174.

   The current licensing basis containment analysis methodology used to confirm the adequacy of the containment heat removal system (which complies with 10 CFR 50 Appendix A GDC-38) described in the UFSAR is different from the proposed methodology which resolves GSI-191 on a risk-informed basis and proposes an exemption request from GDC-38. For example: (a) difference in the single failure assumption in the proposed and current analysis; (b) computer codes RELAP for LOCA mass and energy (M&E) release, and MELCOR for the LOCA sump temperature response are used in the proposed analysis, and SATAN-VI, WREFLOOD, FROTH are used for M&E release and CONTEMPT4/MOD5 is used for sump temperature response in the current analysis; and (c) the proposed analysis inputs and assumptions are required to be conservative from GSI-191 perspective and also required to be conservative for sump temperature response whereas the current analysis inputs and assumptions are conservative for sump temperature response,

   a. Please justify why the UFSAR licensing basis description of the methodology used for confirming the adequacy of containment heat removal system which

complies with GDC-38 should not be replaced with the proposed licensing basis methodology which takes an exemption from GDC-38.

b. Tabulate the differences between the inputs and assumptions between the current licensing basis containment analysis that calculates the most limiting sump fluid temperature profile for available NPSH calculation and the proposed containment analysis performed for risk-informed GSI-191. Please justify that the inputs and assumptions in the proposed methodology are conservative from both GSI-191 and sump temperature response perspectives.

c. In case the UFSAR licensing basis description of the containment heat removal system, including its related mass and energy release analysis methodology, is required to be replaced, please provide the revised UFSAR input for NRC staff review and approval.

4. The current licensing basis methodology for the iodine removal is documented in UFSAR Section 6.5.2, "Containment Spray System - Iodine Removal." The iodine removal is accomplished by the CSS which meets the requirements of 10 CFR 50 Appendix A GDC-41. The proposed risk-informed GSI-191 methodology takes exemption from compliance with GDC-41 requirements.

   a. Please justify why the UFSAR licensing basis description of the iodine removal should not be revised with the proposed methodology which takes exemption from GDC-41.

   b. Please tabulate the differences between the inputs and assumptions between the current licensing basis containment atmosphere cleanup method and the proposed containment atmosphere cleanup which takes exemption from the GDC-41 requirements.

   c. In case the UFSAR licensing basis description of the iodine removal system is required to be replaced, please provide the revised UFSAR input for NRC staff review and approval.

5. In support of Volume 6.2, please list the differences between the heat sinks in the current licensing basis containment analysis documented in the UFSAR Tables 6.2.1.1-7 and 6.2.1.1-8 and in the proposed containment analysis for risk-informed GSI-191. Please provide justification in cases where the conservatism is reduced in the proposed analysis.

6. In support of Volume 6.2, please list the differences between the LOCA surface heat transfer model for heat sinks in the current licensing basis analysis documented in UFSAR Table 6.2.1.1-9 and the model in the proposed containment analysis for risk-informed GSI-191. Provide justification for the differences in case the conservatism is reduced in the proposed analysis.

7. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), Section 6.2.1.5, "Minimum Containment

Pressure Analysis for Emergency Core Cooling System Performance Capability Studies," describes the minimum containment pressure analysis for ECCS performance capability. RG 1.157, "Best Estimate Calculation for Emergency Core Cooling System Performance," May 1989 (ADAMS Accession No. ML003739584), Section 3.12.1, "Containment Pressure," provides guidance for calculating the containment pressure response used for evaluating cooling effectiveness during the post-blow-down phase of a LOCA.

UFSAR Section 6.2.1.5 documents the current minimum containment pressure analysis for performance capability studies of the ECCS. Please describe the proposed containment analysis, including assumptions and inputs, performed for the calculation of minimum containment pressure input for the ECCS analysis that calculates the peak cladding temperature for risk-informed GSI-191. Please justify that the inputs and assumptions are conservative for the purpose.

8. Volume 6.2, page 117, Item 5.a.14, "In-Vessel Thermal Hydraulic Analysis," lists six scenarios simulated using the 3D Vessel-ID Core Model. Please describe and justify the basis for selection of these LOCA breaks scenarios.

9. In support of Enclosure 2-2, "Request for Exemption from Certain Requirements of General Design Criterion 35," please provide the following:

    a. Please list the specific STP plant systems that will not meet the requirements of GDC-35.

    b. Please describe the specific requirements of GDC-35 that will not be met by each of the plant systems listed in response to item (a) above.

Nuclear Performance and Code Review Branch (SNPB)

1. Please provide the following information for the STP Nuclear Steam Supply Systems (NSSSs):

    a. Volume of the lower plenum, core and upper plenum below the bottom elevation of the hot leg, each identified separately. Also, please provide the heights of these regions and the hot-leg diameter.

    b. Loop friction and geometry pressure losses from the core exit through the steam generators to the inlet nozzle of the reactor vessel for steady state full power operation. Also, provide the locked rotor reactor coolant pump (RCP) k-factor. Please provide the mass flow rates, flow areas, k-factors, and coolant temperatures for the pressure losses provided (upper plenum, hot legs, Steam Generators (SGs), suction legs, RCPs, and discharge legs). Please include the reduced SG flow areas due to plugged tubes. Also, provide the loss from each of the intact cold legs through the annulus to a single broken cold leg and the equivalent loop resistance for the broken loop and separately for the intact loop. Please identify the flow area (hydraulic diameter) on which the k-factors are based.

c. Capacity and boron concentration of the RWST.

d. Capacity of the condensate storage tank (CST).

e. Flushing flow rate at the time of switch to simultaneous injection.

f. High pressure safety injection (HPSI) runout flow rate.

g. Capacities and boron concentrations for high concentrate boric storage acid tanks, if part of system.

h. Flow rate into the RCS from the boric acid storage tanks, if applicable.

i. Time to empty the RWST (all pumps operating).

j. Minimum containment pressure or containment pressure versus time graph.

k. Sump boric acid concentration versus time.

l. Minimum RWST temperature.

m. Injection temperature versus time from sump during recirculation.

2. Please provide the following elevation data:

   a. bottom elevation of the suction leg horizontal leg piping and cold leg diameter

   b. top elevation of the cold leg at the RCP discharge

   c. top elevation of the core (also height of core)

   d. bottom elevation of the downcomer

3. Please provide the limiting bottom and top skewed axial power shapes.

4. Please provide the latest analysis or reference showing the timing for boric acid precipitation for the limiting large-break and small-break LOCAs.

5. Justification and description of the methodology used to compute the sump boric acid concentration versus time.

Reactor Systems Branch (SRXB)

The NRC staff requests the licensee to provide the following:

1. *RELAP-3D input decks for these cases with a 3-D vessel and 1-D core:*

    a. *Steady state case in Cold Leg*
    b. *Medium Break LOCA (6") in Cold Leg*
    c. *Double-Ended Guillotine (DEG) Break in Cold Leg*
    d. *Core blockage input file*

2. *RELAP-3D input decks for these cases with a 3-D vessel and 3-D core:*

    a. *Steady state case in Cold Leg*
    b. *Medium Break LOCA (6") in Cold Leg*
    c. *DEG Break in Cold Leg*
    d. *DEG Break in Cold Leg with maximum boron*
    e. *Core blockage input file*

3. *Conversion tables between RETRAN and RELAP-3D ("South Texas Project Power Plant RETRAN-RELAP-3D Conversion Tables")*

4. *Documentation describing model verification ("South Texas Project Power Plant RELAP-3D Steady-state model Verification)*

5. Table 2.2.1 in Volume 3 provides results for sump switchover time based on the break size during a loss of coolant accident (LOCA). The application states that "the timing for switchover to recirculation is dependent on the volume of water in the RWST and the total ECCS and CSS flow rate."

    a. Please provide the assumptions used for the volume of water in the RWST for the results in Table 2.2.1. Please provide justification for use of these assumptions.

    b. Please provide the ECCS flow rate and CSS flow rate for each break size in Table 2.2.1.

    c. Please explain how sump switchover time is calculated based on the responses to a. and b. above.

6. Section 2.2.1 in Volume 3 describes the termination criteria for containment sprays. One of the criteria to terminate containment sprays is that containment pressure has dropped below 6.5 psig. Please provide plots for containment pressure versus time for a range of break sizes to verify pressure drops below the termination criteria of 6.5 pounds per square inch gauge (psig) before 6.5 hours.

7. Switchover to hot-leg injection is started 5.5 hours after the beginning of the LOCA event and is assumed to be completed between 5.75 and 6 hours.

   a. Please summarize the Emergency Operator Procedures (EOPs) that direct the operators to take this action and an associated timeline of the operator key actions if this event were to occur.

   b. Please provide a justification that demonstrates the 15-30 minute response time is achievable during switchover to hot leg injection training scenarios. In the justification, please include the results of simulator runs and training logs.

8. Table 2.2.14 (Volume 3) shows safety injection (SI) flow rates for nominal operating conditions. Please justify the use of nominal conditions versus the use of limiting conditions when analyzing LOCA.

9. Please describe the terms total sump flow, total SI flow, and ECCS flow. Please include if high head safety injection (HHSI), low head safety injection (LHSI), or CSS is a part of each definition.

Safety Issue Resolution Branch (SSIB)

<u>ZOI</u>

1. Volume 3, Table 5.3.1 lists Double Ended Guillotine Break (DEGB) equivalent diameters and Table 5.3.2 lists computer aided design (CAD) DEGB values. These values are calculated by doubling the single sided break area and then calculating an equivalent pipe diameter. Approved ZOIs are based on doubling the volume of single sided break jets (calculated by the American National Standards Institute (ANSI) model) and calculating a radius for a sphere based on that volume. Please describe how the CAD DEGB values calculated by Equation 22 of Volume 3 are used.

<u>Debris Characteristics</u>

2. Please provide the size distributions for Nukon and Thermal Wrap debris created by the postulated LOCA jet (percentage of each size category created). Please provide the methodology used, including the bases, to determine the size distributions. Please provide information regarding whether the distribution is a simple percentage of all generated fibrous debris or based on the distance of the insulation from the break (Volume 3, Sections 2.2.15, "Insulation Debris Size Distribution," 4.2, "Structures Information Process Flow," and 5.4.2, "Insulation Debris Size Distribution Model").

3. Please clarify if the material properties of debris listed in Table 2.2.21 of Volume 3 are used in the head loss correlation. If so, please state if varying the sizes of the particles to a more realistic distribution affect the results significantly. Explain how the particulate debris types that have size distributions are implemented in the correlation. Please explain whether the uncertainty of the size distribution of the materials can affect the permeability of the debris bed and therefore the head loss. Is this uncertainty significant, and if so, please state how is it accounted for in the STP model.

Transport

4. Please provide justification for assumption 6.h.i. of Volume 3 (page 78). According to Table 2.2.22 of Volume 3, line breaks below SG and surge line result in a greater percentage of small debris being blown directly to lower containment. This debris is considered to enter the pool directly while debris blown to upper containment may be held up. Please explain why the SG compartment transport fractions are considered to be conservative compared to these other break locations.

5. Please explain assumption 6.h.v of Volume 3 (page 79). Please describe how the number of strainers in service affects pool fill transport. It appears that the pool fill transport phase would be largely completed prior to strainers being placed in service.

6. For the blowdown transport evaluation, it was not clear how the Drywell Debris Transport Study (DDTS), NUREG/CR-6369, "Drywell Debris Transport Study," Volumes 1, 2, and 3, September 1999 (ADAMS Accession Nos. ML003728226, ML003726871, and ML003728322, respectively), results were applied to the plant condition. (Reference: Volume 3, Section 2.2.17, "Blowdown Transport Fractions," and Volume 6.2, Item 5.a.2 (page 37). Please provide the following information:

   a. The DDTS cautions that if gratings do not cover the entire transport path, they may not be as effective in debris capture. For transport paths where grating does not fully span the transport pathway, please state if the capture metrics was adjusted to account for this potential.

   b. Please state if the calculational methodology account for depletion of debris, as it is captured on upstream objects. Please clarify if the amount reaching the second and third (etc.) objects reflects the debris lost on upstream objects. This was not apparent to the NRC staff upon inspection of the equations used to perform the calculation.

   c. Please explain what was considered to be a 90 degree turn in the plant and how this compared to the 90 degree turns modeled in the DDTS. Please explain how it was determined that the DDTS results are applicable to the STP conditions considered to be 90 degree turns.

   d. Please clarify if there are limits to the mass of debris that can be captured on structures or on specific surface areas of structures and, if applicable, how such limits would affect the calculations for holdup.

   e. Please explain how the ranges of values used in the DDTS were determined to be applicable to the STP conditions.

7. For the washdown transport evaluation it was not clear to the NRC staff that the DDTS test results were applied realistically or conservatively to the STP plant condition as described in Volume 1, Section 1.2.3, "Washdown Transport," Volume 3, Section 2.2.18,

"Washdown Transport Fractions," and Items 5.a.3 and 5.a.5 of Volume 6.2. Please provide the following information:

a. Please state if the washdown evaluation considered that it may be more likely for a piece of debris that has been blown through one or more gratings to subsequently wash down through gratings. It appears that the DDTS did not use debris that had been passed through gratings when studying washdown through gratings.

b. The DDTS washdown tests were run for 30 minutes. The DDTS stated that most washdown occurred in the first 15 minutes. It was not clear to the NRC staff from the test results how washdown over a significantly longer period of time would occur. Please explain why the DDTS results are applicable to significantly longer washdown periods.

c. Please state if debris is washed down through one level of grating, if it is more likely to wash through subsequent levels. If it washes through more than one grating, is it more likely to pass through subsequent levels? Please explain how the transport evaluation accounts for such a likely potential. What was considered when determining the retention fractions for debris on additional levels of grating in the washdown transport evaluation? It was stated that engineering judgment was used in this determination, but the NRC staff did not find an adequate basis documented for the engineering judgment.

d. Please state if the washdown transport evaluation accounted for the significantly higher velocities that may occur with sheeting flow at the beginning of washdown.

e. The submittal provided the calculations for washdown percentages. Item 5.a.3 (Volume 6.2, page 43) uses values of 0.4 and 0.5 for F_{wg}, fraction of debris held up when washed through the first level of grating. The DDTS states that 40-50 percent pass through. Please clarify if 0.4 and 0.5 be reversed or if the 0.4 should be changed to 0.6. The terminology used is not clear and can be misunderstood.

f. Table 2.5.24 of Volume 6.2 is titled, "Washdown transport fractions used in CASA Grande," but the leading paragraph states that the table contains blowdown fractions. Please clarify whether it is blowdown transport fraction or washdown blowdown transport.

8. The evaluation for transport of partially submerged debris on the operating deck makes several unsubstantiated assumptions (Item 5.a.5, Volume 6.2, page 54). Please provide the following information:

a. Please explain if the assumed size distribution considered that most of the debris blown to the operating deck would pass through grating thus likely reducing the size. Please clarify if the assumed size distribution was adjusted for this effect.

b. Please state if the evaluation considered that the initial sheeting flow may be at a higher velocity than the steady state flow and that this may push debris across the floor before a steady state occurs.

c. Please state the basis for the porosity equation. Please explain why the bulk density (as-fabricated) of the fiber relevant after it has been rendered into small pieces and then been blown through grating.

d. Please clarify if the evaluation considered that air may be trapped within the fiber and that it may pick up additional air as it tumbles across the floor.

e. Please state if there is any experimental data available to validate the calculational methodology.

9. It was not clear to the NRC staff if the fibrous debris eroded from large and small pieces of debris were added to the fine transport term. Please clarify that the eroded term was added to the fine source term and is not added as the size category from which they were eroded. For example, in Figure 5.5.3 of Volume 3, the transported debris should be 1.8 percent fines and 35.6 percent small while the total shows 37.4 percent transported (35.6 + 1.8). Please state if the evaluation includes the 1.8 percent as fine debris.

10. The submittal states that unqualified coatings that fail after the sprays are secured cannot transport to the containment sump (Reference Volume 1, Section 1.2.3, "Washdown Transport," Volume 3, Section 2.2.10, "Unqualified Coatings Quantity," Volume 3, Section 5.4.5, "Unqualified Coatings Debris," Volume 3, Section 5.5.7, "Strainer Transport"). Please explain how it was determined that they would not transport. Please clarify if there are transport mechanisms besides washdown from containment spray that could cause some of the coatings to transport. For example, please explain if coatings are located in areas where they could fall directly into the sump or fall relatively freely to the sump. Explain if the flow of condensation on surfaces can carry particles of failed coatings to the sump, etc. (page 173, Section 1.2.3; page 571, Section 2.2.10, page 674, Section 5.4.5, page 680, Section 5.5.7)

11. Table 5.5.5 of Volume 3 lists the recirculation debris time, x(t) and states that it is described in Section 5.8 of Volume 3. The NRC staff was not able to locate a description of this variable. Please provide a definition of the debris recirculation time. The staff found the term x(t) described in STP's initial submittal dared June 19, 2013 (page 74 of 174 of the attachment), which has been superseded in its entirety. Please provide a response to the following questions, which are partially based on the information provided in June 19, 2013, submittal.

a. Please provide a description of the recirculation debris time as implemented in submittal dated November 13, 2013.

b. Please describe the types and sizes of debris to which the recirculation debris time applies.

c. Please state if the equation assumes homogeneous mixing in the pool. If so, please explain if this this is a valid assumption for all debris types.

d. It appears that the x(t) function calculates that all debris is at the strainer at time=0 and decreases as time progresses. This appears to be the reverse of the actual condition expected. Please explain.

e. Please explain the basis for the depletion rate.

12. Based on description of Item 5.a.4 of Volume 6.2 (Page 47), it is assumed that debris will remain in the vicinity in which it was washed down until recirculation starts. Please provide additional justification for this assumption. Please state if the debris would be redistributed during pool fill including by potential sheeting flow and if this would affect the assumption that debris is mixed homogeneously in the pool at the start of recirculation. If so, please describe which types and sizes of debris are affected.

13. Table 2.5.32 of Volume 6.2 includes values for small pieces of Microtherm. This is inconsistent with other statements in the submittal that Microtherm is assumed to fail as 100 percent fines. Please clarify if all Microtherm pieces fail as fines. Please state how small pieces of Microtherm are treated in STP's evaluation.

<u>Head Loss and Chemical Effects Bump Up</u>

14. The STPNOC submittal assumes that no chemical bump up occurs if the debris bed thickness is less than 1/16-inch (Reference: Volume 1, Section 1.1, Step 14; Volume 3, Assumption 7.c; Volume 6.2, Items 5.a.10 and 5.a.11). The NRC staff has previously concluded that a 1/16-inch debris bed is an adequate metric for this purpose for clean plants, where the worst analyzed break could result in 1/16-inch of fiber when conservative methods were used for estimating the amount of debris generated and transported to the strainer. The clean plant criteria also included other restrictions for the use of the metric, such as the lack of problematic debris within any ZOI. The staff has not determined that this limit is appropriate for a more realistic risk-informed evaluation. The staff has reviewed test results conducted with about 1/16-inch of fibrous debris that resulted in some head loss when chemical precipitates were added to the test. It also appears that the STP evaluation has not considered all aspects of the clean plant criteria. Whether or not the clean plant criteria are the basis for the 1/16-inch limit, please provide a justification for its use.

15. The STPNOC submittal states that the strainer debris head loss is calculated using a correlation (References: Volume 3, Section 5.6.2, "Conventional Debris Head Loss Model," Volume 3, Assumption 7.e; Volume 1, Section 1.1, "Structured Information Process Flow.") The NRC staff has generally not accepted correlations for the qualification of PWR strainers for several reasons. Please explain why the following general concerns with the use of correlations are not an issue for the STP application:

a. Correlations have not been validated for the full range of debris loads and morphologies present under plant conditions.

b. Correlations do not address non-homogeneous debris beds which are very likely to occur due to transport timing and non-homogeneous filtering of debris within the bed.

c. Correlations have not been validated for the full range of potential flow conditions and strainer geometries that are present in plants.

d. There is significant uncertainty in the model parameters used to describe the physical attributes of the debris bed constituents.

16. Testing performed to validate the NUREG/CR-6224[(1)] correlation for specific STP conditions does not appear to accomplish the purpose (References: Volume 3, Section 5.6.2, "Conventional Debris Head Loss Model," and Volume 6.2, Item 5.a.10). Additionally, the NRC staff does not have complete information to conclude that the testing adequately represented the plant configuration and range of conditions that could occur at STP following a LOCA. Therefore, the staff was unable to determine that the plant specific vertical loop tests results were representative of head losses that could occur from a debris bed on a prototypical module. The correlation results and the single vertical loop test result that modeled the July 2008 Alden flume test under STP specific conditions were significantly different from each other, and from the results of the flume test. That is, all three results, although modeling similar conditions, had significantly different results. The submittal explained as to why the results were substantially different, but the explanation was not confirmed by testing or by use of accepted theories. Industry head loss tests using similar surrogates on prototypical strainer modules resulted in significantly higher head losses than those reported by STP for their vertical loop testing and those calculated by use of the correlation. The staff is concerned that vertical loop test and module test results from tests conducted under similar conditions may differ due to the differences between debris characteristics in vertical loop test and module test debris beds. These differences could be caused by differences in transport and deposition of the debris onto the perforated surfaces. Therefore, the staff is concerned that the validation testing is not representative of the plant. Please provide the following additional information:

a. If the vertical loop tests conducted by STP are important to the conclusions, please provide details as to why the STP vertical loop tests are valid considering that other module tests conducted in several different facilities under similar conditions, debris loads, and debris characteristics had significantly different results.

b. If the vertical loop tests conducted by STP are important to the conclusions, please provide evidence that vertical loop tests conducted under site specific conditions will correlate to flume tests conducted under similar conditions or to head losses that would occur in the plant. Please include information regarding how it was determined that the debris that transported to the horizontal strainer surface resulted in a debris bed of similar characteristics and morphology to that

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(1) NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," Final Report," October 1995 (ADAMS Accession No. ML083290498).

which would transport to the plant strainer. Please state how was it determined that the head losses would be comparable.

c. Please provide information that demonstrates that the correlation used by STP is valid for plant specific strainer geometries and plant specific conditions. Alternately, please provide a basis for using a correlation that has not been validated specifically for STP plant conditions and geometries.

d. Please discuss how the NUREG/CR-6224 correlation could be used to predict the head losses that would be expected under conditions similar to those in the two flume tests conducted by STP in February and July 2008.

17. The submittal states that all testing performed to validate the NUREG/CR-6224 correlation was bounded by correlation predictions (References: Volume 1, Section 1.2.7, "Conventional Head Loss Model," and Volume 3, Section 5.6.2, "Conventional Debris Head Loss Model"). There have been numerous cases where the correlation severely under-predicted head losses that were carried out under carefully controlled conditions. NUREG-1862, "Development of Pressure Drop Calculation Method for Debris-Covered Sump Screens in Support of Generic Safety Issue 191," February 2007 (ADAMS Accession No. ML071520440), and NUREG/CR-6917, "Experimental Measurement of Pressure Drop Across Sump Screen Debris Beds in Support of Generic Safety Issue 191," February 2007 (ADAMS Accession No. ML071910180), contain data that show that the NUREG/CR-6224 correlation is not conservative in all cases. These NUREGs determined that correlation predictions are highly dependent on the parameters used to describe the physical attributes of the debris bed constituents and that these parameters have significant uncertainty. These NUREGs also determined that head losses are not well predicted by a correlation that assumes a homogenous debris bed. Some of the experimental data involved fine debris, microporous debris, non-homogeneous beds, and other conditions that the NUREG/CR-6224 correlation is not designed to account for. It is very likely that some conditions that NUREG/CR-6224 correlation does not account for, may be present under plant post-LOCA conditions. The submittal states that the head loss correlation from NUREG/CR-6224 has been extensively validated for various conditions. The NRC staff is of the opinion that there is little or no testing that has been conducted under conditions similar to those at STP. The staff is concerned with the validation issues listed below when using a correlation for qualification of strainers. Please state how the STP evaluation accounts for these uncertainties and lack of validation of the correlation under plant conditions.

a. Debris constituents in validation testing are not plant-specific

b. Debris sizes in validation testing are not plant-specific.

c. Very little validation testing was conducted at STP velocities and none validated the correlation.

d. Validation testing did not include prototypical strainer geometries.

e. Validation testing performed in vertical loops does not simulate potentially important aspects of debris bed formation under plant conditions

f. The records of early validation testing are not available or do not contain the information required to determine whether the tests were conducted to adequately represent plant conditions. Therefore, conclusions from early testing must be limited.

18. The implementation of the correlation in the STP model makes specific assumptions and may potentially contain modeling errors that can significantly affect the results of the calculation (References: Volume 3, Assumptions 7.b, 7.e, and 7.f; Volume 3, Section 5.6.2, "Conventional Debris Head Loss Model"; Volume 6.2, Item 5.a.10; and Enclosure 6, Table 1). Please provide the following information to justify that the assumptions and use of the correlation is realistic or conservative for STP plant-specific conditions.

   a. Please provide justification that the beds are homogeneous representative of the plant (Volume 3, Assumption 7.e and Volume 6.2, Item 5.a.10).

   b. It is assumed that fiberglass debris would accumulate uniformly with a density of 2.4 pounds per cubic foot (lb/ft^3). The NRC staff is of the opinion that assuming the debris bed density to be the same as the manufactured density may not be an accurate assumption and is based on the observation of debris beds formed in industry tests and NUREG-1862 testing. The NRC staff is further of the opinion that in the plant, only fine and small fiber will transport and collect at a much higher density. Please describe why the density assumption is valid and why it does not significantly affect the results. Alternately, re-perform the analysis with a density that has been shown to be appropriate. (Volume 3, Assumption 7.f and Section 5.6.2 and Volume 6.2, Item 5.a.10).

   c. Please explain how the NUREG/CR-6224 correlation compression function is applied in the STP model. NUREG-1862 found that the compression relation from NUREG/CR-6224 does not accurately model the compression of the bed, especially at low flow velocities like those at STP. (Volume 3, Section 5.6.2).

   d. The submittal states that STP implemented a linear mass weighted average instead of the volumetric weighted average for implementation of composite surface to volume ratio (S_v) in the 6224 correlation. The submittal states that there are many possible composite weighting methods that could be used, but does not justify the method chosen in the application. NEI 04-07 and NUREG/CR-6371, "Blockage 2.5 Reference Manual," December 1996 (not publicly available), both recommended the volume weighting method. Please explain why mass weighting is acceptable. Please explain if both the methods result in significantly different results. (Volume 3, Section 5.6.2 and Enclosure 6).

   e. Please provide a technical basis for Assumption 7.b regarding coating material packing fractions. Please discuss the effect of the assumption on results.

Please provide the potential ranges of packing factors for coating materials. (Volume 3, Assumption 7.b and Volume 6.2, Item 5.a.10).

19. The application of a multiplier of five (5x) to the result of the head loss correlation used in the STP model appears to indicate uncertainty in the ability of the correlation to predict head losses correctly (References: Volume 1, Section 1.2.7 and Volume 3, Section 5.6.2). If the NUREG/CR-6224 correlation is a robust model as implied in the submittal, the NRC staff is of the opinion that it is unnecessary to use safety factors in the head loss calculations for achieving realistic results. The staff noted that some PNNL testing showed that the 6224 correlation under predicted head loss by more than a factor of 5X. Please provide justification for applying the multiplier to the results of the head loss correlation.

20. The submittal assumes that paint chips or other relatively large debris that may reach the strainer can be accounted for in the correlation as spherical particles (Reference: Volume 3, Section 5.6.2). Large debris may fully or partially block strainer perforations and may deposit non-homogeneously on the strainer. Please provide an experimental basis to confirm that paint chips (or other large particles) may be accurately modeled in the correlation, including the assumption that they can be accurately modeled as spherical particles. In the absence of an experimental justification, please provide an alternate basis for the STP treatment of paint chips and other large particles in the head loss correlation.

21. The STP correlation uses physical properties of materials predicted to be in the debris bed in order to calculate a head loss (References: Volume 3, Section 5.6.2, "Conventional Debris Head Loss Model," and Volume 6.2, Item 5.a.10). Results from the NUREG/CR-6224 correlation are heavily dependent upon the accurate representation of material physical properties. One of the most difficult parameters to accurately determine is the surface to volume ratio (S_v). Please provide the following information:

    a. Please state how S_v values were determined for each material.

    b. It is known that for some debris types, and possibly all debris types expected to be present in PWR debris beds, physical measurements cannot provide S_v values that allow accurate prediction of head loss in existing correlations. This was especially evident for microporous type materials and was shown to be true for other materials by NUREG-1862. Please explain the bases for the S_v values and other material properties used in the STP implementation of the correlation.

    c. Please state how the uncertainty, described on page 184 of Volume 3, is caused by the relationship between experimentally deduced S_v values and head loss, accounted for in the STP model.

    d. The NRC staff does not agree with the statement, on page 185 of Volume 3, which states that the lack of agreement between the correlation and test results using green silicon carbide and tin do not affect the STP calculations. It appears that STP had difficulty determining parameters to input to the correlation to attain

accurate results. Please provide basis for the conclusion that the lack of agreement between the correlation results and test results do not affect STP head loss calculations.

22. The NUREG/CR-6224 correlation, and other similar correlations, use specific surface areas ($S_{v)}$ for cylindrical objects assuming that the fiber is oriented perpendicular to the flow and that the fibers have a uniform diameter. This assumption is used in the STP model (Reference: Volume 1, Section 1.2.7, "Conventional Head Loss Model," and Volume 3, Section 5.6.2, "Conventional Debris Head Loss Model"). NUREG-1862 calculated different specific surface areas for varying diameters of Nukon and noted a difference in S_v between fibers that had binder and those that did not. The NUREG also estimated the S_v of Nukon fiber to be around 250,000 to 300,000 ft^{-1} instead of 180,000 ft^{-1} when corrected for test data. (STP uses for the S_v of 571,429 m^{-1} (174,172 ft^{-1}) for Nukon fiber.) Please explain how the STP evaluation takes these findings into account. (Volume 3, Section 5.6.2).

23. The STPNOC submittal makes the assumption that Microtherm fibers will have properties similar to those of Nukon (bulk density = 2.4 lb/ft^3, microscopic density = 165 lb/ft^3 and S_v = 666,667 m^{-1}) (Reference: Volume 3, Section 5.6.2). Please state the basis for this assumption. Also, please justify the use of the Nukon fiber bulk density as the debris bed density.

24. The physical characteristics used in the head loss correlation can have a significant effect on the results of the head loss calculations. Characteristic values that describe the assumed behavior of STP debris are provided in Tables 5.6.1 and 5.6.2 of Volume 3. NRC research conducted for NUREG-1862 has determined that some of the values that describe the physical characteristics of debris are not well understood. Please provide the bases for the values in Tables 5.6.1 and 5.6.2. Please provide reasonable uncertainty bands for the material properties. Also, please explain how compounded inaccuracies in assumed material properties would affect the head loss values predicted by the correlation.

25. Volume 3, Assumption 7.f, states that it is assumed that fiberglass will accumulate uniformly on the strainers but also states that the amount of debris that can collect on the bottom of the strainer is limited to two inches. This assumption seems to contradict itself.

    a. Please explain how the assumption is accounted for in head loss calculations or provide information that shows it is not significant to the results. Please explain how a non-uniform accumulation of fibrous debris, limited by the floor or pool height, would affect the head loss calculation.

    b. Please provide an evaluation of how this affects Assumption 7.e. of Volume 3 regarding homogeneous bed formation.

26. The submittal calculates circumscribed bed surface areas based on debris loading (Reference: Volume 3, Section 5.6.2). Please provide the following information:

    a. Please state if areas calculated for beds transitioned from thin bed to circumscribed.

    b. When fibrous debris is deposited on the strainer its density will be significantly increased from the manufactured value. Please state how was this accounted for (Volume 3, Page 696, Section 5.6.2).

    c. Please clarify if there are any objects around the strainer that would prevent the debris bed from accumulating uniformly as assumed in the strainer loading (Volume 3, Table 5.6.3).

    d. The NRC staff is of the opinion that it is not realistic to assume the thickness of the debris bed on the strainer can be such that it will exceed the height of the water level in the pool. Please explain how this affects the debris loading calculation (Volume 2, Section 5.6.2).

    e. Please state how often the debris loading algorithm results in a circumscribed bed or one that is transitioning to circumscribed (fully or partially filled interstitial volume).

    f. Please explain the significance of cases that result in the interstitial volume of the strainer becoming partially or completely filled with debris.

27. The submittal states that the clean strainer head loss (CSHL) is 0.220 ft. based on a test (Reference: Volume 3, Sections 2.2.23, "Clean Strainer Head Loss," and 5.6.1, "Clean Strainer Head Loss"). It appears that the value was taken from a test that was conducted using a single module. The CSHL should be reflective of the entire strainer including all modules and connecting piping, fittings, etc. In the STPNOC letter dated December 11, 2008 (ADAMS Accession No. ML083520326), the CSHL was stated to be 1.95 ft. Please explain why the value provided in the risk-informed submittal is significantly different from the previously calculated value and verify that it includes all head losses associated with the entire clean strainer train.

28. Considering the individual uncertainties that result from the issues described in the request for additional information on the subject of head loss, please provide justification that the use of a correlation is acceptable in the risk-informed model. Along with the justification, please provide an analysis of the overall uncertainty and state how this will be incorporated into the overall risk-informed evaluation.

<u>NPSH and Degasification</u>

29. Volume 1, Section 1.1, "Structured Information Process Flow," and Volume 3, Sections 2.2.28, "Pump Gas Limits," and 5.7.4, "Acceptance Criterion: Pump Gas Void Limits," describe the methodology for calculating NPSH margin. The submittal states that if the void fraction exceeds 2 percent that the scenario is recorded as a failure. It is not clear

that NPSH Required (NPSHR) is corrected for degasification that may occur as fluid passes through the debris bed as recommended by RG 1.82, Revision 4, "Water Sources for Long Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 4, March 2012 (ADAMS Accession No. ML111330278). Please clarify whether NPSHR is corrected for the void fraction at the pump inlet. If the NPSHR is not corrected for the void fraction, please provide a justification.

30. The STPNOC submittal states that the degasification caused by the pressure drop through the debris bed is calculated to determine if a pump failure criterion is met (References: Volume 1, Section 1.1, "Structured Information Process Flow"; Volume 3, Assumptions 8 a. through i.; Volume 3, Section 5.7.2, "Degasification"; and Enclosure 6, Table 1). Please state if the degasification calculation credits containment accident pressure. If so, please explain how the pressure for each case or condition is calculated. Please state what temperature is used for the degasification calculation and how this temperature was calculated for each case.

31. The STPNOC submittal does not seem to evaluate the possible effects of the collection of gas bubbles in the strainer or ECCS pump suction piping (Reference: Volume 3, Assumption 8.h. and Section 5.7.3, "Gas Transport and Accumulation"). Please explain how it was determined that gas bubbles would not collect in the strainer, or piping between the strainer and ECCS and CSS pumps and eventually transport as large voids. If gas pockets can become trapped in these locations, please explain its effect.

32. The NRC staff could not determine whether the calculation of NPSH Available (NPSHA) includes containment pressure greater than the saturation pressure of the sump fluid. Volume 3, Assumption 1.c indicates that containment pressure greater than the saturation (above 14.7 pounds per square inch absolute (psia)) is not credited in the NPSH calculations (Reference: Volume 1, Section 1.1, "Structured Information Process Flow"; Volume 3, Sections 3, "Assumptions," and 5.7.2, "Degasification"; and Enclosure 6, Table 1). Please clarify if the calculation for NPSHA includes containment pressure above the saturation pressure of the fluid. If containment pressure greater than the saturation pressure of the fluid is credited in the NPSHA calculation, please provide justification for its use and provide the methodology used to calculate the containment pressure and sump fluid temperature for each case.

33. The pool water level calculation provided in the STPNOC submittal does not appear to account for changes in pool area with elevation or changes in objects that may displace water (Reference: Volume 3, Section 2.2.5, "Pool Water Level"). Please state if there are significant changes in area or objects in the pool that could affect water level. If so, please demonstrate that the methodology used to calculate pool level is realistic or conservative.

34. The submittal lists minimum and maximum values for containment spray flow rates (Reference: Volume 3, Section 2.2.8, "ECCS and CCS Flow Rates"). Please state how these values are used in the evaluation. If flow rates other than the maximum are used, please explain how the appropriate flow rate was determined for each case.

35. The STPNOC submittal calculates an equivalent break size of 38.9 inches for a 27.5 inch-DEGB in Volume 3, Section 2.2.8. Please describe how the equivalent break size of 38.9 inches was calculated and why it was necessary to calculate this value.

36. The submittal states that the NPSHR for the ECCS and CSS pumps is 12 ft (Reference: Volume 3, Section 2.2.24, "Pump NPSH Margin," and Enclosure 6, Table 1). The proposed UFSAR revisions (pages 9 and 11 of Attachment 2 to Enclosure 3) state that the NPSHR for the pumps is between 16.1 and 16.5 ft. A previous STPNOC submittal dated December 11, 2008 (ADAMS Accession No. ML083520326), for response to Generic Letter (GL) 2004-02 stated that the NPSHR values for the LHSI, HHSI, and CS pumps are 16.5 ft., 16.1 ft., and 16.4 ft., respectively. Please provide the basis for the NPSHR values used in the current evaluation.

In-Vessel and Boric Acid Precipitation

37. The STPNOC submittal uses 7.5 grams per fuel assembly as the fiber acceptance limit for cold-leg breaks (References: Volume 1, Section 1.1, "Structured Information Process Flow," Step 18; Volume 1, Sections 1.2.10, "Boric Acid Precipitation," and 1.2.11, "In-Vessel Fiber Limits"; Volume 3, Assumption 11.b; Volume 3, Section 4.2, "Structured Information Process Flow," Step 18; Volume 3, Section 5.11.2, "Acceptance Criteria: Debris Loads"; and Volume 6.2, Items 5.a.13 and 5.a.15). The NRC staff stated in its SE on "Evaluation of Long-Term Cooling Considering Particulate, Fibrous, and WCAP-16793, Revision 2, "Chemical Debris in the Recirculation Fluid," October 2011 (ADAMS Accession No. ML13084A154), that the maximum amount of fiber that would be present in the limiting reactor design following a cold-leg break would be expected to be about 7.5 grams, if the hot-leg break fiber amount did not exceed 15 grams. The staff did not conclude that a fiber load of 7.5 grams was adequate to ensure that boric acid precipitation would not occur. The amount was projected as the potential maximum in the short term until industry completed a separate program on boric acid precipitation (BAP). In its evaluation, the staff considered that the plant calculation of the in-vessel debris load included the worst case debris load for the plant and that most plants would have much less than 7.5 grams of debris following a cold-leg break. Note that testing for the WCAP did show that the flow required to match decay heat boil off would reach the core following a cold-leg break with debris loads greater than 7.5 grams, but did not show that mixing credited to prevent BAP would not be affected. The limit of 7.5 grams per fuel assembly has not been technically justified as an acceptance criterion for BAP. Please provide the technical basis for assuming that 7.5 grams is an acceptable limit for a cold-leg break at STP when considering the potential for boric acid precipitation.

Debris Bypass

38. The submittal discusses the fraction of debris that is "sheddable" from a debris bed (Reference: Volume 3, Section 5.8, "Debris Penetration"). Please explain if V_n, (Fraction of debris that is "sheddable"), is a simple fraction or it is dependent on the amount of debris in the bed.

39. The submittal states that debris bypass or penetration testing was completed to support modeling of the bypass of debris past the STP strainer (Reference: Volume 6.2,

Item 5.a.16). Please provide additional details on how debris penetration testing for fiber was conducted. Specifically, please provide the following information:

a. Provide details on the characteristics of the fiber that was added to the test facility.

   i. How the fiber was prepared.

   ii. State What the percentages were of each fiber classification as described in NUREG/CR-6808, "Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance," February 2003 (ADAMS Accession No. ML030780733), Table 3-2 after the fiber was prepared.

   iii. How was it ensured that agglomeration of the fiber did not occur prior to addition to the test loop?

b. For tests that had more than one batch of fiber added to the test, please state what the timing was of each debris addition.

c. Please describe the design of the test facility.

d. Was the circulation of fluid within the tank turbulent? Did debris settle? If some debris did not reach the strainer, how was this accounted for?

e. How was it ensured that fiber did not bypass the filters during the test?

f. Was the design of the strainer and the design of the test facility (flow rate, etc.) prototypical with respect to the STP strainer?

Defense In Depth and Mitigative Measures

40. Volume 1, Section 2.1.1, "Defense-in-Depth," states that the concerns raised in GSI-191 have no bearing on containment integrity or on the release of radiation (page 18 of Volume 1). Volume 1, Appendix C, page C6 states that the independence of barriers is not degraded. The NRC staff notes that barrier independence is a function of multiple factors (e.g., plant operations, maintenance, environmental conditions) that are not necessarily linked directly to SSC design. Appendix C presents a similar argument with respect to maintaining a balance among core damage prevention, containment, and consequence mitigation. It is not clear to the NRC staff that a lack of design/equipment changes can be equated unconditionally with a balanced approach to prevention, containment, and mitigation. The presence of debris may impact the effectiveness of core damage prevention and containment simultaneously. Implementation of a deterministic solution would result in zero predicted failures of the fuel or containment as a result of debris, following an assumed failure of the RCS barrier. STP's risk-informed solution predicts that some fuel or containment failures may occur. This implies that the independence of barriers may be degraded under the risk-informed approach. Consistent with RG 1.174, please provide discussion on defense in depth contained in

Appendix C, using quantitative assessments to the extent practicable (to supplement the existing qualitative assessment), to demonstrate that the elements of defense in depth described by RG 1.174 are met. Where appropriate, provide a comparison between the hypothetical "clean plant" and the as-built, as-operated plant. Consistent with the RG, please also include an evaluation of the proposed change on affected equipment functionality, reliability, and availability.

41. Volume 1, Appendix C, Section C.5.4, lists mitigative measures that can be taken if the strainer becomes blocked. It is not clear how the mitigative measures identified to address strainer blockage are implemented at STP (note that these actions are also credited for prevention of inadequate core flow). Please explain the following to explain how the mitigative measures are capable of providing alternate flow to the required equipment.

    a. The mitigative actions identified to reduce flow through the strainers appear to actually be designed to conserve RWST volume. These measures may delay the initiation of recirculation, but except for securing CSS pumps will not reduce flow through the strainer. Please state at what point in the recovery these actions are performed. If not performed immediately, will the RWST inventory be conserved? If the reductions of flow through the strainer do not occur until after strainer blockage is evident, please state if these actions are effective.

    b. Please state if STP has implemented operating procedures to secure the third train of ECCS/CSS if all three are initiated following a LOCA.

    c. Please clarify if STP implemented operating procedures or other guidance to backwash the strainers, if necessary. If so, please provide details on the procedural controls for this action.

    d. Please state when the RWST refill is started and how long it takes to refill RWST to the point where injection from the tank is viable. Please note that if the tank is not ready for injection when blockage occurs, this action may not be effective. The NRC staff notes that the STPNOC submittal states that most strainer blockage events occur within the first 24 hours of the LOCA recovery.

42. Volume 1, Appendix C, Section C.5.8, "Mitigation of Inadequate Reactor Core Flow," lists mitigative measures that can be taken if the flow to the core is not adequate to ensure core cooling. It is not clear to the NRC staff how the mitigative measures for inadequate reactor core flow will be effective. Most of the actions attempt to inject coolant through the flowpath that has already been identified as potentially blocked. Other actions do not appear to be effective if the core inlet is blocked. Please provide additional information showing that the mitigative measures are capable of providing coolant to the core.

Technical Specifications Branch (STSB)

1. The proposed LAR provides assumptions of partial flow reduction of certain ECCS equipment during DBA events. Please explain how the proposed LAR ECCS flow

assumptions are affected when the Configuration Risk Management Program (CRMP) Technical Specification (TS) Completion Times are applied to TS inoperable ECCS SSCs. Include in your discussion, how the related ECCS equipment PRA functionality (as defined in NEI 06-09, Revision 0, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," November 2006 (ADAMS Accession No. ML12286A322)) is included in the analysis and how the analysis assumptions are programmatically included the CRMP.

2. The STP CRMP could allow continued power operation with a loss of a TS safety function for up to 30 days. Please explain how a TS loss of function, but PRA functional (as defined by NEI 06-09, Revision 0) ECCS SSC is addressed in the analysis.

3. Please explain how the assumptions and analysis for fibrous material impact on ECCS flow are verified and maintained programmatically (i.e., how and at what frequency are any physical or material changes to the analyzed impact zones evaluated and what physical or material changes would initiate a reevaluation of the affected impact zone).

If you have any questions, please contact me at 301-415-3016 or via e-mail at Balwant.Singal@nrc.gov.

Sincerely,

/RA/

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosure:
Request for Additional Information

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ASallman, NRR/DSS/SCVB
SSmith, NRR/DSS/SSIB
EGeiger, NRR/DSS/SSIB
LWard, NRR/DSS/SNPB
AGuzzetta, NRR/DSS/SRXB
JDozier, NRR/DRA/ARCB
JTsao, NRR/DE/ESGB
PKlein, NRR/DE/ESGB
MYoder, NRR/DE/ESGB
CFong, NRR/DSS/APLA
SLaur, NRR/DSS/APLA
GWaig, NRR/DSS/STSB
BLehman, NRR/DE/EMCB

ADAMS Accession No. ML14087A075 *via Memos dated 11/22/13 and 2/6/14

OFFICE	NRR/DORL/LPL4-1/PM	NRR/DORL/LPL4-1/LA	NRR/DSS/SCVB/BC	NRR/DSS/SSIB/BC (A)
NAME	BSingal	JBurkhardt	RDennig	JStang
DATE	3/28/14	4/4/14	3/31/14	3/28/14
OFFICE	NRR/DSS/SNPB/BC (A)	NRR/DSS/SRXB/BC	NRR/DRA/ARCB/BC	NRR/DE/ESGB/BC (A)
NAME	JDean	CJackson	UShoop	EMurphy
DATE	3/31/14	3/31/14	3/31/14	3/31/14
OFFICE	NRR/DE/EPNB/BC*	NRR/DRA/APLA/BC	NRR/DSS/STSB/BC	NRR/DE/EMCB/BC*
NAME	TLupold	HHamzehee	RElliott	AMcMurtray
DATE	2/6/14	3/31/14	3/31/14	11/22/13
OFFICE	NRR/DORL/LPL4-1/BC	NRR/DORL/LPL4-1/PM	--	--
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