Nuclear Operating Company
South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

June 19, 2013
NOC-AE-13002986
10 CFR 50.12
10 CFR 50.90

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498 and STN 50-499
Revised STP Pilot Submittal and Requests for Exemptions and License Amendment for a Risk-Informed Approach to Resolving Generic Safety Issue (GSI)-191
(TAC Nos. MF0613 and MF0614)

References:

1. Letter, J. W. Crenshaw, STPNOC, to NRC Document Control Desk, "Status of the South Texas Project Risk-Informed (RI) Approach to Resolve Generic Safety Issue (GSI)-191," December 14, 2011, NOC-AE-11002775 (ML11354A386)
2. Letter, D. W. Rencurrel, STPNOC, to NRC Document Control Desk, "GSI-191 Resolution Path Schedule and Commitment Changes," June 4, 2012, NOC-AE-12002858 (ML13025A360)
3. Commission SECY Paper, "Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance," July 9, 2012, SECY-12-0093 (ML121320270)
4. Letter, J. E. Dyer, NRC, to A. W. Harrison, STPNOC, Response to letter requesting an exemption of fees, April 15, 2011, AE-NOC-11002079 (ML111050388)
5. Letter, John W. Crenshaw, STPNOC to NRC Document Control Desk, "STP Pilot Submittal and Request for Exemption for a Risk-Informed Approach to Resolve Generic Safety Issue (GSI)-191," January 31, 2013, NOC-AE-13002954 (ML13043A013)
6. Letter, Balwant K. Singal, NRC, Dennis L. Koehl, STPNOC, "South Texas Project, Units 1 and 2 – Supplemental Information Needed for Acceptance of Requested Licensing Action Re: Request for Exemption for a Risk-Informed Approach to Resolve Generic Safety Issue 191 (TAC Nos. MF0613 and MF0614)," April 1, 2013, AE-NOC-13002417 (ML13066A519)

Pursuant to 10 CFR 50.12, STP Nuclear Operating Company (STPNOC) hereby requests exemptions from certain requirements of 10 CFR 50.46(b)(5), "Long-term cooling", and Appendix A to 10 CFR Part 50 General Design Criteria (GDC) 35, "Emergency Core Cooling", GDC 38, "Containment Heat Removal", and GDC 41, "Containment Atmosphere Cleanup". In accordance with the provisions of 10 CFR 50.90, STPNOC hereby requests an amendment to South Texas Project (STP) Operating Licenses NPF-76 and NPF-80, to revise the STP Units 1 and 2 licensing basis. Approval for the License Amendment Request (LAR) is expected to be contingent upon approval of the exemption requests. This revised submittal supersedes the

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previous submittal in its entirety and includes the additional information identified by the NRC as required for the staff to complete its acceptance review.

By References 1 and 2, STPNOC submitted to the NRC the preliminary results showing that the risks, Core Damage Frequency (CDF) and Large Early Release Frequency (LERF), associated with Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance," for STP Units 1 and 2 are far less than the threshold for Region III, "Very Small Changes," of Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." STPNOC committed STP Units 1 and 2 as the pilot plants for a risk-informed approach, including intent to seek exemption from certain requirements of 10 CFR 50.46 for emergency core cooling systems (ECCS) acceptance criteria.

NRC SECY-12-0093 (Reference 3) discussed closure options for GSI-191 including the STP risk-informed approach and schedule. The STP pilot effort is expected to result in substantial benefit to both the NRC and industry in support of the development and implementation of a risk-informed approach to resolving GSI-191. By Reference 4, the NRC staff concluded that the review of STPNOC's proposed exemption and license amendment submittal with STP Units 1 and 2 as pilot plants warrants a fee exemption in the public interest.

By Reference 5, STPNOC provided a pilot submittal that proposed a risk-informed approach to resolving GSI-191. By Reference 6, the NRC notified STPNOC that the previous application did not provide technical information in sufficient detail, and did not provide adequate discussion of or justification for the requested exemptions. This revised submittal replaces in its entirety the previous application (Reference 5), and includes an accompanying license amendment request (LAR) that is dependent upon approval of the exemption requests. The attachment to this letter provides information and references responsive to each item identified in Reference 6.

The requested licensing actions are for approval of a risk-informed approach for resolving GSI-191 for STP Units 1 and 2 as the pilot plants for other licensees pursuing a similar approach. The results of the reviews will also support closure of Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," for STP Units 1 and 2.

STPNOC seeks NRC approval based on a determination that the STP risk-informed approach and the risk associated with the postulated failure mechanisms due to GSI-191 concerns meets the guidance, key principles for risk-informed decision-making, and the acceptance guidelines in RG 1.174.

The STP piloted risk-informed approach to resolving GSI-191 applies the STP Probabilistic Risk Assessment (PRA) model to quantify the risk associated with GSI-191 concerns by calculating the difference in risk for two cases:

- The actual plant configuration for STP Units 1 and 2, with failures due to GSI-191 concerns, and
- The same plant configuration for STP Units 1 and 2, except for the assumption that there are no failures due to GSI-191 concerns.

The risk associated with GSI-191 concerns includes the effects on long-term cooling due to debris accumulation on Emergency Core Cooling System (ECCS) and Containment Spray

System (CSS) sump strainers in recirculation mode, as well as core flow blockage due to in-vessel effects, following loss of coolant accidents (LOCAs). A full spectrum of postulated LOCAs is analyzed, including double-ended guillotine breaks (DEGBs) for all pipe sizes up to the largest pipe in the reactor coolant system. To inform the PRA with risk insights, the physical processes are modeled as realistically as possible, using results from industry and plant-specific testing, and applying conservatism, where appropriate. The changes to CDF and LERF associated with GSI-191 concerns are then compared to RG 1.174 acceptance guidelines.

Enclosure 1 provides the generic methodology for the proposed risk-informed approach to resolving GSI-191, consistent with RG 1.174 guidance. This enclosure describes the required inputs to the PRA model, the basic structure for appropriately modeling the inputs, and performance criteria used to calculate the risk.

Enclosure 2 provides an introduction and background to the proposed exemptions from certain regulatory requirements in accordance with the provisions of § 50.12. The requests for exemptions address regulatory requirements that concern the ECCS and CSS functions for emergency core cooling, containment heat removal, and containment atmosphere cleanup following postulated LOCAs:

Enclosure 2-1, 10 CFR 50.46(b)(5), *Long-term cooling*

Enclosure 2-2, GDC 35 – *Emergency core cooling*

Enclosure 2-3, GDC 38 – *Containment heat removal*

Enclosure 2-4, GDC 41 – *Containment atmosphere cleanup*

Approval for the requests for exemptions is based on a risk-informed approach demonstrating that the calculated risk associated with GSI-191 concerns for STP Units 1 and 2 is far less than the threshold for Region III, "Very Small Changes," of RG 1.174 acceptance guidelines.

Enclosure 3 provides the License Amendment Request, pursuant to 10 CFR 50.90, for approval of the proposed changes to the STP Units 1 and 2 licensing basis including page markups for the Updated Final Safety Analysis Report (UFSAR). The LAR is contingent upon approval of the requested exemptions. The LAR includes technical and regulatory evaluations for the proposed change, a no significant hazards consideration determination pursuant to § 50.92, and an environmental considerations review. The Plant Operations Review Committee has approved the proposed change. In accordance with 10 CFR 50.91(b), STPNOC has notified the State of Texas by transmitting a copy of this letter and enclosure to the State of Texas Official. Changes to the STP UFSAR are to be implemented pursuant to NRC approval of LAR.

Enclosure 4 provides an introduction and overview of the supporting documentation. Enclosures 4-1, 4-2 and 4-3 support the application by providing summary level and detailed descriptions of the supporting engineering analysis and PRA information. Enclosure 4-1 follows the structure, content and documentation requirements of RG 1.174, and provides references to other supporting documentation. Enclosure 4-1 also provides the details of how the STP piloted approach meets the guidance and conforms to the risk-informed principles in RG 1.174:

- Meets the current regulations except as provided in the enclosed requests for exemptions from certain requirements.
- Is consistent with a defense-in-depth philosophy.
- Maintains sufficient safety margins.

- Shows that for STP Units 1 and 2 the change in risk associated with GSI-191 concerns is very small, approximately 1.1E-8/yr change in CDF and 8.6E-12/yr change in LERF.
- Includes provisions for monitoring the impact of the change.

To support the completion of work and resolution schedule for closure of GSI-191 as described in Reference 3, STPNOC requests approval of the proposed exemption requests and license amendment request by December 2014.

Upon approval of the requested licensing actions, changes to the STP UFSAR will be made as shown in Attachment 2 to Enclosure 3. The licensing commitment for updating the UFSAR is provided as Attachment 1 to Enclosure 3. A 90-day implementation period is requested to provide time to revise the applicable STP licensing documents. There are no other commitments in this letter.

If there are questions regarding this submittal, please contact Jamie Paul at 361-972-7344, or me at 361-972-7867.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 6/19/2013

D. W. Rencurrel
Senior Vice President, Operations

ccc

Attachment: Response to NRC Supplemental Information Items

Enclosures:

1. STP Piloted Risk-Informed Approach to Closure for GSI-191
2. Introduction to Requests for Exemptions for STP Piloted Risk-Informed Approach to Closure for GSI-191

2-1 Request for Exemption from 10 CFR 50.46(b)(5)
2-2 Request for Exemption from General Design Criterion 35
2-3 Request for Exemption from General Design Criterion 38
2-4 Request for Exemption from General Design Criterion 41

3. License Amendment Request for STP Piloted Risk-informed Approach to Closure for GSI-191
   - Attachment 1: List of Commitments
   - Attachment 2: STPEGS UFSAR Page Markups
   - Attachment 3: Technical Specifications Bases Page Markups (Information Only)
4. Risk-Informed Closure of GSI-191 Supporting Engineering Analysis and PRA – Introduction and Overview

4-1 Risk-Informed Closure of GSI-191, Volume 1, Project Summary
4-2 Risk-Informed Closure of GSI-191, Volume 2, Probabilistic Risk Analysis
4-3 Risk-Informed Closure of GSI-191, Volume 3, Engineering (CASA Grande) Analysis

cc: w/o attachment and enclosures except*

(paper copy)

Regional Administrator, Region IV
U. S. Nuclear Regulatory Commission
1600 East Lamar Boulevard
Arlington, TX 76011-4511

*Balwant K. Singal
Senior Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North (MS 8 B1)
11555 Rockville Pike
Rockville, MD 20852

NRC Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 289, Mail Code: MN116
Wadsworth, TX 77483

C. M. Canady
City of Austin
Electric Utility Department
721 Barton Springs Road
Austin, TX 78704

(electronic copy)

Steven P. Frantz, Esquire
*A. H. Gutterman, Esquire
Morgan, Lewis & Bockius LLP

Balwant K. Singal
Stewart Bailey
Jack Davis
Robert Elliott
Michael Markley
John Stang
U. S. Nuclear Regulatory Commission

John Ragan
Chris O'Hara
Jim von Suskil
NRG South Texas LP

Kevin Pollo
Richard Peña
City Public Service

Peter Nemeth
Crain Caton & James, P.C.

C. Mele
City of Austin

Richard A. Ratliff
Robert Free
Texas Department of State Health Services

ATTACHMENT

Response to NRC

Supplemental Information Items

Response to NRC Supplemental Information Items

References:

1. Letter, John W. Crenshaw to NRC Document Control Desk, "STP Pilot Submittal and Request for Exemption for a Risk-Informed Approach to Resolve Generic Safety Issue (GSI)-191," January 31, 2013, NOC-AE-13002954 (ML13043A013)

2. Letter, Balwant K. Singal, NRC, Dennis L. Koehl, STPNOC, "South Texas Project, Units 1 and 2 – Supplemental Information Needed for Acceptance of Requested Licensing Action Re: Request for Exemption for a Risk-Informed Approach to Resolve Generic Safety Issue 191 (TAC Nos. MF0613 and MF0614)," April 1, 2013, AE-NOC-13002417 (ML13066A519)

In Reference 1, STP Nuclear Operating Company (STPNOC) requested an exemption from certain regulations affected by the risk-informed approach to resolution of GSI-191. In Reference 2, the NRC staff identified supplemental information needed for completion of the acceptance review. Reference 1 is superseded in its entirety by this submittal.

In order to facilitate the staff completing its acceptance review of this submittal, responses to each of the supplemental information items identified in Reference 2 are provided. These responses also describe where and how supplemental information requested by the staff is addressed elsewhere in this submittal. Italicized text is as shown in Reference 2.

The NRC staff also concludes that the application does not provide adequate discussion of or justification for the requested exemptions. The licensee submittal requests exemption from Title 10 of the Code of Federal Regulations (10 CFR), Sections 50.46 and 50.67 and General Design Criterion 35, 38, 41, and 19. Each of these regulations require a justification for exemption. Please provide the following information in support of the exemption request for the NRC staff to start its review:

1. *For each exemption request submitted under 10 CFR 50.12, the application should include a narrative as to why the licensee believes that the special circumstances provided in 10 CFR 50.12(a)(2) is present. The licensee in its application has stated that 10 CFR 50.10(a)(2)(ii) and (iii) apply. There appears to be a typographical error and the NRC staff believes licensee meant to invoke 10 CFR 50.12(a)(2)(ii) and (iii). Please confirm this and provide adequate technical basis in support of applicability of 10 CFR 50.12(a)(2)(ii) and (iii). Also, please describe in detail how the special circumstances address 10 CFR 50.12(a).*

   RESPONSE:

   Separate requests for exemption are provided to address the following requirements:

   Enclosure 2-1, 10 CFR 50.46(b)(5), *Long-term cooling*

   Enclosure 2-2, General Design Criterion 35 – *Emergency core cooling*

   Enclosure 2-3, General Design Criterion 38 – *Containment heat removal*

   Enclosure 2-4, General Design Criterion 41 – *Containment atmosphere cleanup*

   Each exemption request includes a discussion as to why special circumstances provided in 10 CFR 50.12(a)(2) apply, and specifically for 10 CFR 50.12(a)(2)(ii) and 10 CFR 50.12(a)(2)(iii). Enclosure 2 provides a background and overview for the four exemption requests. STPNOC has determined that exemptions to 10 CFR 50.67 and General Design Criterion 19 are not required and that basis is discussed in the Enclosures identified above.

2. *The application describes a departure from the method of evaluation described in the Updated Final Safety Analysis Report (UFSAR) used in establishing the design bases in the plant's safety analysis, as defined in 10 CFR 50.59(a)(2) and proposes several draft modifications to the UFSAR. In accordance with 10 CFR 50.59(c)(2)(viii), these modifications would appear to be changes in the design and licensing basis and would require a license amendment in accordance 10 CFR 50.90. Please explain why an amendment is not proposed to accompany this exemption, with the associated draft no significant hazards consideration. The licensee should clearly state the scope and nature of the change to the design and licensing basis.*

RESPONSE:

In accordance with 10 CFR 50.59(c)(2)(viii), the proposed changes to the UFSAR constitute a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the plant's safety analyses, and has been evaluated to require NRC approval. On this basis, a license amendment request (LAR) pursuant to 10 CFR 50.90 is included in Enclosure 3 with the proposed changes to the licensing basis for NRC review and approval. The LAR includes a no significant hazards consideration determination pursuant to 10 CFR 50.92(c).

The proposed risk-informed method of evaluation described in the LAR has been determined to require exemptions from certain regulatory requirements identified in Enclosure 2. Therefore, requests for specific exemptions pursuant to 10 CFR 50.12 are provided in Enclosures 2-1 through 2-4 to support the LAR and RG 1.174 submittal.

As a risk-informed approach to resolving GSI-191 with exemption requests in support of a RG 1.174 application, the STP method is intended to be consistent with previous NRC staff comments in the NRC staff safety evaluation on NEI 04-07 regarding Section 6, Alternate Evaluation, dated December 6, 2004 (ML043280007), and Enclosure 3, "Risk-Informed Approach to Address Generic Safety Issue-191, South Texas Project," to SECY-12-0093, "Closure Options for Generic Safety Issue – 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance," dated July 9, 2012 (ML121310659).

3. *To process the proposed exemption, the NRC staff will need to conduct an environmental review. Please provide the description that will address the special circumstances supporting this review in accordance with 10 CFR 51.41 and 10 CFR 51.45.*

RESPONSE:

In accordance with 10 CFR 51.41 and 10 CFR 51.21, for each exemption request provided in Enclosures 2-1 through 2-4, environmental considerations have been included to support the NRC staff's environmental review. Based on the guidance in RG 1.174 being met, justification is also provided for the actions to qualify for 10 CFR 51.22(c)(9) categorical exclusion. Therefore, an environmental report pursuant to 10 CFR 51.45 is not required.

4. *Please describe how the proposed change will affect the Technical Specifications (TSs). Please indicate whether changes are needed to the operability requirements for the affected systems and any changes to the existing TS Action Statements that may be needed.*

RESPONSE:

An evaluation of how the proposed change will affect the TSs is provided with the LAR in Section 4.1.3 (Enclosure 3). As discussed in more detail in the LAR, the evaluation determines that no changes to operability requirements for affected systems and no changes to the existing TS Action Statements are required. Changes to the TS Bases that conform to the proposed UFSAR changes are included in Attachment 3 to Enclosure 3 for information only.

5. *The basis for the proposed change is that the residual risk from the remaining GSI-191 issues (e.g., those not already addressed in a deterministic manner) satisfies the criteria in 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). However, the application does not appear to provide sufficient detail for the NRC staff to determine whether the criteria of RG 1.174 have been met. Please describe in detail how the principles of RG 1.174 criteria regarding safety margin, defense-in-depth (DID), and change in risk are met. In particular, please include the following:*

   a. *Regarding the technical evaluation that supports the risk metrics, the Project Summary (Enclosure 4 to the application) describes numerous areas where the technical evaluation deviates from the approved guidance for addressing GSI-191. However, the application provides little or no information on how the issues were addressed. Please provide a discussion in sufficient detail to permit NRC staff review of the methods, bases, assumptions, acceptance criteria, and results. If test results are used to develop probability distributions, please describe how these distributions were determined and used in the overall risk evaluation. Please also provide the basis for the acceptance criteria chosen. The NRC staff requires additional information in the following areas:*

      1) *Failure timing, failure amounts, and debris characteristics of unqualified coatings.*
      2) *Capture of small and large pieces of debris on gratings and obstructions.*
      3) *Washdown transport holdups.*
      4) *Non-uniform debris distribution at the onset of recirculation.*
      5) *Time dependent transport.*
      6) *Chemical effects corrosion and dissolution models.*
      7) *Basis for excluding any plant materials from chemical testing.*
      8) *Chemical precipitation models – amount, type, head loss effect.*
      9) *Disposition of chemical effects Phenomena Identification and Ranking Table open items.*
      10) *Head loss model.*
      11) *Chemical effects on head loss (bump-up factor) model.*
      12) *Fiber bypass amounts and amounts reaching the core for various scenarios.*
      13) *Fiber limits for in-vessel evaluations.*
      14) *Thermal-hydraulic analysis for in-vessel evaluations.*
      15) *Boric acid precipitation evaluations.*
      16) *Methodology for determination and implementation of physical effects probability distributions.*

   b. *Regarding DID, please address how DID is maintained to account for scenarios that are predicted to lead to failure. One method of maintaining DID is to demonstrate that the operators can detect and mitigate inadequate flow through the recirculation strainer and inadequate core cooling. Please describe the supporting evaluations that demonstrate DID actions will be effective.*

c. *Please provide supporting evaluations that demonstrate that the barriers for the release of radioactivity will be maintained with sufficient safety margin.*

d. *Please provide sufficient detail necessary to assess the treatment of uncertainty. While several known categories of uncertainty are identified (zone of influence, chemical effects, debris transport, etc.), the mechanistic models and associated parametric factors used in the analysis are not identified, nor are probability density functions for the parameters provided (Enclosure 4, Section 2.5). Please provide this information.*

RESPONSE:

The table below lists sources of information for the responses to Items 5.a through 5.d provided in this Attachment. Volume 3 is included in the submittal as Enclosure 4-3, and the other specific calculations and reports are available for audit. References identified in the responses to Item 5 (indicated in parentheses) are listed at the end of this Attachment.

Table 2.5.1 – References for Responses to Items 5.a through 5.d

5.a Information Item	Reference Sources of Information
1) Failure timing, failure amounts, and debris characteristics of unqualified coatings	• Volume 3 Section 2.2.11 (Enclosure 4-3) • ALION-CAL-STP-8511-06 "STP Unqualified Coatings Debris Generation", Revision 2, November 26, 2012
2) Capture of small and large pieces of debris on gratings and obstructions	• Volume 3 Section 2.2.21 (Enclosure 4-3) • ALION-CAL-STP-8511-08, "Risk-Informed GSI-191 Debris Transport Calculation", Revision 2, January 21, 2013
3) Washdown transport holdups	• Volume 3 Section 2.2.22 (Enclosure 4-3) • ALION-CAL-STP-8511-08, "Risk-Informed GSI-191 Debris Transport Calculation", Revision 2, January 21, 2013
4) Non-uniform debris distribution at the onset of recirculation	• ALION-CAL-STP-8511-08, "Risk-Informed GSI-191 Debris Transport Calculation", Revision 2, January 21, 2013
5) Time dependent transport	• Volume 3 Section 5.6.8 (Enclosure 4-3) • ALION-CAL-STP-8511-08, "Risk-Informed GSI-191 Debris Transport Calculation", Revision 2, January 21, 2013
6) Chemical effects corrosion and dissolution models	• Volume 3 Section 5.5 (Enclosure 4-3) • CHLE-016, "Calculated Material Release", Revision 1, January 10, 2013 • Texas A&M University Department of Nuclear Engineering, "Sump Temperature Sensitivity Analysis", Revision 2.0, January 2013 • CHLE-014, "T2 LBLOCA Test Report", Revision 1, January 12, 2013

5.a Information Item	Reference Sources of Information
7) Basis for excluding any plant materials from chemical testing	• CHLE-006, "STP Materials Calculation", Revision 1, August 15, 2012. • ALION-CAL-STPEGS-2916-002, "GSI 191 Containment Recirculation Sump Evaluation: Debris Generation", Revision 3, October 20, 2008
8) Chemical precipitation models – amount, type, head loss effect	• Volume 3 Sections 5.5 and 5.7.3 (Enclosure 4-3) • CHLE-016, "Calculated Material Release", Revision 1, January 10, 2013
9) Disposition of chemical effects Phenomena Identification and Ranking Table (PIRT) open items	• Volume 3 Sections 5.6, 5.7.3, 5.7.4 and 5.12 (Enclosure 4-3) • CHLE-014, "T2 LBLOCA Test Report", Revision 1, January 12, 2013 • CHLE-012, "T1 MBLOCA Test Report", Revision 3, January 9, 2013 • CHLE-005, "Determination of the Initial Pool Chemistry for the CHLE Test", Revision 1, August 13, 2012 • CHLE-018, "Bench-Scale Test Results of Effect of pH and Temperature on Aluminum Corrosion and Silicon Dissolution", Revision 0, Draft • CHLE-011, "Test 2, Medium Break LOCA Tank Test Parameter Summary", Revision 1, October 30, 2012 • CHLE-013: "T2: Large Break LOCA Tank Test Parameter Summary", Revision 2, January 23, 2013 • ALION-CAL-STP-8511-07, "STP Crud Debris Generation", Revision 0, November 12, 2012 • ALION-SUM-WEST-2916-01, "CAD Model Summary: South Texas Reactor Building CAD Model for Use in GSI-191 Analyses", Revision 3, November 27, 2012 • ALION-CAL-STP-008511-02, "STP Cold Volume Analysis", Revision 0, May 17, 2012
10) Head loss model	• Volume 3 Section 5.7.2 (Enclosure 4-3) • ALION-REP-STP-8511-02, "South Texas Vertical Loop Head Loss Testing Report", Revision 1, January 24, 2013
11) Chemical effects on head loss (bump-up factor) model	• Volume 3 Section 5.7.3 (Enclosure 4-3) • CHLE-012, "T1 MBLOCA Test Report", Revision 3, January 9, 2013 • CHLE-014, "T2 LBLOCA Test Report", Revision 1, January 12, 2013
12) Fiber bypass amounts and amounts reaching the core for various scenarios	• Volume 3 Sections 5.9 and 5.11 (Enclosure 4-3) • University of Texas at Austin, "Filtration as a Function of Debris Mass on the Strainer: Fitting a Parametric Physics Based Model", January 24, 2013 • ALION-REP-STP-8511-03, "South Texas Penetration Test Report", Revision 1, January 24, 2013

5.a Information Item	Reference Sources of Information
13) Fiber limits for in-vessel evaluations	• Volume 3 Sections 5.11 and 5.12 (Enclosure 4-3) • Texas A&M University, Department of Nuclear Engineering, "Core Blockage Thermal-Hydraulic Analysis", Revision 2.1, January 2013
14) Thermal-hydraulic analysis for in-vessel evaluations	• Volume 3 Section 5.11.2 (Enclosure 4-3) • Texas A&M University, Department of Nuclear Engineering, "Core Blockage Thermal-Hydraulic Analysis", Revision 2.1, January 2013 • Texas A&M University, Department of Nuclear Engineering, "RELAP5 Model Input Deck Certification", Revision 3.0, August 1, 2011
15) Boric acid precipitation evaluations	• Volume 3 Section 5.12 (Enclosure 4-3)
16) Methodology for determination and implementation of physical effects probability distributions	• Implementation is described in Volume 3 (Enclosure 4-3) Section 4 generically, and more specifically in Section 2 and other areas of the report.
5.b Defense-in-depth	• Volume 1 Section 2.1 and Appendix C (Enclosure 4-1)
5.c Barriers for release of reactivity safety margin	• Volume 1 Section 2.2 and Appendix C (Enclosure 4-1)
5.d Treatment of uncertainty	• Volume 1 Sections 5.2 through 5.5 (Enclosure 4-1)

Item 5.a: Technical Evaluation

The responses to the request for supplemental information on the 16 specific technical areas are provided below.

Item 5.a.1: Unqualified Coatings

Method: The basic methodology used for the STP unqualified coatings debris generation calculation is shown below:

1. Each component substrate with an unqualified coating was investigated for the coating type, substrate location, and total mass of the coating.
2. The failure fraction of each coating type was analyzed through a survey of applicable literature and test data. The probability of failure fraction was determined for each of the following coatings: IOZ, epoxy, alkyd, and baked enamel. The test data that was used includes testing performed by EPRI (3), GE (4), Comanche Peak (5), and Alion (6).
3. The failure timing of coatings was evaluated and the probability of the coating failing prior to containment spray termination was estimated based on test data. The test data that was used includes testing performed by EPRI (3), GE (4), Comanche Peak (5), and Alion (6).

4. The debris characteristics for each of the unqualified coatings were analyzed through a survey of previous literature. The type and size of debris was determined for IOZ, epoxy, alkyd, baked enamel, and intumescent coatings.

Basis: The following discussion provides a detailed description of how the methodology referred to above was used to develop the unqualified coatings input parameters (7).

Failure Fraction Analysis

Probability distributions were developed for the failure fractions of unqualified epoxy, IOZ, and alkyd coatings. The failure timing analysis was extrapolated from the 7 days of data to accurately represent the full 30-day mission time. As a consequence of this extrapolation, a 152.5% increase of probability statistics was introduced to the failure timing relative frequency analysis (See Failure Timing Analysis). This increase in the failure timing analysis will affect the failure fraction probability. To account for this, the probability of 100% failure for each of the coating types is increased by 152.5%, and the rest of the distribution is fit to this correction with an attempt to keep prior inflection points. This is a significant conservatism because this skews the distribution towards 100% failure, despite the fact that the data shows this is unlikely. For each of the unqualified coatings, the probability distribution based on the data from the Carboline and EPRI testing is provided in contrast to the corrected probability distribution that will be used in the risk-informed analysis.

The statistics for the failure fraction of unqualified epoxy coatings, based on the EPRI and Carboline analysis, is summarized in the following table:

Table 2.5.2 – Epoxy Failure Fraction Probability Statistics

% Failure	Probability
0	0.0088
1	0.0088
5	0.0352
10	0.0088
20	0.0088
100	0.0088

The following figure illustrates the probability distribution of the failure fraction for the epoxy coatings based on the available data:

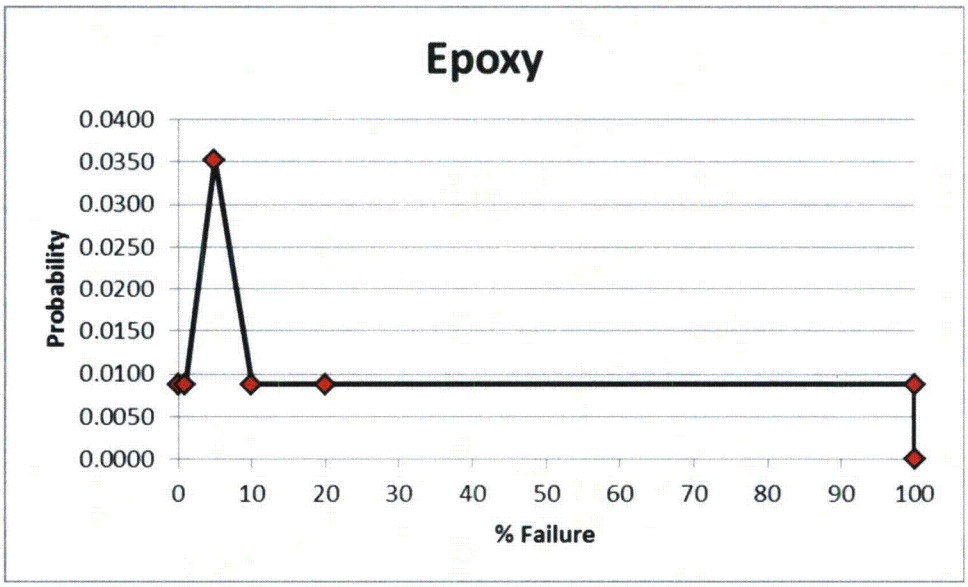

Figure 2.5.1 – Raw Epoxy Failure Fraction Probability Distribution

The data supports the probability of 100% failure as 0.0088. Applying the 152.5% increase to correct for the failure timing extrapolation yields the probability of 100% failure as 0.0222. The rest of the probability distribution is fit to the 100% failure probability. The area under the probability distribution must be equal to 100%: this yields 0% probability of any failure fraction below 10.1%. The following table illustrates the corrected probability statistics that accounts for the failure timing extrapolation:

Table 2.5.3 – Corrected Epoxy Failure Fraction Probability Statistics

% Failure	Probability
0.0	0.0000
10.1	0.0000
100.0	0.0222

These statistics yield the following corrected probability distribution for the failure fraction of unqualified epoxy coatings:

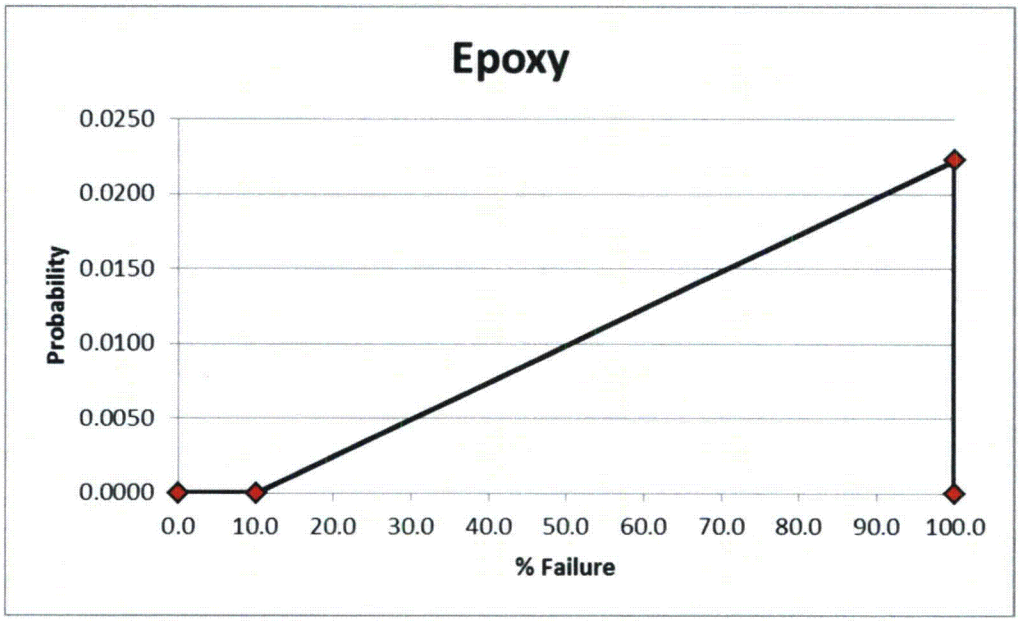

Figure 2.5.2 – Epoxy failure fraction probability distribution

The probability statistics for the failure fraction of unqualified alkyd coatings based on the test data supplied by EPRI and Carboline is summarized in the following table:

Table 2.5.4 – Alkyd Failure Fraction Probability Statistics

% Failure	Probability
0	0.0000
1	0.0127
5	0.0317
20	0.0063
50	0.0127
55	0.0063
80	0.0063
95	0.0063
100	0.0063

The following figure illustrates the probability distribution of the failure fraction for the alkyd coatings based on the available data:

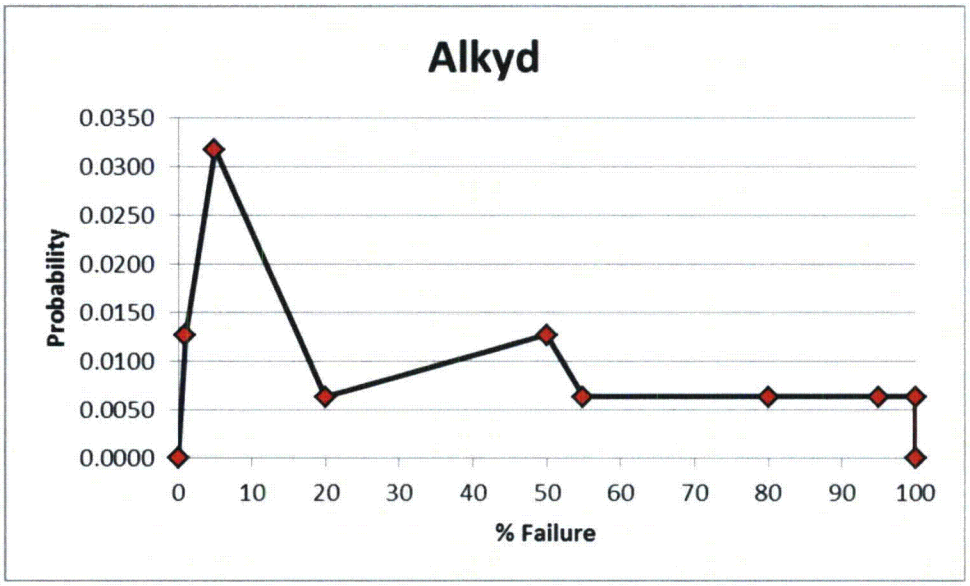

Figure 2.5.3 – Raw Alkyd Failure Probability Distribution

This probability distribution was formulated with the current data available for the failure fraction of unqualified alkyd coatings. However, the two peaks in the distribution are not likely to occur in the natural failure of alkyd coatings. Therefore, the probability distribution was altered to provide a more reasonable distribution (without the two peaks). This was done by keeping the proportional probability between the 5%, 50%, and 100% failure data points, yielding the following probability statistics:

Table 2.5.5 – Altered Alkyd Probability Statistics

% Failure	Probability
0	0
5	0.02054
50	0.008229
100	0.004082

This yields the following probability distribution for the unqualified Alkyd coatings:

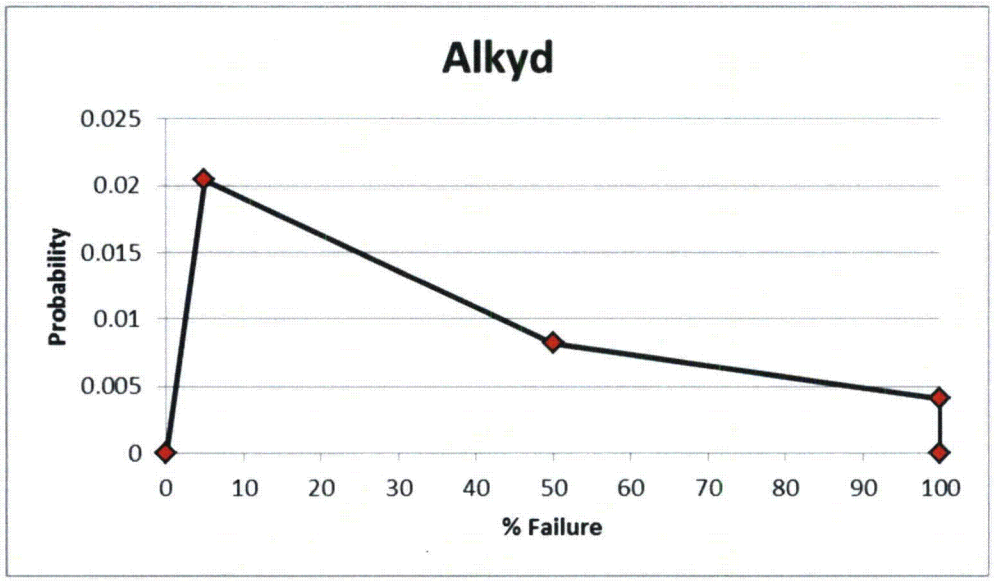

Figure 2.5.4 – Altered Alkyd Failure Fraction Probability Distribution

The following table illustrates the corrected probability statistics that accounts for the failure timing extrapolation:

Table 2.5.6 – Corrected Alkyd Failure Fraction Probability Statistics

% Failure	Probability
0	0.0000
5	0.0102
100	0.0103

These statistics yield the following corrected probability distribution for the failure fraction of unqualified alkyd coatings:

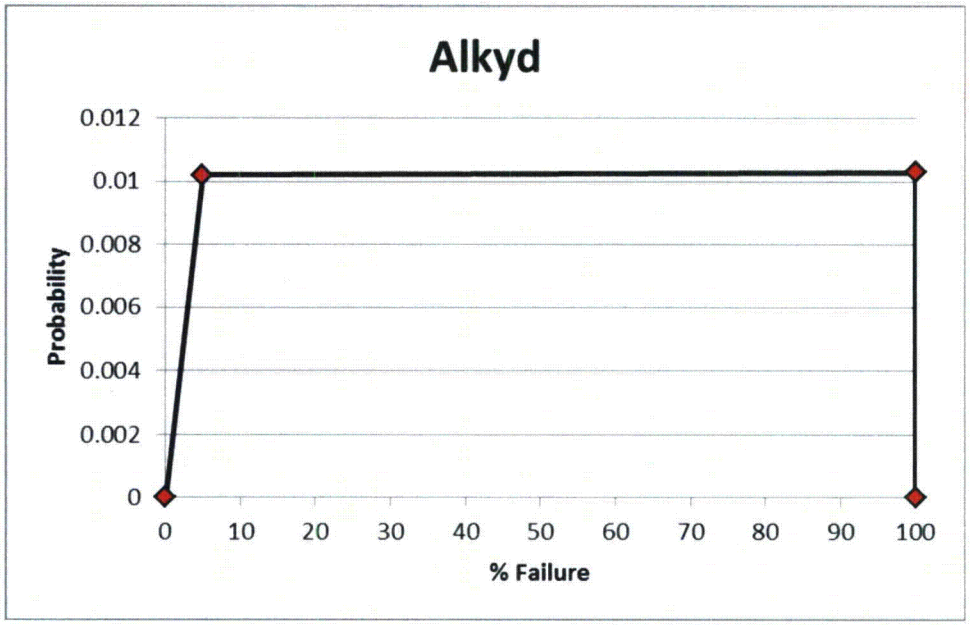

Figure 2.5.5 – Alkyd and baked enamel failure fraction probability distribution

There is not sufficient data for the IOZ failure fracture to perform the same statistical analysis as for alkyds and epoxy. The EPRI sponsored testing shows that the IOZ failure fraction ranges from 1 to 95%. The Carboline testing also supports a similar range of failure: from 0 to 100%. Therefore, the data supports the assertion that the failure fraction probability will be the same over the complete range from 0 to 100%. This yields the following probability distribution for the IOZ failure fraction:

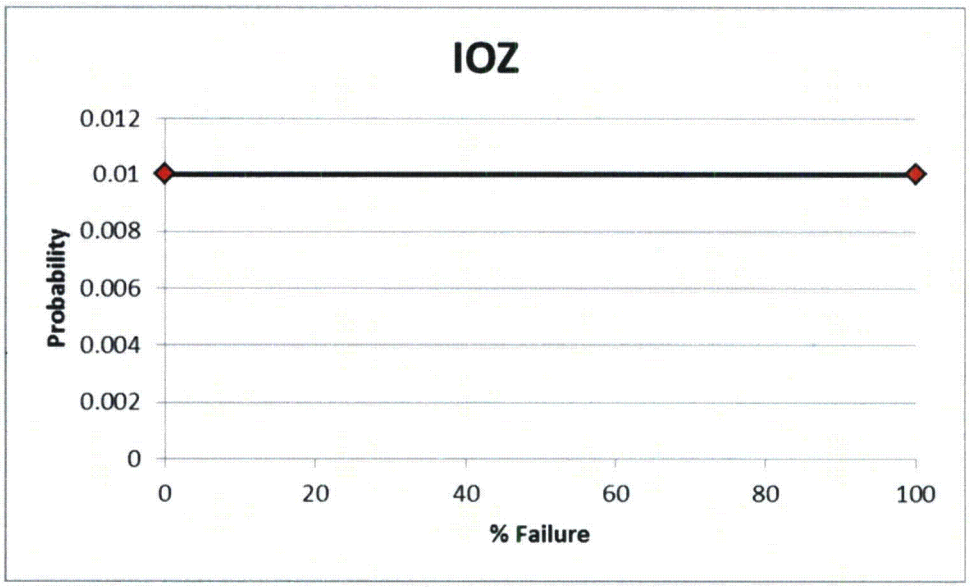

Figure 2.5.6 – Raw IOZ Failure Fraction Probability Distribution

The following table illustrates the corrected probability statistics that accounts for the failure timing extrapolation:

Table 2.5.7 – Corrected IOZ Failure Fraction Probability Statistics

% Failure	Probability
0	0.0000
21	0.0000
100	0.0253

Applying the correction to the 100% failure statistic yields the following corrected probability distribution for the failure fraction of unqualified IOZ coatings:

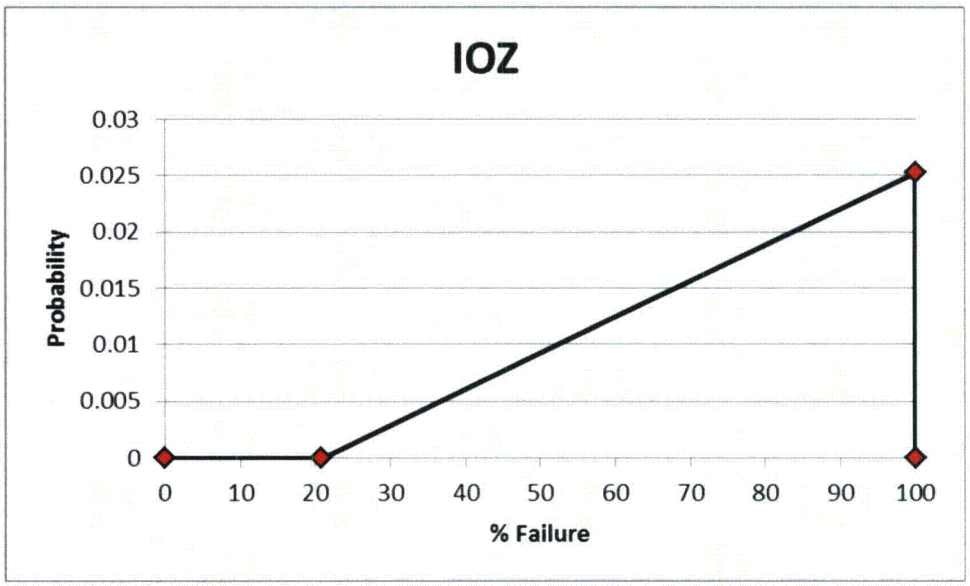

Figure 2.5.7 – IOZ failure fraction probability distribution

Failure Timing Analysis

The other item in the evaluation of unqualified coatings that required probability statistics was the failure timing analysis. In the EPRI sponsored design basis accident (DBA) testing of OEM unqualified coatings (including a combination of epoxy, IOZ, and alkyds), a means of determining the timing of failure was present. The filters used in the autoclave to capture the failed debris were replaced over fifteen times in uneven time increments over the 172 hour test. The time at which these filters were replaced were at 3 hours, 4 hours, 5 hours, 6 hours, 24 hours, 48 hours, 72 hours, 96 hours, 97 hours, 98 hours, 99 hours, 100 hours, 124 hours, 148 hours, and 172 hours. These discarded filters provide a visual timetable of coatings failure. The following figure illustrates the filters that were removed from the autoclave: the filter removal time increases from left to right:

Figure 2.5.8 – EPRI Testing Removed Filters

Test 1 from the previous figure illustrates the filters that captured unqualified coatings debris from the panels that were subjected to irradiation. This test is more prototypical of containment conditions, as the coated surfaces in containment have been subjected to radiation for tens of years. Therefore, the filters from Test 1 will be used to qualitatively determine the timing of failure.

As can be seen from the figure, significant failure of the unqualified OEM coatings starts with the sixth filter from the left: time between 24 and 48 hours. The qualitative estimate of the failure frequency based on visual examination is illustrated in the following figure:

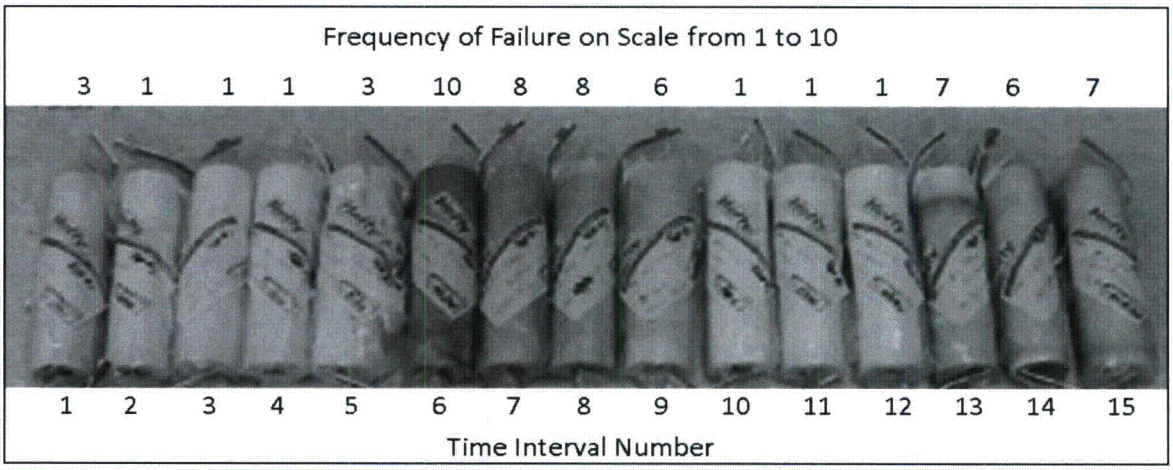

Figure 2.5.9 – Qualitative Frequency of Failure

The most significant failure happens after the 5th time interval (after 24 hours). The following table contains the time interval and its relative frequency of failure:

Table 2.5.8 – Relative Frequency of Failure

Time Interval (#)	Time Interval (hours)	Relative Frequency of Failure
1	0-3	0.047
2	3-4	0.016
3	4-5	0.016
4	5-6	0.016
5	6-24	0.047
6	24-48	0.156
7	48-72	0.125
8	72-96	0.125
9	96-97	0.094
10	97-98	0.016
11	98-99	0.016
12	99-100	0.016
13	100-124	0.109
14	124-148	0.094
15	148-172	0.109

The following histogram shows the coatings failure per time interval as determined by visual inspection:

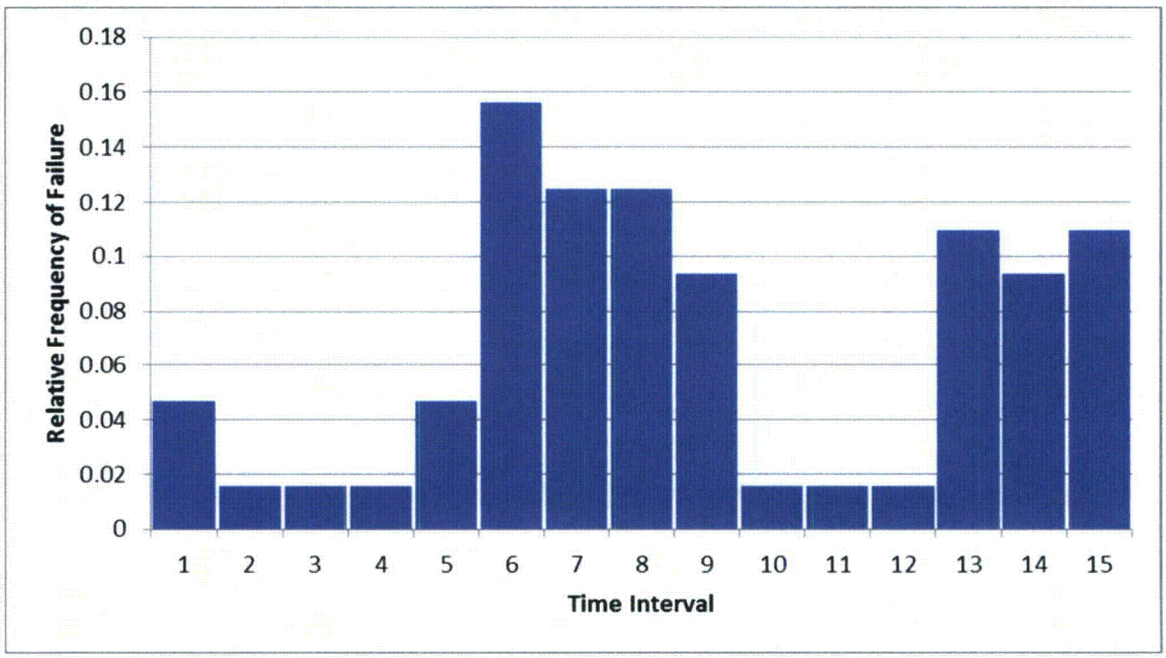

Figure 2.5.10 – Failure Timing Histogram

The time intervals are composed of different time steps. In order to gain a better understanding of the relative frequency of failure timing, the following figure provides an illustration of the normalized failure frequency over the entire 172 hour test (with time interval 9 outlier removed):

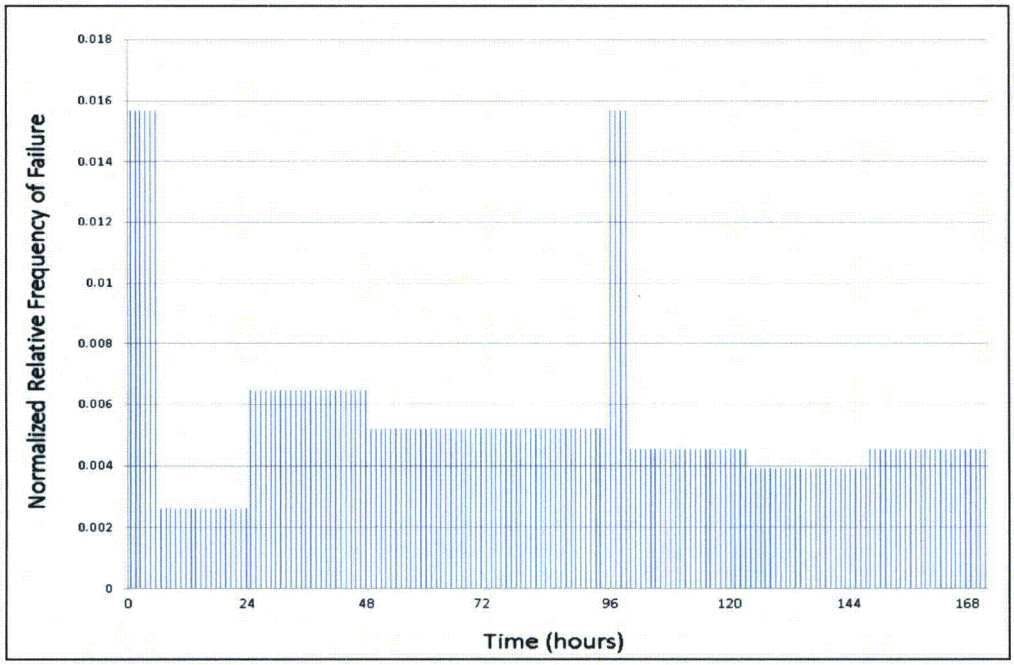

Figure 2.5.11 – Figure Normalized Failure Timing Histogram

This figure shows that although the failure seems to be decreasing, it does not taper off to 0% failure at the end of the 7 days. Therefore, this data has been extrapolated to include the entire 30-day mission time. As can be seen from Figure 2.5.11, there is a slightly declining slope to the failure as time increases. These results have been extrapolated to represent the entire 30-day mission time. The following table illustrates the probability statistics for the extrapolation:

Table 2.5.9 – Failure Timing Statistics

Time (hours)	Probability
1	0.00671
2	0.00671
3	0.00671
4	0.00671
5	0.00671
6	0.00671
7	0.00112
8	0.00112
9	0.00112
10	0.00112
11	0.00112
12	0.00112
13	0.00112
14	0.00112
15	0.00112
16	0.00112
17	0.00112
18	0.00112
19	0.00112
20	0.00112
21	0.00112
22	0.00112
23	0.00112
24	0.00112
25	0.00280
26	0.00280
27	0.00280
28	0.00280
29	0.00280
30	0.00280
31	0.00280
32	0.00280
33	0.00280
34	0.00280
35	0.00280
36	0.00280
37	0.00280
38	0.00280
39	0.00280
40	0.00280
41	0.00280
42	0.00280
43	0.00280
44	0.00280
45	0.00280
46	0.00280
47	0.00280
48	0.00280
49	0.00224
50	0.00224
51	0.00224
52	0.00224
53	0.00224
54	0.00224
55	0.00224
56	0.00224
57	0.00224
58	0.00224
59	0.00224
60	0.00224
61	0.00224
62	0.00224
63	0.00224
64	0.00224
65	0.00224
66	0.00224
67	0.00224
68	0.00224
69	0.00224
70	0.00224
71	0.00224
72	0.00224
73	0.00224
74	0.00224
75	0.00224
76	0.00224
77	0.00224
78	0.00224
79	0.00224
80	0.00224
81	0.00224
82	0.00224
83	0.00224
84	0.00224
85	0.00224
86	0.00224
87	0.00224
88	0.00224
89	0.00224
90	0.00224
91	0.00224
92	0.00224
93	0.00224
94	0.00224
95	0.00224
96	0.00224
97	0.00671
98	0.00671
99	0.00671
100	0.00671
101	0.00196
102	0.00196
103	0.00196
104	0.00196
105	0.00196
106	0.00196
107	0.00196
108	0.00196
109	0.00196
110	0.00196
111	0.00196
112	0.00196

Time (hours)	Probability
113	0.00196
114	0.00196
115	0.00196
116	0.00196
117	0.00196
118	0.00196
119	0.00196
120	0.00196
121	0.00196
122	0.00196
123	0.00196
124	0.00196
125	0.00168
126	0.00168
127	0.00168
128	0.00168
129	0.00168
130	0.00168
131	0.00168
132	0.00168
133	0.00168
134	0.00168
135	0.00168
136	0.00168
137	0.00168
138	0.00168
139	0.00168
140	0.00168
141	0.00168
142	0.00168
143	0.00168
144	0.00168
145	0.00168
146	0.00168
147	0.00168
148	0.00168
149	0.00196
150	0.00196

Time (hours)	Probability
151	0.00196
152	0.00196
153	0.00196
154	0.00196
155	0.00196
156	0.00196
157	0.00196
158	0.00196
159	0.00196
160	0.00196
161	0.00196
162	0.00196
163	0.00196
164	0.00196
165	0.00196
166	0.00196
167	0.00196
168	0.00196
169	0.00196
170	0.00196
171	0.00196
172	0.00196
173	0.00201
174	0.00201
175	0.00201
176	0.00201
177	0.00201
178	0.00201
179	0.00201
180	0.00201
181	0.00201
182	0.00201
183	0.00201
184	0.00201
185	0.00201
186	0.00201
187	0.00201
188	0.00201

Time (hours)	Probability
189	0.00201
190	0.00201
191	0.00201
192	0.00201
193	0.00168
194	0.00168
195	0.00168
196	0.00168
197	0.00168
198	0.00168
199	0.00168
200	0.00168
201	0.00168
202	0.00168
203	0.00168
204	0.00168
205	0.00168
206	0.00168
207	0.00168
208	0.00168
209	0.00168
210	0.00168
211	0.00168
212	0.00168
213	0.00168
214	0.00168
215	0.00168
216	0.00168
217	0.00168
218	0.00168
219	0.00168
220	0.00168
221	0.00168
222	0.00168
223	0.00168
224	0.00168
225	0.00168
226	0.00168

Time (hours)	Probability
227	0.00168
228	0.00168
229	0.00168
230	0.00168
231	0.00168
232	0.00168
233	0.00168
234	0.00168
235	0.00168
236	0.00168
237	0.00168
238	0.00168
239	0.00168
240	0.00168
241	0.00140
242	0.00140
243	0.00140
244	0.00140
245	0.00140
246	0.00140
247	0.00140
248	0.00140
249	0.00140
250	0.00140
251	0.00140
252	0.00140
253	0.00140
254	0.00140
255	0.00140
256	0.00140
257	0.00140
258	0.00140
259	0.00140
260	0.00140
261	0.00140
262	0.00140
263	0.00140
264	0.00140

Time (hours)	Probability
265	0.00140
266	0.00140
267	0.00140
268	0.00140
269	0.00140
270	0.00140
271	0.00140
272	0.00140
273	0.00140
274	0.00140
275	0.00140
276	0.00140
277	0.00140
278	0.00140
279	0.00140
280	0.00140
281	0.00140
282	0.00140
283	0.00140
284	0.00140
285	0.00140
286	0.00140
287	0.00140
288	0.00140
289	0.00140
290	0.00140
291	0.00140
292	0.00140
293	0.00140
294	0.00140
295	0.00140
296	0.00140
297	0.00140
298	0.00140
299	0.00140
300	0.00140
301	0.00140
302	0.00140

Time (hours)	Probability
303	0.00140
304	0.00140
305	0.00140
306	0.00140
307	0.00140
308	0.00140
309	0.00140
310	0.00140
311	0.00140
312	0.00140
313	0.00140
314	0.00140
315	0.00140
316	0.00140
317	0.00140
318	0.00140
319	0.00140
320	0.00140
321	0.00140
322	0.00140
323	0.00140
324	0.00140
325	0.00140
326	0.00140
327	0.00140
328	0.00140
329	0.00140
330	0.00140
331	0.00140
332	0.00140
333	0.00140
334	0.00140
335	0.00140
336	0.00140
337	0.00112
338	0.00112
339	0.00112
340	0.00112

Time (hours)	Probability
341	0.00112
342	0.00112
343	0.00112
344	0.00112
345	0.00112
346	0.00112
347	0.00112
348	0.00112
349	0.00112
350	0.00112
351	0.00112
352	0.00112
353	0.00112
354	0.00112
355	0.00112
356	0.00112
357	0.00112
358	0.00112
359	0.00112
360	0.00112
361	0.00112
362	0.00112
363	0.00112
364	0.00112
365	0.00112
366	0.00112
367	0.00112
368	0.00112
369	0.00112
370	0.00112
371	0.00112
372	0.00112
373	0.00112
374	0.00112
375	0.00112
376	0.00112
377	0.00112
378	0.00112

Time (hours)	Probability
379	0.00112
380	0.00112
381	0.00112
382	0.00112
383	0.00112
384	0.00112
385	0.00112
386	0.00112
387	0.00112
388	0.00112
389	0.00112
390	0.00112
391	0.00112
392	0.00112
393	0.00112
394	0.00112
395	0.00112
396	0.00112
397	0.00112
398	0.00112
399	0.00112
400	0.00112
401	0.00112
402	0.00112
403	0.00112
404	0.00112
405	0.00112
406	0.00112
407	0.00112
408	0.00112
409	0.00112
410	0.00112
411	0.00112
412	0.00112
413	0.00112
414	0.00112
415	0.00112
416	0.00112

Time (hours)	Probability
417	0.00112
418	0.00112
419	0.00112
420	0.00112
421	0.00112
422	0.00112
423	0.00112
424	0.00112
425	0.00112
426	0.00112
427	0.00112
428	0.00112
429	0.00112
430	0.00112
431	0.00112
432	0.00112
433	0.00112
434	0.00112
435	0.00112
436	0.00112
437	0.00112
438	0.00112
439	0.00112
440	0.00112
441	0.00112
442	0.00112
443	0.00112
444	0.00112
445	0.00112
446	0.00112
447	0.00112
448	0.00112
449	0.00112
450	0.00112
451	0.00112
452	0.00112
453	0.00112
454	0.00112

Time (hours)	Probability
455	0.00112
456	0.00112
457	0.00084
458	0.00084
459	0.00084
460	0.00084
461	0.00084
462	0.00084
463	0.00084
464	0.00084
465	0.00084
466	0.00084
467	0.00084
468	0.00084
469	0.00084
470	0.00084
471	0.00084
472	0.00084
473	0.00084
474	0.00084
475	0.00084
476	0.00084
477	0.00084
478	0.00084
479	0.00084
480	0.00084
481	0.00084
482	0.00084
483	0.00084
484	0.00084
485	0.00084
486	0.00084
487	0.00084
488	0.00084
489	0.00084
490	0.00084
491	0.00084
492	0.00084

Time (hours)	Probability
493	0.00084
494	0.00084
495	0.00084
496	0.00084
497	0.00084
498	0.00084
499	0.00084
500	0.00084
501	0.00084
502	0.00084
503	0.00084
504	0.00084
505	0.00084
506	0.00084
507	0.00084
508	0.00084
509	0.00084
510	0.00084
511	0.00084
512	0.00084
513	0.00084
514	0.00084
515	0.00084
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519	0.00084
520	0.00084
521	0.00084
522	0.00084
523	0.00084
524	0.00084
525	0.00084
526	0.00084
527	0.00084
528	0.00084
529	0.00084
530	0.00084

Time (hours)	Probability
531	0.00084
532	0.00084
533	0.00084
534	0.00084
535	0.00084
536	0.00084
537	0.00084
538	0.00084
539	0.00084
540	0.00084
541	0.00084
542	0.00084
543	0.00084
544	0.00084
545	0.00084
546	0.00084
547	0.00084
548	0.00084
549	0.00084
550	0.00084
551	0.00084
552	0.00084
553	0.00084
554	0.00084
555	0.00084
556	0.00084
557	0.00084
558	0.00084
559	0.00084
560	0.00084
561	0.00084
562	0.00084
563	0.00084
564	0.00084
565	0.00084
566	0.00084
567	0.00084
568	0.00084

Time (hours)	Probability
569	0.00084
570	0.00084
571	0.00084
572	0.00084
573	0.00084
574	0.00084
575	0.00084
576	0.00084
577	0.00056
578	0.00056
579	0.00056
580	0.00056
581	0.00056
582	0.00056
583	0.00056
584	0.00056
585	0.00056
586	0.00056
587	0.00056
588	0.00056
589	0.00056
590	0.00056
591	0.00056
592	0.00056
593	0.00056
594	0.00056
595	0.00056
596	0.00056
597	0.00056
598	0.00056
599	0.00056
600	0.00056
601	0.00056
602	0.00056
603	0.00056
604	0.00056
605	0.00056
606	0.00056

Time (hours)	Probability
607	0.00056
608	0.00056
609	0.00056
610	0.00056
611	0.00056
612	0.00056
613	0.00056
614	0.00056
615	0.00056
616	0.00056
617	0.00056
618	0.00056
619	0.00056
620	0.00056
621	0.00056
622	0.00056
623	0.00056
624	0.00056
625	0.00056
626	0.00056
627	0.00056
628	0.00056
629	0.00056
630	0.00056
631	0.00056
632	0.00056
633	0.00056
634	0.00056
635	0.00056
636	0.00056
637	0.00056
638	0.00056
639	0.00056
640	0.00056
641	0.00056
642	0.00056
643	0.00056
644	0.00056

Time (hours)	Probability
645	0.00056
646	0.00056
647	0.00056
648	0.00056
649	0.00056
650	0.00056
651	0.00056
652	0.00056
653	0.00056
654	0.00056
655	0.00056
656	0.00056
657	0.00056
658	0.00056
659	0.00056
660	0.00056
661	0.00056
662	0.00056
663	0.00056
664	0.00056
665	0.00056
666	0.00056
667	0.00056
668	0.00056
669	0.00056
670	0.00056
671	0.00056
672	0.00056
673	0.00056
674	0.00056
675	0.00056
676	0.00056
677	0.00056
678	0.00056
679	0.00056
680	0.00056
681	0.00056
682	0.00056

Time (hours)	Probability
683	0.00056
684	0.00056
685	0.00056
686	0.00056
687	0.00056
688	0.00056
689	0.00056
690	0.00056
691	0.00056
692	0.00056
693	0.00056
694	0.00056
695	0.00056
696	0.00056
697	0.00056
698	0.00056
699	0.00056
700	0.00056
701	0.00056
702	0.00056
703	0.00056
704	0.00056
705	0.00056
706	0.00056
707	0.00056
708	0.00056
709	0.00056
710	0.00056
711	0.00056
712	0.00056
713	0.00056
714	0.00056
715	0.00056
716	0.00056
717	0.00056
718	0.00056
719	0.00056
720	0.00056

These statistics yield the following extrapolated probability of failure timing:

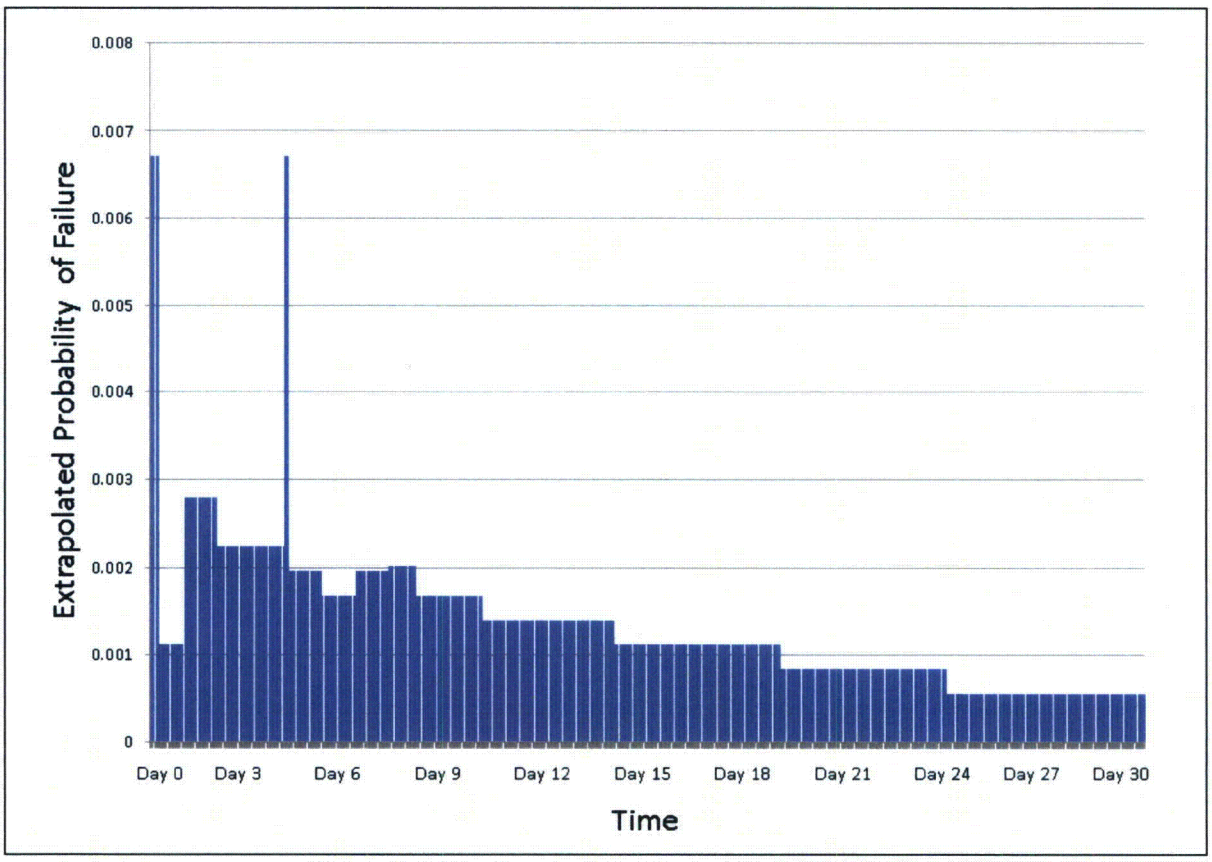

Figure 2.5.12 – Extrapolated Probability of Failure

As a result of this extrapolation, the probability of failure at a time before seven days is 39.6% of the total probability. Therefore, there is a 152.5% increase in probability statistics due to the extrapolation to the 30-day mission time. This increase in probability is applied to the 100% failure statistic of the failure fraction analysis to correct for the extrapolation (See Failure Fraction Analysis). This results in a significant increase in the quantity of failed coatings. Additionally, all of the failed unqualified coatings in upper containment are assumed to be exposed to containment sprays. Therefore, all of the coatings in upper containment that fail when containment sprays are on will transport to the pool. This is conservative since some of the failed coatings in upper containment may be in locations that are shielded from containment sprays. These conservative factors minimize the inherent risk of extrapolation.

Debris Characteristics

The debris characteristics of the failed unqualified coatings in STP are defined in this section. Different types of coating have different failure characteristics. Epoxy coatings are expected to fail as chips while IOZ and Alkyd coatings fail as particulate. The failure mode of each coating determines the debris transportability when exposed to containment sprays.

The IOZ unqualified coatings are expected to fail as particulate. Several studies have shown that the unqualified IOZ fails as powder on the order of 10 microns. The BWROG supported testing that indicated the size range of the IOZ failed coating debris is between 1 and 20 micron (8); with 80% less than 10 micron, and 50% less than 5 micron. Other testing supports these conclusions. The BWR utility resolution guide gives the size range of the failed IOZ coating between 4-20 micron (9). The density of a common IOZ coating (Carbozinc 11) is 208 lbm/ft^3 (10). The weight-averaged density of the unqualified IOZ coatings found at STP is calculated in the following table (11):

Table 2.5.10 – Unqualified IOZ Weight-Averaged Density

Substrate number	**Mass (lbm)**	**Dry Film Density (lbm/ft^3)**
6	5.3	256.6
12a	4.3	121
13	29.1	256.6
16a	1.1	150.1
21	601.3	256.6
23a	37.2	256.6
26a	66.6	150.1
30a	3.5	150.1
31	16.9	150.1
37	8.7	256.6
38a	1.3	256.6
Weighted Average		**243.7**

The epoxy unqualified coatings are expected to fail as chips. There have been several studies to determine the failure mode and size of the epoxy debris. The BWROG supported generic testing that indicated that the thickness of these chips on average is 275 micron (11 mil) (8). Generally, the chip thickness is assumed to be the same as the applied dry film thickness (DFT). The weighted average of DFT for the unqualified epoxy coatings at STP is calculated from information in the unqualified coatings log (11):

Table 2.5.11 – Weight Average of Unqualified Epoxy DFT

Substrate #	Mass (lbm)	DFT (mils)
1a	381.15	10
1b	959.86	22
1c	233.42	5
12b	9.9	11
16b	1.74	8
18b	54.2	8
20	9.57	14
23b	42.23	8
24	2.05	7
26b	110.01	8
27	0.57	6
28	6.55	6
29	0.57	6
30b	4.81	8
32	2.29	12
33	10.4	7
38b	1.56	8
39b	0.42	6.5
Weighted Average		**15**

The Carboline unqualified coatings testing indicated that epoxy chips disbonded at lengths of up to 1" long (4). Moreover, the BWR utility resolution guide states that IOZ with an epoxy topcoat will fail as follows: the epoxy chips with be in the size range of 0.125 to 2.0 inches, while the IOZ will both adhere to the back of the chips and separate as small particulate (9). In addition to this general testing, plant specific testing has been conducted to determine the size distribution of the epoxy chips. Alion characterized samples from Comanche Peak that indicated the length of the failed epoxy chips range from around 6 mils to 2 inches (5). The following table illustrates these results:

Table 2.5.12 – Epoxy Debris Size Distribution by Mass

Size Range of Coating	Mass (g)	Percentage of Total Mass
1-2 inch	3.4657	32.03%
0.5-1 inch	0.9784	9.04%
0.25-0.5 inch	0.4774	4.41%
0.125-0.25 inch	0.5434	5.02%
< 0.125 inch	5.3561	49.50%
Total	10.821	100.00%

The less than 0.125 inch size range includes fines and fine chips. Of the 49.50% of this size range, 12.275% are assumed to be 6 mil particles (fines) and 37.225% are assumed to be 1/64 inch (fine chips) (10). Additionally, 50% of the chips above 0.5 inches are assumed to be curled (5). The following size distributions will be used in the risk-informed analysis:

Table 2.5.13 – Epoxy Debris Size Distribution

Size Designation	Size Range (inch)	Percentage of Total Mass
Fines (particles)	0.006	12.28%
Flat Fine Chips	0.0156	37.23%
Flat Small Chips	0.125-0.5	9.43%
Flat Large Chips	0.5-2.0	20.53%
Curled Chips	0.5-2.0	20.53%

The weight-averaged density of the unqualified epoxy coatings found at STP is calculated based on information from the unqualified coatings log (11):

Table 2.5.14 – Unqualified Epoxy Weight-Averaged Density

Substrate Number	Mass (lbm)	Dry Film Density (lbm/ft^3)
1a	381.2	113.0
1b	959.9	129.4
1c	233.4	138.5
12b	9.9	138.5
16b	1.7	108.5
18b	54.2	84.1
20	9.6	109.4
23b	42.2	109.4
24	2.1	109.4
26b	110.0	108.5
27	0.6	109.4
28	6.6	109.4
29	0.6	109.4
30b	4.8	102.5
32	2.3	109.4
33	10.4	109.4
38b	1.6	109.4
39b	0.4	93.5
Weighted Average		**123.7**

The alkyd coatings are expected to fail as particulate. The debris characteristics of the alkyd coating are shown to be soft pliable pieces and particulate in the BWR utility resolution guide (9). In addition, unqualified alkyd coatings bench top testing conducted by Alion determined that the failure mode was potential delamination and release of particles into solution (6). This testing program showed that the particles were on the order of 10 micron. Due to the similar particle size of the alkyd coatings to IOZ coatings, the size distribution of the particles will be assumed to be the same as IOZ. The weight-averaged density of the specific unqualified alkyd coatings found at STP is calculated based on information from the unqualified coatings log (11):

Table 2.5.15 – Unqualified Alkyd Weight-Averaged Density

Substrate number	Mass (lbm)	Dry Film Density (lbm/ft^3)
2a	13.5	120.4
2b	54.6	97.2
4	25	228.5
8 (zinc rich alkyd)	133.8	268.5
18a	12.3	102.3
22(zinc rich alkyd)	23.7	228.8
35a	1.5	120.4
35b	2.5	97.2
39a	4	120.4
Weighted Average		**207.3**

Baked enamels are assumed to have the same debris characteristics as an alkyd. This is because baked enamel is the common alkyd coating sold for metal finished products (6). The average density of the specific unqualified baked enamel coatings found at STP is calculated based on information in the unqualified coatings log (11):

Table 2.5.16 – Unqualified Baked Enamel Weight-Averaged Density

Substrate number	Mass (lbm)	Dry Film Density (lbm/ft^3)
3	260	93.8
7	7.2	69.5
Weighted Average		**93.1**

There is currently no information on the debris characteristics of intumescent coatings. Therefore, it will be assumed to have the same debris characteristics as epoxy. The weight-averaged density of the intumescent coatings found at STP is calculated based on the unqualified coating logs (11):

Table 2.5.17 – Unqualified Intumescent Weight-Averaged Density

Substrate Number	Mass (lbm)	Dry Film Density (lbm/ft^3)
5a	0.5	83.8
5b	1.8	97.2
19a	10.7	134.0
19b	19.5	75.3
19c	5.0	96.8
Weighted Average		**96.0**

Assumptions: Several assumptions were made in the development of the unqualified coatings calculation (7). The more significant assumptions are listed below:

- For any component substrate location that was indeterminate, it was assumed that the location was at the pool level allowing direct transport to the pool. This is the most conservative alternative.
- The debris characteristics and failure fraction of baked enamel is assumed to be the same as that for the unqualified alkyd coatings. This is a reasonable assumption because baked enamel is a common type of alkyd coating (6).
- It was assumed that the total mass quantities formulated in this calculation are applicable to both STP units. This is a reasonable assumption because the containment buildings for STP Units 1 and 2 are essentially identical.
- The debris characteristics of intumescent coatings are assumed to be the same as epoxy. This is a reasonable assumption because many commercially available intumescent coatings are partially composed of epoxy.

Acceptance Criteria: No acceptance criteria were used for the unqualified coatings evaluation.

Results: The results of the unqualified coatings calculation (7) are used as an input to the overall GSI-191 evaluation (12). These results are described in detail below.

The total quantity and locations of potentially transportable unqualified coatings are shown in Table 2.5.18, and the debris characteristics are shown in Table 2.5.19. Since unqualified coatings can fail outside the zone of influence, these quantities are applicable for all break scenarios (although the failure fraction may be less than 100%).

Table 2.5.18 – Quantity and location of potentially transportable unqualified coatings debris

Coatings Type	Upper Containment Quantity (lb_m)	Lower Containment Quantity (lb_m)	Reactor Cavity Quantity (lb_m)	Total Quantity (lb_m)
Unqualified Epoxy	295 (15%)	36 (2%)	1,574 (83%)	1,905
Unqualified IOZ	305 (83%)	64 (17%)	0 (0%)	369
Unqualified Alkyd	146 (54%)	125 (46%)	0 (0%)	271
Unqualified Baked Enamel	0 (0%)	267 (100%)	0 (0%)	267
Unqualified Intumescent	0 (0%)	2 (100%)	0 (0%)	2

Table 2.5.19 – Material properties of unqualified coatings debris

Debris Type	Debris Size	Macroscopic Density	Microscopic Density
Unqualified Epoxy	Fines: 6 mil particles	-	124 lb_m/ft^3
	Fine Chips: 0.0156"×15 mil		
	Small Chips: 0.125"-0.5"×15 mil		
	Large Chips: 0.5"-2.0"×15 mil		
	Curled Chips: 0.5"-2.0"×15 mil		
Unqualified Alkyd	Fines: 4 - 20 µm particles	-	207 lb_m/ft^3
Unqualified IOZ	Fines: 4 - 20 µm particles	-	244 lb_m/ft^3
Unqualified Baked Enamel	Fines: 4 - 20 µm particles	-	93 lb_m/ft^3

Probability distributions were developed for the failure fractions of unqualified epoxy, IOZ, and alkyd coatings as shown in Figure 2.5.13 through Figure 2.5.15. These probability distributions were sampled for each case evaluated, and were applied for all locations (e.g., if a failure fraction of 85% is selected for unqualified epoxy coatings, the total failure fraction for unqualified coatings in upper containment, lower containment, and the reactor cavity would each be 85%). The quantity of unqualified intumescent coatings was assumed to be negligible.

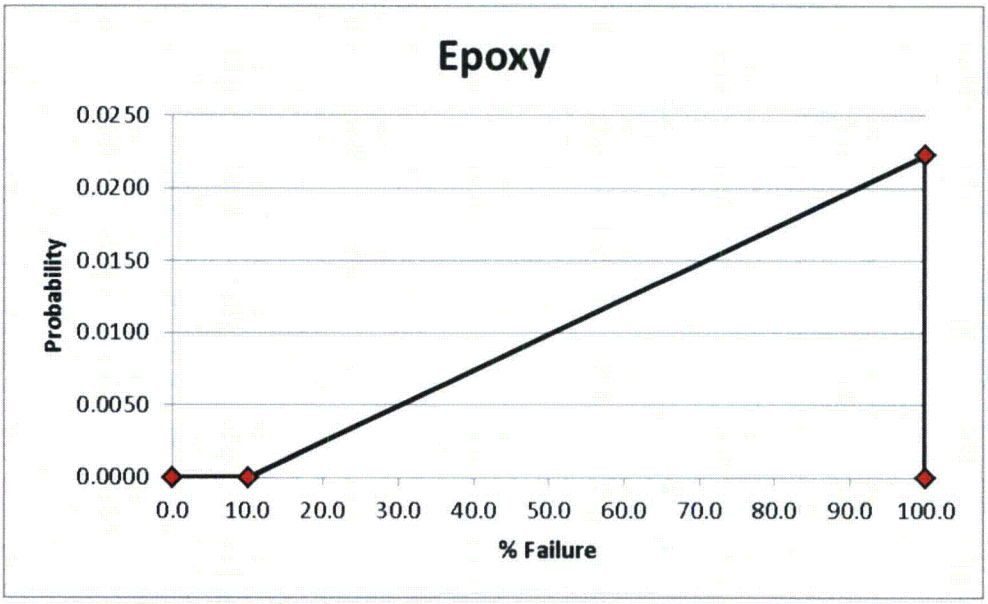

Figure 2.5.13 – Epoxy failure fraction probability distribution

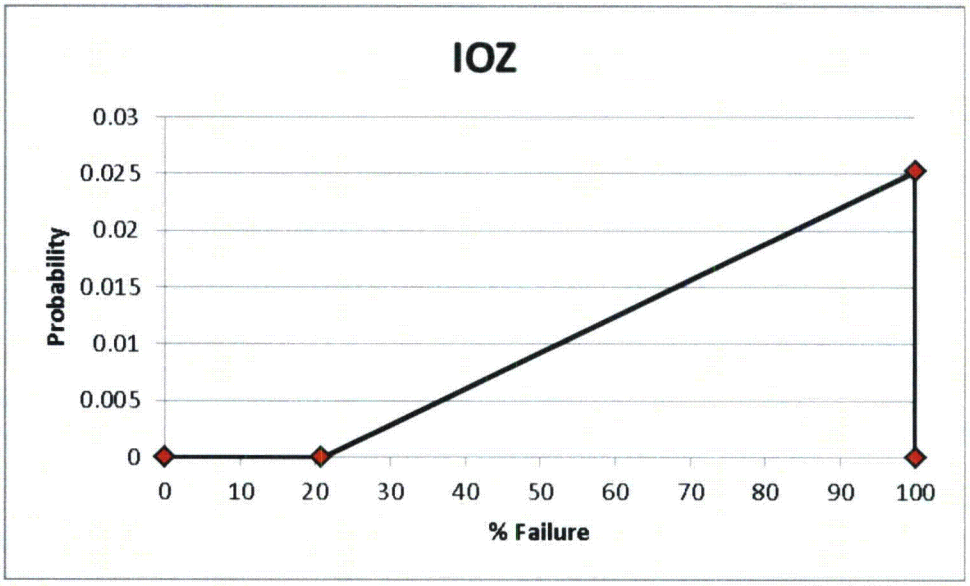

Figure 2.5.14 – IOZ failure fraction probability distribution

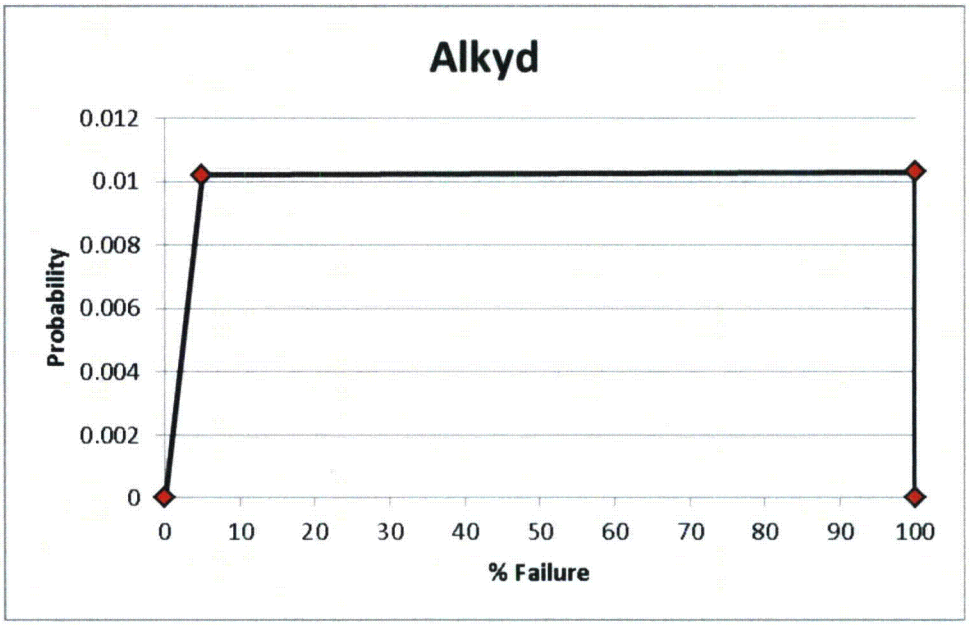

Figure 2.5.15 – Alkyd and baked enamel failure fraction probability distribution

The failure timing that was used for all unqualified coatings is shown in Table 2.5.20, where F_{fail} is the failure fraction (e.g., 85%).

Table 2.5.20 – Time-dependent failure fraction of unqualified coatings

Time (Hours)	Time Dependent Failure
0-24	$0.060 \cdot F_{fail}$
24-48	$0.067 \cdot F_{fail}$
48-72	$0.054 \cdot F_{fail}$
72-96	$0.054 \cdot F_{fail}$
96-124	$0.107 \cdot F_{fail}$
124-148	$0.040 \cdot F_{fail}$
148-172	$0.047 \cdot F_{fail}$
172-192	$0.040 \cdot F_{fail}$
192-216	$0.040 \cdot F_{fail}$
216-240	$0.040 \cdot F_{fail}$
240-264	$0.034 \cdot F_{fail}$
264-288	$0.034 \cdot F_{fail}$
288-312	$0.034 \cdot F_{fail}$
312-336	$0.034 \cdot F_{fail}$
336-360	$0.027 \cdot F_{fail}$
360-384	$0.027 \cdot F_{fail}$
384-408	$0.027 \cdot F_{fail}$
408-432	$0.027 \cdot F_{fail}$
432-456	$0.027 \cdot F_{fail}$
456-480	$0.020 \cdot F_{fail}$
480-504	$0.020 \cdot F_{fail}$
504-528	$0.020 \cdot F_{fail}$
528-552	$0.020 \cdot F_{fail}$
552-576	$0.020 \cdot F_{fail}$
576-600	$0.013 \cdot F_{fail}$
600-624	$0.013 \cdot F_{fail}$
624-648	$0.013 \cdot F_{fail}$
648-672	$0.013 \cdot F_{fail}$
672-696	$0.013 \cdot F_{fail}$
696-720	$0.013 \cdot F_{fail}$
Total	$1.0 \cdot F_{fail}$

Additional details on the basis for the unqualified coatings quantities, locations, failure fractions, and failure timing are provided in the STP unqualified coatings debris generation calculation (7).

In a typical deterministic GSI-191 evaluation, 100% of the unqualified coatings are assumed to fail, and the time-dependence is not considered (i.e. the unqualified coatings are normally assumed to fail at the beginning of the event). The unqualified coatings are often assumed to fail as 10 micron particulate, although some plants have credited a range of chip sizes for unqualified epoxy coatings. The results from the debris characteristics evaluation of unqualified coatings at STP are documented in the following table:

Table 2.5.21 – Debris Characteristics Summary

Coating Type	Debris Type	Size Range	Density
IOZ	Particulate	4-20 micron	244 lbm/ft^3
Epoxy	Chips	10 mils-2 inches	124 lbm/ft^3
Alkyd	Particulate	4-20 micron	207 lbm/ft^3
Baked Enamel	Particulate	4-20 micron	93 lbm/ft^3
Intumescents	Chips	6 mils-2 inches	96 lbm/ft^3

For the STP risk-informed evaluation, the failure fraction for each type of unqualified coating was determined by sampling the failure fraction probability distributions for each of the thousands of scenarios evaluated. The location, failure timing, and debris characteristics are important for several reasons:

- Unqualified coatings in upper containment that fail after containment sprays are secured would not be transported to the containment pool.
- Unqualified coatings in lower containment were assumed to fall directly in the pool and be available for transport. However, delays in the failure timing result in delayed arrival at the strainer and a delayed impact on head loss.
- Unqualified coatings in the reactor cavity would only be available for transport to the strainers if the break is in the reactor cavity.
- Although the unqualified coatings fines would essentially all transport to the strainer, the transport for the chips would be significantly reduced.

Additional details on how the unqualified coatings debris was treated in the overall GSI-191 evaluation are provided in Volume 3 (12).

Item 5.a.2: Debris Capture

Method: The methodology for debris capture on gratings and obstructions during the blowdown phase is documented in an engineering calculation based on plant-specific features (locations of grating, etc.) and applicable test data (13). The test data that was used is documented in the drywell debris transport study (DDTS) (14). The full range of break scenarios were grouped into the following break categories:

- Breaks in the steam generator compartments
- Breaks in the reactor cavity
- Breaks inside secondary shield wall beneath steam generator compartments
- Breaks in the pressurizer compartment
- Breaks outside secondary shield wall in the pressurizer surge line
- Breaks outside secondary shield wall in the RHR compartments
- Breaks outside secondary shield wall in the annulus

For each of these break locations, the fraction of debris blown up toward upper containment or down toward the containment floor was determined based on the relative containment volumes. The fine debris generated inside the zone of influence (ZOI) was assumed to fully transport with the blowdown flow. For small and large pieces of debris, the effects of debris capture were taken into account for miscellaneous structures, grating, and 90° turns in the flow path based on test data from the DDTS.

For each of these break locations, the fraction of debris blown up toward upper containment or down toward the containment floor was determined based on the relative containment volumes. The volume of upper containment (including areas above the operating deck) was calculated to be 2,320,079 ft^3, and the total volume in containment was calculated to be 3,322,040 ft^3. The fine debris generated inside the zone of influence (ZOI) was assumed to fully transport with the blowdown flow. Therefore, the transport fraction for the total fine debris from all areas would be 70% to upper containment. For small and large pieces of debris, the effects of debris capture were taken into account for miscellaneous structures, grating, and 90° turns in the flow path based on test data from the DDTS. The results of the DDTS testing showed that in a wetted, highly congested area, approximately 0% to 13% of small fiberglass debris would be trapped by miscellaneous structures, 15% to 29% would be trapped by grating, and 3% to 29% would be captured at 90° turns in flow path. The amount of small piece debris that gets blown to upper containment can be calculated as shown in the following equation:

$$F_{BD} = \left(\frac{V_{upper}}{V_{total}}\right)(1.00 - F_{misc})(1.00 - F_{90^\circ turns} \cdot N_{turns})(1.00 - F_{grating} \cdot N_{gratings})$$

where:

F_{BD} = fraction of debris blown to upper containment
V_{upper} = volume of upper containment
V_{total} = total volume in containment
F_{misc} = fraction of debris trapped by miscellaneous structures
$F_{90^\circ turns}$ = fraction of debris trapped by changes in flow direction
N_{turns} = number of turns or changes in flow direction debris would pass through
$F_{grating}$ = fraction of debris trapped by grating
$N_{gratings}$ = number of gratings debris would pass through

Large piece debris would be blown away from the break similar to the small piece debris. However, this debris would not pass through grating.

Each break location was analyzed separately to determine the average number of turns and gratings applicable to each location.

Breaks in the Steam Generator Compartments

For breaks in the Steam Generator Compartments, it was determined that the small piece debris blown to upper containment would have to pass through 78% coverage of grating above these compartments, an average of one 90° turn, and a variety of miscellaneous structures. As shown in the following

equations, it was determined that the range for small fiberglass debris blown to upper containment is 33% to 60%.

$$F_{blowdown\,(small\,fiber)} = (0.70)(1.00-0.00)(1.00-0.03)(1.00-0.15\cdot 0.78) = 0.60$$

$$F_{blowdown\,(small\,fiber)} = (0.70)(1.00-0.13)(1.00-0.29)(1.00-0.29\cdot 0.78) = 0.33$$

For the small piece debris blown to lower containment from these compartments, it was determined that debris would have to pass through 1 level of grating at the bottom of the compartments, an average of one 90° turn, and a variety of miscellaneous structures. As shown in the following equations, it was determined that 13% to 25% of small piece debris would be blown to lower containment.

$$F_{blowdown\,SG\,comps.\,(small\,fiber)} = (0.30)(1.00-0.00)(1.00-0.03)(1.00-0.15\cdot 1.00) = 0.25$$

$$F_{blowdown\,SG\,comps.\,(small\,fiber)} = (0.30)(1.00-0.13)(1.00-0.29)(1.00-0.29\cdot 1.00) = 0.13$$

Since the large piece debris would not pass through the floor grating toward lower containment, and a negligible quantity would be blown past miscellaneous structures and the 78% effective grating toward upper containment, it is estimated that 0% to 22% would be blown to upper containment.

Breaks in the Reactor Cavity

For breaks inside the reactor cavity, the transport fractions for small and large fiberglass debris was determined to be the same as for a break in the Steam Generator Compartments, since this debris would essentially follow the same path towards upper containment. Therefore, 33% to 60% of small fiberglass debris would be in upper containment, 13% to 25% in the containment pool, and 15% to 54% remaining in the steam generator compartments. For small and large RMI generated in the reactor cavity, it was estimated that all of this debris would be blown to the containment pool since the RMI debris would be caught up in the reactor cavity and miscellaneous structures more easily. Since Microtherm is in the secondary shield wall penetrations, there wouldn't be any of this type of debris destroyed for a reactor cavity break.

Breaks below the Steam Generator Compartment Floor

For breaks below the Steam Generator Compartment Floor, it was determined that debris would have to pass through one level of grating between the compartments, and the 78% coverage of grating above the steam generator compartments. As shown in the following equations, it was determined that the range of small fiber blown to upper containment would be 21% to 50% based on an average of one 90° turn, and a variety of miscellaneous structures.

$$F_{blowdown\,(small\,fiber)} = (0.70)(1.00-0.00)(1.00-0.03)(1.00-0.15\cdot 1.78) = 0.50$$

$$F_{blowdown\,(small\,fiber)} = (0.70)(1.00-0.13)(1.00-0.29)(1.00-0.29\cdot 1.78) = 0.21$$

The remaining small fiberglass not blown to upper containment would be blown to the containment pool. The large fiberglass would be captured by the grating; therefore, 0% would be transported to upper containment.

Breaks in the Pressurizer Compartment

For breaks in the pressurizer compartment, it was estimated that small piece debris blown to upper containment would have to pass through an average of two 90° turns and a variety of miscellaneous structures. Therefore, as shown in the following equations, the range of small piece debris blown to upper containment would be 26% to 66%.

$$F_{blowdown\,(small\,fiber)} = (0.70)(1.00-0.00)(1.00-0.03\cdot 2.00) = 0.66$$

$$F_{blowdown\,(small\,fiber)} = (0.70)(1.00-0.13)(1.00-0.29\cdot 2.00) = 0.26$$

The small piece debris blown to lower containment was also estimated to pass through an average of two 90° turns and a variety of miscellaneous structures. Therefore, the range of small piece debris blown to the containment floor would be 11% to 28%, as shown in the following equations.

$$F_{blowdown\,PRZRcomp.\,(small\,fiber)} = (0.30)(1.00-0.00)(1.00-0.03\cdot 2.00) = 0.28$$

$$F_{blowdown\,PRZRcomp.\,(small\,fiber)} = (0.30)(1.00-0.13)(1.00-0.29\cdot 2.00) = 0.11$$

The large piece debris would be blown away from the break similar to the small piece debris. However, the transport fraction for this debris would be on the lower end of the range of transport values for the small pieces, since this debris would be more easily held up on structures and the 90° turns in flow path. It was estimated that 16% to 26% would be in upper containment, 1% to 11% in the containment pool, and 63% to 83% remaining in the compartment.

Breaks in the Pressurizer Surge Line

It was estimated that debris blown to upper containment would have to pass through the grating at the 19'-0" elevation or the 37'-3" elevation, and at least two additional levels of grating above these gratings. Therefore, the range of the small piece debris blown to upper containment would be 3% to 36% as shown in the following equations based on debris passing through an average of two 90° turns, 3 levels of effective grating, and miscellaneous structures.

$$F_{blowdown\,(small\,fiber)} = (0.70)(1.00-0.00)(1.00-0.03\cdot 2.00)(1-0.15\cdot 3.00) = 0.36$$

$$F_{blowdown\,(small\,fiber)} = (0.70)(1.00-0.13)(1.00-0.29\cdot 2.00)(1.00-0.29\cdot 3.00) = 0.03$$

Since there is no grating between the pressurizer surge line break and lower containment, it was estimated that the remaining small piece debris would be blown to the containment pool.

The large fiberglass debris would be blown away in the same manner as the small debris. However, since this debris would not pass through grating, it was estimated that 100% would be blown to the containment pool.

Microtherm is present in the surge line penetration in the secondary shield wall. Seventy percent of the Microtherm fines would be blown towards upper containment.

Breaks in the RHR Compartments

The RHR compartments are highly compartmentalized, and debris would be blown down and then back up to upper containment. It was estimated that the range of the percentage of small piece debris blown to upper containment in the RHR compartments would be 3% to 45% as shown in the following equations based on a fraction of debris passing through an average of three 90° turns, 2 levels of effective grating, and a variety of miscellaneous structures.

$$F_{blowdown(small\ fiber)} = (0.70)(1.00-0.00)(1.00-0.03\cdot 3.00)(1-0.15\cdot 2.00) = 0.45$$

$$F_{blowdown(small\ fiber)} = (0.70)(1.00-0.13)(1.00-0.29\cdot 3.00)(1-0.29\cdot 2.00) = 0.03$$

Thirty percent of small piece debris would get blown to the containment floor. Some of this debris would be captured at significant flow turns and by miscellaneous structures. Therefore, the range of small piece debris blown to the containment floor would be 1% to 19% as shown in the following equations:

$$F_{\text{blowdown(smallfiber)}} = (0.30)(1.00-0.00)(1.00-0.03\cdot 3.00)(1-0.15\cdot 2.00) = 0.19$$

$$F_{blowdown(small\ fiber)} = (0.30)(1.00-0.13)(1.00-0.29\cdot 3.00)(1-0.29\cdot 2.00) = 0.01$$

The large piece debris would be blown away from the break similar to the small piece debris. However, since this debris would not pass through grating, the transport fraction to upper containment would be 0%. Some large piece debris could be transported to lower containment, however, since there are locations in the compartment that are located below the lowest level of grating. It is estimated that about 0% to 10% would be in lower containment, and 90% to 100% remaining in the compartments.

Breaks in the Annulus

The break locations in the annulus are between the 19' elevation grating and the 37'-3" elevation grating. Therefore, in order for debris to reach upper containment, it would have to pass through three levels of grating. The range of the transport fractions for small debris blown to upper containment

would be 6% to 37% as shown in the following equations, based on debris passing through an average of one 90° turn, three levels of grating, and miscellaneous structures.

$$F_{blowdown\,(small\,fiber)} = (0.70)(1.00-0.00)(1.00-0.03\cdot 1.00)(1-0.15\cdot 3.00) = 0.37$$

$$F_{blowdown\,(small\,fiber)} = (0.70)(1.00-0.13)(1.00-0.29\cdot 1.00)(1.00-0.29\cdot 3.00) = 0.06$$

Thirty percent of small piece debris would get blown to the containment floor. Some of the debris would get trapped on grating and miscellaneous structures. Therefore, the range of debris that would be blown to lower containment would be 13% to 25% as shown in the following equations, based on an average of one 90° turn, one level of grating, and miscellaneous structures.

$$F_{blowdown\,(small\,fiber)} = (0.30)(1.00-0.00)(1.00-0.03\cdot 1.00)(1.00-0.15\cdot 1.00) = 0.25$$

$$F_{blowdown(small\,fiber)} = (0.30)(1.00-0.13)(1.00-0.29\cdot 1.00)(1.00-0.29\cdot 1.00) = 0.13$$

The large piece debris would be blown away in the same manner as the small debris. However, since this debris would not pass through grating, 100% would remain in the annulus above the pool elevation.

Basis: The methodology used for debris capture during the blowdown phase is based on refined deterministic debris transport methods that have been previously accepted by the NRC (15). The primary difference in the risk-informed evaluation is that several additional break locations are considered, and the retention fractions on grating and other structures is based on the range of values provided in the DDTS rather than a simple bounding value.

Assumptions: The assumptions made in the risk-informed debris transport calculation with respect to blowdown debris capture include the following (13):

- It was assumed that the fines generated by the LOCA blast would be transported to upper containment in proportion to the volume of upper containment compared to the entire volume. This is a reasonable assumption since fine debris generated by the LOCA jet would be easily entrained and carried with the blowdown flow.
- It was assumed that a fraction of small piece debris would also be transported to upper containment in proportion to the relative volume. Each compartment/area where breaks may occur was individually analyzed to determine the percentage of small and large piece debris that would get transported to upper containment.

Acceptance Criteria: No acceptance criteria were used for the blowdown debris capture analysis.

Results: The analysis of debris capture during the blowdown phase is one aspect of the debris transport evaluation. The blowdown transport fractions as well as the transport fractions during other phases of

the event (washdown, pool fill, and recirculation) (13) are used as an input for the overall GSI-191 evaluation (12). The results of the blowdown transport analysis are shown in Table 2.5.22.

Table 2.5.22 – Blowdown transport fractions according to break location

Break Location	Debris Type and Size	Blowdown Transport Fractions		
		Upper Containment	Lower Containment	Remaining in Compartments
1. Steam Generator Compartments	Fines	70%	30%	0%
	Small LDFG	33-60%	13-25%	15-54%
	Large LDFG	0-22%	0%	78-100%
2. Reactor Cavity	Fines	70%	30%	0%
	Small LDFG	33-60%	13-25%	15-54%
	Large LDFG	0-22%	0%	78-100%
3. Below Steam Generator Compartments	Fines	70%	30%	NA
	Small LDFG	21-50%	50-79%	NA
	Large LDFG	0%	100%	NA
4. Pressurizer Compartment	Fines	70%	30%	0%
	Small LDFG	26-66%	11-28%	6-63%
	Large LDFG	16-26%	1-11%	63-83%
5. Pressurizer Surge Line	Fines	70%	30%	NA
	Small LDFG	3-36%	64-97%	NA
	Large LDFG	0%	100%	NA
6. RHR Compartments	Fines	70%	30%	0%
	Small LDFG	3-45%	1-19%	36-96%
	Large LDFG	0%	0-10%	90-100%
7. Annulus	Fines	70%	30%	0%
	Small LDFG	6-37%	13-25%	38-81%
	Large LDFG	0%	0%	100%

The types of debris that would be subject to the blowdown forces include Nukon, Microtherm, qualified coatings, and crud. The Nukon debris would fail as fines, small pieces, large pieces, and intact blankets. The Microtherm, qualified coatings, and crud debris would all fail as fine debris and would transport similar to the Nukon fines. Because the intact blankets would not transport readily, this debris was not included in the transport analysis.

Additional details on how the blowdown transport was incorporated in the overall GSI-191 evaluation are provided in Volume 3 (12).

Item 5.a.3: Washdown Transport

Method: The methodology for calculating debris holdup during the washdown phase is documented in an engineering calculation based on plant-specific features (locations of grating, etc.) and applicable test data (13). The test data that was used is documented in the DDTS (14).

During the washdown phase of a LOCA, debris would be transported down to the containment pool by operation of the containment spray system. Significant amounts of debris could, however, be captured on the concrete floors and grated areas above the containment floor as containment spray water transporting the debris drains through grating to reach the pool.

The debris remaining inside the steam generator compartments would also be washed toward lower containment by the spray flow as well as the break flow. However, some small piece debris and all of the large piece debris would be held up on grating.

The debris blown to upper containment would be scattered around. Therefore, a reasonable approximation of the washdown locations can be made based on the spray flow split in upper containment. As shown in Figure 2.5.16, 25% of the containment sprays were estimated to flow directly into the steam generator compartments, 28% were estimated to flow into the steam generator compartments via the refueling canal (21%) and cable tray chase (7%), and the remaining 47% of the sprays were estimated to flow into the annulus.

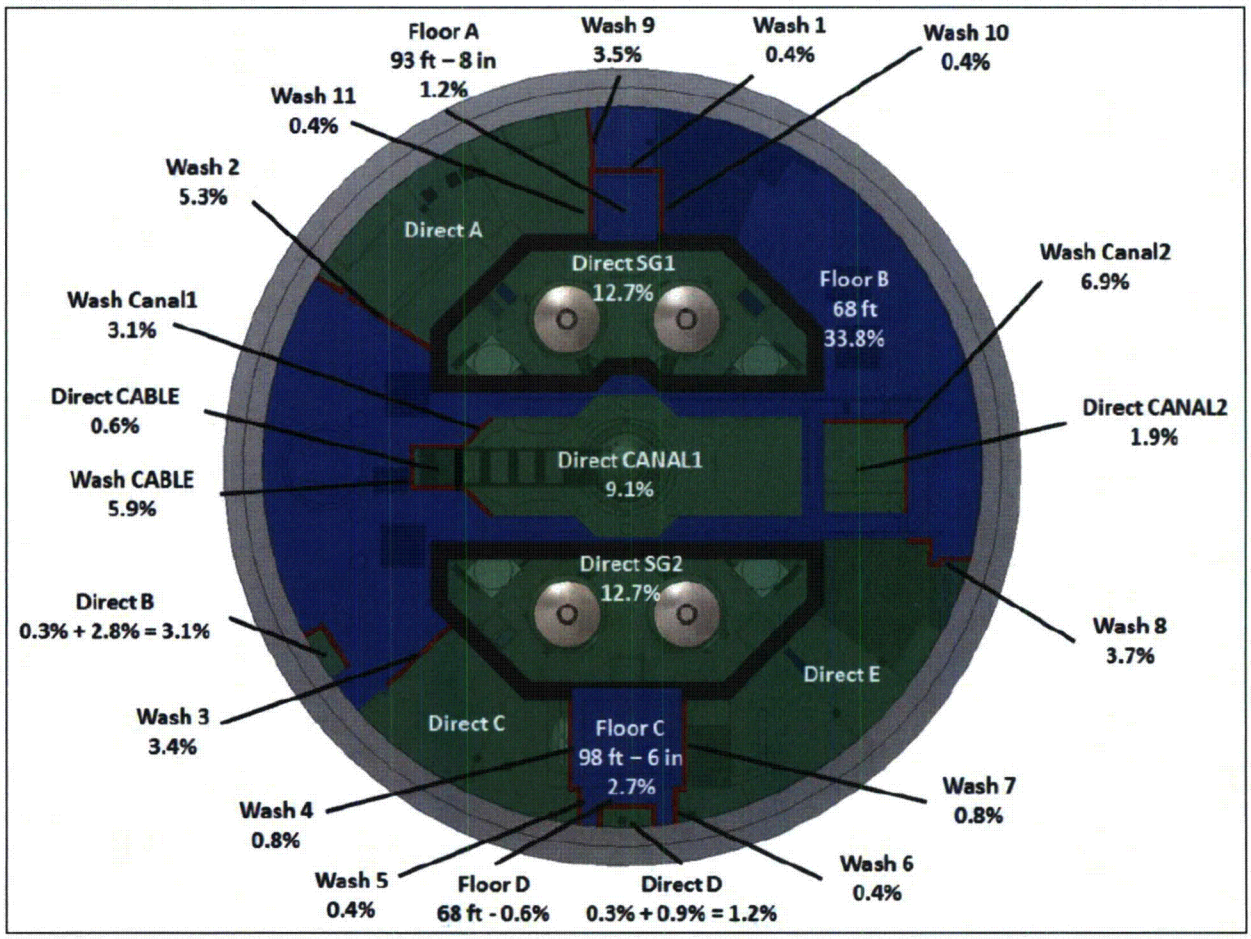

Figure 2.5.16 – 68' Elevation Spray Flow Distribution

The results of the DDTS testing showed that approximately 40-50% of small fiberglass debris landing on grating would be washed through the grating due to spray flows (16). Due to the fact that many of the flow paths to the containment pool would pass through multiple levels of grating, it was assumed that 0-25% of small pieces would be held up on each additional grating level as shown in the following equation. It was conservatively assumed that 100% of fines would transport to the pool. Retention of debris on concrete floors was considered, but was not credited in the final analysis.

$$F_{wash} = F_{CS} \cdot F_{WG} \cdot (1 - F_{AG})^{(N_{gratings} - 1)}$$

where:

F_{wash} = fraction of debris washed down to lower containment
F_{CS} = fraction of debris washed down by containment sprays
F_{WG} = fraction of debris held up when washed through first level of grating
F_{AG} = fraction of debris held up when washed through additional grating
$N_{gratings}$ = total number of gratings debris would pass through

The small piece debris that is present on the operating deck was conservatively assumed to be washed down to lower containment without any retention on grating or structures, since the flow of water over the edge would be concentrated and may be strong enough to push the debris to lower containment through the grating. The small debris washed down in the annulus would pass through a maximum of 5 levels of grating, and a minimum of 1 level. Therefore, the transport fraction for small pieces of fiber washed down in the annulus would be 7% to 19% as shown in the following equations:

$$F_{wash\ annulus\ small\ fiber\ max} = 0.47 \cdot 0.40 \cdot (1.00 - 0.00)^{(1-1)} = 0.19$$
$$F_{wash\ annulus\ small\ fiber\ min} = 0.47 \cdot 0.50 \cdot (1.00 - 0.25)^{(5-1)} = 0.07$$

The small piece debris that is present in the steam generator compartment would have to pass through only one level of grating to reach the pool, so the transport fraction for this debris would be 21% to 27% as shown in the following equations:

$$F_{wash\ SG\ comp\ small\ fiber\ max} = (0.25 + 0.28) \cdot (0.40)(1.00 - 0.00)^{(1-1)} = 0.21$$
$$F_{wash\ SG\ comp\ small\ fiber\ min} = (0.25 + 0.28) \cdot (0.50)(1.00 - 0.25)^{(1-1)} = 0.27$$

The large fiber debris would be washed down in the same locations as the small debris. However, since this debris would not pass through grating, the washdown fraction from upper containment through the annulus and inside the secondary shield wall would be 0%.

Containment sprays would not wash down any debris in the pressurizer compartment or RHR compartments, since the top of these compartments is blocked with a concrete roof or equipment hatches. Therefore, all of the debris remaining in these compartments at the end of blowdown will remain in the compartments during washdown.

Basis: The methodology used for the washdown analysis is similar to refined deterministic debris transport methods that have been used in the past. The retention fraction for the first level of grating is based on the DDTS results, and the retention fraction for each additional level of grating is based on engineering judgment (i.e., if a piece of debris passes through one level of grating, it is more likely to pass through a second level of grating, but still has a non-zero probability of being captured).

Assumptions: The following assumption was made in the risk-informed debris transport calculation with respect to washdown debris holdup (13):

- It was assumed that all debris blown upward would be subsequently washed back down by the containment spray flow with the exception of pieces of debris held up on grating. The fraction of debris washed down to various locations was calculated based on the spray flow split.

Acceptance Criteria: No acceptance criteria were used for the washdown debris holdup analysis.

Results: The analysis of debris holdup during the washdown phase is one aspect of the debris transport evaluation. The washdown transport fractions as well as the transport fractions during other phases of the event (blowdown, pool fill, and recirculation) (13) are used as an input for the overall GSI-191 evaluation (12). The results of the washdown transport analysis are shown in Table 2.5.23.

Table 2.5.23 – Washdown transport fractions according to spray initiation

Sprays Initiated?	Debris Type	Washdown Transport Fraction	
		Washed Down in Annulus	Washed Down inside Secondary Shield Wall
Yes	Fines	47%	53%
	Small LDFG	7-19%	21-27%
	Large LDFG	0%	0%
No	All	0%	0%

The washdown transport fractions do not depend on the location of the break, but only whether sprays are initiated. Since unqualified coatings debris may fail later in the event, this debris would only be washed down to the pool if the sprays are initiated and the coatings fail before the sprays are secured.

Additional details on how the washdown transport was incorporated in the overall GSI-191 evaluation are provided in Volume 3 (12).

Item 5.a.4: Debris Distribution

Method: The methodology for determining the non-uniform debris distribution at the start of recirculation is documented in an engineering calculation based on plant-specific features and careful consideration of break locations, flow paths, and debris types and sizes (13).

Since the various types and sizes of debris transport differently during the blowdown, washdown, and pool fill-up phases, the initial distribution of this debris at the start of recirculation could vary widely.

Insulation debris on the pool floor would be scattered around by the break flow as the pool fills, and debris in upper containment would be washed down at various locations by the spray flow. Due to the fact that the containment pool does not flow preferentially in any given direction after the inactive and sump cavities have been filled and before recirculation begins, it was assumed that the debris washed down by containment sprays would remain in the general vicinity of the washdown locations until recirculation starts.

Basis: The methodology used for determining the initial debris distribution is very similar to the refined deterministic debris transport methods that have been previously approved by the NRC (17). The primary difference is that a more realistic distribution was used for pieces of debris blown to lower containment rather than automatically assuming that these pieces would be preferentially distributed toward the sump strainers.

Assumptions: The following assumptions were made in the risk-informed debris transport calculation with respect to the debris distribution at the beginning of recirculation (13):

- With the exception of latent debris washed to the sump and inactive cavities during pool fill-up, it was conservatively assumed that all latent debris is in lower containment, and would be uniformly distributed in the containment pool at the beginning of recirculation. This is a conservative assumption since no credit is taken for debris remaining on structures and equipment above the pool water level.
- It was assumed that the unqualified coatings outside the reactor cavity would be uniformly distributed in the recirculation pool. This is a reasonable assumption since the unqualified coatings are scattered around containment in small quantities (7).
- It was assumed that the debris washed down inside the secondary shield wall by the break and spray flow would be initially distributed inside the secondary shield wall. It was also assumed that the debris washed down outside the secondary shield wall would be initially distributed in the annulus. These are reasonable assumptions since the debris would be spread out to a certain extent, but there is no preferential pool flow direction during pool fill-up after the inactive and sump cavities have been filled.
- With the exception of debris washed directly to the sump strainers or to inactive areas, it was assumed that the fine debris that is not blown to upper containment would be uniformly distributed in the recirculation pool at the beginning of recirculation. This is a reasonable assumption, since the initial shallow flow at the beginning of pool fill-up would carry the fine debris to all regions of the pool.
- It was assumed that small pieces and large pieces of debris that are blown to the containment pool would be uniformly distributed inside the secondary shield wall for breaks inside the secondary shield wall. For breaks outside the secondary shield wall, it was assumed that small pieces of debris that are blown to the containment pool would be uniformly distributed outside the secondary shield wall, and large pieces would be distributed in the vicinity of the break location. This is a reasonable assumption since the small piece debris would be transported

easily with the blowdown and pool fill flows, and since the large piece debris is less readily transported, this debris is likely to remain in the proximity of the break location.

Acceptance Criteria: No acceptance criteria were used for the initial debris distribution.

Results: The initial debris distribution at the start of recirculation is used to determine the recirculation transport. The recirculation transport fractions as well as the transport fractions during other phases of the event (blowdown, washdown, and pool fill) (13) are used as an input for the overall GSI-191 evaluation (12). The initial debris distributions that were used to determine the recirculation transport fractions are shown in Figure 2.5.17 through Figure 2.5.22.

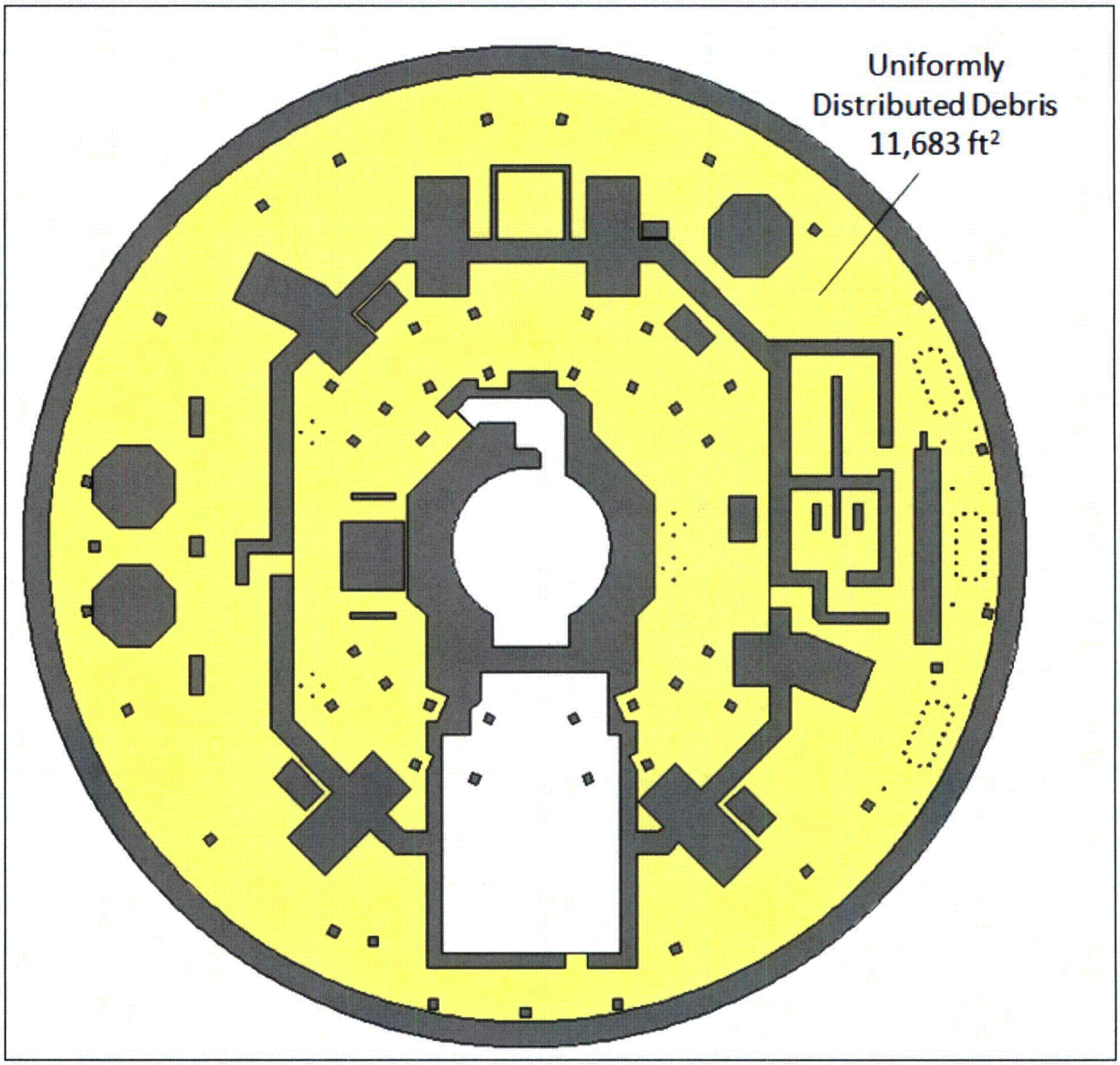

Figure 2.5.17 – Distribution of latent debris, unqualified coatings, and fines in lower containment

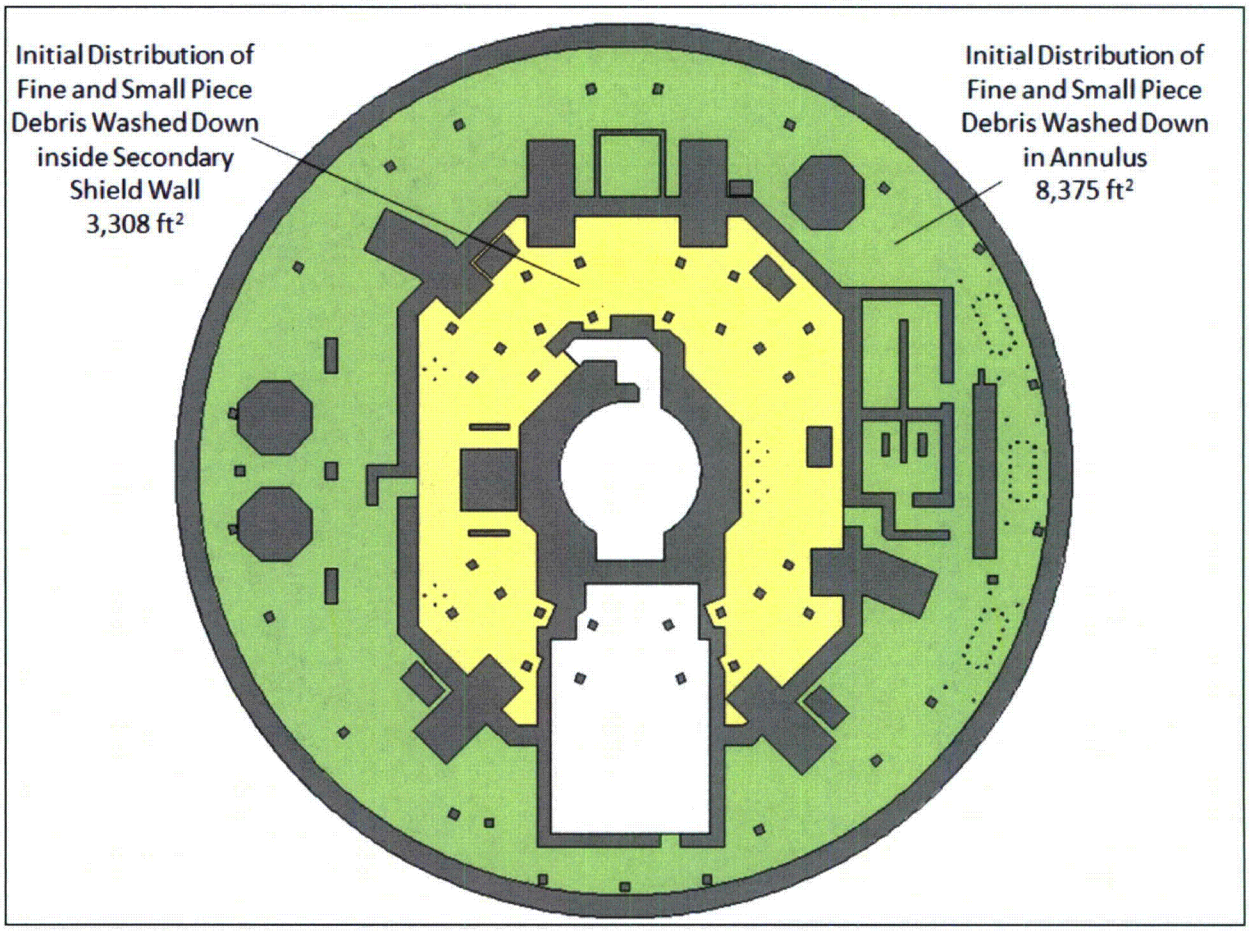

Figure 2.5.18 – Distribution of fines and small piece debris washed down from upper containment and the steam generator compartments

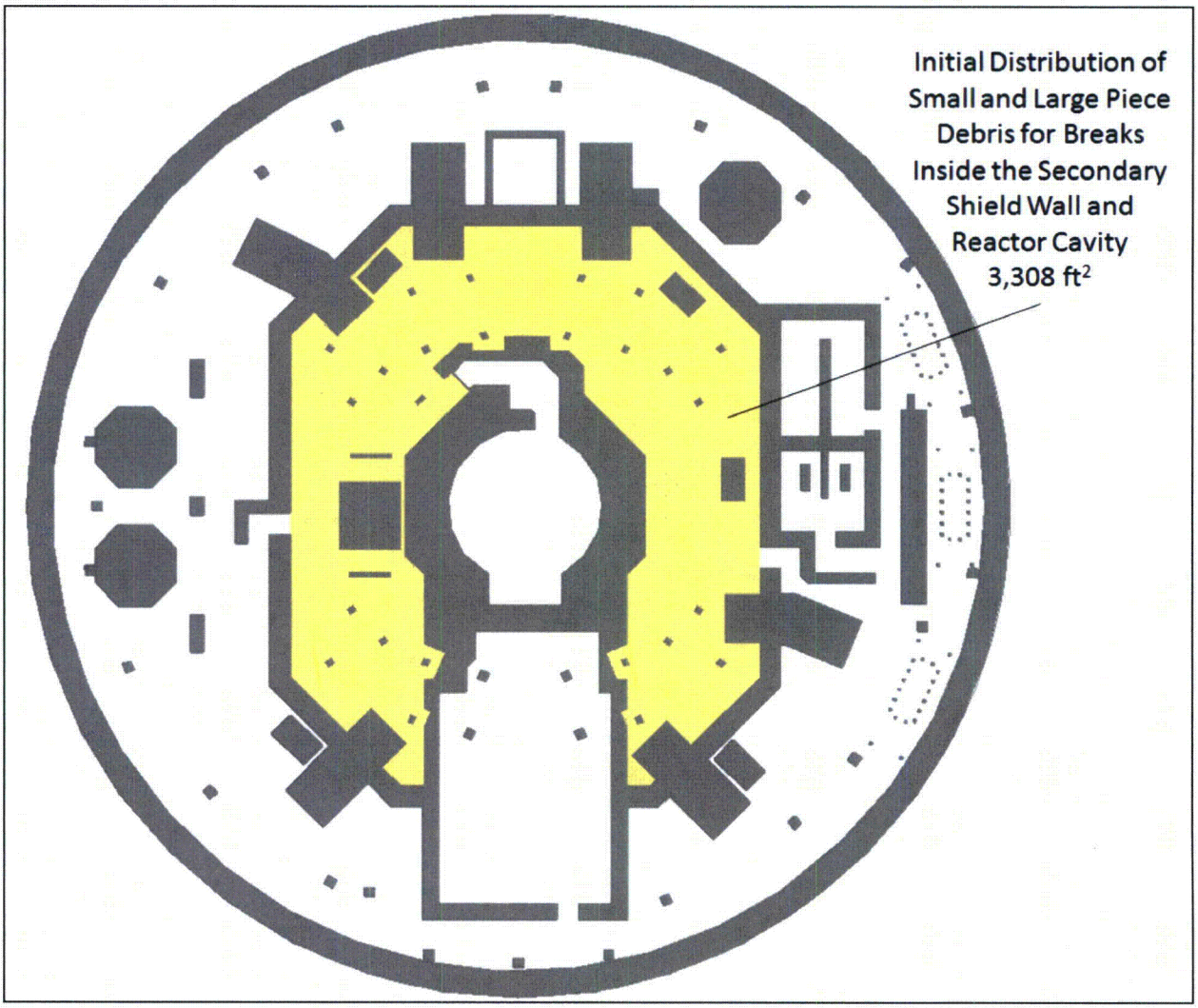

Figure 2.5.19 – Distribution of small & large piece debris in lower containment (breaks inside the secondary shield wall)

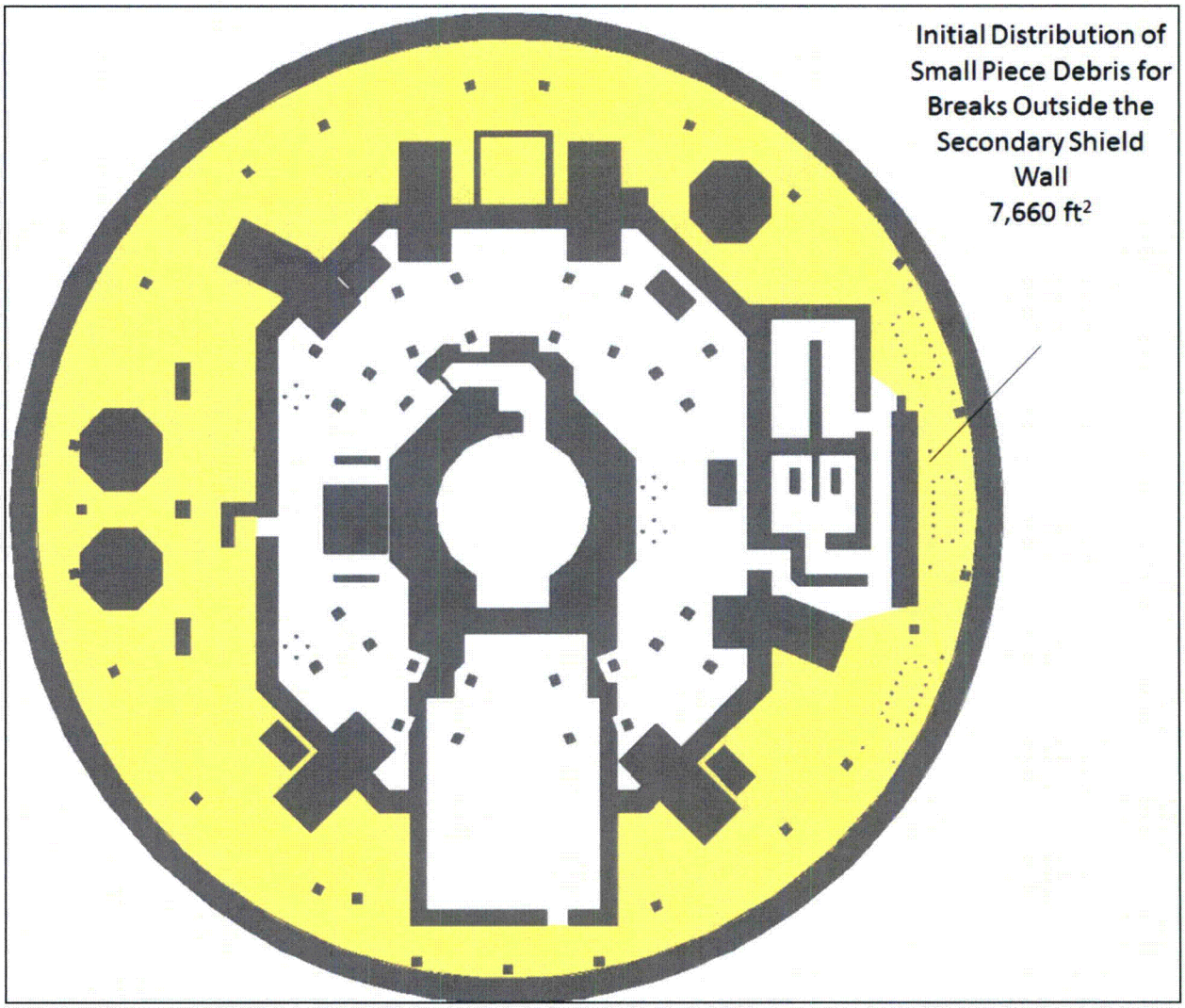

Figure 2.5.20 – Distribution of small piece debris in lower containment (breaks outside the secondary shield wall)

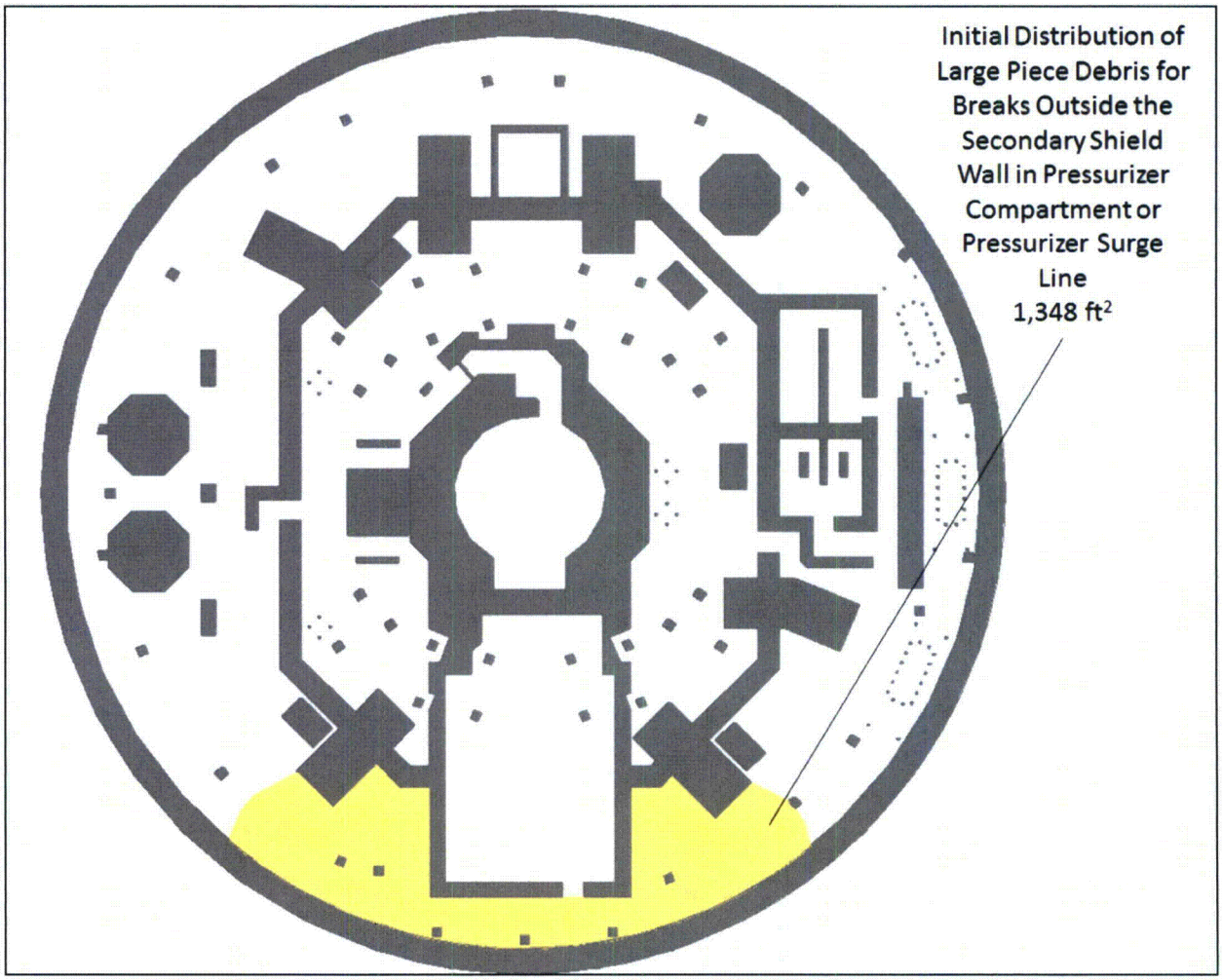

Figure 2.5.21 – Distribution of large piece debris in lower containment (breaks outside the secondary shield in pressurizer compartment and pressurizer surge line)

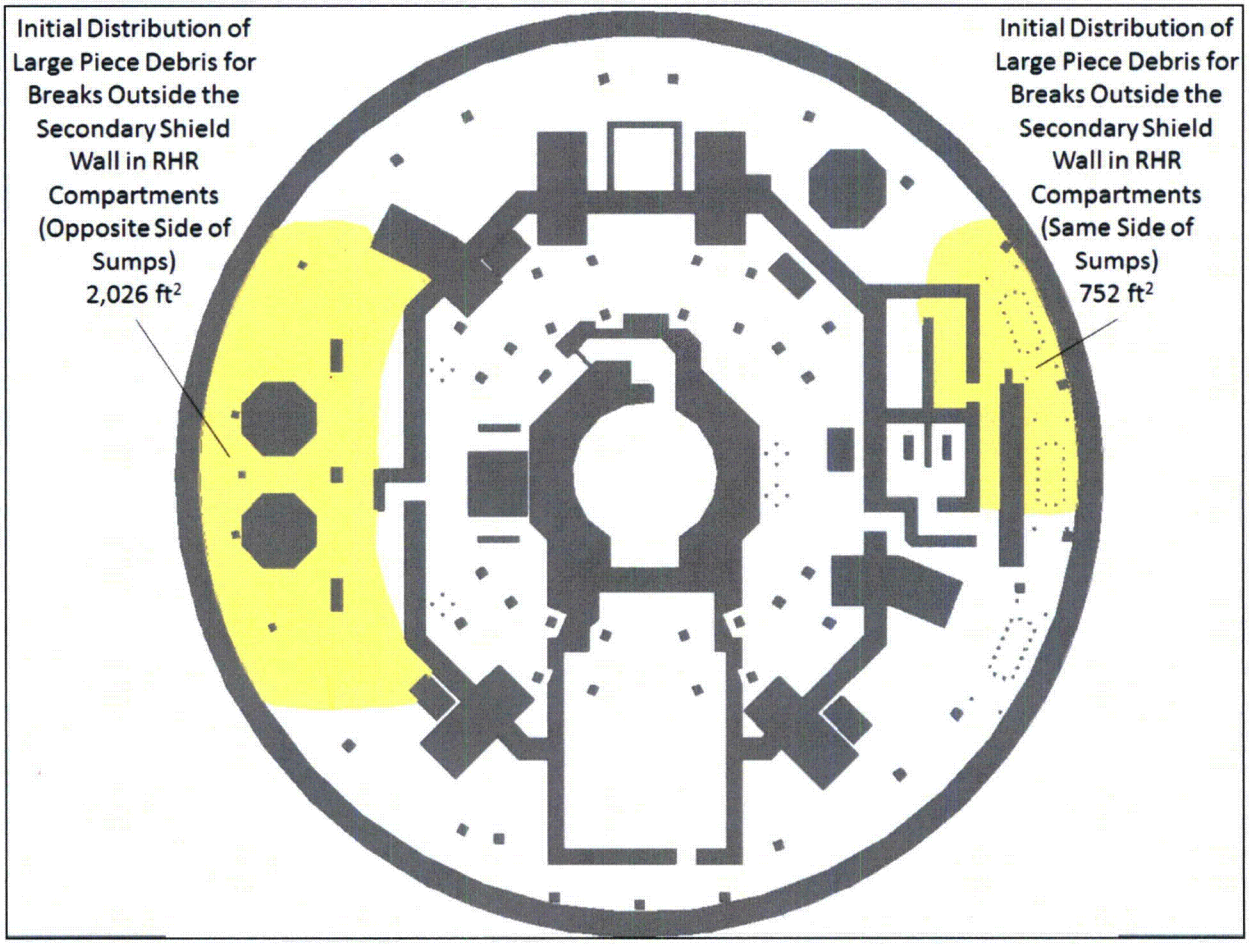

Figure 2.5.22 – Distribution of large piece debris in lower containment (breaks outside the secondary shield in RHR compartments)

Additional details on how the distributions are used to calculate the recirculation transport fractions are provided in the risk-informed debris transport calculation (13).

Additional details on how the recirculation transport fractions were incorporated in the overall GSI-191 evaluation are provided in Volume 3 (12).

Item 5.a.5: Time-Dependent Transport

Method: The methodology for determining time-dependent debris transport is documented in an engineering calculation based on plant-specific features and test data (13). The test data that was used for the time-dependent transport analysis includes the DDTS (14) and Alion erosion testing (18). The methodology includes the following steps:

1. The spray flow path of containment sprays to the pool was determined through the CAD model.

2. The areas and perimeters of each floor that the containment sprays came in contact with were calculated to determine how long containment spray flow would remain on a specific floor level, and if the velocities over the edge would be significant enough to transport debris from the operating deck to the pool level.
3. For each flow path, the number of gratings that flow would have to pass through was used to determine the fraction of debris that would reach the pool during washdown.

Basis: The following discussion provides a detailed description of how the methodology referred to above was used to develop the time-dependent transport (13).

The spray flow paths were determined in the CAD model, and are shown in Figure 2.5.23 through Figure 2.5.30.

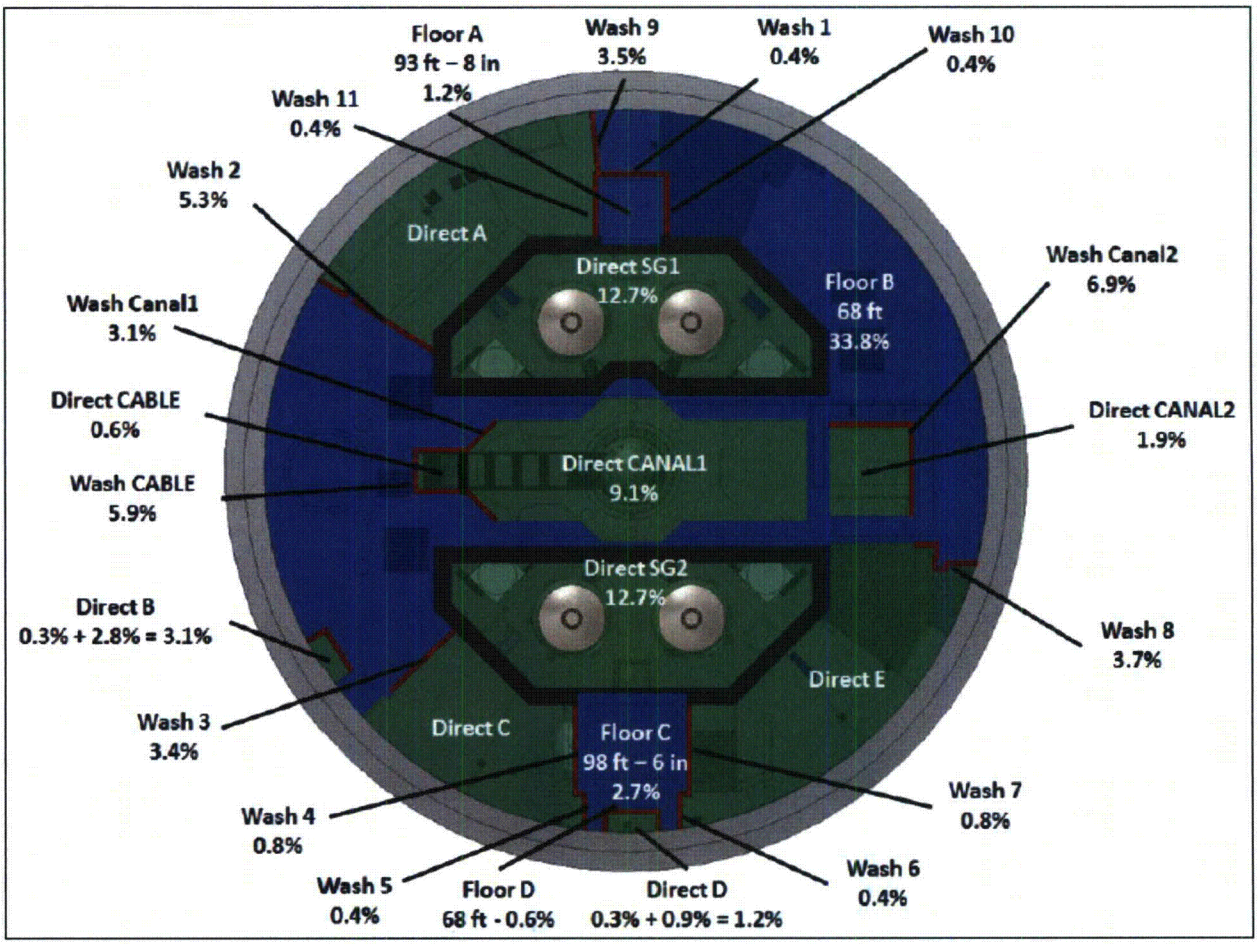

Figure 2.5.23 – 68' elevation spray flow distribution

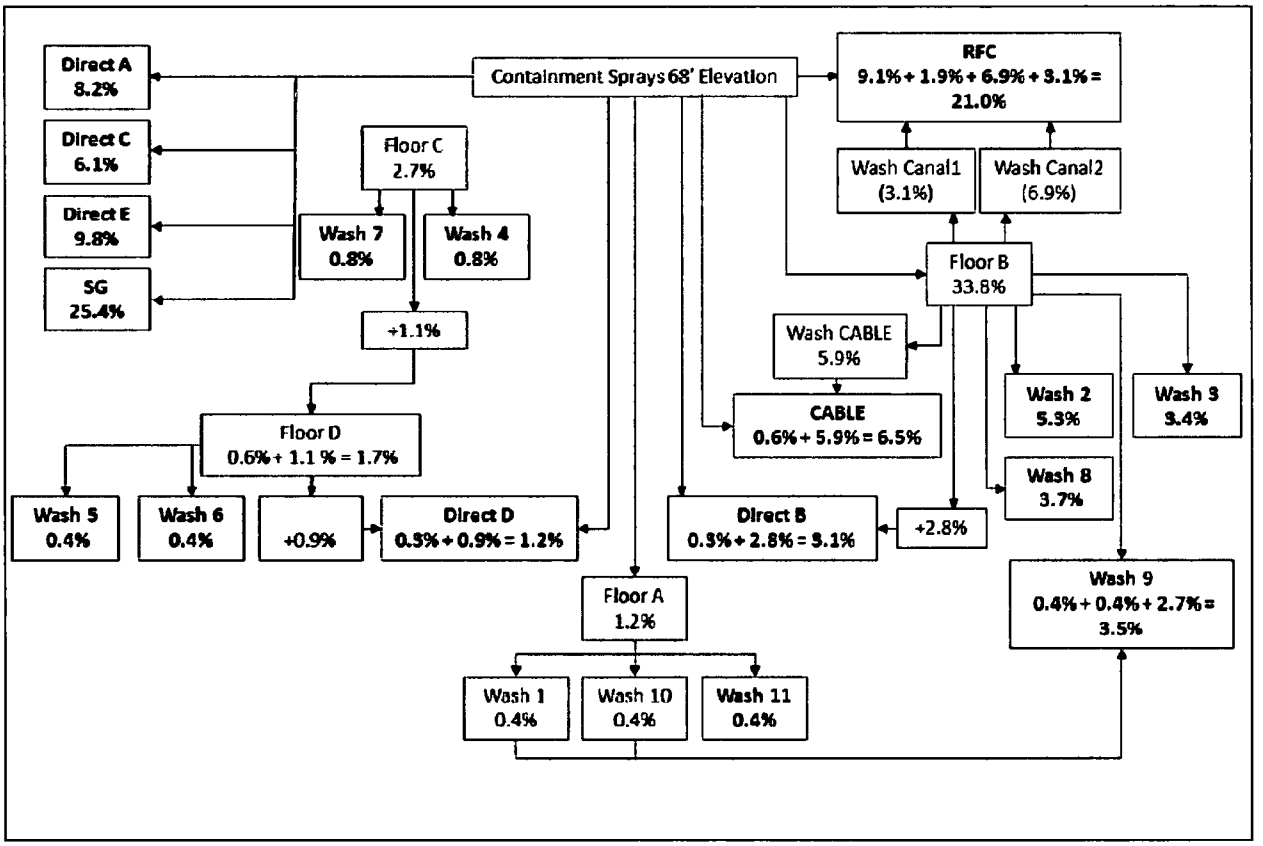

Figure 2.5.24 – 68' elevation spray flow flowchart

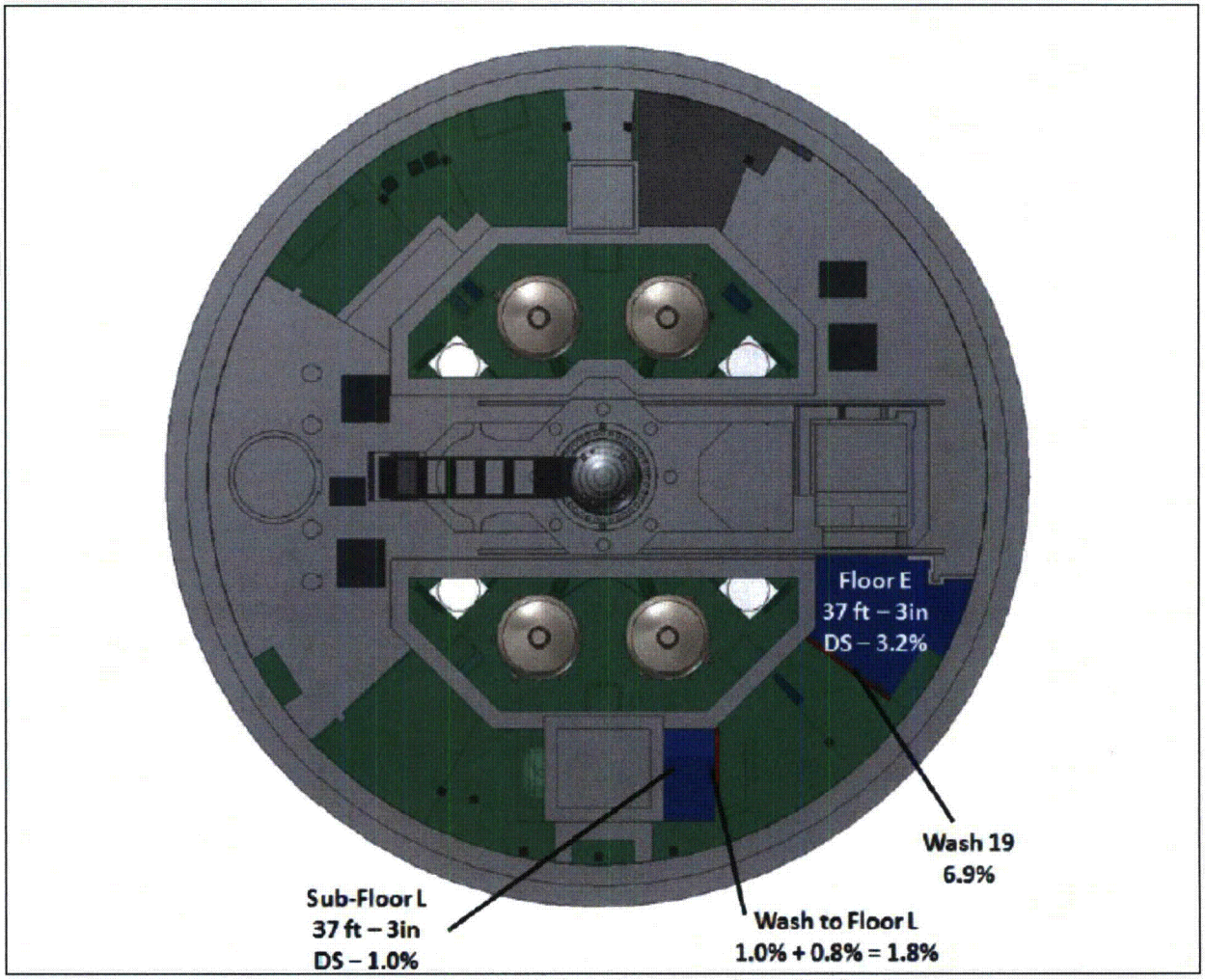

Figure 2.5.25 – 37'-3" elevation spray flow distribution

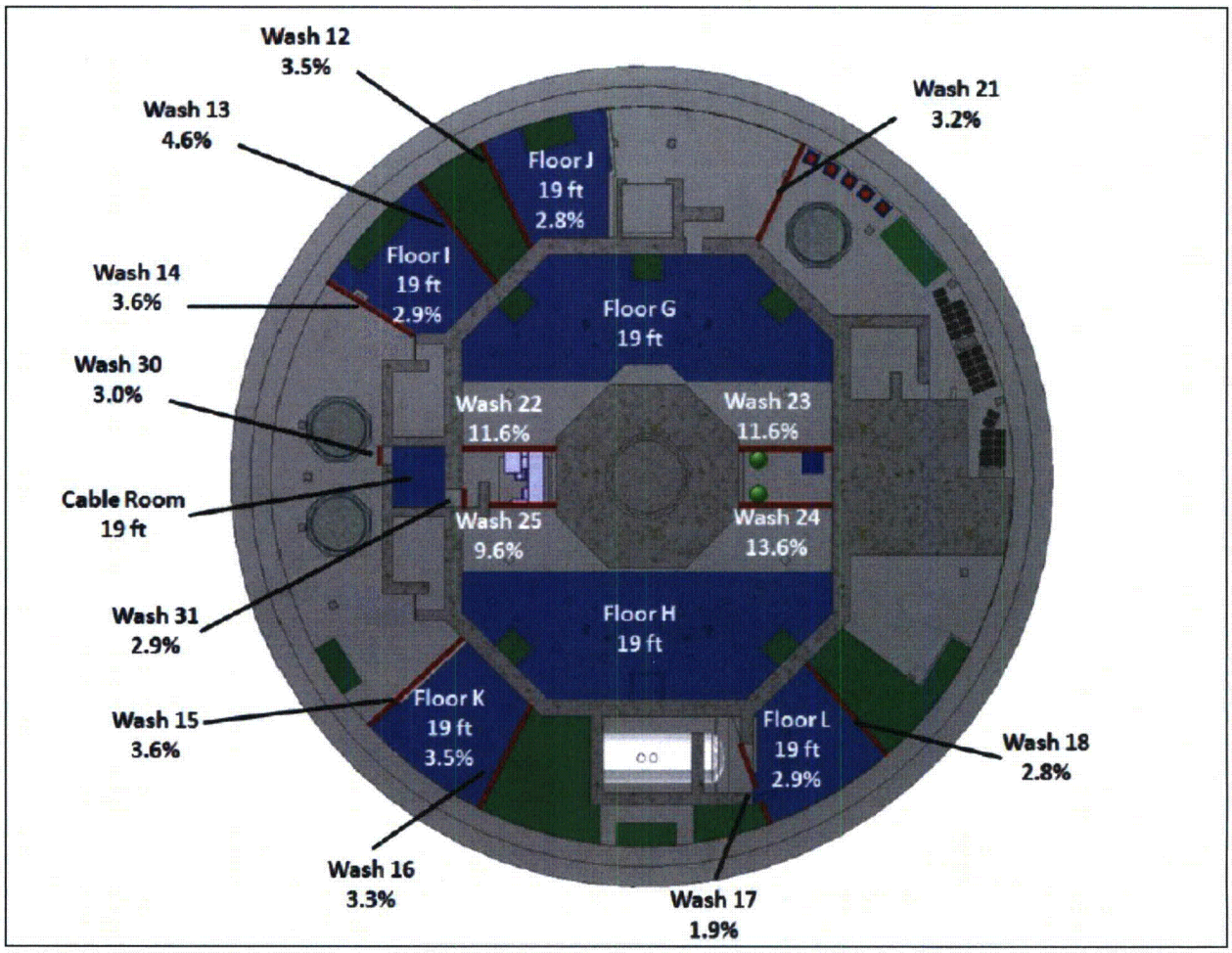

Figure 2.5.26 – 19' elevation spray flow distribution

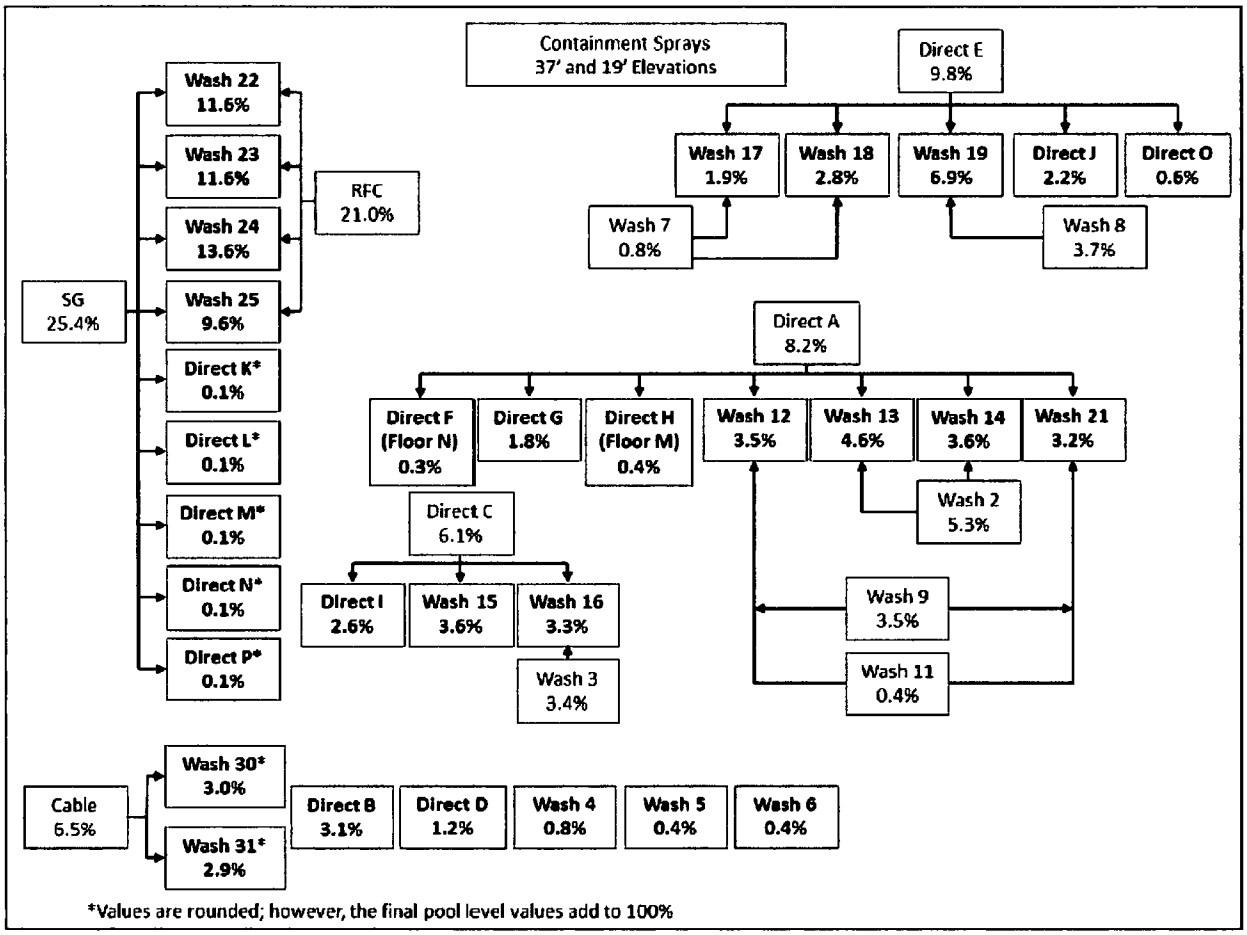

Figure 2.5.27 – 37'-3" and 19' elevation spray flow flowchart

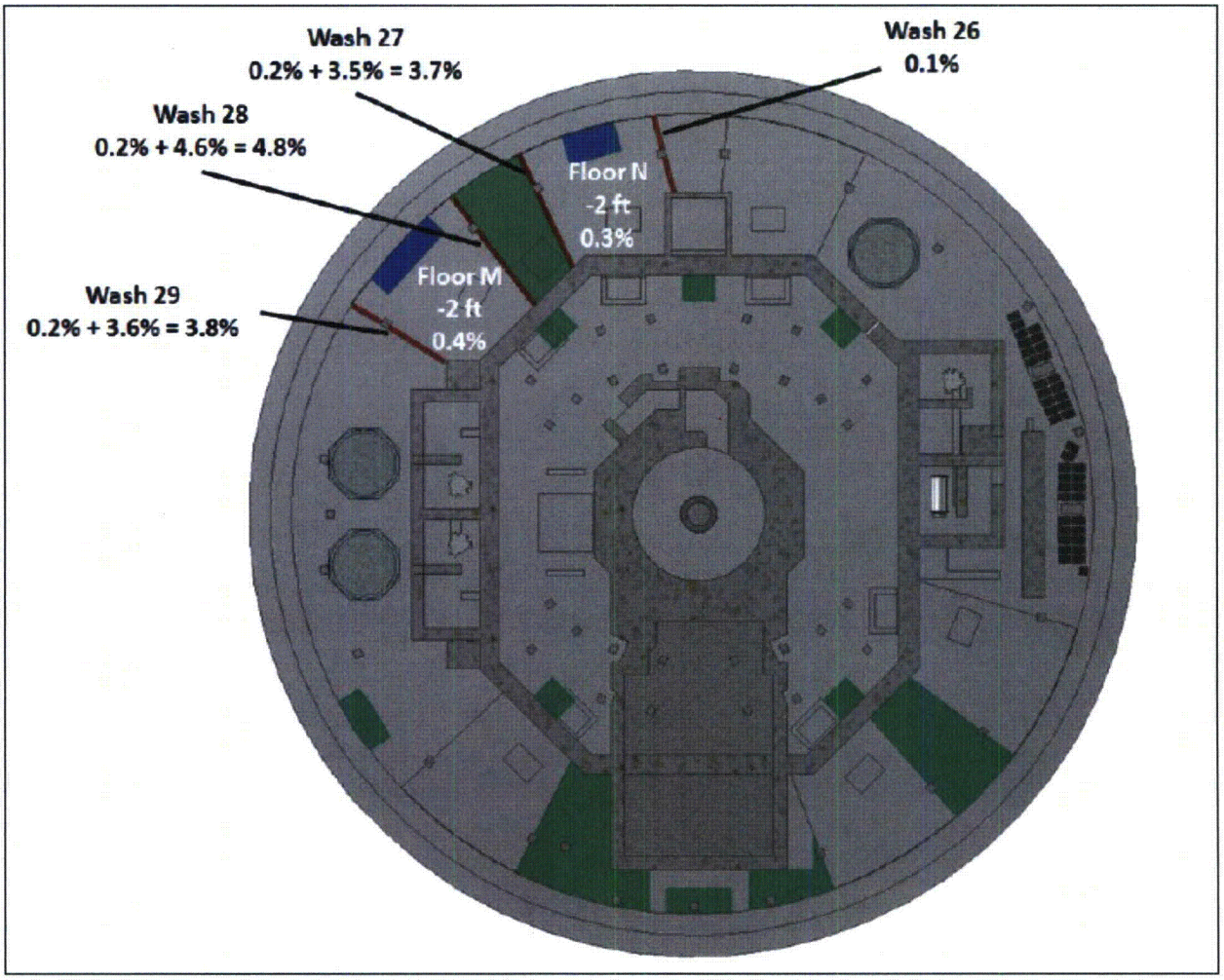

Figure 2.5.28 – (-)2' elevation spray flow distribution

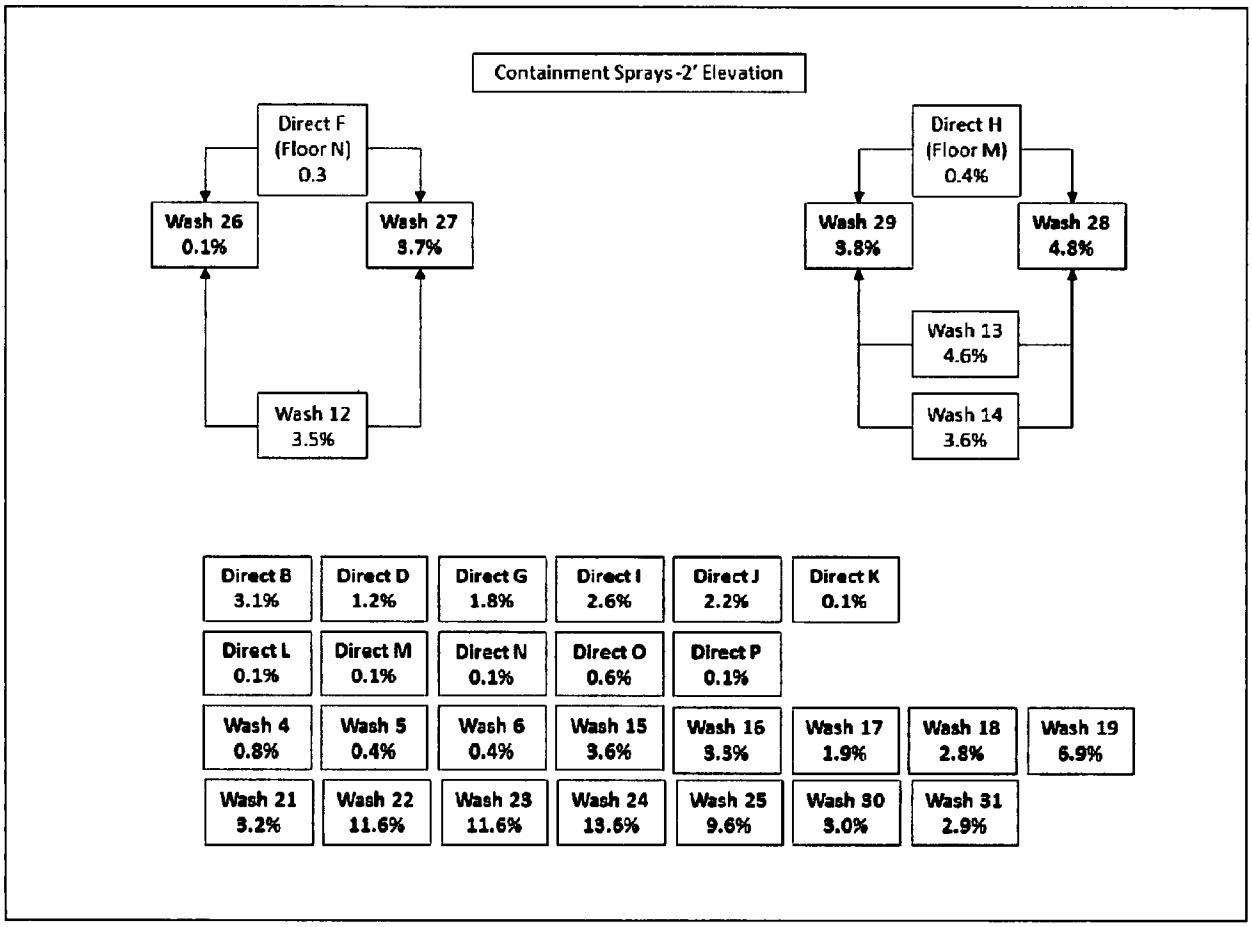

Figure 2.5.29 – (-)2' elevation spray flow flowchart

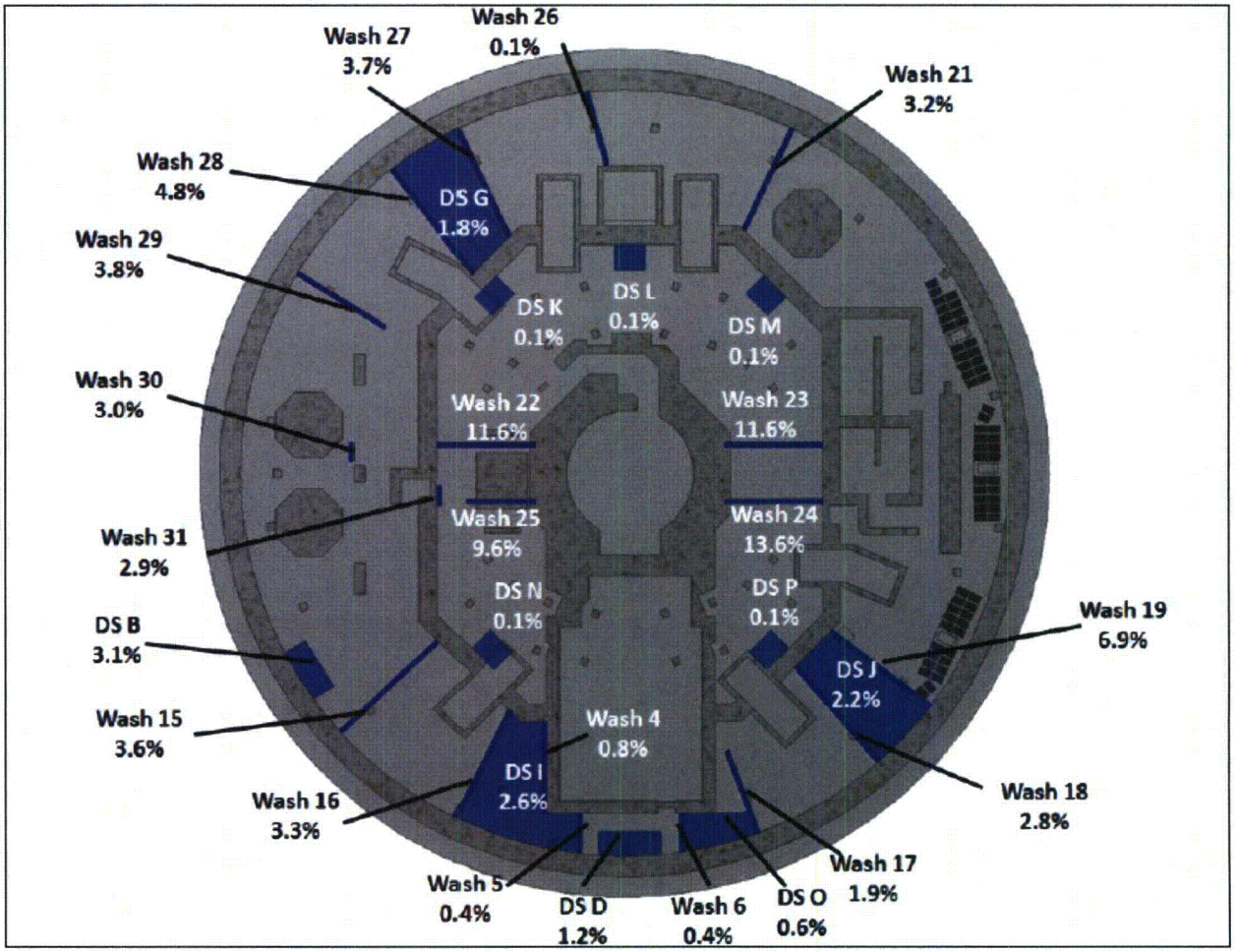

Figure 2.5.30 – Floor elevation spray flow distribution

During the washdown phase of a LOCA, debris would be transported down to the containment pool by operation of the containment spray system. Significant amounts of debris could, however, be captured on the concrete floors and grated areas above the containment floor as containment spray water transporting the debris drains through grating to reach the pool.

The containment sprays would drain to the pool via several flow paths over concrete decks and through grating at elevations 98'-6", 93'-8", 68', 52', 37'-3", 19', and (-)2'. The major flow split through the operating deck elevation (68') would be 24.3% to the refueling cavity, 74.5% off of the concrete operating deck into the annulus through grated openings, and 1.2% directly to the annulus through an opening without grating.

Since a percentage of the debris would land directly on concrete, it was necessary to determine whether it would be transported to the pool. The depth of water on the operating deck and subsequent concrete floors can be approximated as weir flow over a weir opening where the opening length is equal to the open perimeters. The following equation describes weir flow (19).

$$Q = 3.33 \cdot L \cdot H^{1.5}$$

Where Q is the flow rate, L is the perimeter of the floor, H is the height of water on the concrete, 3.33 is an experimentally obtained value with units of $ft^{1/2}/s$. The total perimeter around the operating deck (Floor B) is 198.05 ft. Since the total runoff flow to the operating deck for 3 train operation is 1,835 gpm (4.09 ft^3/s), the water height can be calculated as follows:

$$H = \left(\frac{Q}{3.33 \cdot L}\right)^{2/3} = \left(\frac{4.09\, ft^3/s}{3.33 \cdot 198.05\, ft}\right)^{2/3} = 0.034\, ft = 0.41\, in$$

Taking this depth, along with the flow perimeters and flow rate for 3 train operation, the average velocity on the operating deck would be approximately 0.61 ft/s.

The incipient tumbling velocity for small pieces of fiberglass is 0.12 ft/s (20). However, since this tumbling velocity is for 1" clumps of fiberglass completely submerged in water, the velocity required to tumble clumps of fiberglass sitting on the STP operating deck would be somewhat different since the depth of the water is not sufficient to fully submerge the debris pieces. Assuming that the small pieces of fiberglass on the operating deck are 1 inch clumps with dimensions of approximately 1" × 1" × ½", the clumps would be approximately 82% submerged in the 0.41" water level. As shown in the following calculations, the difference in the submergence level has a significant impact on the transportability of the fiberglass pieces.

The bulk density of Nukon is 2.4 lb_m/ft^3; the material density is 159 lb_m/ft^3 (21). Using the following porosity equation (22), along with an air density of 0.075 lb_m/ft^3 (23) gives the following porosity for Nukon:

$$\phi = \left(\frac{159\, lb_m/ft^3 - 2.4\, lb_m/ft^3}{159\, lb_m/ft^3 - 0.075\, lb_m/ft^3}\right) = 0.985$$

When saturated with water at 272 °F (density of 58.2 lb_m/ft^3), the bulk density of the fiberglass would be:

$$\rho_b = 159\, lb_m/ft^3 - 0.985\left(159\, lb_m/ft^3 - 58.2\, lb_m/ft^3\right) = 59.7\, lb_m/ft^3$$

The horizontal forces acting on the piece of fiberglass include the drag from the water flow (a function of the water velocity and the cross-sectional area of fiberglass), and the friction force between the fiberglass and concrete. The friction force is directly proportional to the normal force which is equal to the weight of the piece of fiberglass minus the buoyancy.

$$\sum F_{horizontal} = F\{velocity, area\} - F_{friction} = 0$$

$$F_{friction} = \mu \cdot N$$

$$N = Weight - Bouyancy$$

$$Weight = V \cdot \rho_b \cdot g$$

$$Bouyancy = V_{submerged} \cdot \rho_{water} \cdot g$$

Since the pieces of fiberglass on the STP operating deck would only be 82% submerged:

$$V_{submerged} = 82\% \cdot V$$

$$N_{submerged} = V \cdot \rho_b \cdot g - 82\% \cdot V \cdot \rho_{water} \cdot g = V \cdot g(\rho_b - 82\% \cdot \rho_{water})$$

And the ratio of the normal forces on a 82% submerged piece of fiberglass versus a fully submerged piece would be:

$$\frac{N_{Partially\ Submerged}}{N_{Fully\ Submerged}} = \frac{V \cdot g(\rho_b - 82\% \cdot \rho_{water})}{V \cdot g(\rho_b - \rho_{water})} = \frac{59.7\ lb_m / ft^3 - 82\% \cdot 58.2\ lb_m / ft^3}{59.7\ lb_m / ft^3 - 58.2\ lb_m / ft^3} = 8$$

Therefore, since the coefficient of friction between fiberglass and concrete would be constant, and the reduced cross-sectional area for a partially submerged piece of fiberglass can be conservatively neglected, an 8 times higher flow velocity would be required to tumble a piece of fiberglass that is 82% submerged compared to a piece of fiberglass that is fully submerged. Given that the incipient tumbling velocity for a fully submerged piece of fiberglass is 0.12 ft/s, a velocity of approximately 0.96 ft/s would be required to tumble the small pieces of fiberglass on the STP operating deck. Since this is more than 50% higher than the actual water velocity on the operating deck, the small fiberglass debris would not transport to the grated openings. This method can be applied to the other concrete floors between the operating deck and the containment floor as shown in Table 2.5.24.

Table 2.5.24 – Velocities and transportability for various floor levels

		1 TRAIN MAX		2 TRAIN MIN		2 TRAIN MAX		3 TRAIN MIN		3 TRAIN MAX	
FLOOR	ELEVATION	VELOCITY ft/s	TRANSPORT yes/no	VELOCITY ft/s	TRANSPORT yes/no	VELOCITY ft/s	TRANSPORT yes/no	VELOCITY ft/s	TRANSPORT yes/no	VELOCITY ft/s	TRANSPORT yes/no
A	93'-8"	0.27	NO	0.30	NO	0.33	NO	0.33	NO	0.36	NO
B	68'	0.49	NO	0.56	NO	0.60	NO	0.61	NO	0.66	NO
C	98'-6"	0.31	NO	0.35	NO	0.37	NO	0.38	NO	0.41	NO
D	68'	0.21	NO	0.25	NO	0.26	NO	0.27	NO	0.29	NO
E	37'-3"	0.61	NO	0.70	YES	0.74	YES	0.76	YES	0.82	YES
G	19'	0.73	YES	0.84	YES	0.89	YES	0.91	YES	0.98	YES
H	19'	0.77	YES	0.88	YES	0.94	YES	0.96	YES	1.03	YES
I	19'	0.49	NO	0.55	NO	0.59	NO	0.60	NO	0.65	NO
J	19'	0.46	NO	0.52	NO	0.56	NO	0.57	NO	0.61	NO
K	19'	0.45	NO	0.51	NO	0.55	NO	0.55	NO	0.60	NO
Sub Floor L	37'-3"	0.33	NO	0.38	NO	0.39	NO	0.41	NO	0.44	NO
L	19'	0.41	NO	0.47	NO	0.51	NO	0.52	NO	0.56	NO
M	(-)2'	0.17	NO	0.19	NO	0.21	NO	0.21	NO	0.24	NO
N	(-)2'	0.18	NO	0.20	NO	0.20	NO	0.20	NO	0.23	NO
CABLE	19'	0.84	YES	0.96	YES	1.02	YES	1.05	YES	1.13	YES

It is possible to approximate the time it takes for debris to transport from various floors to the pool level. It is simplest to split this time into two categories: the time it takes the debris to flow over concrete floors, and the time it takes to fall between floor levels.

The time it takes for water to drop between floor levels can be determined using the kinematic equations of motion with uniform acceleration, in conjunction with the terminal velocity of a water droplet (24). These times are not dependent upon flow rate and are listed in Table 2.5.25. Most notably, it takes 5.8 seconds for direct sprays to reach the pool level.

Table 2.5.25 – Falling times for containment sprays (s)

		Falling Elevation						
		143 ft	**98 ft 8 in**	**93 ft 8 in**	**68 ft**	**37 ft 3 in**	**19 ft**	**(-) 2 ft**
Landing Elevation	**98.67 ft**	2.0						
	93.67 ft	2.2						
	68 ft	3.0	1.5					
	37 ft 3 in	4.1	2.6	2.4	1.5			
	19 ft	4.7		3.0	2.1	1.1		
	(-) 2 ft	5.5						
	(-) 11 ft 3 in	5.8	4.2		3.2	2.1	1.5	0.8

The refueling canal and cavity have areas of hold-up that need to fill before sprays can wash through the canal drains to the 19' elevation. As these cavities are fairly large, the time to fill them would be considerably longer than the time it takes for sprays to follow the other flow paths, and since there are multiple flow sources to these areas, it is reasonable to estimate the fill times using the aggregate flow rates and cavity volumes. Table 2.5.26 shows the hold-up times for different containment spray rates.

Table 2.5.26 – Cavity fill times

			1 TRAIN MAX		2 TRAIN MIN		2 TRAIN MAX		3 TRAIN MIN		3 TRAIN MAX	
	Hold-up Volume ft^3	**% CS Flow**	**Flow Rate (ft^3/s)**	**Fill Time (min)**	**Flow Rate (ft^3/s)**	**Fill Time (min)**	**Flow Rate (ft^3/s)**	**Fill Time (min)**	**Flow Rate (ft^3/s)**	**Fill Time (min)**	**Flow Rate (ft^3/s)**	**Fill Time (min)**
Lower Internals	605	11.50%	0.67	15.0	0.99	10.2	1.20	8.4	1.27	7.9	1.58	6.4
Refueling Cavity	5545	7.50%	0.43	214.9	0.65	142.2	0.79	117.0	0.83	111.3	1.03	89.7

A steady state volume for each floor can be determined using the floor area and the water depth previously calculated. The time it takes to wash fines from a concrete floor can be estimated by doubling the time it takes to fill this steady state volume. That time is doubled so that there is time to fill the floor with enough water to spill over the edge, and then enough time for that entire volume of water to be replaced once. These times are tabulated in Table 2.5.27 for fine debris and in Table 2.5.28 for small piece debris. It should be noted that the spray flow from the reactor cavity was not included in the flow to Floors G & H, as the time of holdup in the cavity was much longer than the direct spray.

Table 2.5.27 – Time for fine debris to wash from specific concrete floors

			1 TRAIN MAX		2 TRAIN MIN		2 TRAIN MAX		3 TRAIN MIN		3 TRAIN MAX	
	Percent CS	Concrete Area (ft^2)	Flow Rate (ft^3/s)	Average Time (s)	Flow Rate (ft^3/s)	Average Time (s)	Flow Rate (ft^3/s)	Average Time (s)	Flow Rate (ft^3/s)	Average Time (s)	Flow Rate (ft^3/s)	Average Time (s)
Floor A	1.20%	197	0.07	37.1	0.10	33.0	0.13	30.3	0.13	30.3	0.17	27.5
Floor B	33.80%	6004	1.96	134.8	2.91	118.0	3.54	110.6	3.74	108.8	4.65	101.2
Floor C	2.70%	472	0.16	51.4	0.23	45.6	0.28	42.8	0.30	41.9	0.37	39.0
Floor D	0.60%	98	0.03	25.3	0.05	21.2	0.06	20.0	0.07	19.1	0.08	18.3
Floor E	6.90%	594	0.40	100.1	0.59	88.0	0.72	82.3	0.76	80.8	0.95	75.0
Floor G	23.20%	2184	1.34	157.1	2.00	137.6	2.43	128.9	2.57	126.5	3.19	117.8
Floor H	23.20%	2146	1.34	171.4	2.00	150.0	2.43	140.6	2.57	137.9	3.19	128.4
Floor I	8.20%	512	0.48	45.9	0.71	40.2	0.86	37.6	0.91	37.0	1.13	34.4
Floor J	6.70%	1082	0.39	105.4	0.58	92.5	0.70	86.9	0.74	85.1	0.92	79.3
Floor K	6.90%	614	0.40	55.6	0.59	48.7	0.72	45.5	0.76	44.8	0.95	41.6
Sub Floor L	1.00%	179	0.06	59.7	0.09	52.0	0.10	50.4	0.11	48.9	0.14	45.0
Floor L	4.70%	505	0.27	58.4	0.40	51.0	0.49	47.8	0.52	46.8	0.65	43.5
Floor M	0.40%	563	0.02	146.0	0.03	127.3	0.04	115.5	0.04	115.5	0.06	101.3
Floor N	0.30%	540	0.02	157.0	0.03	140.7	0.03	140.7	0.03	140.7	0.04	127.0
Cable Room	6.50%	146	0.38	49.4	0.56	43.4	0.56	40.7	0.72	39.9	0.9	37.0

Table 2.5.28 – Total time for small debris to transport to the pool

	Total Time to Pool		
Wash	1 TRAIN (min)	2 TRAIN (min)	3 TRAIN (min)
19	NA	1.8-10.5	1.7-9.7
22	8.1-56.4	6.9-49.5	6.3-45.6
23	8.1-56.5	6.9-49.6	6.3-45.7
24	8.9-57.2	7.4-50.2	6.8-46.1
25	8.9-57.2	7.4-50.2	6.8-46.1
30	2.8	2.3-2.5	2.2-2.3
31	2.8	2.3-2.5	2.2-2.3

Based on the DDTS testing, approximately 40-50% of small pieces of debris would pass through one level of grating (16). Due to the fact that many of the flow paths to the containment pool would pass through multiple levels of grating, it was assumed that 0-25% of small pieces would be held up on each additional grating level as shown in the following equation. It was conservatively assumed that 100% of fines would transport to the pool. For the purposes of calculating the transport fractions, it was assumed that the small pieces of debris passing through grating would be in the middle of the range, 45%.

$$F_{wash} = F_{CS} \cdot F_{WG} \cdot (1 - F_{AG})^{(N_{gratings} - 1)}$$

Where:

F_{CS} = fraction of debris washed down by containment sprays
F_{WG} = fraction of debris washed through first level of grating
F_{AG} = fraction of debris held up when washed through additional grating
$N_{gratings}$ = total number of gratings debris would pass through

Table 2.5.29 and Table 2.5.30 show the transport fractions of washdown to the pool for specific washes, and is separated into transport to the area inside the secondary shield wall and transport to the annulus.

Table 2.5.29 – Number of gratings and transport fractions for individual washes

Wash	Number of Gratings	Transport Fraction
19	5	0.94%
22	1	4.99%
23	1	4.99%
24	1	5.85%
25	1	4.13%
30	1	1.29%
31	1	1.25%
Total Inside SSW		21.21%
Total In Annulus		2.23%

Table 2.5.30 – Number of gratings and transport fractions for direct sprays

Direct Spray	Number of Gratings	Transport Fraction
B	4	0.56%
D	0	1.20%
G	5	0.24%
I	4	0.47%
J	5	0.30%
K	1	0.04%
L	1	0.04%
M	1	0.04%
N	1	0.04%
O	4	0.11%
P	1	0.04%
Total Inside SSW		0.20%
Total In Annulus		2.88%

Table 2.5.31 shows the ranges for the fraction of debris in upper containment and the steam generator compartments that would be expected to transport to the pool floor inside and outside the secondary

shield wall during 1, 2, and 3 Train operation. Note that washdown transport fractions for the latent debris and degraded qualified coatings outside the ZOI were not quantified.

Table 2.5.31 – Total washdown transport fractions

Debris Type	Fines	Small Pieces			Unjacketed Large Pieces	Jacketed Large Pieces
		1 Train	2 Train	3 Train		
LDFG	100%	26%	26%	25%-26%	0%	0%
Microtherm	100%	26%	26%	25%-26%	NA	NA
Qualified Coatings (inside ZOI)	100%	26%	26%	25%-26%	NA	NA
Unqualified Miscellaneous Coatings (outside ZOI)	NA	NA	NA	NA	NA	NA
Unqualified Epoxy in Reactor Cavity (outside ZOI)	NA	NA	NA	NA	NA	NA
Dirt/Dust	NA	NA	NA	NA	NA	NA
Latent Fiber	NA	NA	NA	NA	NA	NA

For time-dependent washdown, Figure 2.5.31 through Figure 2.5.34 summarize the time it takes for containment sprays to reach the pool.

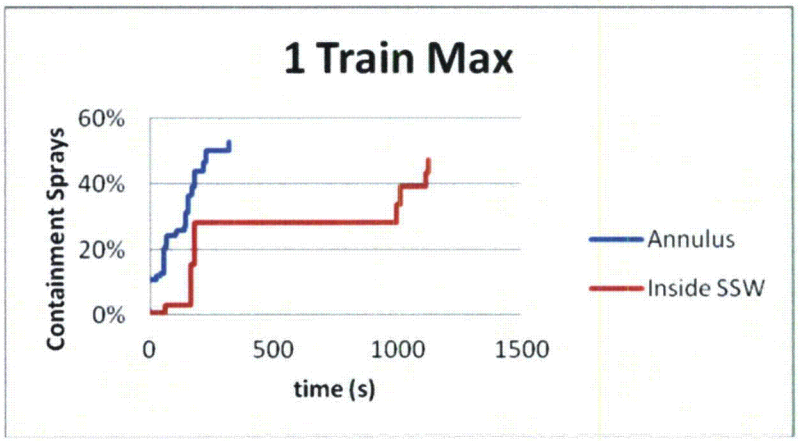

Figure 2.5.31 – Time for containment sprays to wash to pool for 1 train max operation

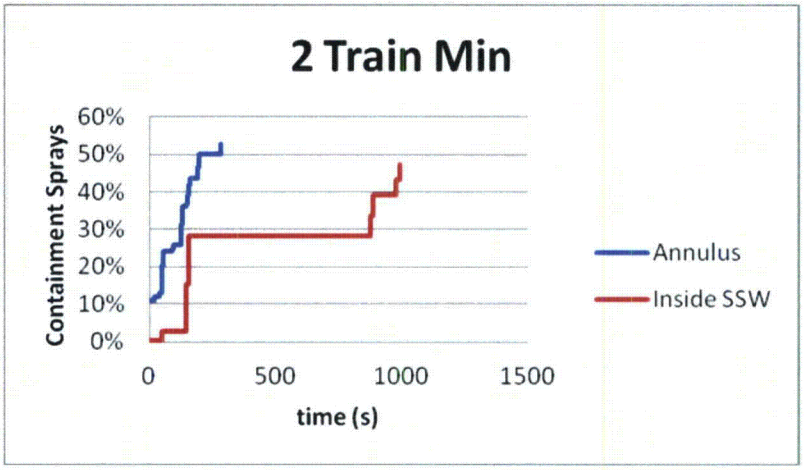

Figure 2.5.32 – Time for containment sprays to wash to pool for 2 train min operation

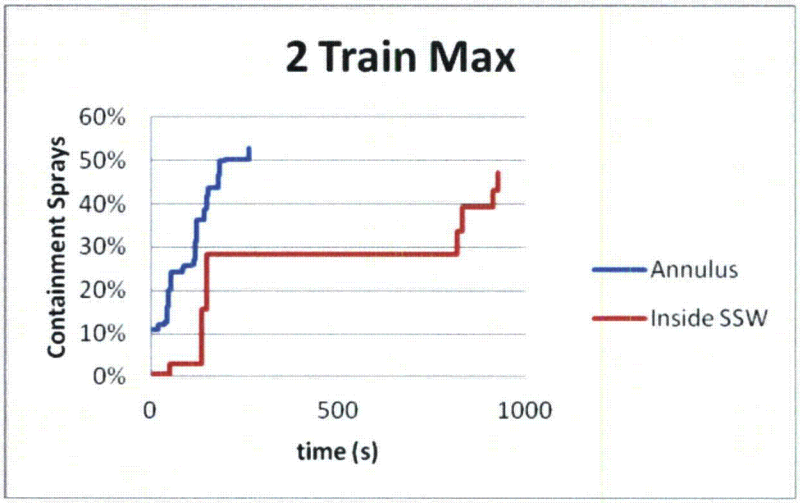

Figure 2.5.33 – Time for containment sprays to wash to pool for 2 train max operation

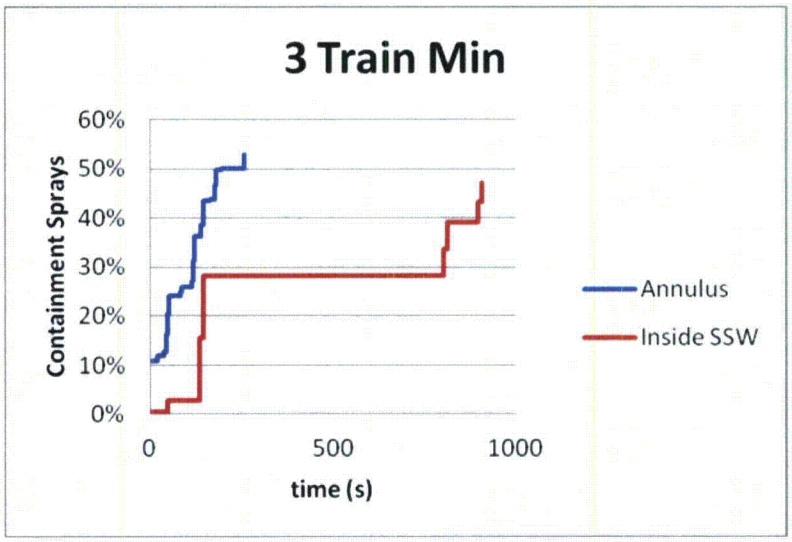

Figure 2.5.34 – Time for containment sprays to wash to pool for 3 train min operation

Assumptions: The following assumptions related to time-dependent transport were made in Volume 3 (12):

- It was assumed that debris washed down from upper containment reaches the pool after the inactive and sump cavities are filled, but before recirculation is initiated. This is a conservative assumption since it neglects transport of any washdown debris to inactive cavities during pool fill, but accelerates the time that debris would reach the strainer during the recirculation phase.
- It was assumed that unqualified coatings in upper containment would wash down to the pool immediately after failure if sprays are still on at the time of failure. This is a conservative assumption since it accelerates the time that debris would reach the strainer.
- It was assumed that the fine debris that is initially in the pool at the start of recirculation as well as the fine debris that transports to the pool during recirculation would be uniformly distributed in the pool. This is a reasonable assumption since the fine debris in lower containment prior to the start of recirculation would be well mixed in the pool as it fills, and the fine debris washed down from upper containment during recirculation would be well mixed due to the dispersed locations where containment sprays enter the pool.
- It was assumed that debris generated due to erosion by containment sprays would be transported to the pool prior to the start of recirculation. This is a conservative assumption since it accelerates the time that debris would reach the strainers.
- It was assumed that all debris that penetrates the strainer and bypasses the core (either through the containment sprays or directly out the break) would immediately be transported back to the containment pool. This is a conservative assumption since it neglects potential hold-up of debris in various locations and neglects the time that it would take for debris to transport through the systems and wash back to the pool.

Acceptance Criteria: No acceptance criteria were used for the time-dependent transport analysis.

Results: Evaluating time-dependent transport requires an analysis of several different factors. The results of the analysis are summarized in Table 2.5.32 and Figure 2.5.35.

Table 2.5.32 – Time-dependent transport

Source	Time or Equation	Comments
Inactive Cavity Fill	t = ~0 s (no curbs around inactive cavity entrances)	Assume only applies for debris blown to pool and latent debris
Sump Strainer Fill	t ~ 425 s (based on a flow rate of 14,040 gpm and a pool volume of 13,325 ft^3)	Assume only applies to debris blown to pool and latent debris
Total Fill (Switchover)	t ~ 20 min (LBLOCA)	
Initial Washdown	6 s – 1000 s (fines); 2 min – 50 min (small pieces)	Assume washdown occurs after inactive and sump cavities are filled, but before recirculation is initiated
Unqualified Coatings Failure	0 min – 30 days	Assume instant washdown at time of failure if sprays are on
Recirculated Spray Flow Debris Washdown	t ~ 300s	Assume instant washdown
Recirculated Break Flow Debris Washdown	t < 300s	Assume instant washdown
Spray Erosion Washdown	t < 15 min	Assume during pool fill
Pool Erosion Recirculation	0-30 days	
Initial Debris in Pool at start of recirculation (x_i)	x_i = blowdown + initial washdown debris	Total debris in pool from blowdown and initial washdown
Debris Recirculation Time (x(t))	$x(t) = x_i e^{-t(Q/Vpool)}$	Q = flow rate; V_{pool} = Pool Volume

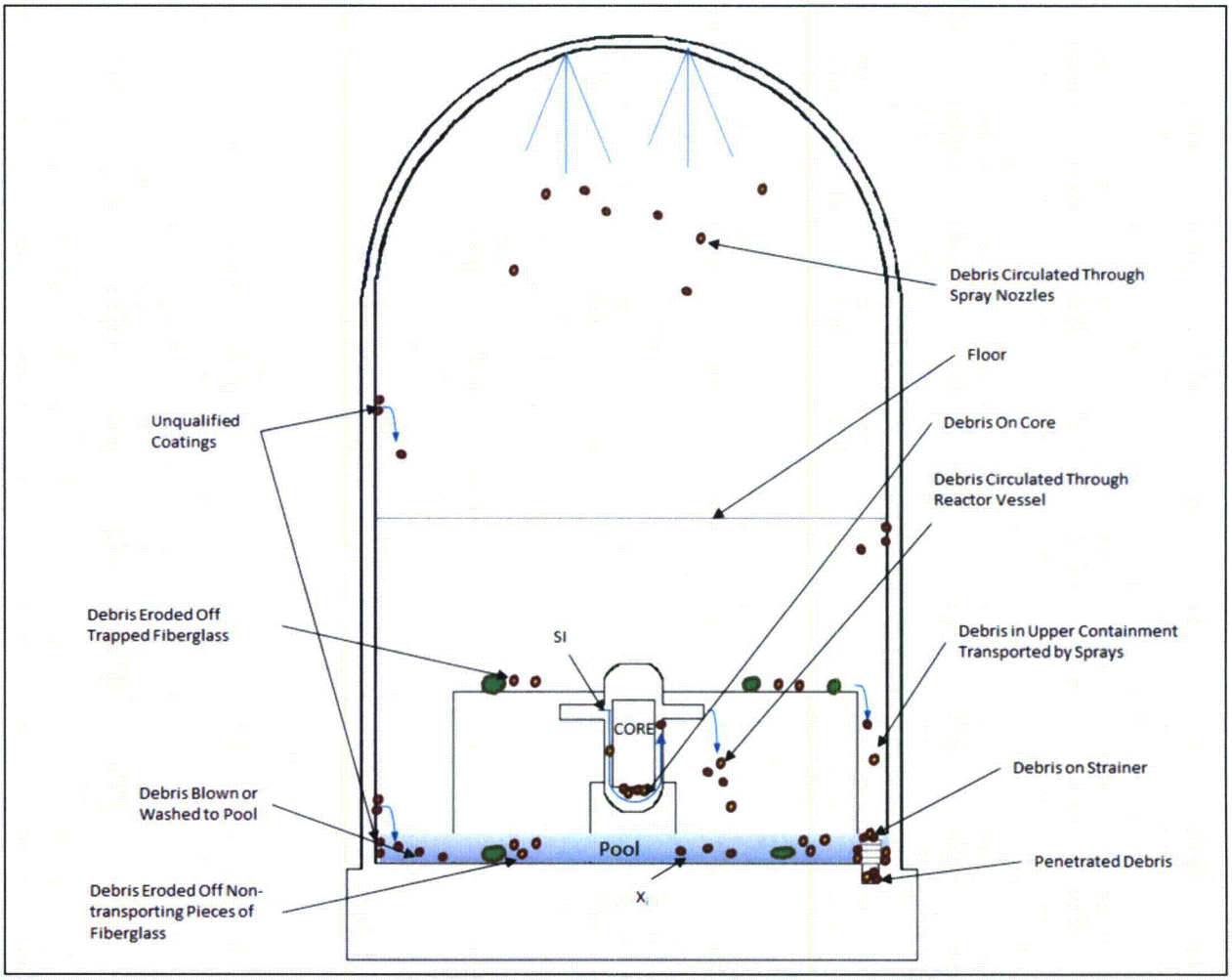

Figure 2.5.35 – Illustration of time-dependent transport

As shown in Table 2.5.32, the transient transport of debris in the pool at the start of recirculation can be calculated using the following equation. This equation can also be applied to debris that enters the pool later in the event (i.e. failed unqualified coatings or debris circulated through the break or containment sprays).

$$x(t) = x_i \cdot e^{-t\left(\frac{Q}{V_{pool}}\right)}$$

where:

x(t) = Time dependent arrival of debris at the strainer(s)
x_i = Initial quantity of debris in the pool at the start of recirculation
t = time
Q = Total sump flow rate
V_{pool} = Pool volume

Additional details on how the time-dependent transport was incorporated in the overall GSI-191 evaluation are provided in Volume 3 (12).

Item 5.a.6: Corrosion and Dissolution Model

Method: One of the goals of the STP chemical effects test program was to develop a new corrosion and dissolution release model specific to STP conditions. However, this model was not fully developed for the submittal. Instead, the material release for aluminum, silicon, and calcium was calculated for a series of scenarios and documented in an engineering calculation based on plant-specific conditions (25). The scenarios that were evaluated included small, medium, and large breaks ranging from 1.5 inches to 27.5 inches, a range of fiberglass debris quantities from 0 ft^3 to 2,385 ft^3, and a range of water volumes from 1,775,000 L to 2,255,000 L. The analysis also looked at a range of pH profiles from minimum to maximum conditions and a range of temperature profiles from nominal to maximum conditions. The total quantity of Al, Si, and Ca released for each scenario was determined based on the release equations developed through bench-top testing as documented in WCAP-16530-NP (26). The solubility limit for aluminum and calcium products was estimated for specific conditions based on limited thermodynamic modeling. These solubility limits were compared to the release quantities to determine which scenarios would result in a chemical product. The way this information was used to address chemical effects in the overall analysis is described in more detail in Volume 3 (12).

Basis: The approach for calculating the release of aluminum, silicon, and calcium is based on the standard deterministic methodology in WCAP-16530-NP, which has been approved by the NRC (27). 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 (28) 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; however only the aluminum hydroxide solubility limit (Log K of 10.8 (29)) was considered in this analysis since it was determined as a suitable substitute for sodium aluminum silicate in head loss testing (28). Calcium phosphate (Log K of -28.25 (29)) 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. Different temperature bounds were required for this evaluation because a decrease in temperature results in a decrease of aluminum product solubility over the given pH range as seen in Figure 2.5.36; while it produces an increase in calcium product solubility over the same pH range as seen in Figure 2.5.37. The temperature bound for aluminum product solubility was set at 140 °F (60°C) since this temperature has been used by U.S. nuclear power plants in past analyses. The temperature bound for the calcium product solubility was set at 185°F (85°C). The chosen bound was lower than the LOCA peak temperatures because these peaks occur over a very short duration (minutes) of a 30-day event and return to temperatures ≤185°F (85°C) for appreciable durations before declining (30; 31). Using this approach, the concentration of aluminum expected to result in formation of a chemical product is

approximately 4.9 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.

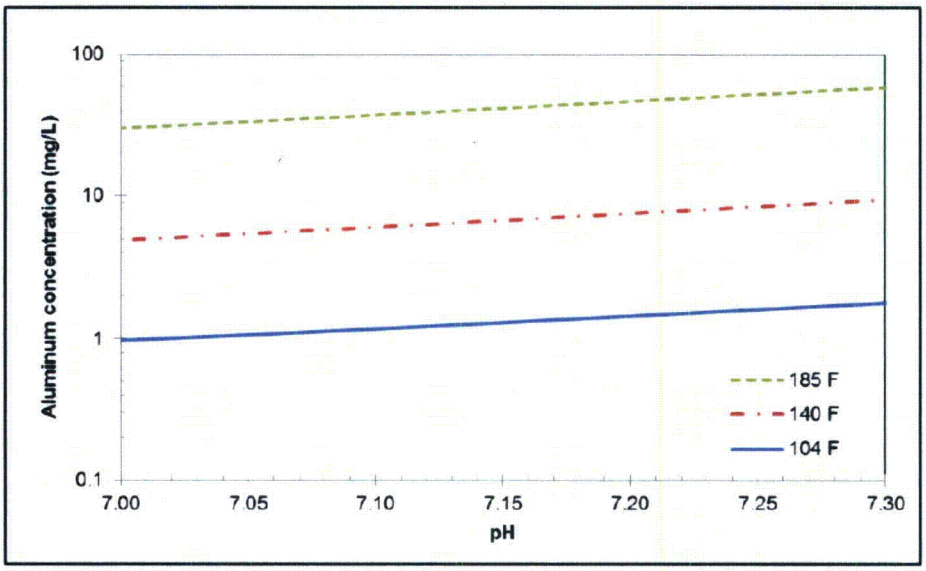

Figure **2.5.36** – Aluminum Hydroxide Solubility in Borated-TSP Solution

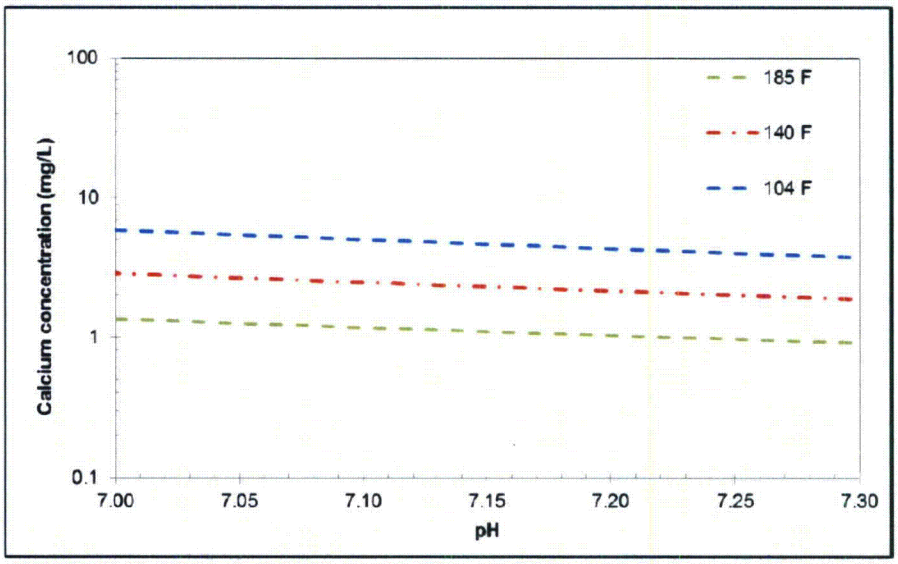

Figure **2.5.37** – Calcium Hydroxide Solubility in Borated-TSP Solution

Assumptions: The following assumptions were made as part of the chemical release analysis (25):

- The nominal temperature profiles, which were generated for the first 10 hours from the thermal-hydraulic analysis (32), were estimated from 10 hours to 30 days by linearly interpolating between the final simulation temperatures at 10 hours to a temperature of 110 °F at 30 days. This gives a conservatively high temperature profile for the majority of the event, which maximizes the total release quantities.
- 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 (33).

Acceptance Criteria: A chemical product was judged to form for scenarios where the aluminum concentration is greater than 4.9 mg/l or the calcium concentration is greater than 0.8 mg/l (25).

Results: For nominal temperature profiles, chemical products (aluminum and calcium precipitates) were not predicted to form for any of the small breaks evaluated. However, some of the medium and large break cases evaluated had total aluminum concentrations that were approximately equal to or slightly higher than the estimated solubility limits (25). The calcium concentration was relatively high for cases where a maximum fiberglass quantity of 2,385 ft^3 was assumed. However, for cases with 60 ft^3 of fiber or less, the calcium concentration was approximately equal to the solubility limit (25). As discussed in Volume 3, the quantity of fiberglass insulation debris generated was less than 10 ft^3 for 99.9% of the scenarios evaluated (12). This indicates that even if chemical products form for the nominal scenarios, the effects on strainer head loss would be relatively benign. An evaluation of the chemical concentrations for a maximum temperature profile, however, indicated that the concentration of aluminum would be significantly higher (on the order of 20 times greater than the nominal scenarios). It is possible that these scenarios could result in significant chemical head loss. However, the maximum temperature profiles were developed based on a highly unlikely scenario where the CCW temperature is at the maximum level, four out of six fan coolers fail to operate, and all of the RHR heat exchangers fail (32). Extreme temperature profiles like this have not been fully evaluated yet, so the current limited testing does not completely preclude the possibility that chemical products may form and arrive at a debris-laden strainer in sufficient quantity to cause unacceptable head loss. The detailed results of the chemical release analysis are shown in Table 2.5.33 for the nominal temperature profiles.

Table 2.5.33 – Nominal temperature profile material release results

Case	pH	Fiber Quantity	Water Volume	Break Size (in)	Ca (mg/l)	Si (mg/l)	Al (mg/l)	Product
1	Min	Min	Min	1.5	0.2	1.7	1.3	-
				2	0.2	1.7	3.5	-
				4	0.3	2.8	4.3	-
				6	0.3	2.8	1.7	-
				8	0.9	8.4	1.3	Ca only
				15	0.9	7.2	1.1	Ca only
				27.5	0.9	7.7	1.1	Ca only
2	Min	Min	Max	1.5	0.1	1.4	1.1	-
				2	0.1	1.4	2.9	-
				4	0.2	2.3	3.6	-

Case	pH	Fiber Quantity	Water Volume	Break Size (in)	Ca (mg/l)	Si (mg/l)	Al (mg/l)	Product
				6	0.2	2.3	1.5	-
				8	0.8	7.0	1.1	-
				15	0.8	6.2	0.9	-
				27.5	0.8	6.6	0.9	-
3	Min	Max	Min	1.5	0.3	2.9	1.4	-
				2	0.3	2.9	3.6	-
				4	0.9	8.4	4.5	Ca only
				6	0.9	8.4	1.8	Ca only
				8	30.0	161.5	2.6	Ca only
				15	25.0	41.6	1.5	Ca only
				27.5	30.0	72.4	1.7	Ca only
4	Min	Max	Max	1.5	0.3	2.4	1.1	-
				2	0.3	2.4	2.9	-
				4	0.8	7.0	3.8	-
				6	0.8	7.0	1.5	-
				8	25.0	154.5	2.3	Ca only
				15	25.0	37.5	1.3	Ca only
				27.5	25.0	65.9	1.5	Ca only
5	Max	Min	Min	1.5	0.2	1.7	1.6	-
				2	0.2	1.7	4.1	-
				4	0.3	2.8	5.0	Al only
				6	0.3	2.8	2.1	-
				8	0.9	8.4	1.5	Ca only
				15	0.9	8.0	1.3	Ca only
				27.5	0.9	8.4	1.3	Ca only
6	Max	Min	Max	1.5	0.1	1.4	1.3	-
				2	0.1	1.4	3.4	-
				4	0.2	2.3	4.2	-
				6	0.2	2.3	1.7	-
				8	0.8	7.0	1.3	-
				15	0.8	6.9	1.1	-
				27.5	0.8	7.0	1.1	-
7	Max	Max	Min	1.5	0.3	2.9	1.6	-
				2	0.3	2.9	4.1	-
				4	0.9	8.4	5.3	Ca and Al
				6	0.9	8.4	2.2	Ca only
				8	30.0	173.1	2.9	Ca only
				15	26.4	45.4	1.7	Ca only
				27.5	30.0	78.6	2.0	Ca only
8	Max	Max	Max	1.5	0.3	2.4	1.3	-
				2	0.3	2.4	3.4	-
				4	0.8	7.0	4.4	-
				6	0.8	7.0	1.8	-
				8	25.0	167.3	2.6	Ca only
				15	25.0	41.2	1.5	Ca only
				27.5	25.0	72.0	1.7	Ca only

The analysis was repeated using the maximum temperature profile for 6-inch breaks. As shown in Table 2.5.34, the release quantities were significantly larger for the higher temperature conditions based on the higher release rates.

Table 2.5.34 – Maximum temperature profile material release results

Case	pH	Fiber Quantity	Water Volume	Break Size (in)	Ca (mg/l)	Si (mg/l)	Al (mg/l)	Product
1	Min	Min	Min	6	0.3	2.8	37.0	Al only
2	Min	Min	Max	6	0.3	2.3	30.8	Al only
3	Min	Max	Min	6	0.9	8.4	37.6	Ca and Al
4	Min	Max	Max	6	0.8	7.0	31.3	Al only
5	Max	Min	Min	6	0.3	2.8	41.8	Al only
6	Max	Min	Max	6	0.3	2.3	34.9	Al only
7	Max	Max	Min	6	0.9	8.4	42.4	Ca and Al
8	Max	Max	Max	6	0.8	7.0	35.4	Al only

Additional details on the methodology and basis for the chemical release and product formation analysis are provided in the material release calculation (25).

Additional details on how this analysis was incorporated in the overall GSI-191 evaluation are provided in Volume 3 (12).

Item 5.a.7: Basis for Excluding any Plant Materials from Chemical Testing

The potentially relevant materials not included in the large break LOCA integrated chemical effects test (33) include copper, lead, carbon steel, Microtherm, alkyd coatings, and epoxy coatings. The basis for excluding these plant materials from the chemical effects testing is described below.

Copper: Various sources of copper are found in containment at STP. These sources include wiring, cables, and tubes of the fan coolers (34). While copper is present in STP containment, none of it will be submerged during a LOCA. In addition, significant quantities of the unsubmerged copper will be protected from spray impingement. Copper cable and wiring will not be subjected to spray as long as some insulation is in place. As a result of all these factors, copper was excluded from the long-term CHLE tests.

Lead: Lead exists in STP containment in two forms: (1) lead blankets in storage containers and (2) permanently installed lead blankets on piping. There are approximately 500 lead blankets (1 ft x 3 ft) in storage containers (45% are submerged and 55% not submerged) (34). The equivalent thickness for a lead sheet in the blanket is 3/16" (34). These lead blankets are stored in drums with holes to prevent them from floating if containment floods, but the sources of lead are sealed within vinyl-laminated nylon covers which provide a protection barrier between the material and pool solution.

The permanently installed lead blankets are sparsely present on only three pipes in containment. The probability that these blankets will be in the zone of influence is relatively low (34). Since the contribution of lead from the pipe insulation is not a likely occurrence in a LOCA, the probable contribution from this material to the pool solution was neglected.

Carbon Steel: Uncoated carbon steel is generally present in containment as structural supports. While there is a significant amount of carbon steel in containment, previous research found that carbon steel corrosion occurred in insignificant amounts (35). The ICET tests contained 0.15 ft^2/ft^3 of carbon steel, with 34 percent of the material submerged and 66 percent in the vapor space. The unsubmerged uncoated steel coupons had very little change in weight, with changes ranging from +1.3 to -0.4 g, compared to a mean pre-test weight of 1025 g. The submerged uncoated steel coupons in Test #1 (high pH) had a weight change of -23.3 g, but had very little weight change in the remainder of the tests (ranging from +1.4 to -1.1 g). In ICET Test #2, which corresponded most closely to the STP conditions, the unsubmerged coupons gained 1.3 g and the submerged coupons gained 1.4 g of weight. Iron concentrations remained nearly undetectable throughout the full duration of all the ICET tests. The highest concentrations of iron were less than 0.1 mg/L, during the first few days of ICET Test #3. Iron was undetectable during the entire ICET Test #2. Based on these results, uncoated carbon steel was not included in the CHLE tank tests.

Microtherm: Microtherm was excluded from the CHLE tank tests due to the relatively insignificant quantity in containment (36).

Alkyd Coatings: Alkyd coatings were excluded from the CHLE tank tests based on testing that indicated that the coatings would not play a significant role in the creation of chemical precipitates (6).

Epoxy Coatings: Similar to the alkyd coatings, epoxy coatings were excluded from the CHLE tank tests based on testing that indicated that the coatings would not play a significant role in the creation of chemical precipitates (26; 37; 38).

Item 5.a.8: Chemical Precipitation Model

The methodology, basis, and results for the chemical precipitation model are described in the response to Item 5.a.6.

Item 5.a.9: Chemical Effects Phenomena Identification and Ranking Table

The following discussion provides details of the specific PIRT issues and how they were addressed. The italicized text was copied directly from the NRC's March 2011 report describing each of the issues (39).

PIRT Item 1.1: RCS coolant chemistry conditions at break

The reactor coolant system (RCS) coolant chemistry varies over the fuel cycle. Boron concentrations vary from approximately 2,000 to 4,000 parts per million (ppm) at the beginning of the fuel cycle to approximately 50 ppm at the end of the fuel cycle. Therefore, the initial reactor water chemistry spewing out of the break and forming the containment pool will have variable boron concentration while the ratio

of lithium to boron is approximately constant. The two-phase jet emanating from the break is initially at 315 degrees Celsius (C) (599 degrees Fahrenheit (F)) and then cools to 120 degrees C (248 degrees F). The main concern raised by the peer reviewers relates to how variations in the initial RCS chemistry will affect the interaction with containment materials and whether these variations have been appropriately addressed. Variations may influence corrosion rates of metals, leaching of species from nonmetallic materials, formation of chemical precipitates, and ultimately, plant-specific chemical effects.

The following root issues are contained in this item:

1. The break jet impacts different materials, and chemistry variations may have different effects.
2. Boron concentration in the RCS fluid varies over the fuel cycle.
3. Lithium concentration in the RCS fluid varies over the fuel cycle.
4. The temperature of the water exiting the break varies over the duration of the event.

The blowdown phase is brief (less than a minute for large break conditions), so chemical effects issues associated with impact by the break jet are negligible. After the blowdown phase ends, the water from the break would simply spill into the pool with minimal contact of containment materials.

Boron concentration is an important factor for chemical effects—partly due to potential chemical reactions, and partly due to its effect on pH. Therefore, a realistic boron concentration should be used to determine a realistic pH level, and an appropriate concentration should be added at the start of an integrated test.

Lithium is not likely to be a major contributor to chemical effects since the concentration is generally low (ranging from negligible quantities to a few ppm). However, since it is relatively easy to include, a representative concentration of lithium should be added at the start of an integrated test.

Temperature is an important factor since it has a direct effect on corrosion rates and solubility limits. Therefore, a realistic analysis should take into consideration temperature variations over the duration of the event.

<u>STP Resolution</u>: Long-term (30-day) tests were run representing medium and large LOCA scenarios (31; 33). The quantities of materials in each test were determined from the quantities of materials present in STP containment that are impacted by each size break, as determined by break modeling.

The boron and lithium concentrations for each test were selected by determining the concentrations in the RCS, RWST, and accumulators at STP and calculating the concentration based on the contribution from each source for each LOCA scenario (40). The boron and lithium contributions from the RCS were based on time-averaged concentrations. The impact of higher and lower concentrations of boron and lithium on the pH of the system was evaluated using chemical equilibrium modeling (Visual MINTEQ). Based on the ranges observed over two operating cycles (fall 2009 through summer 2012), the pH would not be significantly affected by the variation in boron and lithium concentrations. The results of this evaluation showed that pH is controlled tightly within a range of approximately 7.1 – 7.3 (40).

The temperature for the two long-term tests was varied over the 30-day duration to match the temperature profile of the selected break scenarios (31; 33).

PIRT Item 1.2: pH Variability

The normal operating pH of the RCS is typically in the range of 6.9–7.4. The pH adjusted to 25 degrees C (77 degrees F) changes during the course of the fuel cycle from acidic at the beginning of the cycle to closer to neutral by the end of a fuel cycle. There are implications similar to those discussed in Section 1.1 of this report with respect to how pH variations may affect the interactions between containment materials and the post-LOCA environment. These variations may influence corrosion rates of metals, leaching of species from nonmetallic materials, formation of chemical precipitates, and ultimately plant-specific chemical effects.

The following root issue is contained in this item:

1. pH level in the RCS fluid varies over the fuel cycle.

pH is an important factor since it has a direct effect on corrosion rates and solubility limits. Therefore, a realistic analysis should take into consideration pH variations in the RCS fluid and the resulting impact on the overall pH in the pool.

STP Resolution: The issue of pH variability over the fuel cycle is addressed by the selection of boron and lithium concentrations in PIRT Item 1.1. Bench tests were run at pH values of 6.0, 7.2, and 7.7 to investigate the effects of pH variability (41).

PIRT Item 1.3: Hydrogen Sources within Containment

Dissolved hydrogen may play a significant role in the containment pool water chemistry. Hydrogen sources within the containment include the RCS inventory; the corrosion of metallic materials, including the reactor fuel cladding; and the Schikorr reaction. Containment pool reduction-oxidation (redox) potential is a function of the dissolved hydrogen resulting from these sources. Higher H_2 concentrations may decrease the redox potential. However, containment conditions are expected to foster H_2 evaporation, which could raise the redox potential. This issue could be important if H_2 concentrations have a significant effect on the redox potential in the post-LOCA containment water. The redox potential determines which materials will corrode or dissolve within the pool. A higher redox potential (i.e., more oxidizing) promotes metallic corrosion. As the concentration of dissolved constituents increases, so does the potential for solid species precipitation that could affect ECCS performance. The NRC or industry testing has not attempted to accurately simulate post-LOCA H_2 concentrations. However, the Schikorr reaction, by itself, may be beneficial by converting compounds that could form gelatinous-type chemical species into the mineral magnetite.

The following root issue is contained in this item:

1. Dissolved hydrogen may increase corrosion or dissolution of materials in the containment pool.

As discussed in the March 2011 review (39), H_2 is considered insignificant since there will be limited amounts of H_2 in solution, and higher concentrations could actually reduce potential corrosion.

STP Resolution: No action required.

PIRT Item 1.4: Containment spray CO_2 scavenging and CO_2/O_2 air exchange

Air entrainment within the containment pool beginning soon after the LOCA will cause carbon dioxide (CO_2) absorption within the containment pool. This entrainment increases the amount of CO_2, which could produce higher carbonate precipitate concentrations than would otherwise be present. These precipitates could also enhance nucleation and precipitation of other chemical species. Consequently, the air/liquid interactions within containment may increase the amount of chemical precipitates and degrade ECCS performance more than if these interactions were not considered.

The following root issue is contained in this item:

1. Dissolved carbon dioxide may result in carbonate precipitates such as $CaCO_3$.

This is more of an issue for plants that do not use TSP as a buffer since dissolved calcium can react with the TSP to form calcium phosphate precipitates. As discussed in the March 2011 review (39), tests that are open to the atmosphere would generally have a higher concentration of dissolved CO_2 than an air-tight containment. Therefore, although this is a potentially significant issue that should be considered for air-tight tests, no additional analysis is required for tests that are not air-tight.

STP Resolution: The STP containment contains sufficient air space that the pool solution will be in equilibrium with the CO_2 and O_2 in the atmosphere (chemically, the solution behaves as an open system). The CHLE loop is not air tight, and the lid will periodically be removed during sampling, so the tests will have sufficient opportunity for air to interact with the solution and will also behave as an open system. Calculations were performed to determine the dissolved gas concentrations that would be present in the solution in the STP containment, and it was determined that the concentration of carbonate in the CHLE tank would be slightly higher than it would be in the post-LOCA pool (40). No special provisions were performed to control the dissolved CO_2 and O_2 concentrations during the tests.

PIRT Item 1.5: Emergency Core Cooling System Injection of Boron

After a pipe break, RWST inventory with a boron concentration of approximately 2,800 ppm is injected into the RCS to cool the reactor core. This provides for a large boron source, which may affect chemical reaction products in the containment pool. Specifically, the boron source will serve as a pH buffer. This may influence corrosion rates of metals, leaching of species from nonmetallics, and ultimately formation of chemical precipitates.

The following root issue is contained in this item:

1. Boron concentration in the RWST will affect the pH in the pool.

Boron concentration is an important factor for chemical effects—partly due to potential chemical reactions, and partly due to its effect on pH. Therefore, a realistic boron concentration should be used to determine a realistic pH level, and an appropriate concentration should be added at the start of an integrated test.

STP Resolution: The concentration of boron used for the testing is addressed in PIRT Item 1.1.

PIRT Item 2.1: Radiolytic Environment

Radiolysis is the dissociation of molecular chemical bonds by a high energy radiation flux. The largest source of this radiation flux is the gamma radioactive decay of the reactor fuel. When the ECCS fluid passes through the reactor core, it is subjected to this radiation flux. Radiolysis reactions may change the pH of the ECCS containment pool, the fluid's redox potential, or both. Hence, chemical species which differ from those evaluated may form or the fluid may be more corrosive than that evaluated in all previous chemical effects testing.

The following root issues are contained in this item:

1. Radiolysis can affect pool pH through the creation of H_2O_2 and OH radicals.
2. Radiolysis can break down electrical cable insulation or dissolved nitrogen to form strong acids.

As discussed in the March 2011 report, the formation of H_2O_2 and OH radicals is not considered to be a significant issue based on previous analyses (39). The formation of strong acids due to the breakdown of cables may have a non-negligible impact on the long-term pH, and therefore should be considered. As discussed in the March 2011 report, one licensee determined that acid formation would reduce the pH by 0.2 (39).

STP Resolution: The quantity of acid formation due to radiolysis at STP has been calculated to be 0.8 mM of hydrochloric acid and 0.25 mM of nitric acid over a 30-day LOCA duration. This quantity of acid has been calculated to depress the pH by approximately 0.15 pH units based on the chemical system and buffers in STP during a LOCA (40). While 0.15 pH units may not be significant, a decrease in pH decreases the solubility of aluminum hydroxide precipitates, and may cause precipitation if the solution in the CHLE tests is near the precipitation threshold. Therefore, the quantity of acid as projected by calculations for STP was included over time in the CHLE tests (42; 43).

PIRT Item 2.2: Radiological Effects: Corrosion Rate Changes

Radiolysis of water bearing the chloride ion (Cl^-) can elevate the post-LOCA corrosion rate through formation of hypochlorite (ClO^-) or hypochlorous (HOCl) acid. The presence of these acids could increase the corrosion rate of metallic and nonmetallic species in containment, which in turn could alter the chemical byproducts formed. Hence, the chemical precipitates that form could differ from those previously evaluated. These different precipitates could subsequently affect ECCS performance in a manner that has not been considered previously.

The following root issue is contained in this item:

1. Radiolysis of water with chloride ions can create strong acids.

Chloride ions may be in solution primarily due to the breakdown of electrical cable insulation, but also due to potential leaching from coatings. As discussed for Item 2.1, the formation of strong acids may have a non-negligible impact on long-term pH, and therefore should be considered.

STP Resolution: The addition of acid to the tests to simulate radiolysis is addressed in PIRT Item 2.1.

PIRT Item 2.3: Hydrolysis

Nickel oxide (NiO), as well as other oxides, resulting from the corrosion of stainless steel and Alloy 600 metals can become a catalyst for producing H_2 from radiolysis of water. This process occurs more readily at higher water temperatures (i.e., hydrothermal environments). The hydrothermal hydrolysis of various organic/inorganic coating and insulation materials could partially depolymerize polymeric materials, producing materials ranging from small molecules to colloids. The colloids could subsequently aggregate into larger particles and gels. If this were to occur, the aggregated depolymerized materials may be more likely to transport to the sump strainer and affect pump performance or create chemical precipitates with different characteristics than those evaluated.

The following root issue is contained in this item:

1. Hydrolysis may cause H_2 formation.

As discussed in the March 2011 report (39), hydrolysis is a chemical reaction that causes water molecules to split into hydrogen and hydroxide ions. Hydrolysis is more significant at higher temperatures (generally above boiling). Since the containment pool temperature would only be above 200°F for a few hours, and the formation of H_2 due to hydrolysis is a gradual process, this is an insignificant issue.

STP Resolution: No action required.

PIRT Item 2.4: Conversion of N_2 to HNO_3

One panelist was concerned about the effects of nitric acid (HNO_3) formed in the containment pool due to radiolysis of dissolved nitrogen (N_2). This panelist was mostly concerned that the HNO_3 concentration may overwhelm the buffering capacity and cause the containment pool pH to drop precipitously to a range within 1–3. If the containment pool pH were this acidic, the redox potential becomes strongly oxidizing and corrosive and would lead to significant metallic corrosion and leaching of inorganic ions from other materials (e.g., concrete). Most previous NRC and industry-sponsored research has evaluated the chemical effects and their implications associated within the neutral-to-alkaline pH range (i.e., 7–10) that is expected within the buffered post-LOCA containment pool. Therefore, if the containment pool pH were highly acidic (i.e., 1–3), the chemical effects that would occur may differ significantly from those previously evaluated. The implications of these effects on ECCS performance would also be largely unknown.

The following root issues are contained in this item:

1. Radiolysis of dissolved N_2 may result in the formation of nitric acid.
2. Nitric acid may cause the pool pH to become strongly acidic.

As discussed in the March 2011 report (39), the formation of nitric acid due to radiolysis is expected to be relatively low due to the low solubility of N_2 in water. The assumption that the pool could become strongly acidic did not take into account the presence of the buffers. Therefore, the pool is not expected

to become strongly acidic. However, similar to the other issues regarding the formation of strong acids, the effects on long-term pH due to the formation of nitric acid should be considered.

STP Resolution: The addition of acid to the tests to simulate radiolysis is addressed in PIRT Item 2.1.

PIRT Item 2.5: Additional Debris Bed Chemical Reactions

The concentration of radionuclides, postulated to be hundreds of Curies, available within the sump strainer fiber bed acts as a "resin bed" or chemical reactor potentially altering the local chemical conditions, such as pH. A number of possible radiolytic reactions could occur which may directly alter the chemical byproducts formed. This effect may lead to the formation of different, or a larger quantity of, chemical products than those evaluated, which could have a different impact on head loss than that considered.

The following root issues are contained in this item:

1. Radionuclides trapped in the debris bed may change the local chemistry and cause precipitation.
2. Radionuclides trapped in the debris bed may cause the bed to break down.

As discussed in the March 2011 report (39), local changes in the chemistry (i.e. the formation of H_2O_2 due to radiolysis) will not have a significant effect since the constant flow through the debris bed will effectively flush it out. Also, the concern that the fiber bed may break down due to the radionuclides is not considered to be significant since materials similar to fiberglass insulation are routinely used as a filtration media for high activity particulate.

During the chemical effects summit, the NRC questioned whether other types of insulation or coatings debris besides fiberglass may break down in the debris bed due to the radionuclides (44).

STP Resolution: The non-fiberglass debris at STP includes Microtherm and coatings debris. Coatings have been extensively tested in DBA conditions including high radiation. Although unqualified coatings can break down due to the heat, humidity, and radiation in a post-LOCA environment, the size distribution for unqualified coatings already takes these effects into consideration. The Microtherm debris quantity at STP is minor compared to the quantity of fiberglass debris. Therefore, even if radiolysis did have an effect on Microtherm particulate, it would not significantly change the structure of the overall debris bed at STP. Therefore, no additional evaluation is required.

PIRT Item 3.1: Crud Release

A PIRT panelist postulated that iron and nickel corrosion oxides up to 125 microns thick may exist on the interior of the RCS piping, fuel, and components. These oxides could be released by the hydraulic shock of the LOCA event. After release, the reduced Fe and Ni ions can be dissolved in the RCS (aided by radiolysis) and, when combined with air, can form oxides of hematite, maghemite, and magnetite. The crud release can create a localized radiolytic environment on materials caught on the sump screens, which could affect subsequent chemical reactions. The crud particles would also add to the debris concentration within the containment pool.

The following root issues are contained in this item:

1. The crud may influence the localized radiolytic environment.
2. A significant quantity of crud could be released as another source of particulate debris.

As discussed in the March 2011 report (39), the radiolytic effects of crud are insignificant compared to other sources. The March 2011 report estimated that the total quantity of crud in the RCS could be on the order of 400 kg (39). This is a potentially significant source of particulate debris, but it is not likely that 100% of the crud would be released by the oxidation effects and hydraulic shock of a LOCA. The March 2011 report concluded based on transport considerations that this is not a significant issue (39).

At the chemical effects summit, the NRC questioned whether the RCS crud could transport and have a significant impact on head loss (44).

STP Resolution: This item was not addressed in the 30-day CHLE tests. The crud is a source term for particulate debris and is not expected to affect the chemical environment. Crud release was addressed by considering it as a potential additional source term for particulate debris that can contribute to head loss. The total quantity of crud debris released was calculated to be within a range of 5 – 24 lb_m (45).

PIRT Item 3.2: Jet Impingement

The two-phase jet, and fine debris within the jet, will impact surfaces and could chip coatings, cause metallic erosion, or ablate materials like concrete. This phenomenon will govern the contributions of these materials in the early post-LOCA time period, before corrosion and leaching become important. Jet impingement could also initiate pitting corrosion, which could accelerate the corrosion of normally passivated materials like stainless steel. Most of the discussion from the peer review panel describes the jet interaction with materials as the primary source for post-LOCA debris. Jet impingement could result in a potential chemical effects debris source term that is greater than currently anticipated.

The following root issues are contained in this item:

1. Debris can be generated by the jet blast.
2. Pitting due to jet impingement could accelerate corrosion.

The generation of debris and subsequent effects of that debris in terms of both debris bed head loss and chemical effects is an important issue that should be considered.

As discussed in the March 2011 report (39), jet impingement during blowdown has a very short duration, and any pitting that occurs would be localized and have a minimal effect on the overall quantity of corrosion products.

STP Resolution: The approach for determining the quantity of materials during each test takes debris generation into account and is addressed in PIRT Item 1.1.

PIRT Item 3.3: Debris Mix Particulate/Fiber Ratio

Breaks in different locations will create different debris characteristics with respect to the total mass of debris, debris constituents, and the ratio of particulates to fiber. Depending on the specific break location, significantly different types and quantities of debris (e.g., Cal-Sil and fiberglass insulations) can alter the type and quantity of chemical effects. Ultimately, the debris bed characteristics determine the chemical product capture efficiency and the total pressure drop across the sump screen strainer.

The following root issues are contained in this item:

1. Different mixtures of debris can have a different impact on chemical effects.
2. Variations in the particulate/fiber ratio impact the chemical precipitate capture efficiency.
3. Variations in the particulate/fiber ratio impact the debris bed head loss.

In an integrated environment, the presence of some materials may inhibit the corrosion or dissolution of other materials. For example, silicon that is released into solution from the dissolution of fiberglass may inhibit the corrosion of aluminum. In some cases, therefore, scenarios with lower quantities of certain types of debris could potentially result in more severe chemical effects.

Fiber beds act as very effective filters and can capture small particles. As the particulate to fiber ratio increases, the debris bed is compacted and the filtration efficiency increases (along with the head loss). Therefore, the particulate to fiber ratio is a significant parameter.

STP Resolution: The mixture of debris to be used in each test is addressed in PIRT Item 1.1. The long-term tests used a fiber-only debris bed to assess the relative impact of chemical effects under a standardized condition (31; 33). The impact of variations in the particle/fiber ratio on chemical precipitate capture efficiency was not evaluated during the test program. The method used to address chemical effects head loss is described in Volume 3 (12).

PIRT Item 3.4: Effects of Dissolved Silica from Reactor Coolant System and Refueling Water Storage Tank

Dissolved silica is present in the water storage systems and the RCS during normal operation. This silica can react with other chemical constituents (most prominently magnesium, calcium, and aluminum) that form as a result of material dissolution or corrosion, or both, within the containment pool after the LOCA occurs. This reaction may result in a greater concentration of the chemical precipitates than would otherwise exist. The reaction may also alter the nature of the chemical precipitates by creating amorphous materials or gels or precipitates with retrograde solubility (i.e., they become more insoluble as temperature increases). The creation of additional chemical precipitates, amorphous materials, and retrograde soluble species could degrade ECCS performance by increasing head loss at the sump strainer or decreasing in the heat transfer rate from the reactor fuel if significant quantities of silica-containing precipitates are formed.

The following root issue is contained in this item:

1. The dissolved silica initially in the water may precipitate with other materials later in the event.

Silicon is an important factor for chemical effects. In some cases, it may help inhibit corrosion of aluminum, and also can contribute to precipitate formation. Therefore, the initial concentration of dissolved silica in the RCS, RWST, and accumulators should be considered.

STP Resolution: The quantity of silica present in the RCS, RWST, and accumulators at STP was evaluated along with the boron and lithium as described in PIRT Item 1.1. The concentration of silica in the RWST was determined to be between 1 and 6 mg/l based on operating history (40). However, it was decided not to include silica in the integrated tests since silicon can passivate aluminum and reduce the corrosion rate (40). The effects of the ratio of aluminum to silicon on corrosion were investigated in bench-scale tests (41).

PIRT Item 3.5: Containment Spray Transport

Following a LOCA, the containment spray will tend to wash latent debris, corrosion products, insulation materials, and coating debris into the containment pool. This changes the containment debris sources (types, amounts, compositions) and chemical species reaching the containment pool environment which could affect the sump strainer debris bed and the formation of chemical precipitates.

The following root issues are contained in this item:

1. Corrosion products generated above the pool could be washed down into the pool.
2. Debris above the pool could be washed into the pool.

Both of these items are potentially significant and should be considered.

STP Resolution: The debris quantities used for the CHLE tests (42; 43) took into account debris transport due to containment sprays for each accident scenario. The amount of material transported to the pool versus the debris held up above the pool surface was evaluated as part of the larger risk-informed approach (12).

PIRT Item 3.6: Initial Debris Dissolution

Typical debris generated by the LOCA (within the first 20 minutes) includes Cal-Sil insulation, cement dust, organic fiberglass binders, and protective coatings. Initial debris dissolution could indicate potential important contributors to the chemical containment pool environment. It is possible that the dissolved, ionic species could react and precipitate to form new, solid phases that were not originally in the containment pool.

The following root issue is contained in this item:

1. Dissolution of debris can form chemical precipitates.

This is the chemical effects issue and should be appropriately modeled in realistic chemical effects tests.

STP Resolution: The relevant materials and debris determined to be present at STP and contribute to chemical effects in each of the LOCA scenarios were included in the CHLE loop at the beginning of each test (42; 43). Determination of the quantities of debris is addressed in PIRT Item 1.1.

PIRT Item 3.7: Submerged Source Terms: Lead Shielding

Acetates present in the containment pool will corrode any submerged lead existing in containment, which could lead to formation of lead carbonate particulate or dissolved lead within the containment pool. Lead blanketing or lead wool is used to shield radiation hot spots during refueling outages and may remain in the containment building during the fuel cycle. In addition, several plants may still use small quantities of lead wool for insulation.

Lead carbonate contributions would provide additional particulate loading within the containment pool that could contribute to head loss at the sump strainer screen. Dissolved lead could also lead to cracking of submerged stainless steel structural components within containment. Neither the testing conducted to date nor do the licensee evaluations of ECCS performance consider these contributions. These omissions are potentially non-conservative if significant quantities of lead carbonate or dissolved lead are formed.

The following root issues are contained in this item:

1. Lead could dissolve and precipitate with other materials.
2. Dissolved lead may lead to cracking of submerged stainless steel components.

Generally, the quantity of lead exposed to the pool or sprays would be low. However, the dissolution of lead and subsequent precipitation is a potentially significant issue that should be considered.

As discussed in the March 2011 report (39), relatively low lead concentrations will not induce cracking in stainless steel components within the 30-day mission time.

STP Resolution: The lead in containment at STP includes lead blankets in storage barrels and lead blankets that are permanently installed on a few pipes. The lead blankets in the storage barrels are at different elevations including the containment floor. The storage barrels have holes in them so that they would be filled with water during a LOCA event. However, the lead blankets are sealed within vinyl-laminated nylon covers that provide a protective barrier between the material and pool solution. The lead blankets are inspected both before and after use, and blankets that exhibit signs of wear (such as cracking of the blanket material, damaged or corroded grommets, or other signs of physical damage) are removed from service (46). The jacketing material would not be damaged during a break, and the water in the barrels would be relatively stagnant, so it is reasonable to assume that this source of lead will not have an impact on chemical effects. The other source of lead is lead blankets installed on three pipes in the steam generator compartments (47). The lead blankets are robust and would only be damaged by larger breaks in the near vicinity of the blankets. Also, even for the cases where the lead blankets could be damaged, the pieces of lead wool would not be easily transported to lower containment. Therefore, lead was not included in the 30-day CHLE tests.

PIRT Item 3.8: Submerged Source Terms: Copper

Copper present in containment can accelerate or inhibit corrosion of other metals. One way in which Cu can alter the corrosion rate of other materials is by forming a galvanic couple. Galvanic effects can accelerate corrosion of less noble material while inhibiting corrosion of more noble materials. Dissolved

copper can also enhance the rate of corrosion of other metals within an oxygenated environment. Different corrosion rates can impact the amount of corrosion products formed and therefore could have different effects on ECCS sump head loss.

The following root issues are contained in this item:

1. Galvanic couples can accelerate or inhibit corrosion of other metals.
2. Dissolved copper can enhance the corrosion rate of other metals by forming local galvanic cells.
3. Copper can inhibit corrosion of other metals by depositing and creating a passivation layer.

As discussed in the March 2011 report (39), the potential effect of galvanic couples in containment is insignificant. Local galvanic cells may enhance corrosion of aluminum, but this would only apply to the submerged aluminum. Also, as discussed in the March 2011 report, copper deposits were observed on aluminum samples in some of the ICET tests, which may have helped inhibit aluminum corrosion since the tests had negligible aluminum concentrations (39). Copper corrosion is expected to be relatively minor, but is a potentially significant issue that should be considered.

As discussed at the chemical effects summit (44), only the second root issue is important for chemical effects—potential enhancement of metal corrosion due to a local galvanic cell. The NRC also stated that the corrosion of zinc (from galvanized steel or other sources), and subsequent formation of zinc precipitates is a potentially significant issue that should be evaluated.

<u>STP Resolution</u>: The sources of copper at STP include copper in the fan coolers and copper wiring in the electrical cables. The fan coolers are above the containment pool elevation and the copper would be shielded from containment sprays. The copper in the electrical cables would only be exposed if the cables are damaged by the break jet. The cable insulation is robust and the cable trays would help protect the cables from damage, so the exposed surface area of copper wiring would be negligible.

The ICET tests had significant quantities of copper in both the submerged and unsubmerged portion of the tank. The copper coupons in the ICET tests exhibited very little change in weight: the mean change in weight ranged from +0.2 to -0.3 g for the submerged copper coupons and from +0.3 to -0.2 g for the unsubmerged copper coupons in all tests, compared to a mean pre-test weight of 1318 g. In ICET Test #2, which is most representative of the STP chemical conditions, the mean change in weight of both submerged and unsubmerged coupons was <0.1 g (48). Throughout all ICET tests, the copper concentration in solution was below about 1 mg/L. In ICET Test #2, the copper concentration was below the limit of detection for the entire 30 day duration of the test (48). Since the exposure of copper in a LOCA at STP would be substantially less than the copper included in the ICET tests, copper was not included in the CHLE tests.

Galvanized steel and zinc granules (representing inorganic zinc coatings debris) were included in the large break test to evaluate the effects (43).

<u>PIRT Item 3.9: Concrete Material Aging</u>

The PIRT panelists raised questions about the effect of aging on the leaching process for nonmetallic materials such as concrete. Neither the exposed concrete faces nor concrete dust in the containment

building is likely to be fresh. After 30 years of exposure to the atmosphere, a substantial fraction of both the exposed calcium silicate hydrate (C-S-H) gel and the portlandite ($Ca(OH)_2$) constituents of the concrete would have been carbonated. Carbonation or other aging processes of concrete could affect the leaching rates and dissolved species as compared to relatively fresh concrete samples used in the ICET experiments and other research programs.

The following root issue is contained in this item:

1. Aged concrete may release a larger quantity of calcium.

Concrete surfaces in containment are generally coated, which would prevent carbonation due to aging. However, this may be a significant issue for plants with large uncoated concrete surfaces; especially if the plant uses a TSP buffer. Therefore, this issue should be addressed for realistic testing.

During the chemical effects summit, the NRC stated that the difference in dissolution between aged and fresh concrete is not significant and it is not necessary to use aged samples for chemical effects testing.

STP Resolution: No action required.

PIRT Item 3.10: Alloying Effects

Another issue raised by the PIRT is the effect of different alloys on the quantity of corrosion products. Corrosion rate data exhibit wide variability depending on the specific corrosion conditions and the nature of the alloy being subject to corrosion. Alloying could affect dissolution and corrosion rates, thereby affecting the solid species precipitates that are formed.

The following root issue is contained in this item:

1. Differences in alloys may affect dissolution and corrosion rates.

As discussed in the March 2011 report (39), alloys would generally exhibit lower corrosion rates than pure metals. In realistic testing, it may be beneficial to use the actual alloys that exist in containment. Regardless, it is important to appropriately justify all surrogate materials (including metal coupons) that are used in chemical effects tests.

At the chemical effects summit, the NRC stated that there is not a large difference between corrosion rates for pure materials and alloys. However, it is appropriate to use materials that are representative of what is in containment.

STP Resolution: Pieces of aluminum scaffolding from STP were used in the CHLE tests to represent the aluminum sources (42; 43). Galvanized steel coupons were not included in the medium break test, but were included in the large break test (43).

PIRT Item 3.11: Advanced Metallic Corrosion Understanding

The PIRT panel raised several other issues related to the understanding of metallic corrosion in the post-LOCA environment. These issues include enhanced Al corrosion caused by hypochlorite or other catalytic

effects (e.g., jet impingement), synergistic effects on corrosion, and corrosion inhibition. These effects could substantially affect corrosion rates and therefore could have different effects on ECCS sump head loss.

The following root issues are contained in this item:

1. Enhanced corrosion due to acid formation.
2. Enhanced corrosion due to pitting from jet impingement.
3. Synergistic effects on corrosion.
4. Corrosion inhibition.

As discussed previously, the long-term effects on pH due to acid formation may be an important factor that should be considered. Also as discussed previously, pitting from jet impingement is considered to be an insignificant factor due to the localized impact of the jet. Generally, synergistic effects tend to inhibit corrosion, but both synergistic effects and corrosion inhibition are inherently considered in integrated testing.

STP Resolution: Synergistic effects and corrosion inhibition are address due to the selection of materials in the proper proportions relative to the STP containment, as described in PIRT Item 1.1. Acid formation is addressed in PIRT Item 2.1.

PIRT Item 3.12: Submerged Source Terms: Biological Growth in Debris Beds

The PIRT considered the propensity for bacteria or other biota to grow in preexisting debris beds located on the sump strainer screen or elsewhere within the ECCS system. Significant bacterial growth may be important if it creates additional debris that contributes to sump screen clogging or detrimental performance of downstream components like pumps and valves.

The following root issue is contained in this item:

1. Biological growth in the post-LOCA environment may contribute to clogging issues.

As discussed in the March 2011 report (39), most microorganisms cannot survive under high temperature, low or no light, and high radiation conditions. Any microorganisms that do survive would be highly unlikely to experience significant growth under the harsh post-LOCA conditions. Therefore, biological effects can be reasonably neglected for a realistic chemical-effects analysis.

During a public conference call on September 6, 2012, the NRC stated that STP should examine any wet sumps in containment to determine whether there is a significant source of existing biological material (49).

STP Resolution: A review of the corrective action records was made to determine whether there were ever any problems with biological debris in the secondary sump. Two issues were identified for Unit 2. In the first case, a sensor was caked with "sludge" (50), and in the second case a sensor was covered with "rusty slime" (51). There was no indication that the foreign material was biological, and in both cases it was cleaned up at the time of discovery. Note also, that the secondary sump is flushed with hydrogen peroxide during refueling outages, which is very effective in killing any bacteria that could be present.

PIRT Item 3.13: Reactor Core: Fuel Deposition Spall

Spall of reactor fuel cladding oxides (ZrO_2) and deposited chemical products could be a potential source of activated materials that could affect chemical reactions in the post-LOCA containment pool. Also, precipitates of post-LOCA chemical products (organics, Al, B, Ni, Fe, Zn, Ca, Mg, silicates (SiO_3^{2-} and SiO_4^{4-}), and CO_3^{2-}-based products) could deposit on the fuel clad and spall, contributing either to clogging within the reactor core, or head loss across the sump strainer.

The following root issues are contained in this item:

1. Spall of activated fuel cladding oxides could affect chemical reactions in the containment pool.
2. Precipitation and spall of chemical products on the fuel could contribute to fuel or strainer clogging.

As discussed previously, the effect of activated particles on chemical effects due to radiolysis is considered to be insignificant. However, this debris could contribute to the source term for particulate debris with an effect on the overall head loss across the strainer or fuel channels. This issue is addressed in PIRT Item 3.1.

Some precipitates, particularly certain calcium precipitates, exhibit retrograde solubility. As water flows through the reactor vessel, the high temperature in the vicinity of the fuel rods may cause some of these materials to precipitate. The precipitates may form on the fuel rods themselves, or in solution where they can be swept out of the reactor vessel and potentially contribute to strainer clogging. This is a potentially significant issue that needs to be addressed for materials that exhibit retrograde solubility.

STP Resolution: The effects of chemical precipitation on the fuel rods have been previously addressed in a conservative manner for STP using the LOCADM software (52). Calcium phosphate and other calcium products demonstrate retrograde solubility characteristics. Although they could potentially form under STP conditions, there was no observation of any calcium products in the CHLE tests (31; 33). A detailed evaluation of these or other products that may exhibit retrograde solubility was not performed. However, the methodology used to address strainer head loss and in-vessel head loss (12) is believed to encompass the potential effects from chemical products formed due to retrograde solubility.

PIRT Item 4.1: Polymerization

The PIRT panelists expect polymerization to occur after molecular precipitation as a precursor to solid species agglomeration in post-LOCA environments. Molecular precipitation refers to the formation of bonds between metallic species and oxygen to form monomers. Polymerization is the ripening of these bonds to form covalent bonds and the growth of the monomers through one of many types of polymerization reactions. Chain polymerization, which is the most common, consists of initiation and propagation reactions and may include termination and chain transfer reactions. Step-growth and condensation polymerization are two additional mechanisms. Polymerization occurs until approximately nanometer-sized particles have formed. These particles can then continue to grow to larger sizes through agglomeration mechanisms.

The PIRT panelists expect polymerization is needed to form large enough particles to tangibly affect ECCS performance. The fact that chemical precipitates have formed during testing to simulate post-LOCA conditions provides evidence that polymerization is likely occurring. The issue is important only if the differences in polymerization mechanisms in the simulated and actual post-LOCA environments are significant enough to alter head loss or downstream effects associated with the chemical precipitates.

The following root issue is contained in this item:

1. Polymerization processes may cause initial precipitate growth.

As discussed in the March 2011 report (39), polymerization is expected to be an important process in the formation of precipitates, but is appropriately represented in testing and does not need to be further evaluated.

STP Resolution: No action required.

PIRT Item 4.2: Heat Exchanger: Solid Species Formation

Chemical species having normal solubility profiles may be dissolved in the containment pool at higher temperatures. However, these chemical species may precipitate in the heat exchanger because of a drop in temperature of approximately 30 degrees F. Some possible solid species that could form include $Al(OH)_3$, FeOOH, and amorphous SiO_2. The lower temperature at the heat exchanger outlet could also facilitate the development of macroscale coatings or suspended particulates, or both, that can continue to transport in the circulating fluid. Possible implications of this scenario include (1) species remain insoluble at higher reactor temperatures and affect the ability to cool the reactor core, (2) solid species formed may clog the reactor core and degrade heat transfer from the fuel, (3) species remain insoluble at higher containment pool temperatures and cause additional head loss upon recirculation, and (4) particulates act as nucleation sites for other compounds to precipitate.

The following root issue is contained in this item:

1. The temperature drop at the heat exchanger may reduce the solubility limit sufficiently to cause precipitate formation.

This is a potentially significant issue that should be evaluated in realistic testing. Timing is an important factor here. Early in the event while the pool temperatures are hot, the temperature drop across the heat exchangers may be significantly higher than 30°F. Since it takes time for containment materials to corrode and dissolve, precipitation may not be possible until much later in the event when the concentration in the pool starts to approach the solubility limit. Timing may also be important with respect to the kinetics of precipitate formation since the duration that coolant flow is exposed to lower temperatures downstream of the heat exchangers is relatively brief.

STP Resolution: At STP, only the LHSI flow passes through a heat exchanger. The LHSI flow rate ranges from 0 gpm for an SBLOCA to a maximum of 2,800 gpm per pump for an LBLOCA (12). The HHSI flow (1,620 gpm per pump) mixes with the LHSI flow downstream of the heat exchangers prior to reaching the cold or hot leg pipes. Based on the maximum flow rates and the volume of the piping and reactor vessel lower plenum and downcomers (53), the water would be at the cold heat exchanger discharge

temperature for approximately 4 seconds and at the cool LHSI and HHSI mixture temperature for approximately a minute before reaching the core.

The CHLE testing included a loop in which the temperature of the solution was decreased, passed through an analytical system to detect whether precipitation occurred, and then increased back to the tank temperature. The turbidity measurements and membrane filters both before and after the heat exchanger were used to identify whether any precipitates formed due to the temperature drop. No significant differences were observed in the filters or turbidity measurements (31; 33).

PIRT Item 4.3: Reactor Core: Precipitation

The increased temperature in the reactor vessel (i.e., 70 degrees C higher than the containment pool) and retrograde solubility of some species (e.g., Ca silicate, Ca carbonate, zeolite, sodium calcium aluminate) causes precipitation and additional chemical product formation. This could result in the following: (1) additional precipitate could be created and transported to the sump screen that would then contribute to head loss and (2) precipitate or spall (see Section 3.13 of this report) passing through the sump screen may degrade the performance of ECCS components downstream from the screen.

The following root issue is contained in this item:

1. High localized temperatures in the reactor vessel may cause precipitation of materials with retrograde solubility.

This is a potentially significant issue that should be evaluated in realistic testing. It should be noted, however, that the bulk flow temperature in the reactor vessel would generally not be 70°C (158°F) higher than the pool temperature. It is possible for local temperatures within the core (i.e. next to the fuel cladding) to be significantly hotter than the pool, which could result in localized precipitation. Also, under certain scenarios (such as a cold leg break during cold leg injection), it is possible for the water in the core to boil. Even under these conditions, however, the maximum bulk temperature in the core would be limited to the saturation temperature, which would never approach a level that is 158°F hotter than the pool. Therefore, the focus of this issue should be on localized high temperatures in the reactor vessel rather than overall high temperatures in the bulk flow.

STP Resolution: This issue is addressed in the response to PIRT Item 3.13.

PIRT Item 4.4: Particulate Nucleation Sites

Particles within containment create the nucleation sites required for chemical precipitation. Examples of particles that could serve as nucleation sites include irradiated particles, dirt particles, coating debris, insulation debris, biological debris, and other materials within the post-LOCA containment pool. These particles then grow through polymerization (see Section 4.1 of this report) and agglomeration (see Sections 5.1 and 6.2 of this report) into solid species that are large enough to possibly degrade ECCS performance.

This issue identifies a fundamental aspect of the formation of solid species. Implications only arise if the nucleation sites in the post-LOCA environment are not appropriately simulated in testing. That is, the

quantities and types of nucleation sites used in testing should be representative of the post-LOCA environment to ensure that solid species formation is not suppressed.

The following root issue is contained in this item:

1. Heterogeneous nucleation sites are required for precipitation to occur.

As discussed in the March 2011 report (39), both containment and test conditions contain numerous nucleation sites. Therefore, this is not a significant issue.

STP Resolution: No action required.

PIRT Item 4.5: Coprecipitation

Coprecipitation occurs when a normally soluble ion becomes either included or occluded into the crystalline structure of a particle of insoluble material. Precipitation of one species could lead to increased precipitation of another species (which, if taken separately, are each below their solubility limit). Thus, more solid species could form, which could lead to a greater concentration of chemical precipitates at the sump strainers or downstream of the strainers. Additionally, the species that form could differ in size from those observed in the ICET tests (i.e., 1 to 100 microns) such that they affect the head loss at the sump strainer more significantly.

The following root issue is contained in this item:

1. Precipitation of one material may result in precipitation of another material that would not otherwise have precipitated.

Coprecipitation does not reduce the solubility limit of precipitates, and therefore would not cause precipitation of two materials that are both below their solubility limit as suggested above. Although it is a potentially significant issue, in an integrated test environment, the various reactive materials are present together and coprecipitation can occur naturally. Therefore, this issue is inherently included in an integrated test.

STP Resolution: The issue of coprecipitation is addressed by inclusion of all materials that participate in chemical reactions in the same proportions that they are present at STP, as described in PIRT Item 1.1.

PIRT Item 5.1: Inorganic Agglomeration

Inorganic agglomeration is the formation of larger clumps of smaller particulates. This phenomenon depends upon the pH of the point of zero charge (PZC) of the species and the ionic strength (the higher the ionic strength, the smaller the distance for agglomeration) of the fluid. This phenomenon is sensitive to many factors, including particle shape factors, and maximum particle size. Inorganic agglomeration of small particles into larger sized particulates could degrade strainer performance.

The following root issue is contained in this item:

1. Agglomeration of chemical precipitates, insulation particulate, and/or latent particulate may form larger particles that would be more easily captured in a debris bed.

In general, agglomeration of particles will make the debris less transportable. Also, as shown in NUREG/CR-6224 (54), smaller particles have a larger impact on head loss due to the larger surface-to-volume ratio. Since head loss testing has shown that fiber beds can very effectively capture 10 micron particles, and the majority of insulation, latent, and coatings particulate debris would be larger than 10 microns, agglomeration of this material with each other or chemical precipitates is not a significant issue.

STP Resolution: No attempt to either stimulate or prevent agglomeration of particles was incorporated in the CHLE tests.

PIRT Item 5.2: Deposition and Settling

Chemical products formed in the post-LOCA containment environment could either settle within the containment pool or be deposited on other surfaces. Chemical species which attach to or coat particulate debris may enhance settling. Examples are aluminum coating on NUKON® fiber shifting the PZC or formation of a hydrophobic organic coating. This could result in less particulate debris and chemical product transporting to the sump screen and either accumulating on or passing through it. The possible implications of this issue are that the chemical precipitates added to the plant-specific chemical effects tests could result in increased settling during the tests compared to actual plant conditions.

The following root issue is contained in this item:

1. Chemical precipitates may settle or enhance settling of other particulate in the containment pool.

Given their small size, chemical precipitates can readily transport under relatively low flow conditions, and it is not expected that significant settling would occur. Therefore, this is not considered to be a significant issue.

STP Resolution: No action required.

PIRT Item 5.3: Quiescent Settling of Precipitate

Quiescent flow regions within the containment pool promote settling. The low flow rate within most of the containment pool also allows larger size, more stable particles and precipitates to form, which promotes settling. Settling of nonchemical debris and precipitate could be beneficial with respect to the pressure drop across the sump strainer.

The following root issue is contained in this item:

1. Chemical precipitates may settle or enhance settling of other particulate in the containment pool.

As discussed above, this is not considered to be a significant issue.

STP Resolution: No action required.

PIRT Item 5.4: Transport Phenomena: Precipitation and Coprecipitation

Precipitation or coprecipitation and ripening of solid species within the containment pool would create solid species which are less likely to transport. Decreased transportability will result in less product migrating to or through the sump screen.

The following root issue is contained in this item:

1. Chemical precipitates may settle in the containment pool.

As discussed above, this is not considered to be a significant issue.

STP Resolution: No action required.

PIRT Item 6.1: Break Proximity to Organic Sources

The pipe break location plays an important role in debris generation. If the break occurs in close proximity to organic sources, it could introduce a significant amount of organic materials into the containment pool. Organic sources could then affect the nature, properties, and quantities of chemical byproducts that form in the post-LOCA containment environment. The scenario evaluated by the PIRT considered failure or leakage of oil and other organics from either the RCP oil collection tanks or lube oil systems resulting from LOCA-induced damage. If the pipe break occurs in close proximity to the organic sources, up to approximately 250 gallons of oil may be released to the containment pool. If this should occur, head loss and downstream effects may be altered, either beneficially or negatively, by these organic materials.

The following root issues are contained in this item:

1. Certain breaks may result in a significant quantity of oil being released into the containment pool.
2. Other organic materials may be present due to failure of coatings and the organic binders in insulation debris.

As discussed in the March 2011 report (39), one licensee added a large quantity of oil (representative of the quantity from one RCP motor) to an integrated chemical effects head loss test. The oil addition had no impact on the head loss, and is not considered to be a significant factor.

Similarly, the presence of smaller quantities of organic material from other sources is not expected to have a significant effect on the pool chemistry conditions.

During the chemical effects summit, the NRC stated that although the issue of RCP motor oil does not appear to be a significant concern, the results of the integrated chemical effects test where oil was added were not clear since there was no differential test to compare against (44). They also stated that in general, qualified and unqualified coatings particulate debris could be important for chemical effects, although intact qualified coatings and failed epoxy paint chips do not need to be considered for leaching.

STP Resolution Plan

The cases where a significant quantity of oil would be introduced to the containment pool would be limited to a few larger breaks in the vicinity of one of the RCP motors. Since the majority of break cases would not have significant quantities of oil, it was not included in the 30-day CHLE tests. Additional analysis of the effects of oil was not performed. However, the methodology used to address strainer head loss and in-vessel head loss (12) is believed to encompass the potential effects of oil for the limited scenarios where a significant quantity of oil could be released into the containment pool. The issue of organic materials from coatings failure is addressed in PIRT Item 6.4.

PIRT Item 6.2: Organic Agglomeration

Organic agglomeration is the process of small organic colloidal particles (1 to 100 nanometers in size) joining together, or coagulating, to form larger particles and precipitates. Coagulated particles can collect on sump strainers, decreasing ECCS flow; they could also collect on other wetted surfaces, such as walls or structural steel, and decrease the debris loading on the sump screen. Hence, head losses and downstream effects could differ from those evaluated during plant-specific testing.

The following root issue is contained in this item:

1. Organic agglomeration may form larger particles that would be more easily captured in a debris bed.

As discussed in the March 2011 report (39), this issue is similar to the issue of inorganic agglomeration. During the chemical effects summit, the NRC agreed that organic agglomeration is probably not an issue, but could be important if chemical precipitates are amorphous (44).

STP Resolution: Amorphous precipitates were not observed in either the medium or large break CHLE tests (31; 33). However, the method used to address chemical head loss encompassed the extreme effects of head loss increases due to full capture of amorphous chemical precipitates. This is described in detail in Volume 3 (12).

PIRT Item 6.3: Organic Complexation

Organic complexing agents act to inhibit agglomeration either by adsorption onto solid surfaces or by interaction in solution with metal ions. Organic surface complexation occurs if organic molecules (i.e., amines, acids, and heterocycles) adsorb on surfaces of ions or solids and inhibit the subsequent precipitation or growth of those species. The implications of organic complexation are counter to those associated with organic agglomeration. Organic complexation could reduce the effects associated with chemical precipitates and therefore may be beneficial to ECCS performance if this phenomenon is not credited or addressed during plant-specific testing.

The following root issue is contained in this item:

1. Organic complexation may inhibit agglomeration.

Since both inorganic and organic agglomeration are not considered to be significant issues, organic complexation would be an insignificant factor also.

STP Resolution: No action required.

PIRT Item 6.4: Coating Dissolution and Leaching

Coatings existing within containment represent possible additional physical debris sources. Generally conservative guidance for considering the effects of physical coating debris is provided for the evaluation of ECCS performance. However, dissolution and leaching of coatings can impact the chemical effects that occur within, or are transported to, the ECCS cooling water. Both inorganic (e.g., zinc-based) and organic (e.g., epoxy-based) coatings exist within containment. One concern is that these coatings leach chemicals as a result of being submerged in the containment pool environment after the LOCA. Coatings may create additional chemical species (e.g., chlorides or organics) within the containment pool that could potentially increase sump screen head loss or promote more deleterious downstream effects.

The following root issue is contained in this item:

1. Materials may leach from coatings affecting the overall pool chemistry.

As discussed in the March 2011 report (39), the amount of material that dissolves or leaches from coatings is expected to be relatively low. However, this is a potentially significant issue and should be appropriately addressed in realistic testing.

STP Resolution: Existing literature was reviewed to assess the effects of leaching from coated surfaces. Alkyd coatings were excluded from the CHLE tank tests based on testing that indicated that the coatings would not play a significant role in the creation of chemical precipitates (6). Similar to the alkyd coatings, epoxy coatings were excluded from the CHLE tank tests based on testing that indicated that the coatings would not play a significant role in the creation of chemical precipitates (26; 37; 38). Zinc particles were included in the CHLE tests to address potential concerns regarding the effects of inorganic zinc coatings on the formation of zinc products (42; 43).

PIRT Item 7.1: Emergency Core Cooling System Pump: Seal Abrasion and Erosion or Corrosion

Abrasive wearing of pump seals (e.g., magnetite—high volume or concentration of mild abrasive) creates additional materials that contribute to containment pool chemistry. In addition, chemical byproducts cause erosion or corrosion of pump internals, especially close-clearance components (e.g., bearings, wear rings, impellers). The possible implications of these phenomena are (1) additional particles could contribute to reactor core clogging, (2) particles could add additional sump screen loading, (3) particles could affect chemical species formation, and (4) pump performance degrades, possibly to the point of being inoperable.

The following root issue is contained in this item:

1. Particulate debris generated by abrasive wearing of pump seals may cause additional downstream problems.

As discussed in the March 2011 report (39), the quantity of particulate material generated by wearing of the pump seals is insignificant compared to other particulate sources. Also, the pump materials are not unique, and the surface area of similar metals and materials in containment are large enough that the impact of the pump internals on chemical effects is considered to be negligible. Therefore, this issue is insignificant.

STP Resolution: No action required.

PIRT Item 7.2: Heat Exchanger: Deposition and Clogging

Solid species which form in the heat exchanger lead to surface deposition or clogging, or both, within close-packed heat exchanger tubes (5/8-inch in diameter). This could cause decreased flow through the heat exchanger core or diminished heat transfer between the ECCS and heat exchanger cooling water, or both. Diminished cooling of the ECCS water could ultimately decrease the capacity of the ECCS water to remove heat from the reactor core.

The following root issue is contained in this item:

1. Precipitation within the heat exchanger may affect the heat exchanger performance.

As discussed in the March 2011 report (39), chemical precipitates would not have enough shear strength to block flow through the heat exchanger tubes. It's possible that some precipitates could create a thin coat on the tube walls. However, since the precipitates would generally form later in the event when the heat exchangers have ample margin, any slight degradation in performance due to the precipitates is negligible.

During the chemical effects summit, the NRC agreed that clogging of the heat exchanger is not likely to be a significant issue. However, they suggested that a post-test inspection of the heat exchanger would be a good idea (44).

STP Resolution: The efficiency of heat exchange and performance of the heat exchanger was not monitored during the CHLE tests. Based on the diagnostic methods used during the CHLE tests (turbidity measurements and membrane filters both upstream and downstream of the heat exchanger), there was no reason to suspect formation of precipitates within the heat exchanger (31; 33). The ends of the heat exchanger tubes were also examined, with no indication of any chemical products on the tube surfaces.

PIRT Item 7.3: Reactor Core: Fuel Deposition and Precipitation

The increased temperature (+70 degrees C from containment pool) and retrograde solubility of some species (e.g., Ca silicate, Ca carbonate, zeolite, sodium calcium aluminate) causes scale buildup on the reactor core. Zn, Ca, Mg, and CO_2-based deposits, films, and precipitates may form at higher temperatures within the reactor core. This may lead to (1) a decrease in heat transfer from the reactor fuel, (2) localized boiling due to insufficient heat removal, and (3) spallation of deposits, creating additional debris sources which could clog the reactor core or contribute to sump screen head loss.

The following root issue is contained in this item:

1. High localized temperatures in the reactor vessel may cause precipitation of materials with retrograde solubility.

As discussed previously, precipitation of materials with retrograde solubility on the fuel surfaces or in solution within the core is a significant issue that needs to be addressed.

STP Resolution: This issue is addressed in the response to PIRT Item 3.13.

PIRT Item 7.4: Reactor Core: Diminished Heat Transfer

Physical and chemical solid debris within the ECCS coolant water could diminish the fluid's heat transfer capacity and degrade the ability of the coolant to remove heat from the core.

The following root issue is contained in this item:

1. Concentrated materials in the reactor vessel may reduce the water's heat removal capacity.

The highest debris concentrations would occur under cold leg break conditions during cold leg injection since the water entering the core would boil off raising the concentration of boron, other dissolved materials, and suspended solids. As discussed in the March 2011 report (39), the relatively dilute concentration of dissolved solids would not significantly affect the rate of boiling and rate of heat removal. The effects of high boron concentration on heat removal are not fully understood, but a PWROG program investigating this issue is currently in progress and is expected to be completed by 2015. Although the outcome of the PWROG research may change the acceptable limit for boron concentration in the reactor vessel, it would not affect the physical processes that must be evaluated in realistic chemical effects testing. Therefore, the PWROG progress should be monitored for potential plant modifications that may be required (i.e. timing for switchover from cold leg to hot leg injection), but is not a significant issue for realistic chemical effects testing.

At the chemical effects summit, the NRC announced that the boron precipitation issue must now be addressed as part of the overall resolution of GSI-191 (44).

STP Resolution: The resolution of the boron precipitation issue was addressed as part of the larger risk-informed approach (12).

PIRT Item 7.5: Reactor Core: Blocking of Flow Passages

Fuel deposition products and precipitated retrograde soluble chemical species spall and settle within the reactor vessel. Settling can be potentially deleterious if flow passages to the fuel elements are either globally or locally impeded. Reduced flow within the RPV, if significant, has the potential to diminish heat transfer from the fuel.

The following root issue is contained in this item:

1. Debris may spall and settle within the reactor vessel causing blockage.

As discussed previously, precipitates that form due to retrograde solubility within the reactor vessel must be properly addressed. This item raises an additional issue of the potential settling of precipitates or other debris spall under low flow conditions within the reactor vessel. During cold leg injection, the flow would move upward through the core and would tend to lift the debris and transport it out of the reactor vessel. If the settling velocity is high enough for the debris to settle, it would not be expected to create any significant head loss since the flow would simply have to overcome the "weight" of the debris to continue injecting into the core. During hot leg injection, the flow would move downward through the core in the same direction that settling debris would be moving. The debris could accumulate in various locations where it could form a bed and cause higher head losses. However, this issue would occur regardless of debris settling. Therefore, debris settling concerns are insignificant for realistic chemical effects testing.

STP Resolution: No action required.

PIRT Item 7.6: Reactor Core: Particulate Settling

Relatively low, upwards flow (for cold leg injection) within the reactor causes particulates to settle. Compacted deposits form and may impede heat transfer and water flow, especially for lower portions of reactor fuel.

The following root issue is contained in this item:

1. Particulate debris may settle during cold leg injection causing flow path blockage or inhibiting heat transfer.

As discussed previously, debris that settles during cold leg injection would not result in significant head loss. Also, as discussed in the March 2011 report (39), the higher flow through the core for a hot leg break, and the turbulence due to boiling for a cold leg break would be expected to keep the particulate debris from blocking heat transfer to the lower portions of the fuel. Therefore, debris settling concerns are insignificant for realistic chemical effects testing.

STP Resolution: No action required.

Item 5.a.10: Conventional Head Loss Model

Method: The methodology for determining the conventional debris head loss as a function of time-dependent parameters is based directly on the NUREG/CR-6224 head loss correlation (54). Limited testing was conducted to ensure that the correlation provided a reasonable prediction of head loss under STP-specific conditions (55).

These tests were conducted in a high temperature vertical loop (HTVL) which circulated water down through a vertical section of 6-inch transparent piping where an installed screen was used to entrap debris introduced from above. An installed heat exchanger was used to control the water temperature, which was also measured and recorded. Additives were used to control the water chemistry. The debris was introduced at the open top of the loop and was uniformly distributed on the horizontal screen located in the middle section of the vertical pipe. Water was circulated at prescribed flow rates

while the temperature and the differential pressure were measured. The test screen was a perforated plate that supports the debris bed while imparting minimal clean screen head loss. The test loop instrumentation included three temperature thermocouples, two flow meters, and three differential pressure sensors. The temperature thermocouples were located upstream of the debris screen, downstream of the debris screen, and the room environment. Instrumentation was calibrated before the first test in each test series and rechecked before the last test in each series.

A total of eleven tests were conducted. The HTVL tests were performed with the same water chemistry as determined for the CHLE tank tests. The strainer area, the size of the screen holes, and the screen orientation necessarily remained constant for all of the tests. The varied parameters included the flow velocity, the water temperature, and the masses of the fiber and particulate debris.

Due to previously raised concerns regarding the NUREG/CR-6224 correlation, however, a bump-up factor of 5x was used to account for uncertainties in the head loss predictions.

Basis: The correlation is based on theoretical and experimental research for head loss across a variety of porous and fibrous media carried out since the 1940s. The NUREG/CR-6224 head loss correlation was developed in support of the NRC evaluation of the strainer clogging issue in BWRs and has been extensively validated for a variety of flow conditions, water temperatures, experimental facilities, types and quantities of fibrous insulation debris, and types and quantities of particulate matter debris. The types of fibrous insulation material tested include Nukon, Temp-Mat, and mineral wool. The particulate matter debris tested includes iron oxide particles from 1 to 300 μm in characteristic size, inorganic zinc, and paint chips. In all of these cases, the NUREG/CR-6224 head loss correlation has bounded the experimental results. Due to the semi-empirical nature of the correlation, STP performed confirmatory head loss tests to demonstrate the applicability of the correlation to STP conditions (55). Table 2.5.35 illustrates the debris loads and objectives for the STP conventional head loss tests:

Table **2.5.35** – Basic Test Parameters

Test No.	Nukon Bed	Particulates	Mass Ratio	Specific Objectives
1	NEI Protocol 13.4 g Nominal 2" (at 0.9 lbm/ft^3)	67 g F600 SiC introduced after testing with the fiber alone	5	(1) Establish uniform fiber bed using standard NEI protocol. (2) obtain data for fiber without particulates, & (3) assess the filterability of the fine F600 SiC particulates.

Test No.	Nukon Bed	Particulates	Mass Ratio	Specific Objectives
2	NEI Protocol 13.4 g Nominal 2" (at 0.9 lbm/ft^3)	(1) 67 g F400 SiC mixed in with fiber (2) Follow on addition of 30 g F500 SiC (3) Follow on addition of 67 g F320 SiC	(1)5.0 (2)7.2 (3)12	(1) Improve NEI protocol fiber bed uniformity by introducing particulate mixed in with the fiber and assess the filterability of the F400 SiC particulates 2) follow on addition of F500 SiC. (3) follow on addition of easily filtered coarse F320 SiC to enhance bed compaction and subsequent effect on head loss. (SiC consistently did not result in a characteristic head loss.)
3	NEI Protocol 4.5 g Nominal 1/4" thin-bed (at 2.4 lbm/ft^3)	(1) 88 g SiC (before fiber) (a) 20% F320 (b) 30% F400 (c) 30% F500 (d) 20% F600 (2) Follow on addition of 30 g F320 SiC	(1) 20 (2) 26	(1) Develop a procedure for establishing a uniform thin layer of fiber and establish thin-bed of SiC particulates, (2) follow on addition of easily filtered coarse F320 SiC to force further head loss and bed compression. (SiC consistently did not result in a characteristic head loss.)
4	NEI Protocol 18 g Nominal 1" (at 2.4 lbm/ft^3)	88 g SiC (mixed in with fiber) (a) 20% F320 (b) 30% F400 (c) 30% F500 (d) 20% F600	4.9	Establish a thick bed of fiber replicating successful procedure developed in Test 3 and further assess the ability of SiC to generate head loss. (SiC consistently did not result in a characteristic head loss.)
5	NEI Protocol 18 g Nominal 1" (at 2.4 lbm/ft^3)	138 g Iron Oxide (BWR Specific Distribution) (mixed in with fiber)	7.7	Using successful fiber bed procedure from Test 4, assess the head loss for alternate type of particulate for direct comparison with SiC in Test 4.
6	NEI Protocol 18 g Nominal 1" (at 2.4 lbm/ft^3)	44 g Pulverized Acrylic Coatings (mixed in with fiber)	2.4	Using same fiber bed procedure used in Test 4 and 5, assess the head loss of pulverized acrylic coatings particulate.
7	NEI Protocol 18 g Nominal 1" (at 2.4 lbm/ft^3)	180 g Tin Particulate (mixed in with fiber)	10	Using same fiber bed procedure used in Test 4, 5, and 6, assess the head loss of tin particulate

Test No.	Nukon Bed	Particulates	Mass Ratio	Specific Objectives
8	NEI Protocol 4.0 g Nominal 0.22" thin bed (at 2.4 lbm/ft^3)	(a) 33.5 g tin (b) 14.9 g acrylic (c) 0.92 g Microtherm (d) 4.5 g Marinate	16	Replicate the August 2008 ARL prototype test
9	NEI Protocol 2.8 g Nominal 0.15" thin bed (at 2.4 lbm/ft^3)	(a) 28 g acrylic (b) 33.5 g tin (c) 20 g Microtherm	(a) 10 (b) 12 (c) 7.1	(1) Additional acrylic particulate data (2) Additional tin particulate data (3) Fist Microtherm data
10	NEI Protocol 9 g Nominal 0.5" (at 2.4 lbm/ft^3)	44 g Pulverized Acrylic Coatings (mixed in with fiber)	4.9	Additional acrylic particulate data
11	NEI Protocol 9 g Nominal 0.5" (at 2.4 lbm/ft^3)	44 g Pulverized Acrylic Coatings (mixed in with fiber)	4.9	Repeat of Test 10

Assumptions: The following assumptions related to debris bed head loss were made in Volume 3 (12):

- It was assumed that miscellaneous debris would partially overlap and would fully block strainer flow over an area equivalent to 75% of the miscellaneous debris surface area. This assumption is consistent with the guidance in NEI 04-07 (56).
- It was assumed that small and large pieces of fiberglass debris can be treated as 0.5 inch thick cubes and 1 inch thick cubes respectively to calculate the surface to volume ratio. This is a conservative assumption since small pieces range in size up to six inches and large pieces are greater than six inches.
- It was assumed that all coatings materials would have a packing fraction similar to acrylic coatings. It was also assumed that non-coatings particulate debris would have a packing fraction similar to iron oxide sludge. These assumptions are based on engineering judgment due to limited data.
- It was assumed that a fiber bed of at least 1/16th of an inch is necessary to capture chemical precipitates. This is a reasonable assumption since a thinner debris bed would not fully cover the strainer and would not support appreciable head losses due to chemical debris.
- It was assumed that 100% of the transported particulate debris would be captured on the strainer at the time of arrival. This assumption does not imply that no particulate would

penetrate the strainer. However, since the in-vessel effects acceptance criteria that were implemented in CASA are independent of the particulate quantity, this assumption is conservative.

- It was assumed that the debris on the strainers would be homogenously mixed. This is a reasonable assumption since much of the debris would arrive at the strainer simultaneously.

Acceptance Criteria: The time-dependent total strainer head loss (the combination of clean strainer, conventional debris, and chemical debris head losses) was compared to the strainer structural margin and the time-dependent NPSH margin. If the strainer head loss exceeded either of these values, the ECCS was assumed to completely fail.

Results: Discussion of the calculated total strainer head losses is provided in the response to Item 5.a.11.

The application of a head loss correlation to head loss data requires the measurements of head loss, water temperature, and flow velocity for a relatively uniform and homogeneous fibrous/particulate debris bed of known composition at relatively stable conditions. Turbidity measurements, as well as water clarity, are used to judge the completeness of the filtration process.

The correlation validation process depends on knowing the input hydraulic characteristics of each type and size category of debris introduced into the test. Debris size characterization can be used to approximate the hydraulic characteristics of simple forms of debris, such as Nukon fibers, but not for complex particulates. A typical particulate consists of roughly shaped particles of varied sizes making the analytical assessment of the specific surface area, S_v, somewhat difficult and uncertain. Some insulation materials such as calcium silicate, Microtherm, Min-K, and amorphous chemical precipitates have complex forms that simply cannot be assessed analytically, and their impact on head loss has to be addressed experimentally. The solid density of a particle is based on the material properties and the particulate bulk density can be deduced by weighing a known volume of the particulate. The S_v value is deduced by applying a head loss correlation to head loss test data where all parameters are known except the S_v value for the material in question. As such, inaccuracies in the form of the correlation become inherent in the experimentally deduced input parameters. Therefore, the correlation and the hydraulic characteristics become somewhat interdependent.

A total of eleven exploratory head loss tests were performed (57). All testing was done using fibers from a single-side baked Nukon blanket, which was processed using the NEI debris preparation process. All testing was conducted starting at 200 °F at the STP buffered and borated water conditions. The particulate types tested were green silicon carbide, iron oxide (the BWR sludge simulant used in the development of the NUREG/CR-6224 head loss correlation), tin, and ground acrylic paint. Flow and temperature sweeps were performed at the end of some of the experiments to examine the impact of different flow conditions and temperatures.

The NUREG/CR-6224 head loss correlation was used to replicate the measured head loss of the test conducted with iron oxide and a debris bed thickness similar to the test parameters used in the development of the NUREG/CR-6224 head loss correlation (57). The iron oxide S_v value was adjusted until the calculated head loss matched the measured head loss. The final S_v value was in reasonable agreement with the specifications of the size distribution of the sludge simulant indicating that the

NUREG/CR-6224 head loss correlation was a reasonable predictor of head losses at STP water and temperature conditions. The iron oxide test, however, was limited to the lowest approach velocity of 0.02 ft/s due to equipment limitations. The NUREG/CR-6224 head loss correlation also generated reasonable estimates of the head loss experiments conducted with ground acrylic paint and extended the approach velocity down to the STP strainer approach velocity of 0.0086 ft/s.

The NUREG/CR-6224 head loss correlation, however, could not replicate the low head losses observed in the tests with tin and/or green silicon carbide. The test report provides a hypothesis for this behavior based on observations of the difference in smooth surfaces noted on SEMs of green silicon carbide and tin as compared to the rough surfaces of iron oxide and ground acrylic paint (57). Further experiments would need to be conducted to confirm this hypothesis. This lack of agreement between the NUREG/CR-6224 head loss correlation and testing with green silicon carbide and tin does not impact the STP head loss calculations since there is no green silicon carbide or tin in the STP debris mixture. The green silicon carbide has been used in the past as a simulant of paint, and the tin has been used as a simulant of IOZ coatings. Most of the STP particulate debris comes from coatings, either from qualified coatings in the ZOI or from unqualified coatings elsewhere.

Another anomaly observed in the STP head loss tests was the absence of a direct correlation of the head losses observed in the temperature sweeps with the water viscosity. The test report provides a hypothesis that the temperature also impacts the compression of the fiber debris bed due to the temperature impact on the malleability of the fibers (57). An analytical model was developed to couple the compression to temperature that showed good agreement with the experimentally determined temperature sweep data. The compression algorithm implemented in the NUREG/CR-6224 head loss correlation used in CASA was not modified to incorporate the temperature dependence suggested by the tests. The experiments showed that the measured head losses at lower temperature were lower than the head losses calculated by the NUREG/CR-6224 head loss correlation, hence the CASA calculated head losses are conservative. Additional experiments and analysis need to be performed to validate the temperature dependent compression algorithm prior to its implementation in CASA.

One of the tests conducted (Test 8) was designed to replicate the August 2008 ARL STP prototype test (57; 58). However, this test completely failed to replicate the head losses observed in the previous testing. Both tests used the same primary surrogates of Nukon fibers along with tin and acrylic particulates. Three differences in the tests are: 1) Test 8 had a greater thickness of fiber than was reported in the ARL test, 2) Test 8 used Alion supplied Microtherm and Marinate board particulate instead of the same materials used at ARL, and 3) the ARL fiber debris preparation protocol used a food processor whereas Test 8 used the NEI debris preparation protocol. Based on the experience of the CHLE tests (59), fiber beds with food processor prepared fiber tended to exhibit higher head losses than fiber beds prepared in accordance with the NEI debris preparation protocol. Comparisons of the beds prepared with food processor prepared debris and the NEI debris protocol revealed that the NEI protocol fibers tended to bridge the perforated plate holes and form a debris bed over the perforated plate, while the food processor fibers tended to form low porosity "dimples" at the perforated plate holes. The higher head losses observed with food processor beds was attributed to the formation of the low porosity "dimples". The food processor prepared fibers used in the ARL test could have also formed low porosity "dimples", and allowed the particulate to pack tighter in the ARL test than in Test 8 resulting in a lower porosity bed with higher head losses. The formation of "dimples" in the strainer

holes instead of a fiber bed over the perforated plate could also explain the very thin bed observed in the ARL test. The lack of reproducibility of the head losses observed in the Alion vertical loop test compared with the ARL test does not impact the applicability of the NUREG/CR-6224 in calculating the CASA head losses since the differences in the results are attributable to different debris preparation methods. The NUREG/CR-6224 head loss correlation assumes the formation of a debris bed over a perforated plate as was observed with the debris beds prepared in accordance with the NEI debris preparation protocol. Therefore, the NUREG/CR-6224 head loss correlation is considered to be applicable to the debris beds formed with STP prototypical debris.

The test report also addresses the impact of the three main ACRS comments of the NUREG/CR-6224 head loss correlation (57). These ACRS comments were mainly directed at debris beds containing calcium silicate, a known problematic insulation. The test report provides suggested modifications to the NUREG/CR-6224 head correlation to address the three main ACRS concerns (57). Note that all Marinite (similar to calcium silicate) has been removed from containment at STP. Therefore, as shown in the test report, the three main ACRS comments are not significant for STP conditions (57).

Overall, these tests demonstrated that the NUREG/CR-6224 head loss correlation provided reasonable predictions of head loss for the prototypical STP debris types and loads, water chemistry, temperature, and strainer approach velocities. However, due to the generic concerns regarding the NUREG/CR-6224 correlation, the head loss calculated using the correlation was increased by a factor of five in CASA Grande to account for uncertainties in the head loss predictions.

Additional details on the STP conventional debris head loss testing are provided in the head loss test report (55).

Additional details on how the test results were used to justify use of the NUREG/CR-6224 correlation and how the correlation was incorporated in the overall GSI-191 evaluation are provided in Volume 3 (12).

Item 5.a.11: Chemical Effects Head Loss Model

Method: As discussed in the response to Item 5.a.6, there are a relatively limited number of scenarios where significant chemical effects would be observed. However, since the corrosion and dissolution release model and the solubility model were not directly implemented in CASA Grande, a set of chemical effects bump-up factor probability distributions were developed and applied for all breaks. To account for the presence of extreme conditions in the scenario sample space, exponential probability distributions were defined and applied as direct multipliers to the estimated conventional head loss. The probability distributions were developed based on the current results from the CHLE testing (31; 33), WCAP-16530-NP calculations (25), and reasonable engineering judgment.

Basis: The magnitude and probability distributions for the chemical effects bump-up were developed using engineering judgment. In chemical effects testing, a wide range of effects have been observed. In some cases, there are no chemical products, or chemical products form but have a negligible impact on head loss. In other cases, chemical products have been observed to cause head loss to spike dramatically. For STP, the actual formation and effect of chemical products based on realistic conditions

has been relatively minor. However, since not all scenarios have been fully evaluated, the probability distributions included conservative extremes as discussed below.

Assumptions: The following assumptions related to chemical head loss were made in Volume 3 (12):

- It was assumed that chemical products would not form before the pool temperature drops below 140 °F. This is a reasonable assumption for the purposes of this evaluation since the solubility limit for aluminum precipitates increases significantly at higher temperatures, and calcium precipitates are not expected to form in large quantities for most of the scenarios evaluated (25). Note that the temperature profiles used in the CASA Grande evaluation conservatively minimize the temperature and therefore minimize the time that it would take for chemical products to form.

Acceptance Criteria: The time-dependent total strainer head loss (the combination of clean strainer, conventional debris, and chemical debris head losses) was compared to the strainer structural margin and the time-dependent NPSH margin. If the strainer head loss exceeded either of these values, the ECCS was assumed to completely fail.

Results: The chemical effects model that was implemented in CASA Grande is described below:

- No bump-up factor is applied if the fiber quantity on a given strainer is less than 1/16 of an inch.
- No bump-up factor is applied prior to the temperature dropping below 140 °F. Note that since only two temperature profiles were implemented in CASA Grande, the increase in head loss would occur approximately 5 hr after the start of the event for large breaks, and approximately 16 hr after the start of the event for small and medium breaks.
- As shown in Table 2.5.36 and Figure 2.5.38 through Figure 2.5.40, the probability distributions for the chemical effects bump-up factors were developed with mean bump-up factors of approximately 2x for small breaks, 3x for medium breaks, and 3x for large breaks, and maximum bump-up factors of approximately 15x for small breaks, 18x for medium breaks, and 24x for large breaks.

Table 2.5.36 – Exponential probability distribution parameters applied to chemical effects bump-up factors for each LOCA category

Parameters		SBLOCA	MBLOCA	LBLOCA	Tail Probability
Formal	Min	0	0	0	~1e-5
	Mean	1.25	1.5	2.0	~1e-5
	Max	14.3	17.2	23	~1e-5
Shifted	Min	1	1	1	~1e-5
	Mean	2.25	2.5	3.0	~1e-5
	Max	15.3	18.2	24	~1e-5

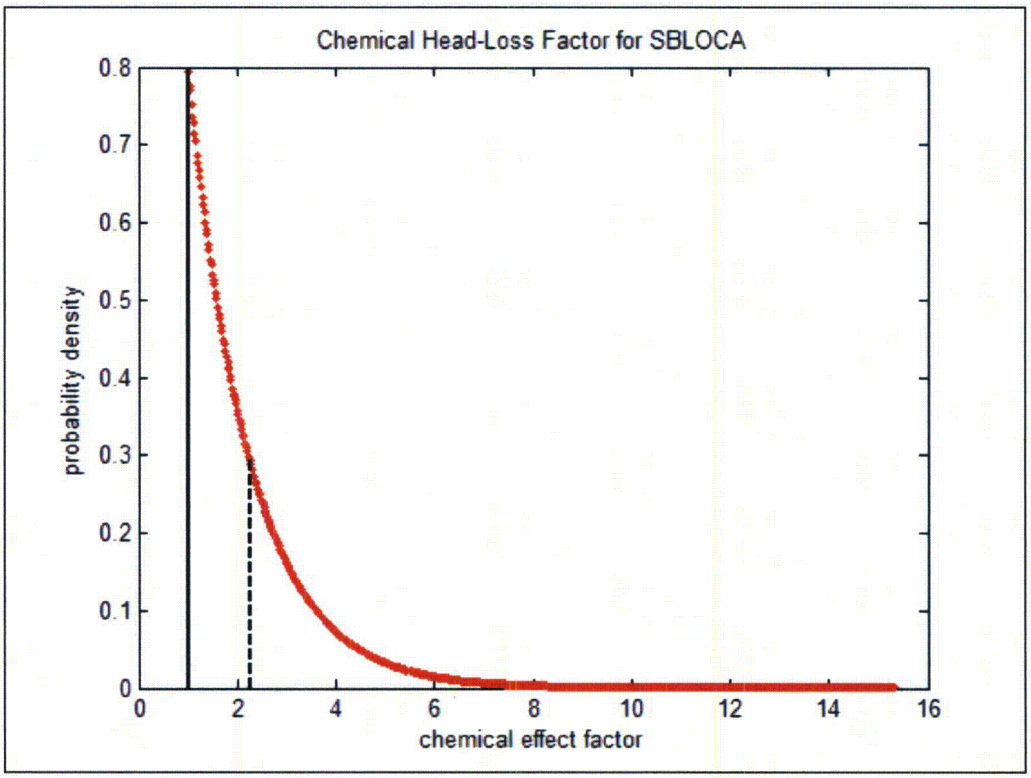

Figure 2.5.38 – Exponential probability density function for chemical effects bump-up factors applied to SBLOCAs

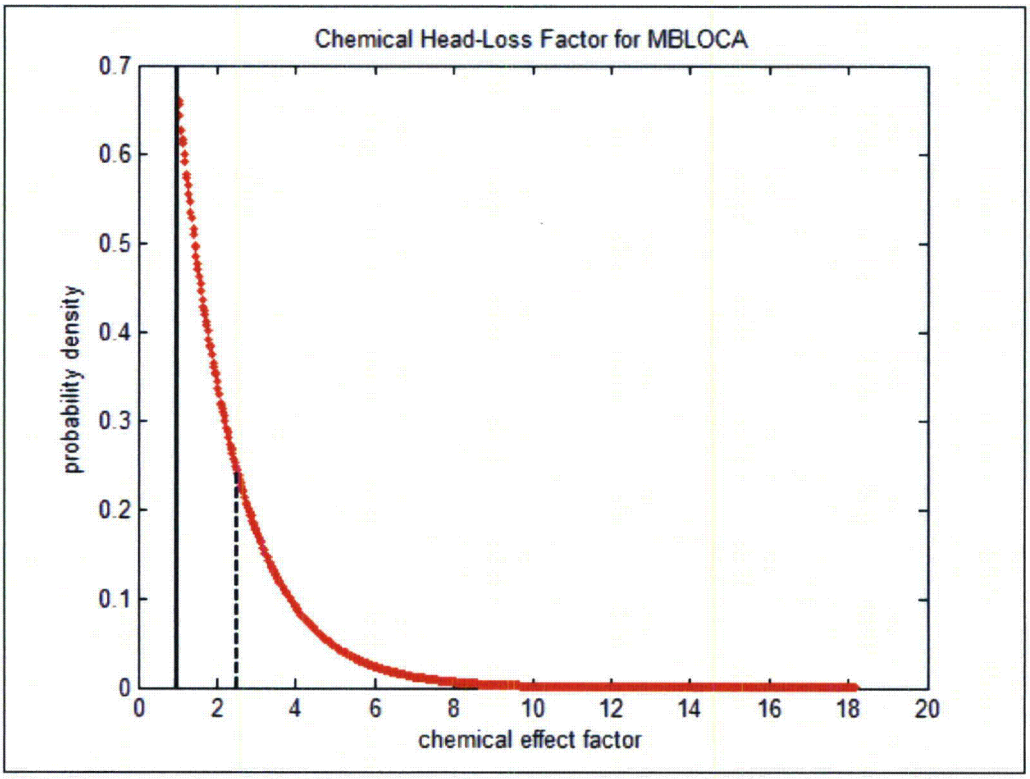

Figure 2.5.39 – Exponential probability density function for chemical effects bump-up factors applied to MBLOCAs

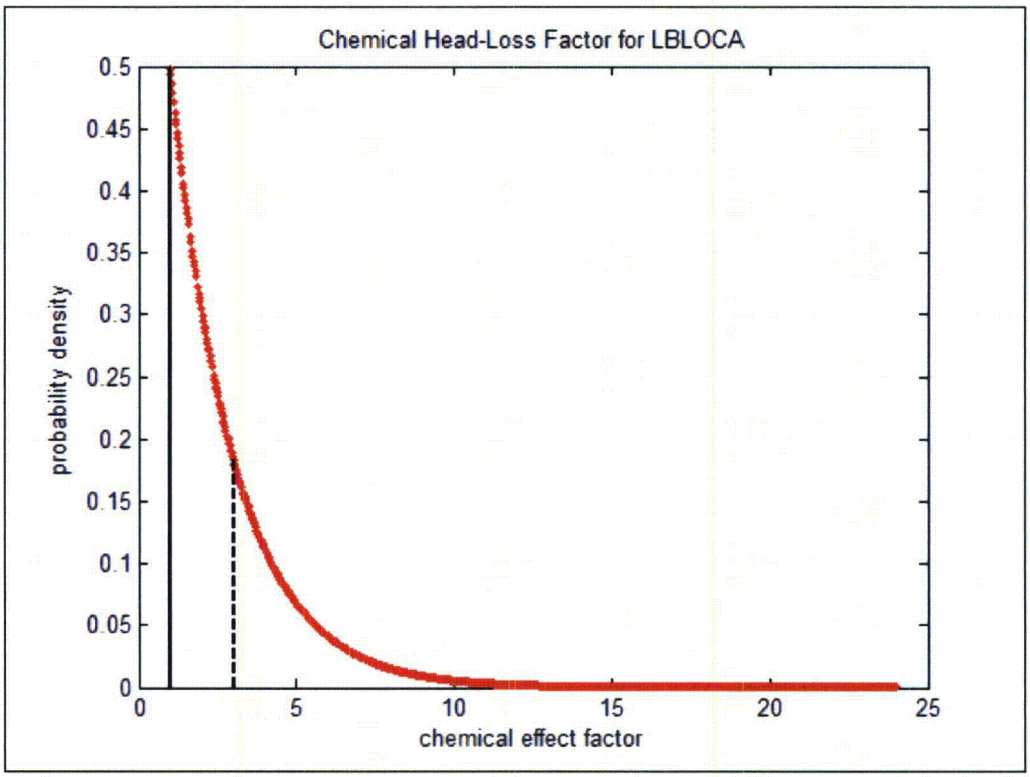

Figure 2.5.40 – Exponential probability density function for chemical effects bump-up factors applied to LBLOCAs

Figure 2.5.41 shows an example of the time-dependent total strainer head losses calculated for various random scenarios. In the final evaluation, strainer failure was not predicted for any of the small or medium breaks. The conditional probability of strainer failure given a large break scenario was only 2.45E-04 for the baseline case where all trains of ECCS are operating.

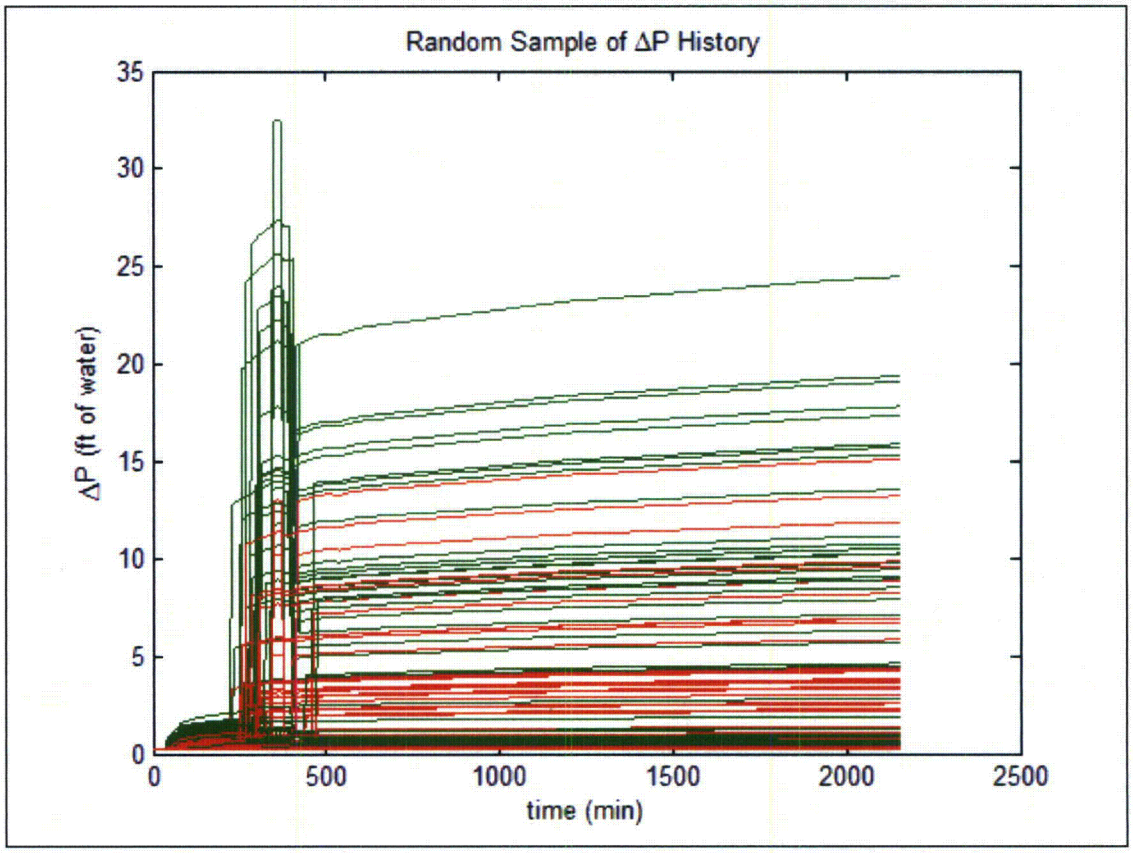

Figure 2.5.41 – Typical sample of sump-strainer head loss histories generated under the assumption of exponential chemical effects factor and artificial head-loss inflation

Additional details on the chemical effects bump-up factor probability distributions and the impact on the overall GSI-191 evaluation are provided in Volume 3 (12).

Item 5.a.12: Fiber Bypass

Method: Debris penetration is a function of two mechanisms. The first mechanism is direct passage of debris as it arrives on the strainer. A portion of the debris that initially arrives at the strainer will pass through, and the remainder of the debris will be captured by the strainers. The direct passage penetration is inversely proportional to the combined filtration efficiency of the strainer and the initial debris bed that forms. The second mechanism is shedding, which is the process of debris working its way through an existing bed and passing through the strainer. By definition, the fraction of debris that passes through the strainer by direct penetration will go to zero after the strainer has been fully covered with a fiberglass debris bed. Shedding, however, is a longer term phenomenon since particulate and small fiber debris may continue to work its way through the debris bed for the duration of the event.

Debris that penetrates the strainer can cause both ex-vessel and in-vessel problems. The most significant downstream effects concern is related to the quantity of fiberglass debris that accumulates in the core. This is a highly time-dependent process due to the following time-dependent parameters:

- Initiation of recirculation with cold leg injection
- Switchover to hot leg recirculation
- Arrival of debris at the strainer
- Accumulation of debris on the strainer
- Direct passage
- Debris shedding
- Flow changes when pumps are secured
- Decay heat boil-off

The various parameters associated with time-dependent debris accumulation on the strainer and core are illustrated in Figure 2.5.42, where $S_n(t)$ is the source rate for initial introduction of debris type n, V(t) is the pool volume, $m_n(t)$ is the mass of debris n in the pool, $f_n(t)$ is the filtration efficiency for debris n at the strainer, $s_n(t)$ is the shedding rate for debris n from the existing debris bed, Q(t) is the volumetric flow rate passing through the strainers, γ is the fraction of SI flow compared to the total flow, λ is the fraction of flow passing through the core compared to the total SI flow, and $g_n(t)$ is the filtration efficiency for debris n at the core.

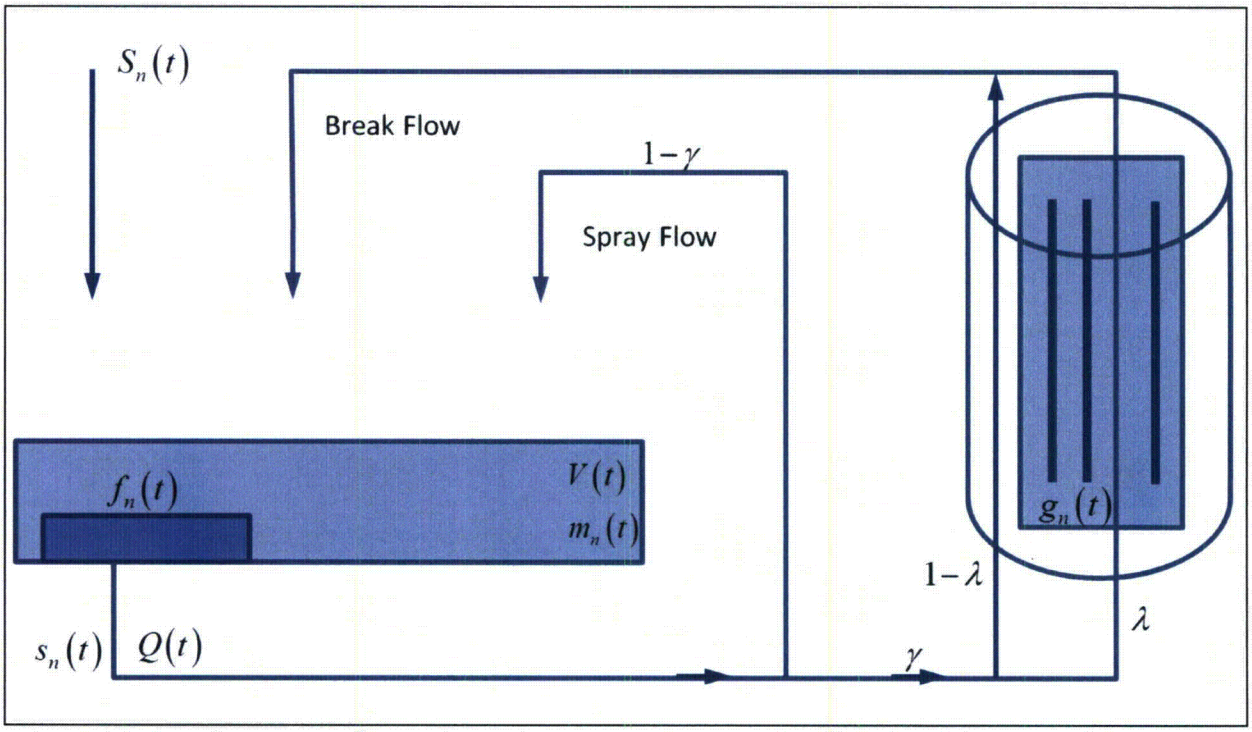

Figure 2.5.42 – Illustration of time-dependent parameters associated with debris accumulation on the strainer and core

As illustrated by Figure , debris that passes through the strainer will not necessarily end up on the core. A portion of the debris could pass through the containment spray pumps, and a portion could either

bypass or pass directly through the core and spill out the break. The debris that doesn't accumulate in the core may end up back in the pool where it could transport and potentially pass through the strainer again.

The methodology for determining the coefficients associated with the direct penetration and shedding terms is described in an evaluation (60) based on STP-specific testing (61). The methodology for calculating the time-dependent arrival of debris at the strainer, time-dependent penetration through the strainer, and time-dependent accumulation of debris on the core is described in Volume 3 (12). Additional discussion is also provided in the responses to Item 5.a.2 through Item 5.a.5 and Item 5.a.16.

Basis: The bases for the penetration correlation and the time-dependent accumulation on the core are described in detail in Volume 3 (12).

Assumptions: The following assumptions related to debris penetration and core accumulation were made in Volume 3 (12):

- It was assumed that the debris beds on the strainers would not be disrupted after the debris initially accumulates. This is a reasonable assumption since the strainers are not located in the immediate vicinity of any potential breaks.
- It was assumed that debris that penetrates the strainers would be uniformly distributed in the flow and would transport proportional to the flow split to the SI pumps vs. CS pumps (γ) and the flow split to the core vs. bypass paths (λ). This is a reasonable assumption since the fiber that penetrates the strainer would be very fine and would easily transport with the flow.
- It was assumed that all debris that penetrates the strainer and transports through the core would be trapped on the core (i.e. 100% filtration efficiency). This is a conservative assumption since it maximizes the debris load on the core.
- It was assumed that all debris that penetrates the strainer and bypasses the core (either through the containment sprays or directly out the break) would immediately be transported back to the containment pool. This is a conservative assumption since it neglects potential hold-up of debris in various locations and neglects the time that it would take for debris to transport through the systems and wash back to the pool.

Acceptance Criteria: The acceptance criteria for debris accumulation on the core are described in the response to Item 5.a.13.

Results: Discussion of the calculated debris accumulation on the core is provided in the response to Item 5.a.13.

Additional details on the STP penetration testing and statistical analysis are provided in the test report (61) and data analysis report (60). This is also discussed in response to Item 5.a.16.

Additional details on how the penetration correlation was implemented in the overall GSI-191 evaluation are provided in Volume 3 (12).

Item 5.a.13: In-Vessel Fiber Limits

Method: In-vessel fiber limits were selected based on an evaluation of the realistic phenomena associated with the core blockage and boron precipitation for the various scenarios (cold and hot leg breaks during the cold and hot leg injection phases). Thermal-hydraulic modeling was used to determine the potential for core damage to occur given full blockage at the bottom of the core at the start of recirculation for small, medium, and large cold leg and hot leg breaks (62). For the cases that could lead to core damage, the core blockage acceptance criteria in WCAP-16793-NP was used (63). In addition, a more stringent boron precipitation fiber limit criterion was used for medium and large cold leg breaks based on the draft NRC safety evaluation on WCAP-16793-NP (64).

Basis: The basis for the in-vessel acceptance criteria is a combination of STP-specific thermal-hydraulic model results, generic PWROG fuel blockage test results, and boron precipitation considerations. This is described in more detail in Volume 3 (12).

Assumptions: The following assumptions related to core blockage were made in Volume 3 (12). Additional assumptions related to boron precipitation are provided in the response to Item 5.a.15.

- It was assumed that a debris bed would not form at the top of the core (blocking flow to the core) for a hot leg break during the hot leg injection phase. This is a reasonable assumption since the majority of debris penetration would generally occur during the cold leg injection phase. Although it is possible that debris may continue to penetrate the strainer and reach the core even after switchover to hot leg injection, if a debris bed formed and blocked coolant from reaching the core, the water in the core would begin to boil. The boiling and counter-current flow would disrupt the debris bed and allow the SI flow to reach the core. Also, since one train of SI is always left on cold leg injection, it would be necessary for both the top and bottom of the core to be fully blocked by debris for core damage to occur during the hot leg injection phase. This is not considered to be a plausible scenario since the cross flow would prevent a debris bed from forming on either one side or the other.
- It was assumed that a debris bed would not form at the top of the core (blocking flow to the core) for a cold leg break during the hot leg injection phase. Similar to the previous assumption, limited debris penetration later in the event and countercurrent flow due to the buoyancy of the hot water in the core would break up any debris bed that starts to form on the top of the core.
- To calculate the boil-off flow rate for a cold leg break during cold leg injection, it was assumed that the RCS pressure is 14.7 psia, and the SI flow entering the reactor vessel is saturated liquid (i.e. 212 °F). This assumption conservatively maximizes the boil-off flow rate since a lower inlet temperature and/or a higher RCS pressure would increase the enthalpy required to boil the water.

Acceptance Criteria: The acceptance criteria for debris loads on the core were defined based on the break location, injection flow path, and fiberglass debris loads that could potentially cause issues for

debris blockage. Based on the thermal-hydraulic modeling, which showed that full blockage at the bottom of the core would not result in core damage for hot leg breaks, the acceptance criterion was set to essentially an infinite fiber quantity. For cold leg breaks, an acceptance criterion of 15 g/FA was used based on the conservative results of testing by the PWROG (63). Note, however, that the core blockage acceptance criteria are bounded by the boron precipitation acceptance criteria. As discussed in the response to Item 5.a.15, boron precipitation was not considered to be an issue for hot leg breaks. For medium and large cold leg breaks, the acceptance criterion for boron precipitation was assumed to be 7.5 g/FA of fiber debris on the core.

Results: In the final evaluation, failure due to core blockage or boron precipitation was not predicted for any of the small or medium breaks. The conditional probability of in-vessel failure given a large break scenario was only 6.94E-04 for the baseline case where all trains of ECCS are operating.

Additional details on the STP thermal-hydraulic analysis are provided in the core blockage calculation (62).

Additional details on how the in-vessel fiber acceptance criteria were defined for the overall GSI-191 evaluation are provided in Volume 3 (12).

Item 5.a.14: In-Vessel Thermal Hydraulic Analysis

Method: RELAP5-3D was selected to perform the simulation of the reactor system. The reactor containment response was simulated using MELCOR.

As described in the response to Item 5.a.13, a series of simulations were run to investigate worst case scenarios with full core blockage at the start of the ECCS recirculation phase. Two RELAP5-3D models were developed and used to conduct the simulations. The *3D Vessel – 1D Core* model was selected to run the basic simulations of the LOCA transients under a hypothesized full core and core bypass blockage. This model was selected because it combines the detailed nodalization of the vessel (using multi-dimensional components available in RELAP5-3D) accounting for more realistic injection flow paths, with the one-dimensional core and core bypass to minimize the simulation time. The following basic scenarios were simulated using the *3D Vessel – 1D Core* model:

- Small Break (2") in Cold Leg.
- Small Break (2") in Hot Leg.
- Medium Break (6") in Cold Leg.
- Medium Break (6) in Hot Leg.
- Double-Ended Guillotine Break (27.5") in Cold Leg.
- Double-Ended Guillotine Break (29") in Hot Leg.

Additional simulations were conducted to study the thermal-hydraulic behavior of the core under partial core blockage for a selected case (medium cold leg break), using the *3D Vessel – 3D Core* model. This model simulates the reactor core with multi-dimensional components, allowing partial core blockage (by fuel channel) with a relatively larger simulation time. The additional cases that were run include:

- Full core blockage and full core bypass blockage (confirmation of *3D Vessel – 1D Core* model results)
- Full core blockage with open core bypass
- Core bypass blockage and core blockage except for one fuel channel (center)
- Core bypass blockage and core blockage except for one fuel channel (periphery)

Both models used for these simulation sets were originated from a *Full 1D* model, which is described in detail in the input certification report (65).

Basis: The basis for the RELAP5 simulations are described in the core blockage calculation and input certification report (62; 65).

Reactor System Response

A set of RELAP5-3D models were developed to perform the analysis of the STP reactor system during LOCA scenarios of different break size and locations.

Full 1D model

This model uses one-dimensional components such as single-volumes, branches, and pipes to simulate the regions of the reactor system. The primary cooling loops were simulated independently to account for the expected flow asymmetry during the phases of the injection. The power plant was modeled using a total of 283 nodes. The total number of junctions defined in the model was 293. The reactor core was modeled using two one-dimensional pipe components. One pipe component was used to simulate the average channels, where 192 assemblies were lumped together, and one pipe component was used to simulate the hot channel. Heat structures were defined and connected to the hot and average channels to simulate the fuel pins, including the fuel, gap, and cladding regions. Radial and axial peaking factors were defined to distribute the total power of the reactor within the average channels and the hot channel (hot rod and average rods in hot channel). The model nodalization diagram is shown in Figure 2.5.43 (65).

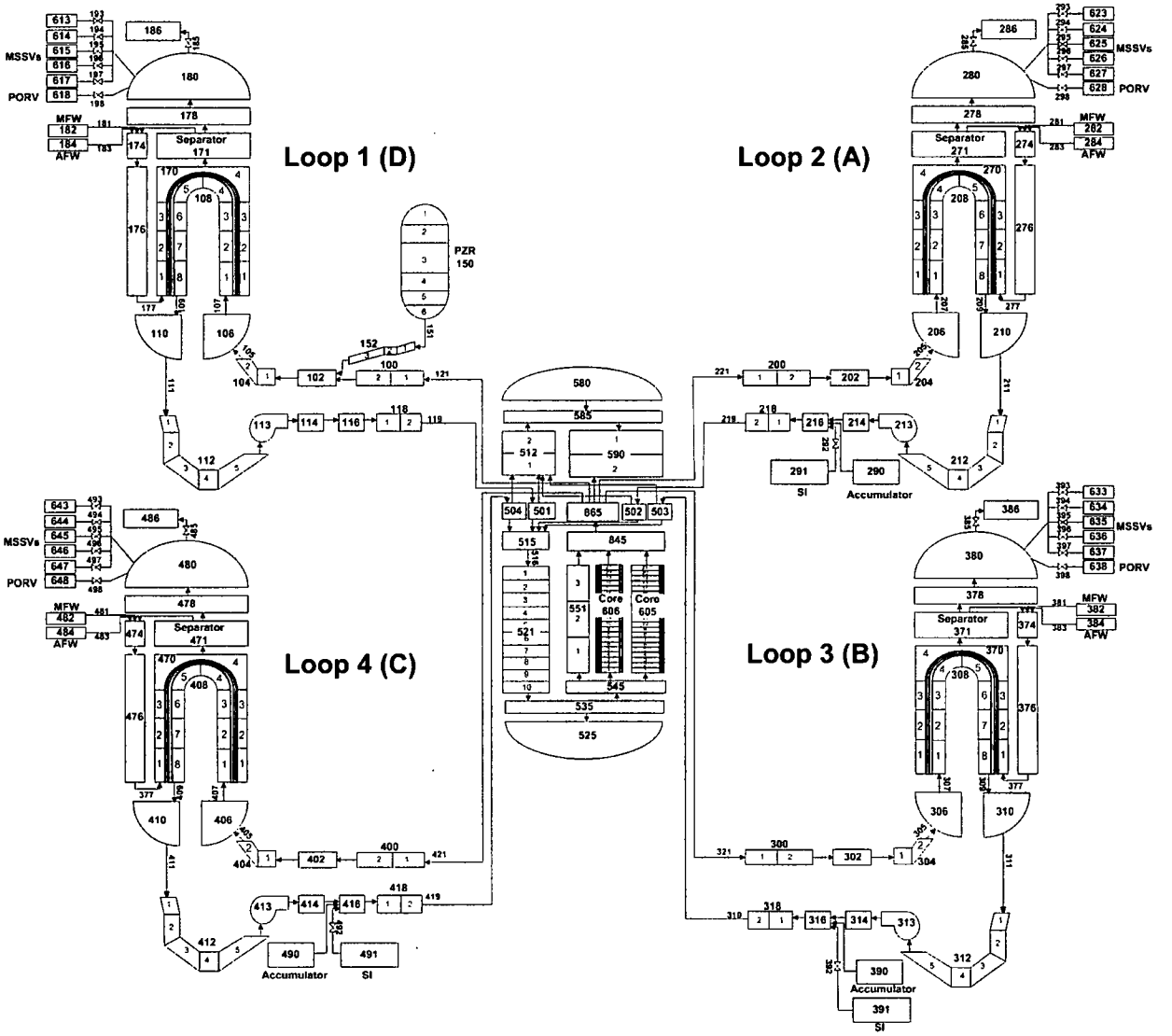

Figure 2.5.43 – RELAP5-3D - 1D Model Nodalization Diagram

The Safety Injection system of the power plant consists of three independent trains, which were simulated according to the realistic geometry. Each train contains:

- One Accumulator simulated using the accumulator component available in RELAP5-3D. A control variable was defined in order to isolate the accumulator during special LOCA manual operation procedures.
- One HPSI modeled with a time-dependent junction, where a table of the velocity of the liquid to be injected as a function of the pressure of the primary system injection location was defined.
- One LPSI modeled with a time-dependent junction, where the same approach applied to the HPSI was used.
- One RHR connected downstream the LPSI pump simulated.

- The RWST and sump, modeled using a common time-dependent volume, where the condition of the thermal-hydraulic conditions of the water before and after the sump switchover were defined using a table controlled by a trip function (defined true at sump switchover and false otherwise).
- A set of two trip valves to control the injection location (cold or hot leg).

The nodalization adopted for the injection system is shown in Figure 2.5.44.

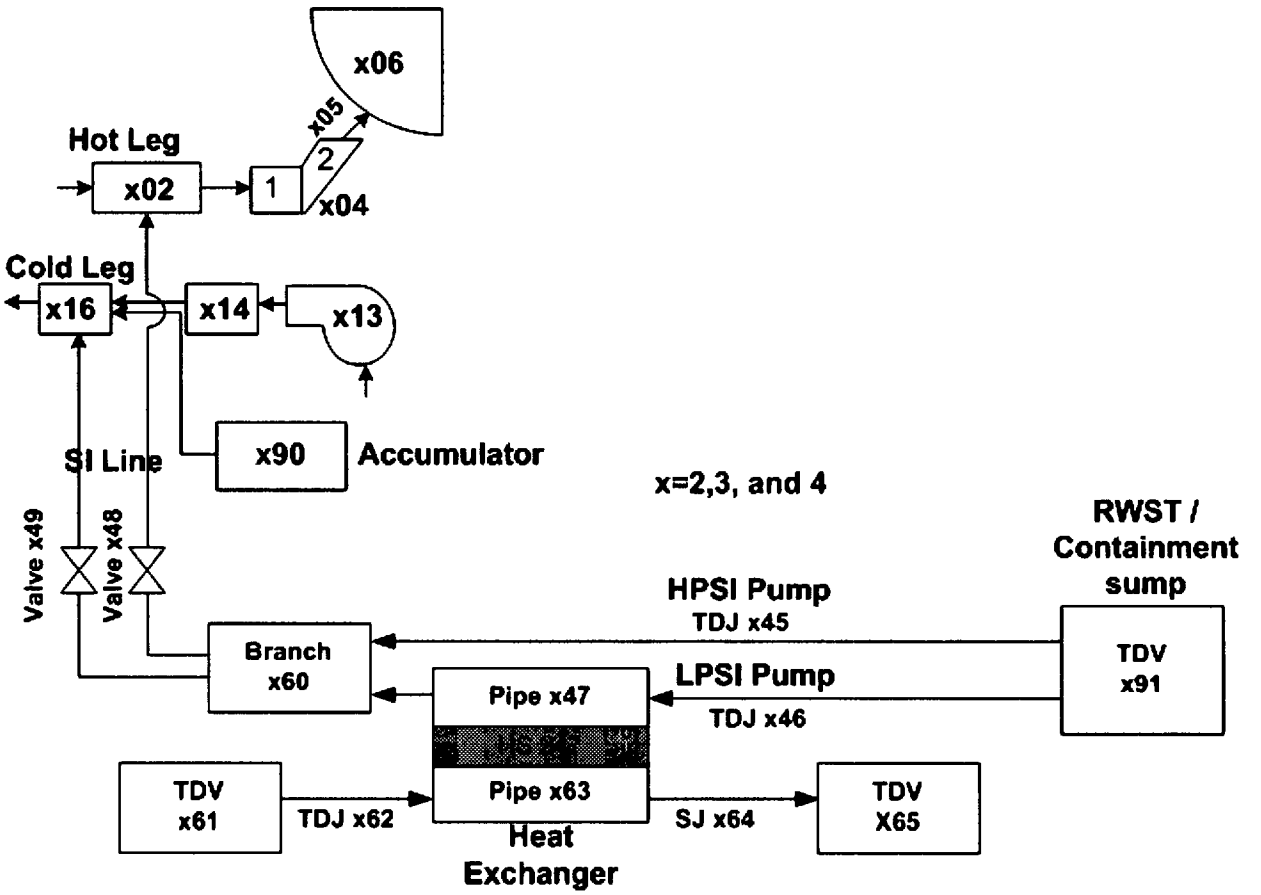

Figure 2.5.44 – SI System Nodalization Diagram

3D models

Two 3D models (3D Vessel – 1D Core and 3D Vessel – 3D Core) were prepared to perform selected simulations of LOCA scenarios, to achieve a more realistic representation of selected regions of the vessel and the core. In particular, multi-dimensional Cartesian components were used to simulate 193 fuel channels with cross-flow. The regions outside the vessel (loops, steam generators, and SI system was simulated with one-dimensional components as described for the 1D Model. This specific model was used to perform selected simulation of LOCA scenarios under hypothetical core blockage. The total number of nodes defined in the 3D Vessel – 3D Core model is 4235. The number of junctions is 11627. The nodalization diagram for these models is shown in Figure 2.5.45.

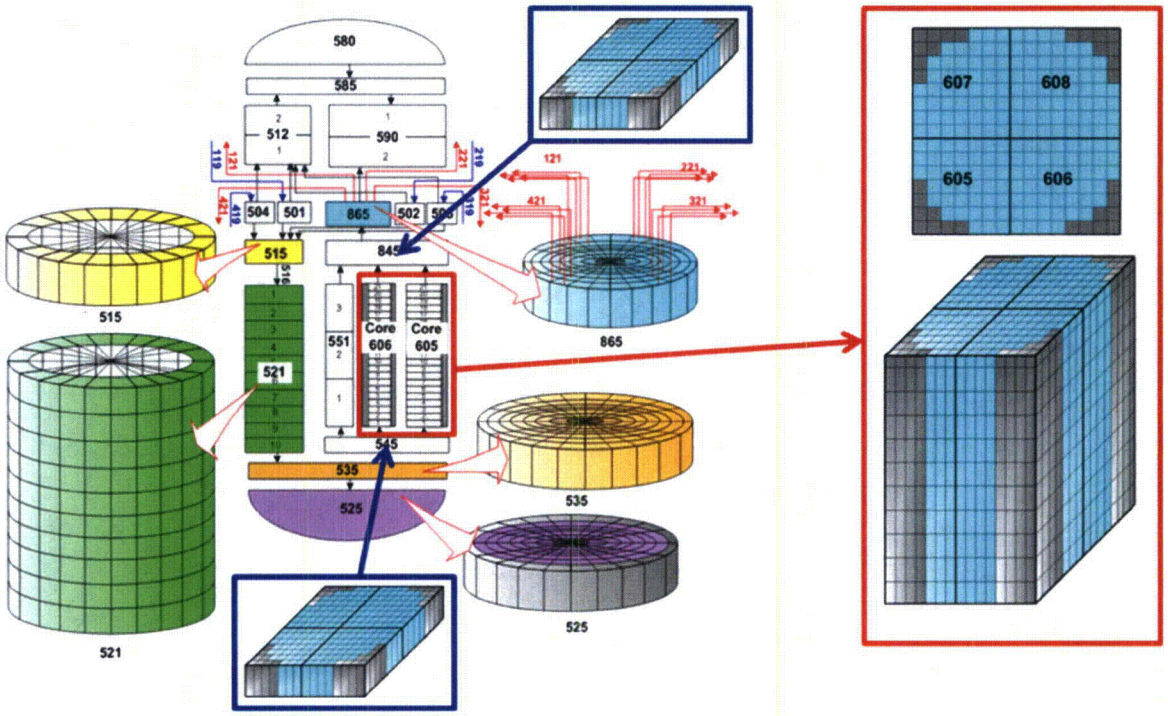

Figure 2.5.45 – RELAP5-3D - 3D Models Nodalization Diagrams (Left: 1D Core; Right: 3D Core)

Containment Response

The MELCOR containment model of the STP PWR large, dry containment was prepared to run the simulations of the containment response during the LOCA scenarios. The model consists of six control volumes, eleven flow paths, and forty-nine heat structures that represent ceilings, walls, and floors among other condensing surfaces. Cavity compartment, lower compartment, steam-generator compartment (where the break in the primary system is assumed to discharge), upper compartment, annular compartment (where the sump is located), and pressurizer compartment were created in the containment model. Each compartment is a single control volume with geometry defined by volume/altitude tables. Additional flow paths were included in the model to represent drainage pathways from the refueling canal (lower portion of the upper compartment) to the steam generator compartment. Heat structures were defined to simulate walls, floor and other structures in the reactor containment. Pressure suppression features (engineered safety features) such as six fan coolers and three containment sprays were included in the model and behaved realistically according to actuation set-points, delays, ramp-ups, coast-downs, etc. Control functions were defined to control the fan coolers and sprays actuation and manual operations (sprays shutoff). Figure 2.5.46 shows the MELCOR nodalization diagram adopted (66).

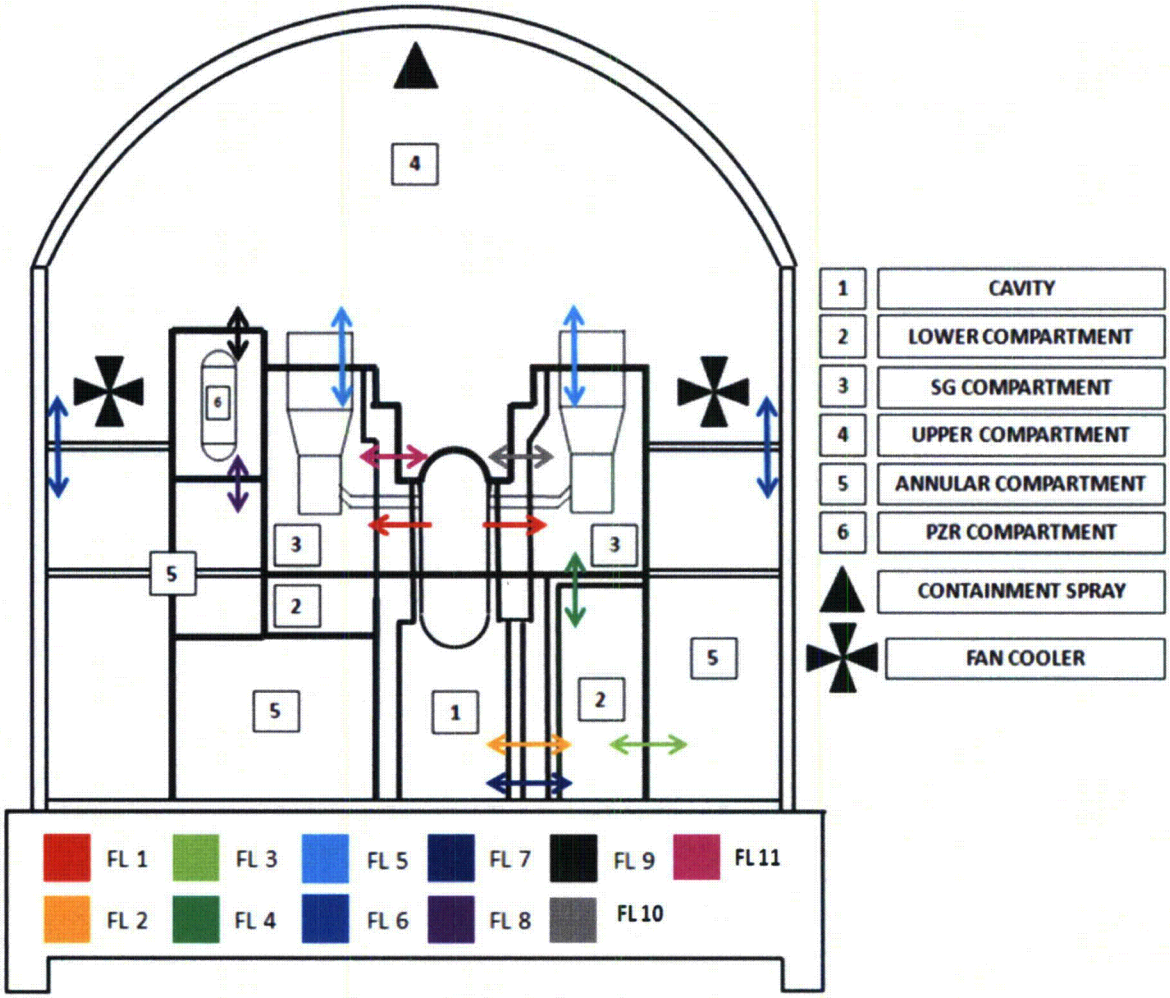

Figure 2.5.46 – MELCOR Nodalization Diagram

Validation Approach

The RELAP5 and MELCOR models were based on existing certified input files of the reactor system and containment.

The RELAP5-3D model was created starting from the RETRAN input file of the reactor system (65). All the geometrical information of the regions of the system were extracted from the RETRAN input file and specified in the RELAP input deck. Additional information was extracted from the MAAP input file for selected components (Steam Generators, SI system). The validation of the model was conducted by:

- Comparing the steady-state results for selected parameters with the plant normal operating conditions.
- Comparing the simulation results for selected LOCA scenarios with the plant simulator results.

The MELCOR model of the reactor containment was based on the existing MAAP input file (66). The model was modified to account for additional details retrieved from the CAD drawings and other plant-specific information.

Assumptions: The assumptions associated with the thermal-hydraulic simulations include the following (62):

- In all the cases, the break orientation was imposed at the bottom of the leg. In all the models used for the simulations, all the safety systems were assumed to be available throughout the transient.
- For the blowdown phase, it was assumed that the break would open instantaneously.
- For the six cases run with the *3D Vessel – 1D Core* model, it was assumed that the bottom of the core and baffle bypass flow paths would be fully blocked just after the start of ECCS recirculation. This is a conservative assumption because of the time-dependence associated with debris transport and accumulation. Also, significant quantities of debris would be required to reach the point where the core could be considered fully blocked (in most cases, full blockage would never occur).
- For the three additional cases run with open flow through the core bypass channel, center fuel assembly channel, or periphery fuel assembly channel, it was assumed that blockage of the other channels would occur just after the start of ECCS recirculation similar to the initial six cases.
- All the cases which produced a peak cladding temperature increase due to the core blockage which did not exceed 800 °F were assumed to be successful cases which may not lead to core damage.

Acceptance Criteria: The acceptance criterion for the thermal-hydraulic simulations was a peak cladding temperature of 800 °F.

Results: The models prepared were used to perform selected simulations of the reactor system and containment response under specific conditions. Three sets of simulations were performed:

1) Core Blockage Simulations to Support the In-Vessel Effects Analysis (62)
2) Sump Temperature Sensitivity Analysis to Support CASA Grande Calculations (32)
3) Sump Temperature Analysis for and Medium (6") and Large (DEG) LOCA to Support the CHLE Tests (67; 68)

Core Blockage Simulations to Support the In-Vessel Effects Analysis

The simulations were conducted using the RELAP5 model (3D Vessel – 1D core and 3D Vessel – 3D Core) to analyze the reactor system response under hypothetical core blockage scenarios during selected Loss of Coolant Accident (LOCA). The purpose of these calculations was to:

1) Identify the scenarios which may produce an increase in the peak cladding temperature and, subsequently, a potential core damage among selected LOCAs of different break sizes and locations under full core and core bypass blockage.
2) For the cases identified in 1), analyze the system response under a partial core blockage hypothesis.

The simulations performed for this task are listed below:

- Small Break (2") in Cold Leg.
- Small Break (2") in Hot Leg.
- Medium Break (6") in Cold Leg.
- Medium Break (6) in Hot Leg.
- Double-Ended Guillotine (DEG) Break in Cold Leg.
- Double-Ended Guillotine (DEG) Break in Hot Leg.

Table 2.5.37 summarized the basic assumptions and boundary conditions for the simulations:

Table 2.5.37 – Boundary Conditions

Parameter	Simulation Condition
ECCS	3 Trains Running (A, B, C)
Break Location	Cold Leg B, bottom
Core Blockage Methodology	Istantaneous Increase of the k-loss after Sump Switchover
Reactor Core Power (MWt)	3853
Axial Power Shape	Double Peak (0.15 and 0.8 Core Height)
Actinides	RELAP5-3D Default Actinide Model
Decay Heat Model	ANS73 +0%
RWST Temperature	85 F
ECCS Flow	Realistic

The blockage of the core was assumed to be instantaneous after the sump switchover. Both core and core bypass (baffle flow) were assumed to be blocked at the bottom. The peaking cladding temperature was used as figure of merit to determine the success of failure of the scenario simulated. All the cases which produced a peak cladding temperature after the core blockage which did not exceed 800 °F were assumed to be successful cases, which may not lead to core damage. The cases where the maximum peak cladding temperature was found to diverge after the core blockage time (exceeding the limiting temperature of 800 °F) were considered failing cases, which may lead to core damage. The peak cladding temperature for the scenarios analyzed is plotted in Figure 2.5.47 (small breaks), Figure 2.5.48 (medium breaks), and Figure 2.5.49 (DEG breaks).

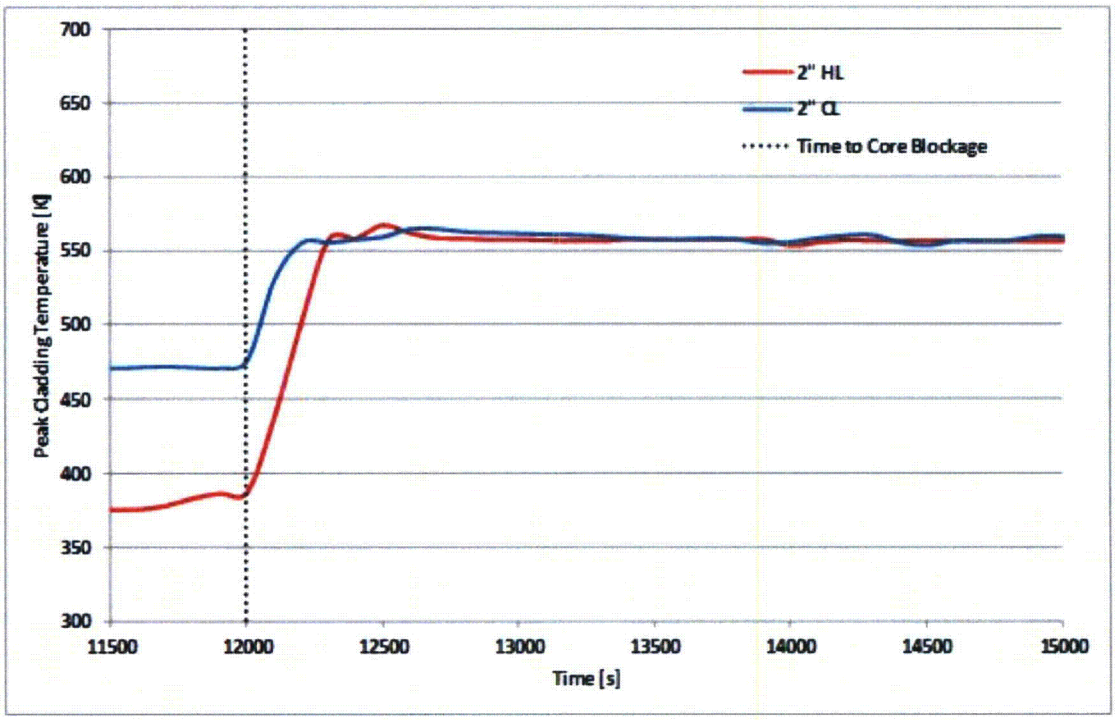

Figure 2.5.47 – Small break LOCA peak cladding temperature

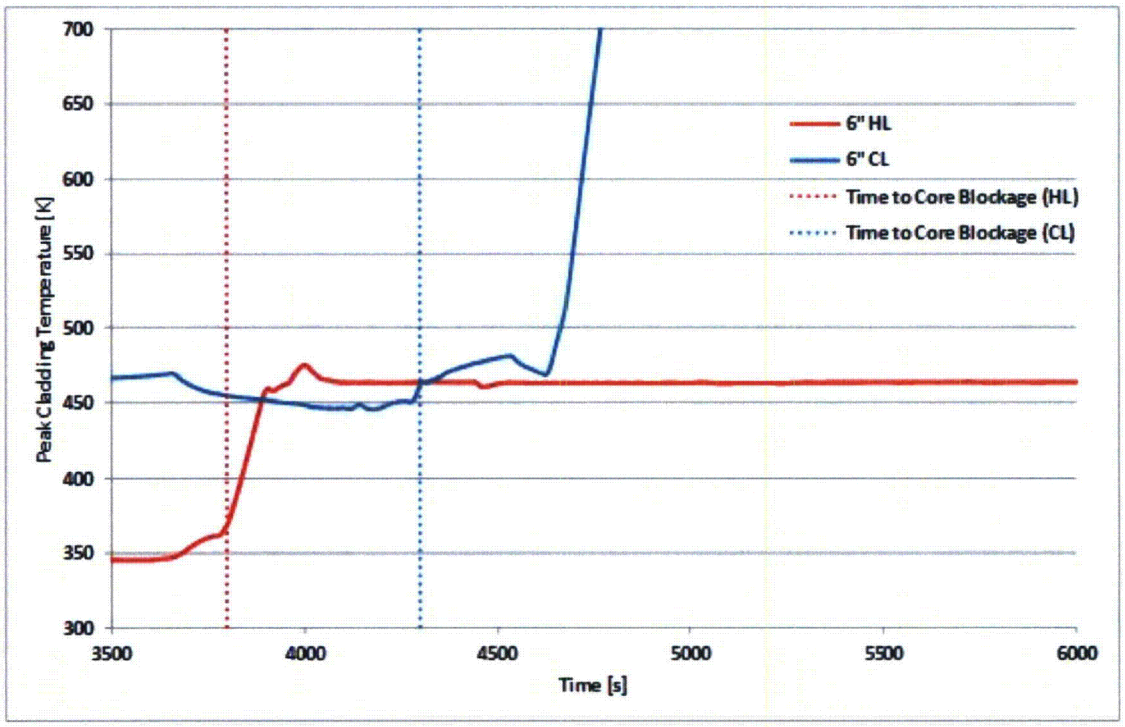

Figure 2.5.48 – Medium break LOCA peak cladding temperature

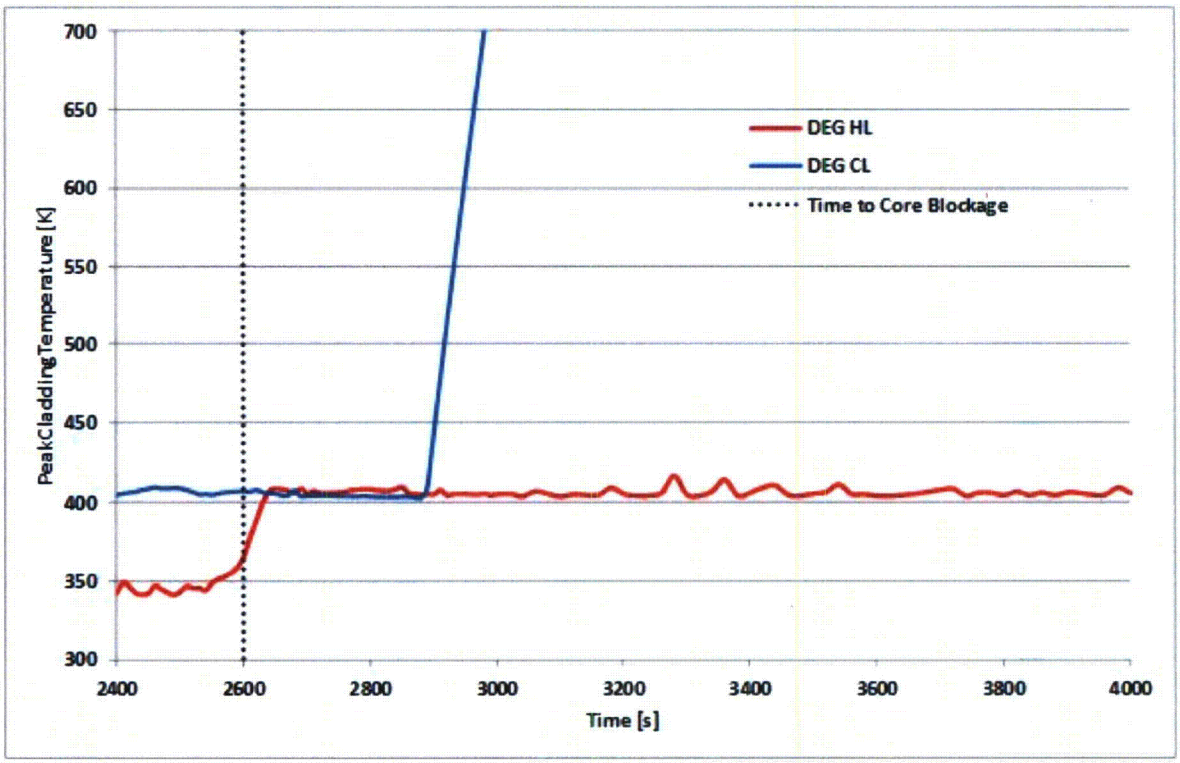

Figure 2.5.49 – DEG break LOCA peak cladding temperature

Table 2.5.38 summarizes the results obtained.

Table 2.5.38 – Core Blockage Scenarios - Summary

	Break Location	
Break Size	**Cold Leg**	**Hot Leg**
Small (2")	Pass	Pass
Medium (6")	Fail	Pass
Large (DEG)	Fail	Pass

Cold Leg Break Scenarios

For smaller breaks (2"), the injection system was found to be able to refill the steam generators with liquid water so that, at the time of core blockage, an alternative flow path was already available for the cooling water to reach to top of the core (from the cold leg injection point, through the steam generators tubes, and to the top of the core via the hot legs).

For larger break sizes (6" and DEG), the break flow takes most of the cooling water coming from the two intact injection loops. The water injected through the cold leg preferentially moved in the downcomer toward the broken cold leg. The steam generators were found to be empty at the time of core blockage so no available alternative flow paths were observed for these cases. The core peak cladding temperature was found to diverge starting from the core blockage time. The simulations were stopped when the maximum limit of 800 °F was reached.

Hot Leg Break Scenarios

Due to the break location compared to the loop injection location (cold leg) at the time to core blockage, the injected cooling water was forced to flow through the steam generators and reach the upper plenum before leaving the vessel through the broken hot leg.

Additional Simulations

Assuming the maximum peak cladding temperature as figure of merit, the scheme presented in Figure 2.5.50 summarizes the results of the simulations performed with the *3D Vessel – 3D Core* model (pass/fail).

Confirmed 3D Vessel – 1D Core Model Simulation Results

Case	Description	Result
1	Full Core + Bypass Blockage	Fail
2	Full Core Blockage, Free Bypass	Pass
3	Full Core Blockage except 1 FA (Center)	Pass
4	Full Core Blockage except 1 FA (Periphery)	Pass

Core Bypass (between Baffle and Barrel) may provide alternative flow path and allow decay heat removal

The coolant flow through one Fuel Assembly (FA) was found to be enough to remove the decay heat

Figure 2.5.50 – Summary of thermal-hydraulic simulations for partial blockage

Case 1 showed what was previously found using the *3D Vessel – 1D Core* model: the peak cladding temperature steadily increases reaching the maximum limit, confirming that the full core and core bypass blockage assumption imposed in this case may lead to core damage. Case 2 showed that the flow through the core bypass is sufficient to provide the required coolant flow at the top of the core and minimize the peak cladding temperature, even if the core is assumed to be fully blocked. Cases 3 and 4 predicted a sufficient flow through only one free fuel assembly to supply the required coolant flow and maintain the peak cladding temperature under the limit.

Sump Temperature Sensitivity Analysis to Support CASA Grande Calculations

Simulations were conducted to study the effects of selected thermal-hydraulic parameters on the containment water temperature during the phases of Loss of Coolant Accidents (LOCA). LOCA scenarios of different break sizes (very small, small, medium, and large) were analyzed. Different conditions were also considered in the analysis to investigate the behavior of the reactor containment response. This included engineering features unavailability (fan coolers and containment sprays) or other operating conditions such as Component Cooling Water temperature and Residual Heat Removal heat exchanger's availability. RELAP5-3D was used to perform the simulations of the primary system (1D Model). The containment response was simulated using MELCOR.

The independent parameters selected for this analysis were:

- Break Size
- Number of Operating Containment Fan Coolers (2 or 6 in operation)
- Number of Operating Containment Sprays (3 trains or no trains available)

- Number of Operating Residual Heat Removal (RHR) Heat Exchangers (3 or no exchangers available)
- Component Cooling Water (CCW) Temperature (60 °F, 85.84 °F or 150 °F)

From the same sets of simulations, selected thermal-hydraulic parameters related to the primary system were also analyzed, such as:

- Time to Sump Switchover
- Total Safety Injection (SI) Flow Rate

Table 2.5.39 summarizes the cases simulated. All the cases assumed the break to be in cold leg. The maximum sump temperature (Column "Maximum" in Table 2.5.39) was achieved by minimizing the system heat removal capabilities (only two operating fan coolers, no containment sprays and RHR heat exchangers available, maximum CCW temperature). The minimum sump temperature (Column "Minimum" in Table) was achieved by maximizing the system heat removal capabilities (all the engineering features available, minimum CCW temperature). Complete engineering features availability and nominal CCW temperature were assumed to simulate the nominal (Column "Nominal" in Table) conditions.

Table 2.5.39 – Simulation matrix

	Operating Conditions (see Table 1.)		
Break Size	**Minimum**	**Nominal**	**Maximum**
1.5"		X	
2"	X	X	X
4"		X	X
6"	X	X	X
8"		X	X
15"		X	X
DEG (27.5")		X	X

Results were summarized in terms of containment pressure and sump temperature profiles as a function of the break size and implemented into CASA Grande. Figure 2.5.51 shows an example of the sump temperature profiles for the nominal case (by break size) used for the CASA Grande calculations. Examples of additional parameters extracted, such as sump switchover time and total SI flow rate, are shown in Table 2.5.40.

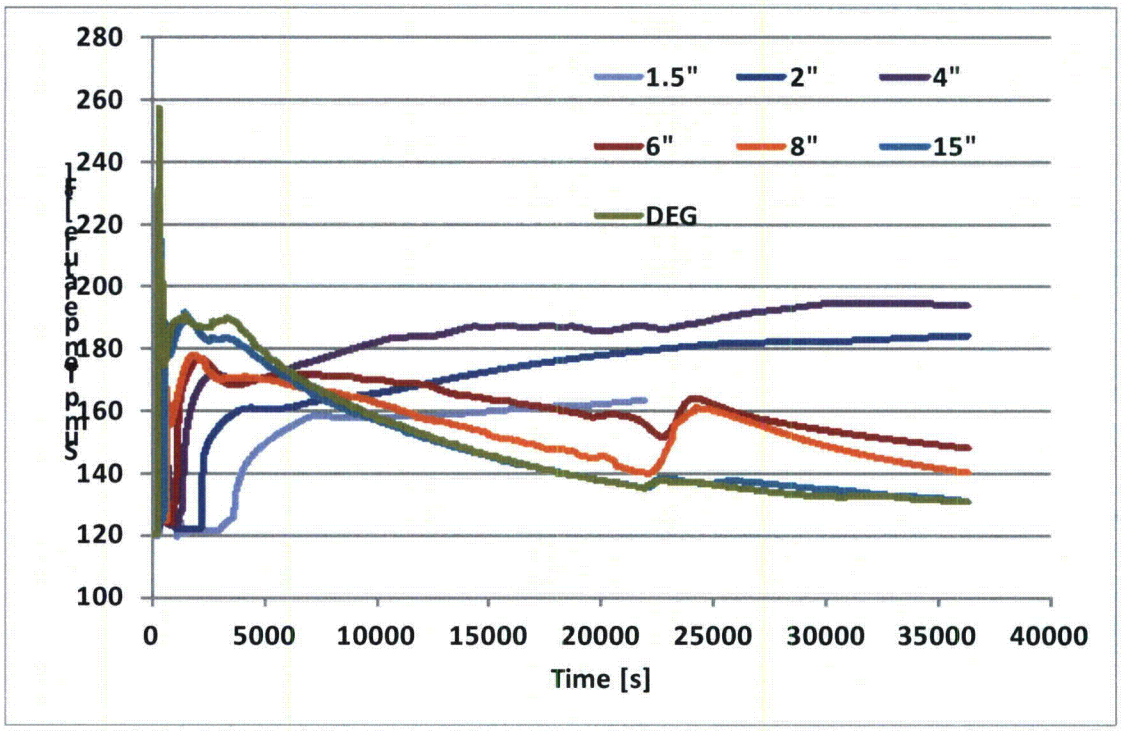

Figure 2.5.51 – Example of the sump sensitivity results (nominal cases)

Table 2.5.40 – Results summary (nominal cases)

Break Size	1.5"	2"	4"	6"	8"	15"	DEG
Sump Switchover Time	5.6h	1.3h	55.9m	44.2m	37.8m	31.2m	29.5m
Total SI (US gal/min)	1230.686	2075.797	4119.646	7950.749	10285.35	11779.73	11988.22

Sump Temperature Analysis for a Medium (6") and Large (DEG) LOCA to Support the CHLE Tests

The simulation of the containment response during a medium (6") and large (DEG) break Loss of Coolant Accident (LOCA) scenario in the cold leg was performed in order to predict the sump water temperature profile during a period of 30 days from the break event, to support the CHLE tests. The MELCOR model was used to perform the calculations of the reactor containment response. The boundary conditions for the MELCOR simulations were calculated with the RELAP5-3D (1D Model) model. The simulation approach was similar to the one described for the nominal cases for the sump temperature sensitivity analysis. Selected thermal-hydraulic parameters of reactor containment and primary system were provided to the CHLE test team, such as:

- Sump Compartment Water Temperature
- Water Temperature Variation through the Residual Heat Removal (RHR) Heat Exchangers

The simulations were extended to comprehend:

a) A 300s *Steady-State* Phase

b) The *Safety Injection* phase (from break opening at t = 300s to the sump switchover time)
c) The *Long-Term Cooling* phase with *Cold Leg Injection* (from the sump switchover time to the Hot Leg switchover time)
d) The *Long-Term Cooling* phase with the *Hot Leg Injection* (from the Hot Leg switchover time up to t_{end} = 30 days)

Figure 2.5.52 and Figure 2.5.53 show the sump pool temperature profile calculated with MELCOR for the 6" cold leg break scenarios. The temperature profile for the 15" cold leg break scenario is shown in Figure 2.5.54 and Figure 2.5.55.

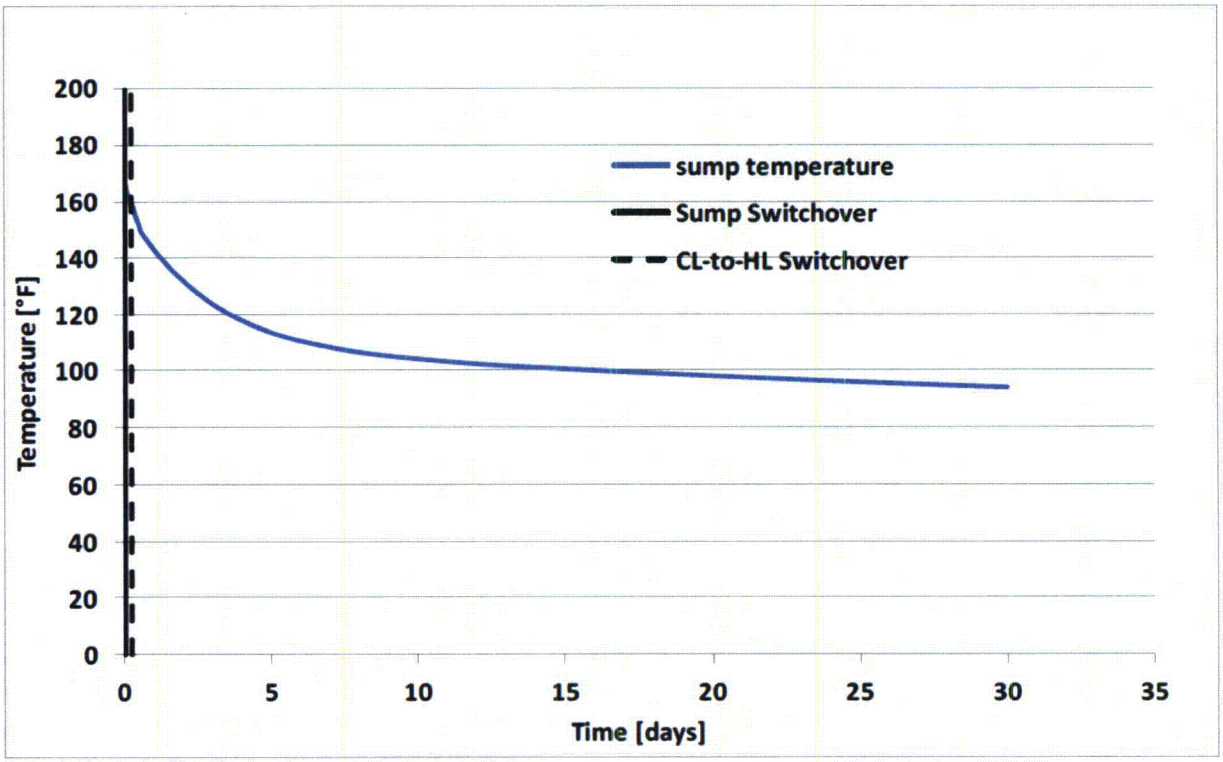

Figure 2.5.52 – 6" break sump temperature profile (30-day overview)

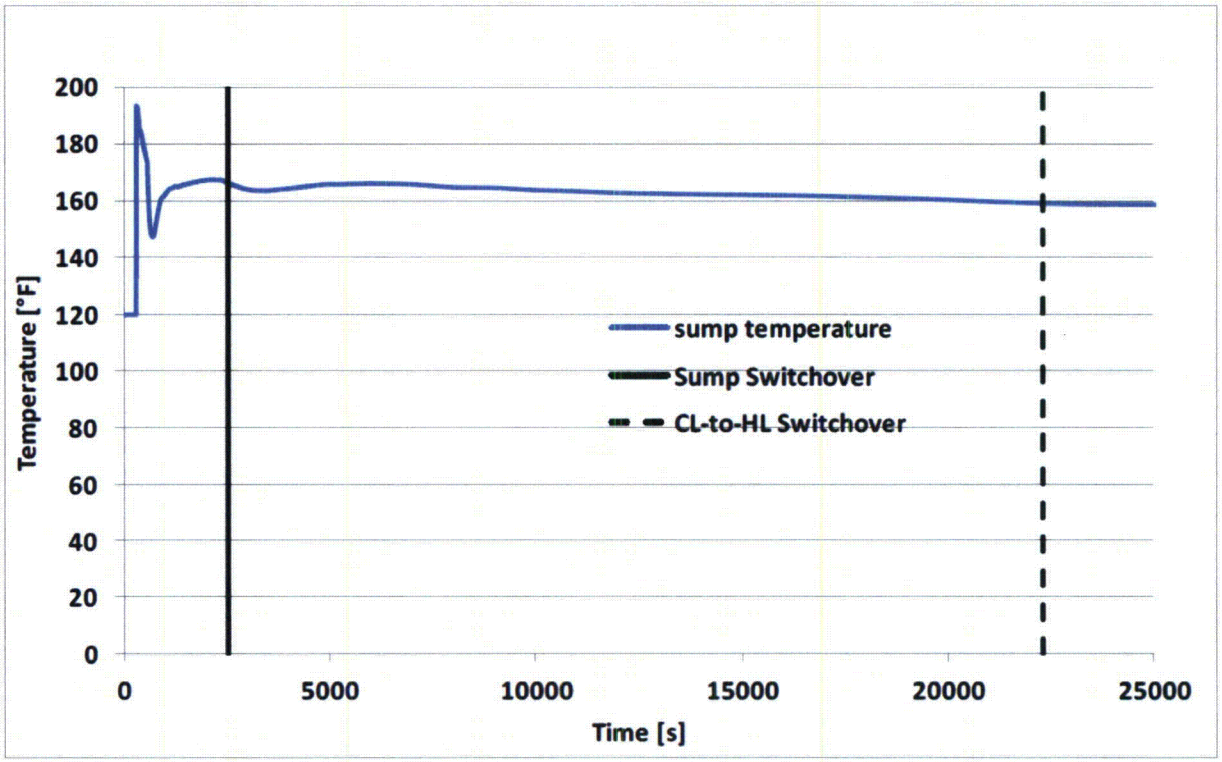

Figure 2.5.53 – 6" break sump temperature profile (zoom)

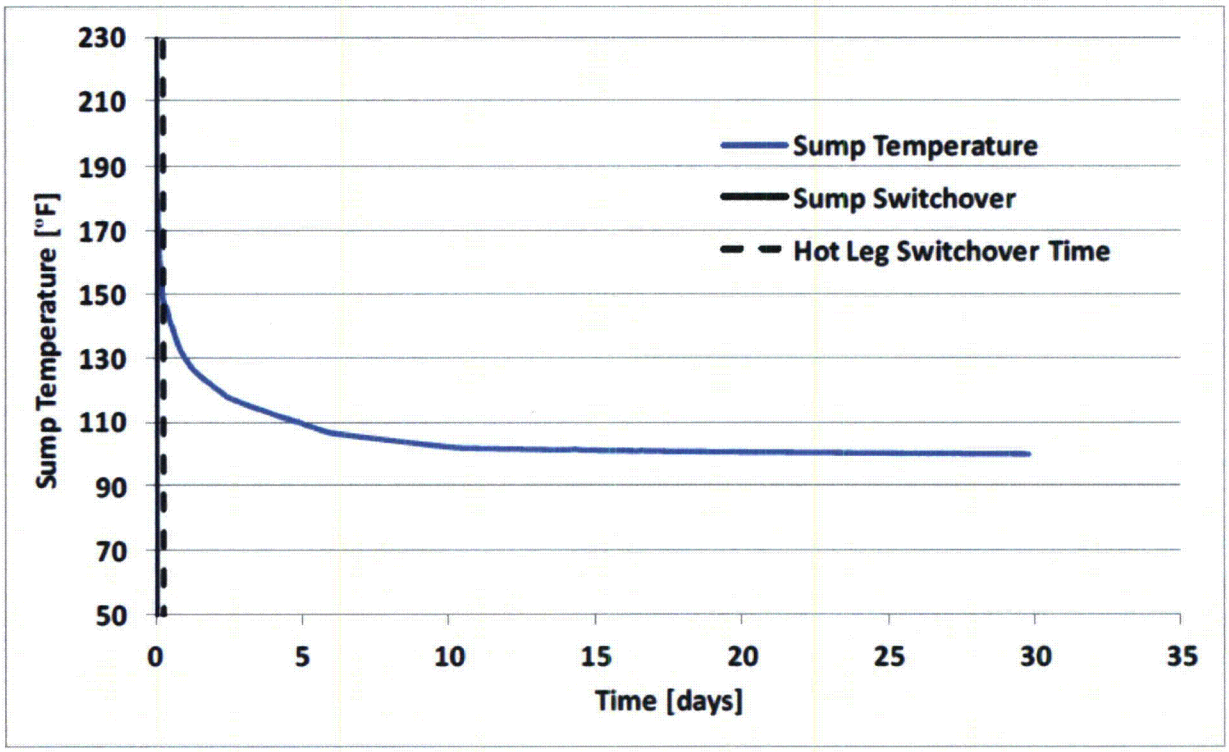

Figure 2.5.54 – 15" break sump temperature profile (30-day overview)

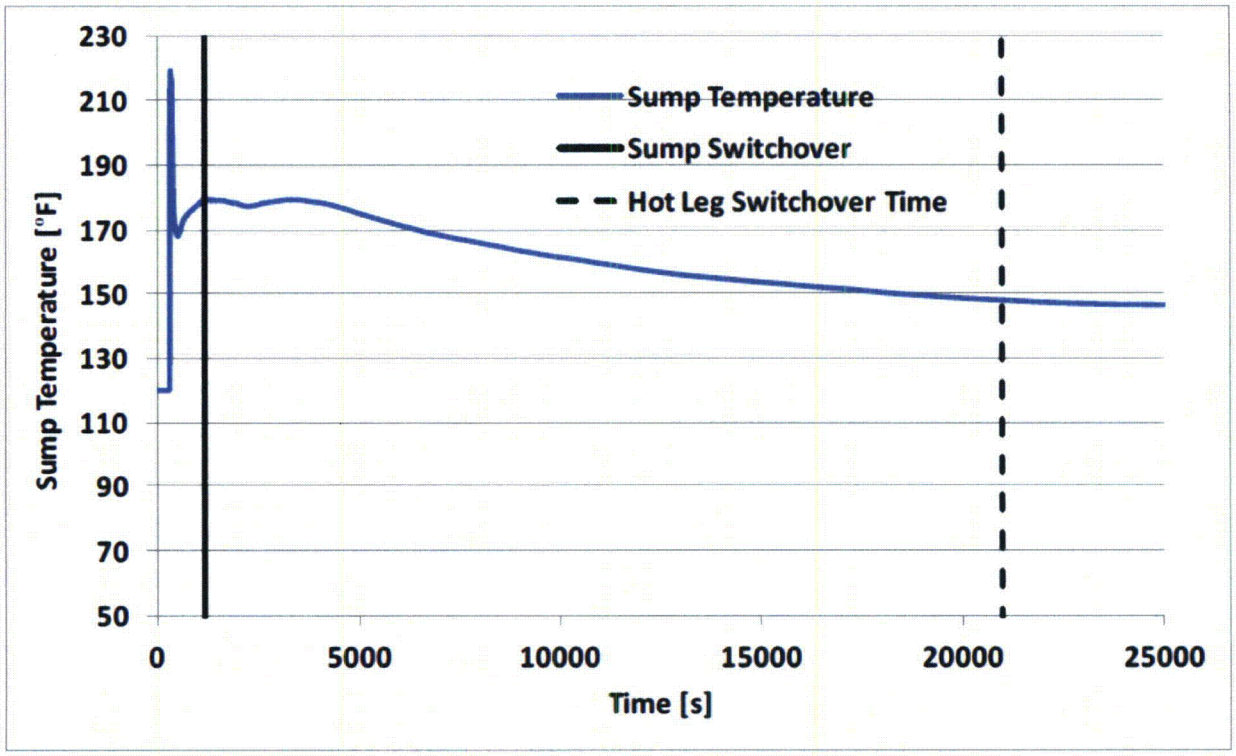

Figure 2.5.55 – 15" break sump temperature profile (zoom)

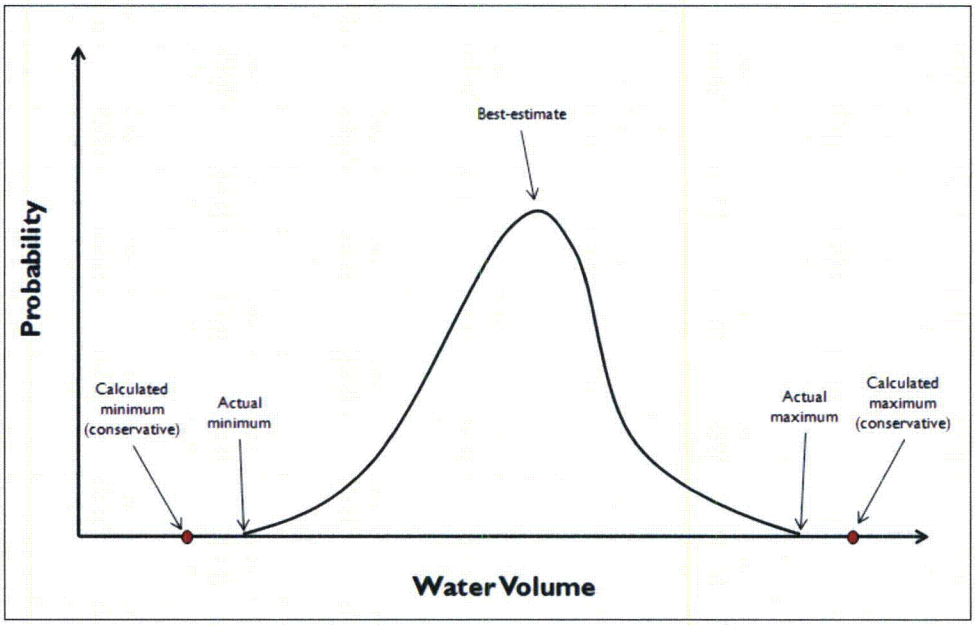

Figure 2.5.56 – Example of realistic probability distribution for an input variable

In addition to using realistic inputs, it is also important to perform a time-dependent evaluation to capture the time-dependent factors and events that are significant to GSI-191. This includes time-dependent failure for unqualified coatings, time-dependent transport of debris to the strainers, time-dependent corrosion and subsequent precipitation of chemical products, time-dependent operator actions such as securing pumps or switching over to hot leg injection, etc.

Some specific examples of how the probability distributions are developed have been provided in the response to Item 5.a.16. A summary of the input parameters used in the STP risk-informed GSI-191 evaluation is shown in Table 2.5.60. This table indicates whether a distribution or a fixed value (or values) were used in the evaluation, along with references to the Volume 3 sections (12) and original source documents.

Table 2.5.60 – Input parameter summary

Input Parameter	Distribution(s) or Fixed Value(s)	Dependencies	Volume 3 Section(s)	Source Reference(s)
Time to recirculation	Fixed values	Break size	Section 2.2.1	(32)
Time to secure containment spray	Fixed value	None	Section 2.2.1	(81; 82)
Time to hot leg switchover	Fixed value	None	Section 2.2.1	(83; 84)
Containment geometry	Fixed values	None	Section 2.2.2, Section 5.2	(47)
Break size and frequency	Distributions	Break location	Section 2.2.3, Section 5.3	(80; 77; 47; 75)
Pool volume	Distributions	Break size	Section 2.2.6	(85)
Pool area	Fixed value	None	Section 2.2.6	(85)
Pool temperature	Fixed values	Break size	Section 2.2.7	(32)
Containment pressure	Fixed values	Pool temperature	Assumption 1.c	None

Input Parameter	Distribution(s) or Fixed Value(s)	Dependencies	Volume 3 Section(s)	Source Reference(s)
Operating pumps	Fixed values	None	Section 2.2.4, Section 2.2.8, Section 5.1	(86)
Low head safety injection flow rate	Fixed values	Break size, pumps running	Section 2.2.9	(87; 32)
High head safety injection flow rate	Fixed values	Break size, pumps running	Section 2.2.9	(87; 32)
Containment spray flow rate	Distributions	Pumps running	Section 2.2.9	(87; 88)
Qualified coatings quantity	Fixed values	Break size	Section 2.2.10	(10)
Unqualified coatings quantity	Distributions	None	Section 2.2.11	(7)
Unqualified coatings failure time	Fixed values	None	Section 2.2.11	(7)
Crud quantity	Fixed value	None	Section 2.2.12	(45)
Latent debris quantity	Fixed values	None	Section 2.2.13	(36)
Miscellaneous debris quantity	Fixed value	None	Section 2.2.14	(36)
Miscellaneous debris failure time	Fixed value	None	Assumption 4.b	None
Insulation ZOI size	Fixed values	Break size, insulation location	Section 2.2.15	(89; 56)
Fiberglass size distribution	Fixed values	Break size, insulation location	Section 2.2.16	(90)
Debris characteristics	Fixed values	None	Section 2.2.17	(36; 10; 7; 45)
Chemical product formation time	Fixed values	Pool temperature	Assumption 5.a, Section 5.7.3	(25)
Blowdown transport	Distributions	Break location, debris size	Section 2.2.21	(13)
Washdown transport	Distributions	Sprays initiated, debris size	Section 2.2.22	(13)
Pool fill transport	Fixed values	Break location, debris size	Section 2.2.23	(13)
Recirculation transport	Fixed values	Break size, break location, debris type, debris size	Section 2.2.24	(13)
Fiberglass spray erosion	Fixed value	Sprays initiated	Section 2.2.25	(13)
Fiberglass pool erosion	Distribution	None	Section 2.2.25	(13; 18)
Fiberglass pool erosion time	Fixed values	None	Section 2.2.25	(13; 18)
Transport time	Fixed values	Sump flow rate, pool volume, failure time	Section 5.6.8	(13)
Strainer geometry	Fixed values	None	Section 2.2.26	(91; 92; 93; 94; 95; 96)
Clean strainer head loss	Fixed values	Pool temperature, sump flow rate	Section 2.2.27, Section 5.7.1	(58)
Thin-bed thickness	Fixed value	None	Assumption 7.d	None
Conventional head loss bump-up	Fixed value	None	Appendix 1	None
Chemical head loss bump-up	Distributions	Break size	Section 5.7.3	(25)
Pump NPSH required	Fixed values	Void fraction	Section 2.2.28, Section 5.7.5	(97)

Input Parameter	Distribution(s) or Fixed Value(s)	Dependencies	Volume 3 Section(s)	Source Reference(s)
Pump NPSH available	Fixed values	Pool temperature, pump flow rate, pool level, containment pressure	Section 5.7.5	(97)
Strainer structural margin	Fixed value	None	Section 2.2.29	(98; 99)
Containment relative humidity	Fixed value	None	Assumption 8.e	None
Pump gas void limits	Fixed value	None	Section 2.2.32	(100)
Fiber filtration parameters	Distributions	None	Section 2.2.33, Section 5.9	(60)
Fiber shedding parameters	Distributions	None	Section 2.2.33, Section 5.9	(60)
Boil off flow rate	Fixed values	None	Section 2.2.34, Section 5.11.3	(101; 102)
Number of fuel assemblies	Fixed value	None	Appendix 1	(101)
Core blockage fiber limits	Fixed values	Break location, injection path	Section 2.2.35, Section 5.11.5	(63)
Boron precipitation fiber limits	Fixed values	Break location, injection path	Section 5.12.2	(71)

The fixed values listed in the previous table do not require additional detail to assess the treatment of uncertainty. This is because of the previously discussed factors (results are not heavily dependent on the variation of the parameter value, or the range of parameter values is close enough that a best estimate value is adequate to provide realistic results). The values listed as having a distribution associated with them are described below.

Break Size and Frequency

See discussion for Item 5.a.16.

Pool Volume

The volume of the recirculation pool during mitigation of a LOCA is documented in a plant-specific calculation (85). The following discussion documents the methods, basis, and results for the distribution used to define the pool volume.

Method: The basic methodology used for the STP post LOCA water volume analysis is shown below:

1. The range of water inventory associated with injection from the RWST was developed based on operating procedures and plant data. The injection volume was constructed as the difference of the initial and final volume in the RWST. The injection mass was developed by applying accurate densities to the volumetric values. The range of this inventory is significant and substantially affects subsequent evaluations; therefore, this input required the development of probability distributions.

2. The water inventory attributed to the RCS was evaluated based on operating procedures and design inputs. The volume and state of the inventory at hot-full power operating conditions was evaluated to determine the best estimate RCS volume and mass. The method of developing the RCS water inventory input allowed for a representative best estimate value.
3. The SI accumulator water inventory was derived through analysis of the technical specifications for volume, temperature and pressure. Due to the tight range between the minimum and maximum values, it has been determined that a best estimate value is adequate for all postulated break conditions.
4. The total water in containment was developed as the sum of the RWST, RCS, and SI accumulator water inventories. Due to the variation of the RWST, the total water in containment is presented as a probability distribution.
5. The active water in containment value was determined as the difference between the total water in containment and any inactive cavities. The amount of inventory designated as inactive depends on the break size. Due to the variation of the RWST injection inventory, the active water in containment is presented as a probability distribution.
6. The pool volume was established by subtracting any hold-up volumes from the active water in containment. The applicable amount of hold-up volume varies based on break size, break elevation, containment spray activation, containment spray flow rate, break flow rate, and pool temperature. To account for all of these variables, a function was derived to define the pool volume in containment.

Basis: The following is a detailed description of how the methodology referred to above was used to develop the appropriate input to the risk-informed evaluation.

RWST Distribution

The RWST is the largest source of water available for post-LOCA mitigation: therefore, it is the largest source of variance between minimum and maximum values. For this reason, a best-estimate value may not accurately represent the actual amount of water injected from the RWST. Therefore, a probability distribution for the RWST inventory was formulated to fully encompass the range of injection volume/mass.

The following figure illustrates the level alarms and volume capacities of the RWST that were used to evaluate the minimum and maximum RWST values:

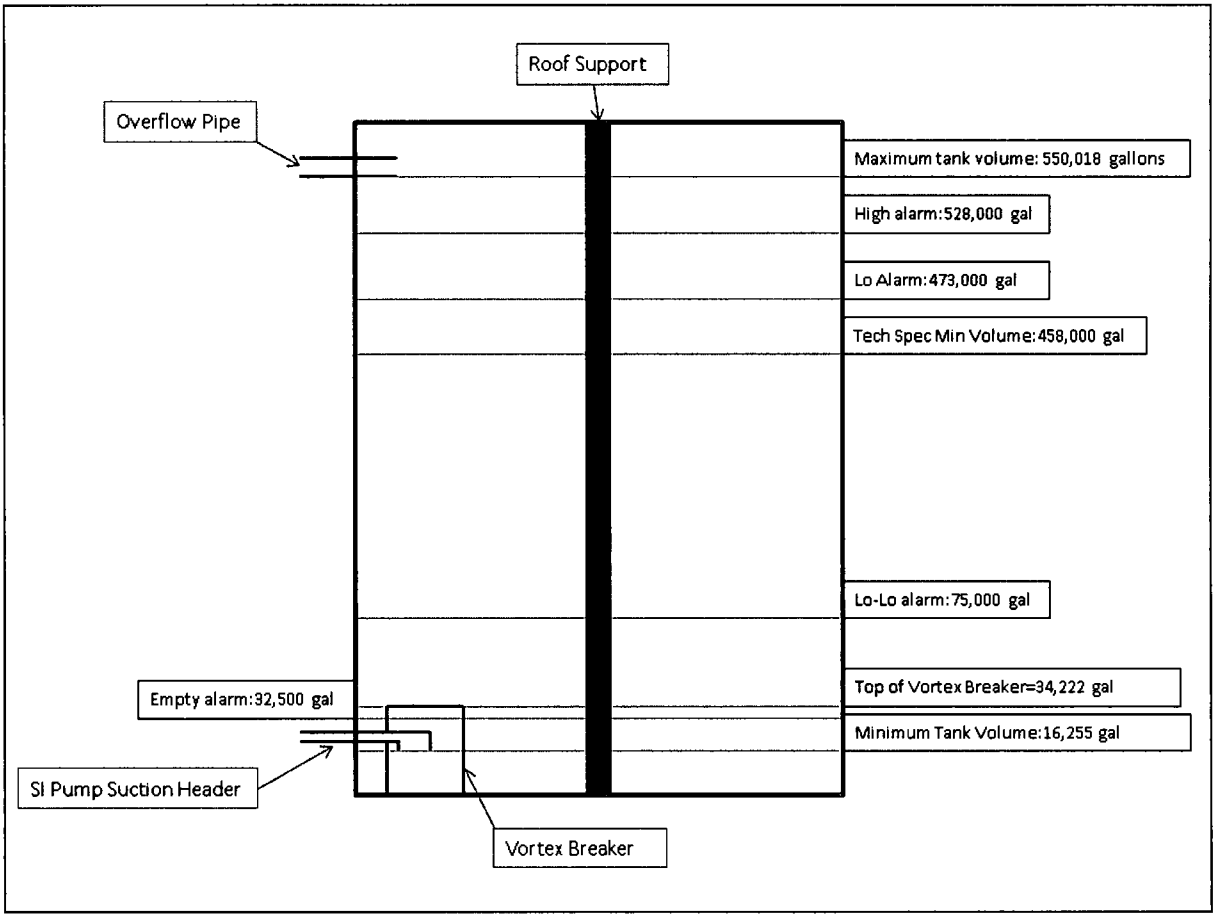

Figure 2.5.57 – RWST Water Levels and Alarms

The initial maximum injection volume is the volume corresponding to the Hi alarm. The minimum volume in the tank is the volume at the top of the vortex breaker. This is a suitable volume because the switchover must be completed before the water level drops below this to avoid possibly excessive air entrainment. The actual maximum RWST injection volume is calculated as follows:

$$V_{RWST}(Act.Max) = V_{initial}(Max) - V_{final}(\mathbf{Min})$$

$$V_{RWST}(Act.Max) = (528{,}000 - 34{,}222)gal = 493{,}778\ gal = 66{,}009\ ft^3$$

This value is converted to mass using the density of water at the minimum acceptable temperature during normal operating conditions: 50 ^{0}F.

$$M_{RWST}(Act.Max) = V_{RWST}(Act.Max) * \rho$$

$$M_{RWST}(Act.Max) = \frac{493{,}778\ gal}{7.4805\frac{gal}{ft^3}} * \left(62.414\frac{lbm}{ft^3}\right) = 4{,}119{,}866\ lbm$$

The actual minimum injection volume was formulated by evaluating the input parameters of the RWST. The initial minimum volume in the tank was set at the Lo alarm. The maximum final volume in the tank was evaluated at the volume corresponding to the Lo-Lo alarm: this is due to operating mandates (103). Although there is automatic valve realignment where all of the valves to the containment sumps are opened, the procedures require a manual action to close the valves to the RWST (104) (103). This would result in possibly pulling flow from both locations. However, if the containment pressure of the RCB is higher than 5 psig (which it normally would be for large and medium breaks), the containment pressure would be greater than the RWST pressure and would cause the RWST check valves to close (104) (103). Therefore, no adjustment to the Lo-Lo alarm volume was made, and it was used as the maximum final volume in the tank. The actual minimum injection volume was calculated as follows:

$$V_{RWST}(Act.Min) = V_{initial}(Min) - V_{final}(Max)$$

$$V_{RWST}(Act.Min) = (473{,}000 - 75{,}000)gal = 398{,}000\ gal = 53{,}205\ ft^3$$

This value is converted to mass using the density of water at the maximum acceptable temperature during normal operation: 104 ^{0}F.

$$M_{RWST}(Act.Min) = V_{RWST}(Act.Min) * \rho$$

$$M_{RWST}(Act.Min) = \frac{398{,}000\ gal}{7.4805\frac{gal}{ft^3}} * \left(61.944\frac{lbm}{ft^3}\right) = 3{,}295{,}730\ lbm$$

With the minimum and maximum values established, the probability distribution of injection volume from the RWST was established using operating data on the water level in the RWST during normal operation (See Figure 2.5.58). The volume of water left in the tank was set at the volume associated with the Lo-Lo alarm (for the same reason as discussed in the actual minimum volume calculation). The probability distribution of RWST injection volume uses a trapezoidal distribution: this is used to fully encompass the operating data and the minimum and maximum values. The top of the trapezoidal distribution encompasses the 25th and 75th percentile values, while the bottom of the trapezoidal distribution is bounded by the actual minimum and maximum injection values.

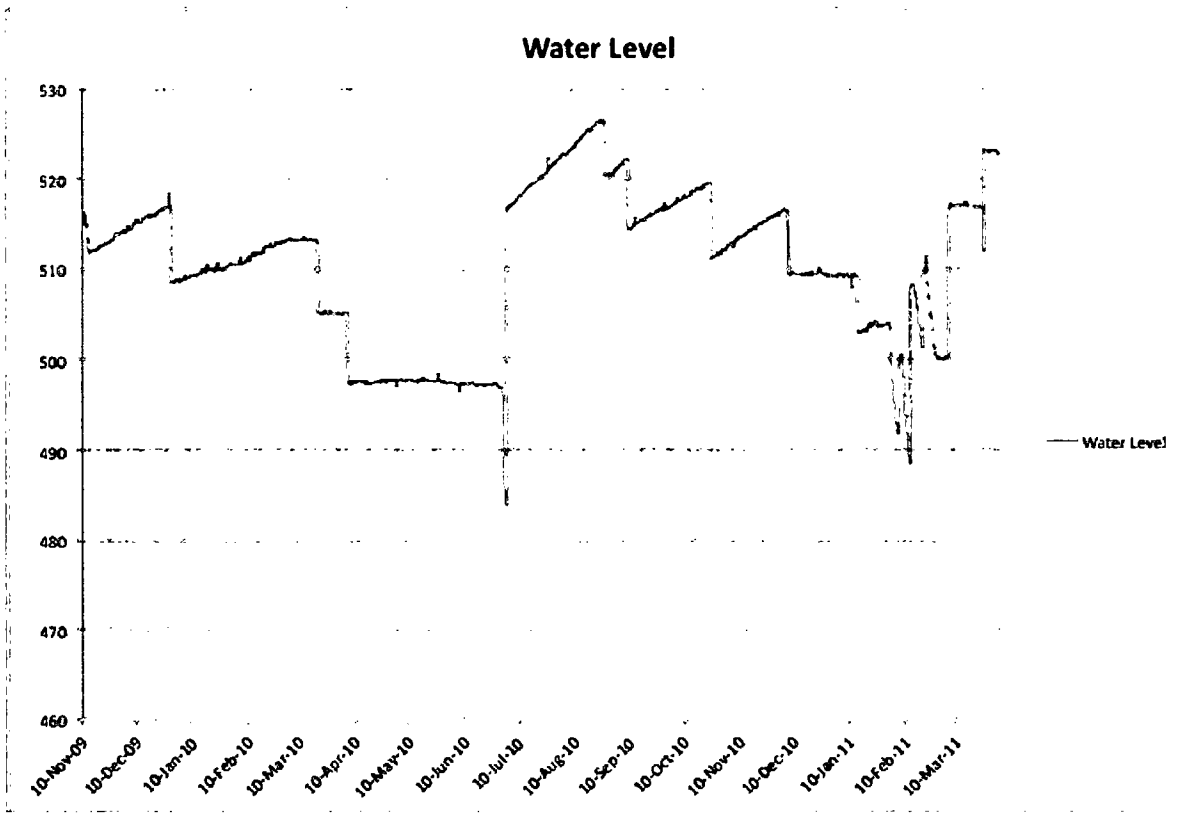

Figure 2.5.58 – RWST Operating Data

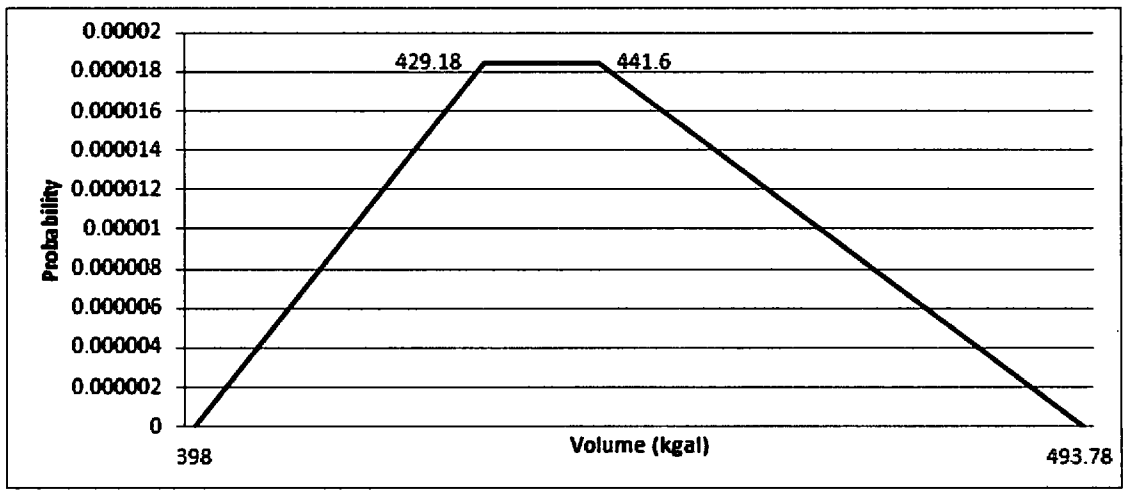

Figure 2.5.59 – RWST Injection Volume Probability Curve

The probability distribution is characterized by five input values:

a= the minimum value bounding the bottom left portion of the trapezoid
b= the 25th percentile value bounding the top left portion of the trapezoid
c= the 75th percentile value bounding the top right portion of the trapezoid
d= the maximum value bounding the bottom right portion of the trapezoid
h= the probability of the top portion of the trapezoid

The following table illustrates the five input values for the RWST injection volume probability curve:

Table 2.5.61 – RWST Volume Probability Function Inputs

Input	RWST Volume Function
a	398,000
b	429,179
c	441,596
d	493,778
h	0.000018485

The RWST injection mass probability curve follows the same trapezoidal pattern as the volume probability distribution. To apply this to a mass distribution, the 25th and 75th percentile values were converted to mass values with the average temperature; 77 ^{0}F temperature corresponding to a density of 62.28 lbm/ft^3.

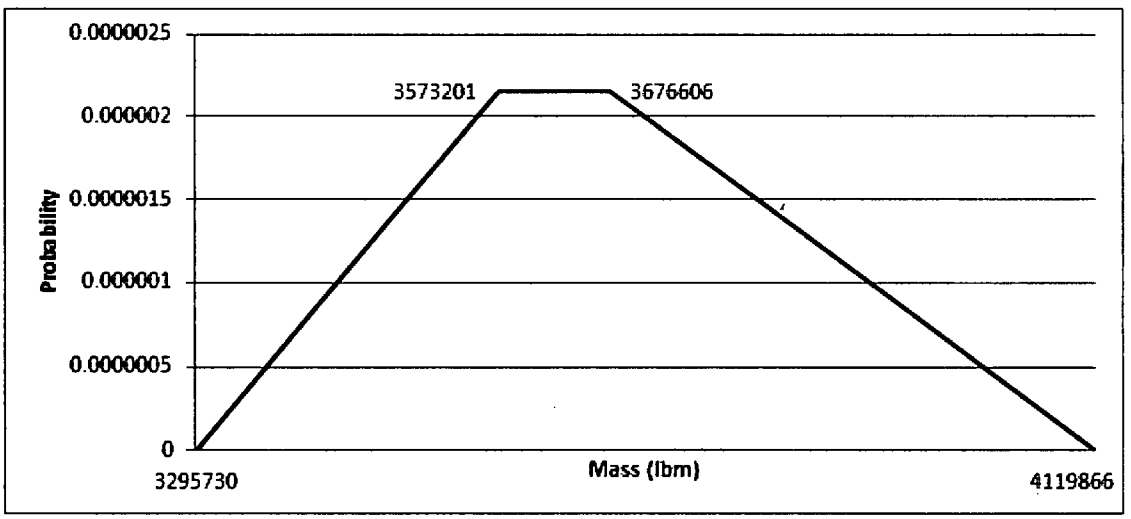

Figure 2.5.60 – RWST Injection Mass Probability Curve

The inputs for the RWST injection mass distribution are illustrated in the following table:

Table 2.5.62 – RWST Mass Probability Function Inputs

Input	RWST Volume Function	RWST Mass Function
a	398,000	3,295,730
b	429,179	3,573,201
c	441,596	3,676,606
d	493,778	4,119,866
h	0.000018485	0.000002156

RCS and SI Accumulator Fixed Values

The RCS and SI accumulator volumes are significantly small than the RWST injection volume, and the range between minimum and maximum values is small. Therefore, fixed values were used as a best estimate volume for these water sources. The RCS was evaluated at the volume associated with hot-full power operation because the plant is in full power production mode for the majority of the year (the only exceptions are for refueling outages. The best-estimate value of RCS volume is documented as 14,044 ft^3. The best estimate RCS mass is 612,644 lbm. The best estimate of the volume in the SI accumulators was calculated using operating data over a nearly two year span. The average water level for each of the three accumulators was summed yielding a total injection volume of 3,711 ft^3. The mass of the accumulators was calculated with the average temperature and average absolute pressure to define the density of the water inventory. The best estimate mass was calculated as 231,334 lbm.

Total Mass in Containment Distribution

The total mass in containment probability distribution was formulated using the trapezoidal distribution for the RWST mass, and adding the best estimate mass values of the RCS and SI accumulators. This distribution may be converted to volume using the time dependent pool temperature profiles formulated using thermal-hydraulic modeling.

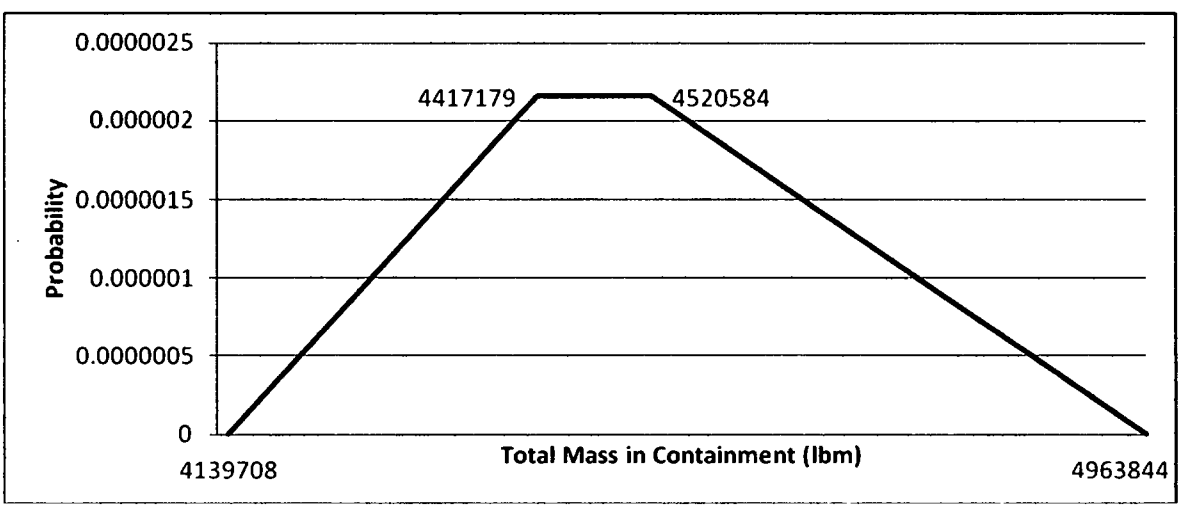

Figure 2.5.61 – Total Mass in Containment Probability Curve

The inputs for the total mass in containment probability curve are documented in the following table:

Table 2.5.63 – Total Mass in Containment Probability Function Inputs

Input	RWST Volume Function	Total Mass Function
a	398,000	4,139,708
b	429,179	4,417,179
c	441,596	4,520,584
d	493,778	4,963,844
h	0.000018485	0.000002156

Active Water in Containment Distribution

The active water in containment is defined as the total water in containment minus any inactive cavities. The inactive cavities would remove water from the recirculation inventory; the water in these cavities is effectively stagnant. Through engineering analysis of the containment building, it was determined that there are no inactive cavities for an MBLOCA and an LBLOCA. Therefore, the active water in containment would be the same as the total water in containment for these break conditions.

For an SBLOCA, the SI accumulators are an inactive cavity. During normal operating conditions, there are a series of check valves isolating the SI accumulators from the RCS. Depressurization of the RCS below the nitrogen pressure head of the accumulators causes injection into the core (103). For an MBLOCA and an LBLOCA, this depressurization of the RCS would occur. Therefore, there needs to be no adjustment to the active water in containment as all of the accumulators would inject their inventory to the reactor core for immediate cooling. An SBLOCA may not result in rapid full depressurization of the RCS below the nitrogen pressure head of the accumulators. In addition, before the long-term depressurization of the RCS, the SI accumulators may be isolated (105). Therefore, it is reasonable to assume that the accumulators would not inject to the reactor core for the SBLOCA condition. The total water in containment was reduced by the value of the SI accumulators to form the active water in containment for an SBLOCA.

The probability distribution of the active water in containment uses the trapezoidal distribution.

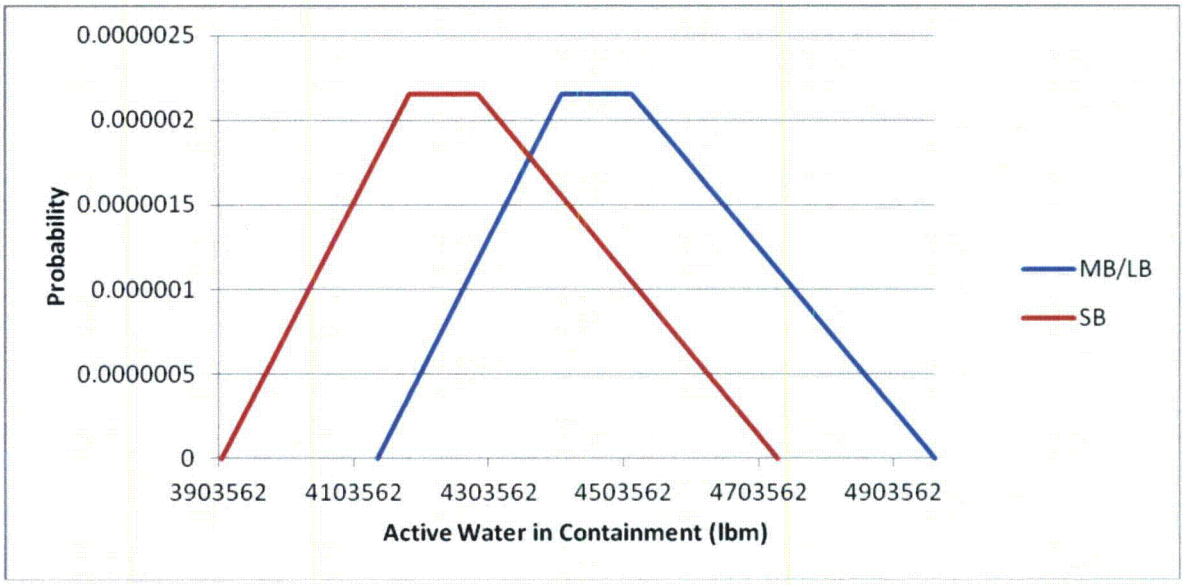

Figure 2.5.62 – Active Water in Containment Probability Curve

The inputs for the active water in containment probability curve are documented in the following table:

Table 2.5.64 – Active Water in Containment Probability Function Inputs

Input	RWST Volume Function	AWC Function-SBLOCA	AWC Function-MBLOCA	AWC Function-LBLOCA
a	398,000	3,908,374	4,139,708	4,139,708
b	429,179	4,185,845	4,417,179	4,417,179
c	441,596	4,289,250	4,520,584	4,520,584
d	493,778	4,732,510	4,963,844	4,963,844
h	0.000018485	0.000002156	0.000002156	0.000002156

The active water in containment is the basis for the base pool volume in containment. A value from this distribution is sampled as the base volume from which transitory hold-up volumes were subtracted. These transitory hold-up volumes include the inventory held up in the RCS, containment spray falling through containment, vapor in the atmosphere, etc. These values are a function of break size, flow rates, pool temperature, etc.

Containment Spray Flow Rate

If containment sprays are initiated, the flow rate is not dependent on the size of the break. However, it would vary depending on the number of trains in operation. The maximum spray flow rate for a single train is 2,600 gpm. If all three trains are operating, the maximum flow rate is approximately 2,060 gpm per train (40). If two trains are operating, the maximum flow rate is approximately 2,350 gpm per train

(41). The minimum probable CS flow rates are approximately 1,657 gpm per train for three train operation and 1,932 gpm per train for two train operation (41). The minimum spray flow rate for one train operation was not available in STP documentation, but was assumed to be 80% of the maximum flow rate consistent with the range of flow rates for two and three train operation. This is a reasonable assumption since the minimum spray flow rate for two train operation is 82% of the maximum spray flow rate for the two train operation, and the minimum spray flow rate for three train operation is 80% of the maximum spray flow rate for three train operation. This gives a minimum spray flow rate of 2,080 gpm for single train operation. Table 2.5.65 provides a summary of the range of containment spray flow rates.

Table 2.5.65 – Containment Spray Flow Rate Information

Number of Operating Spray Pumps	Minimum Spray Flow per Train (gpm)	Maximum Spray Flow per Train (gpm)
One Train	2,080	2,600
Two Trains	1,932	2,350
Three Trains	1,657	2,060

Unqualified Coatings Quantity

See discussion for Item 5.a.1.

Blowdown Transport

See discussion for Item 5.a.2.

Washdown Transport

See discussion for Item 5.a.3.

Fiberglass Pool Erosion

Small or large pieces of fiberglass debris that settle in the containment pool would be subject to erosion by the flow of water moving past the debris. The average erosion fraction for debris that settles in the recirculation pool would be a range below 10% as documented in Table 6.6 of the STP risk-informed debris transport calculation (13). As shown within Table 6.6 of the debris transport calculation, the majority of erosion would occur within the first 24 hours, but some erosion would continue at reduced rates over the duration of the event (13). The specific erosion values that were used in the analysis are not documented here since they are based on proprietary test data. However, the debris transport calculation, which provides a full description of the erosion values, is available for audit (13).

Chemical Head Loss Bump-up

See Discussion for Item 5.a.11

Fiber Filtration Parameters

See Discussion for Item 5.a.16

Fiber Shedding Parameters

See Discussion for Item 5.a.16

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92. **TDI-6005-01.** *SFS Surface Area, Flow and Volume Calculations.* Revision 1 : August 31, 2006.

93. **SFS-STP-GA-00.** *South Texas Project Units 1 & 2 Sure-Flow Strainer General Arrangement.* Revision 4 : August 25, 2006.

94. **SFS-STP-PA-7103.** *South Texas Project Units 1 & 2 Sure-Flow Strainer Sections and Details.* Revision 2 : August 4, 2006.

95. **2F369PSI0572 Sheets 3, 4 & 6.** *Safety Injection 'SI'.*

96. **5L019PS0004.** *Specification for Criteria for Piping Design and Installation.* Revision 23 : s.n.

97. **ALION-CAL-STP-8511-05.** *STP Net Positive Suction Head Margin.* Revision 0 : November 19, 2012.

98. **EC-PCI-STP-6005-1001.** *AES Document No. PCI-5473-S01 Rev 2 "Structural Evaluation of Strainers for Containment Emergency Sumps".* Revision 2 : January 7, 2010.

99. **EC-PCI-STP-6005-1004.** *AES Document No. PCI-5473-S03 Rev 0 "Structural Evaluation of Strainers for Containment Emergency Sumps for Long Term Post LOCA Case".* Revision 0 : January 7, 2010.

100. **Regulatory Guide 1.82.** *Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident.* Revision 4 : March 2012.

101. **5N079NB01000 (WCAP-12381).** *STPNOC Design Basis Document Accident Analysis.* Revision 15 : July 29, 2009.

102. **Technical Specifications Section 1.27.** *Rated Thermal Power.* Unit 1 Amendment No. 154; Unit 2 Amendment No. 142 : s.n.

103. **South Texas Project Electric Generating Station.** *STPEGS UFSAR-6.3-ECCS.* s.l. : Revision 15.

104. **STP Procedure 0P0P05-EO-ES13.** *Transfer to Cold Leg Recirculation.* July 2008 : Revision 10.

105. **WES010-Calc-001.** *South Texas Project Pos-LOCA containment Water Level Calculation.* Revision 0 : November 2008.

106. **Email from Dave Morton (UT Austin) to Tim Sande (Enercon).** *RE: penetration analysis.* : January 15, 2013.

107. **STPEGS UFSAR.** *Chapter 6.3: Emergency Core Cooling System.* Revision 15.

108. **OG-07-129.** *Pressurized Water Reactor Owners Group Resonses to the NRC Second Set of Requests for Additional Information (RAI's) on WCAP-16530, "Evaluation of Chemical Effects in Containment Sump Fluids to Support GSI-191".* : April 3, 2007.

ENCLOSURE 1

STP Piloted Risk-Informed Approach to Closure for GSI-191

STP Piloted Risk-Informed Approach to Closure for GSI-191

I. Introduction

This enclosure provides a generic methodology for licensees planning to use a risk-informed approach to resolving Generic Safety Issue (GSI)-191, as discussed in SECY Paper, "Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance" (Reference 1). The South Texas Project (STP) risk-informed approach is intended to be applied to STP Units 1 and 2 as pilot plants.

The risk-informed approach is consistent with the guidance in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 2). The risk associated with GSI-191, which includes the modeling of the containment sumps and sump strainers, is determined using a plant-specific, Probabilistic Risk Assessment (PRA) model, and the results are compared to the acceptance guidelines in RG 1.174. Based on meeting the acceptance guidelines, the sump design is determined to be acceptable, thereby reconstituting the licensing basis for the supported Emergency Core Cooling System (ECCS) and Containment Spray System (CSS).

The generic methodology for the STP risk-informed approach, consisting of the required inputs to the plant-specific PRA, the basic structure for modeling the inputs, and the performance criteria used, are described below.

Implementation for STP Units 1 and 2 is documented in Enclosure 4-1, "Volume 1 Project Summary," Enclosure 4-2, "Volume 2 Probabilistic Risk Analysis," and Enclosure 4-3, "Volume 3 Engineering (CASA Grande) Analysis," to this letter. These enclosures provide more detailed descriptions and explanations for the approach.

Background

GSI-191 concluded that debris generated during a postulated loss of coolant accident (LOCA) could clog the containment sump strainers in pressurized-water reactors (PWRs), leading to the loss of net positive suction head (NPSH) for the ECCS and CSS pumps. The NRC issued Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors" requesting that licensees address the issues raised by GSI-191. GL 2004-02 was focused on demonstrating compliance with ECCS acceptance criterion 10 CFR 50.46 (b)(5) for long term cooling.

In response, the industry completed plant modifications, such as installing larger sump strainers and replacing fibrous insulation inside containment, and implemented other compensatory measures to reduce the risk of strainer clogging and to mitigate strainer blockage and inadequate core cooling. Sump designs have been previously evaluated, using deterministic methods, to meet the current licensing basis for debris blockage and related effects, complying with the applicable regulatory requirements. Considerable effort has also been made to reduce the uncertainties and conservatisms in the standard

models used to evaluate GSI-191 concerns. However, the complexity of the debris effects has challenged the ability to use these methods to fully resolve GSI-191.

Summary of the STP Risk-Informed Approach

The STP risk-informed approach to resolving GSI-191 uses the plant-specific PRA with realistic modeling to quantify the residual risk associated with GSI-191 and to evaluate for acceptable sump design in support of successful ECCS and CSS operation in recirculation mode following postulated LOCAs with the debris effects discussed in GSI-191. The residual risk associated with GSI-191 for the as-built, as-operated plant represents those issues not previously resolved using the deterministic methods. The approach follows the guidance and meets the key principles of RG 1.174.

A full spectrum of postulated break sizes is analyzed, including double-ended guillotine breaks (DEGBs) for all pipe sizes up to and including the largest in containment. The required inputs to the PRA, the basic structure for modeling the inputs, and performance criteria used to calculate the risk are described. The physical processes are modeled as realistically as possible, using results from industry and plant-specific testing, and applying conservatism, where appropriate. Debris accumulation effects on the containment sumps and core flow are evaluated.

Acceptable sump design in support of long term cooling is based on a high probability that net positive suction head for the pumps is maintained, and that other limits for acceptable sump performance are not exceeded. High probability is confirmed by showing that the residual risk associated with GSI-191 concerns meets the RG 1.174 acceptance guidelines for changes in Core Damage Frequency (CDF) and Large Early Release Frequency (LERF).

This risk-informed approach is expected to be applicable to plants with substantial fibrous insulation, and also may be beneficial to plants with low to medium fibrous insulation. Risk-informed insights into the plant design may also be used to assess and prioritize plant modifications, if required.

II. Methodology

Define the Proposed Change

The proposed change is to use the risk-informed method, rather than deterministic methods, to evaluate for acceptable sump design based on a quantification of the residual risk associated with GSI-191, and to revise the plant's Updated Final Safety Analysis Report (UFSAR) as appropriate to describe the method and its results. The sump design is required to support ECCS and CSS system functions following postulated LOCAs, including long term cooling, containment heat removal, and containment atmosphere cleanup.

Approval for the proposed change is based on implementation of a risk-informed approach that meets the key principles of RG 1.174, and a determination that the residual risk associated with GSI-191 meets the acceptance guidelines of RG 1.174. If the acceptance guidelines are not met, insights obtained from the model may be used

incrementally to target modifications to the facility that would further reduce risk in order to achieve acceptable results.

The plant licensing basis considers the requirement for ECCS to meet the long term cooling criterion 10 CFR 50.46(b)(5) and requires ECCS to operate with high level of confidence following a postulated LOCA. A determination of acceptable sump design, using the risk-informed method, thereby reconstitutes the long term cooling licensing basis. Acceptable sump design also supports the licensing basis for meeting the Generic Design Criteria (GDC) applicable to ECCS and CSS system design requirements following postulated LOCAs

PRA Model Assessments and Supporting Engineering Analysis

The required inputs to the PRA model and the method for assessing the risk associated with GSI-191 using the PRA model are shown in Table 1 and Table 2 below.

The risk-informed method uses an integrative approach to explicitly provide the probabilities for post-LOCA events. This is accomplished by modeling the underlying physical phenomena of the basic events and by propagating uncertainties in the physical models. These inputs are used to risk-inform the plant-specific PRA model utilized to assess the risk associated with GSI-191.

To apply the inputs, the demand recirculation failure probability in the plant-specific PRA model is replaced with basic events (strainer failures, core flow blockage with chemical effects, and boron precipitation in the core) as shown in Table 2. Failure modes leading to core damage are explicitly modeled, excluding those that were previously addressed for the plant using deterministic evaluations. The inputs to the model include a full spectrum of pipe breaks, up to and including the design basis accident (DBA) LOCA.

Failure probabilities and associated uncertainties determined in the supporting engineering analysis provide inputs to the three new top events added to the PRA to accommodate composite GSI-191 failure processes (sump strainer failure, core flow blockage, and boron precipitation in the core). The outcome of a full spectrum of LOCA events is tested against appropriate performance thresholds for the top events (Table 2).

The plant-specific PRA model, informed with the risk-insights associated with GSI-191 failure modes, as described in Table 2, is then used to assess the difference in risk for two cases:

- Case 1: the actual plant configuration, risk-informed to model the failure mechanisms associated with the concerns raised by GSI-191, and
- Case 2: a hypothetical plant assuming no failure mechanisms associated with the concerns raised by GSI-191, otherwise identical to the actual plant.

The PRA model is required to be RG 1.200 compliant for assessing the risk of internal events associated with GSI-191.

III. Results and Conclusions

The risk-informed method uses the plant-specific PRA model to assess and quantify the residual risk associated with GSI-191 concerns. This approach is used to evaluate the design of the containment sump in support of ECCS and CSS recirculation modes following postulated LOCAs.

A determination that the risk results meet the acceptance guidelines in RG 1.174 confirms acceptable sump design and provides justification for allowing fibrous insulation to remain in containment. The results provide a basis for NRC approval of the proposed change in order to reconstitute the licensing basis for long term cooling, demonstrate that the Commission's safety goals for maintaining public health and safety continue to be met, and provide closure for GSI-191.

IV. References

1) Commission SECY Paper, "Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance," July 9, 2012, SECY-12-0093 (ML121320270)

2) Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2 (ML100910006).

3) NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Volume 1 "Pressurized Water Reactor Sump Performance Evaluation Methodology," Revision 0, dated December 2004 (ML050550138)

4) NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Volume 2 "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02, Revision 0, December 6, 2004," Revision 0, dated December 2004 (ML050550156)

5) Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities"

V. List of Tables

Table 1: Generic Methodology for Determining Risk Associated with GSI-191 Concerns

Case 1 In summary, use the plant-specific probabilistic risk assessment (PRA) model to assess the risk associated with GSI-191 concerns for the as-built, as-operated plant, based on realistic assessments to the extent practical. Modeling of basic events, failure modes, and new top events to accommodate composite GSI-191 failure processes with appropriate performance thresholds is described in Table 2.

- The inputs to the risk model encompass the concerns raised in GSI-191, including the major topical areas discussed in NEI 04-07 (Reference 3), as appropriate:
  - pipe break characterization
  - debris generation/zone of influence (ZOI), including latent debris
  - debris transport
  - chemical effects
  - strainer head loss, including structural margin
  - air intrusion
  - debris penetration
  - ex-vessel downstream effects
  - in-vessel downstream effects
  - boron precipitation

- For each input to the risk model, any differences between the methods to be used in the model and NRC-approved methods (refer to Reference 4 for an example) are defined.

- For each input to the risk model, an uncertainty quantification process is used to add detail (basic events, refer to Table 2) to the PRA model for the LOCA initiating sequences. Examples of appropriate sources of information include, but are not limited to:
  - applicable risk assessments
  - results obtained from generic industry and/or plant-specific testing
  - expert elicitation
  - assumptions, realistic or conservative
  - qualitative insights based on engineering judgment

- For each input to the risk model, interdependencies between the inputs to the model are considered and appropriately described in the risk model.

- The risk is determined using a plant-specific PRA that meets the necessary requirements identified in RG 1.200 (Reference 5), including the capability to model a full spectrum of LOCA events, and the capability for Level 1 and Level 2 risk assessments, including internal and external events.

Case 2 Evaluate the risk assuming no long term cooling failure contributors associated with GSI-191 concerns, and assuming no additional failures. Other than the basic events details associated with GSI-191 concerns, the Case 2 assessment model is identical to model used for Case 1.

Calculate the risk associated with GSI-191 concerns

The risk associated with GSI-191 concerns is the difference in risk between Cases 1 and 2, for comparison with the acceptance guidelines in RG 1.174.

Table 2: Modeling Basic Events, Failure Modes, and Top Events with Performance Thresholds

Using the inputs noted below, applied within the framework described in Table 1, the PRA uses risk insights to address the risk associated with failure modes resulting from GSI-191 concerns.

Basic Events

In the plant-specific PRA model, the demand recirculation failure probability is replaced with the following:

- Pressure drop due to debris build-up on the sump strainers with chemical effects resulting in loss of net positive suction head (NPSH) margin for pumps.
- Strainer mechanical collapse. P-buckle is the strainer structural design limit for the differential pressure (DP) across the ECCS strainers at which they are analyzed to be within code design allowable stresses.
- Air ingestion through the sump strainers. F-void is the vapor fraction of the liquid just downstream of the ECCS strainers.
- Core blockage with chemical effects.
- Boron precipitation in the core.

Failure Modes

For input into the plant-specific PRA, accident sequences from a full spectrum of LOCAs are analyzed in a realistic time-dependent manner with uncertainty propagation to determine the probabilities of various failures potentially leading to core damage.

The failure modes shall be explicitly modeled in the PRA analysis, except for failure modes that were addressed with no issues of concern as part of previous deterministic evaluations for the plant.

Top Events and Performance Thresholds

Failure probabilities and associated uncertainties determined in the supporting engineering analysis are passed to the plant-wide PRA, which determines the incremental risk associated with GSI-191 failure modes with three new top events added to accommodate composite GSI-191 failure processes. The engineering analysis supports the three new top events by testing the outcome of every postulated LOCA scenario against seven performance thresholds, discussed in detail in Enclosure 4, and summarized below.

New Top Events	Performance Thresholds
1. Failure at sump strainers	1. Strainer DP ≥ NPSH margin 2. Strainer DP ≥ P-buckle 3. Strainer F-void ≥ 0.02
2. Boron precipitation in the core	4. Core fiber load ≥ cold leg break fiber limit for boron precipitation 5. Core fiber load ≥ hot leg break fiber limit for boron precipitation
3. Core flow blockage	6. Core fiber load ≥ cold leg break fiber limit for flow blockage 7. Core fiber load ≥ hot leg break fiber limit for flow blockage

ENCLOSURE 2

Introduction to Requests for Exemptions for STP Piloted Risk-Informed Approach to Closure for GSI-191

Introduction to Requests for Exemptions for STP Piloted Risk-Informed Approach to Closure for GSI-191

Introduction

In support of the South Texas Project (STP) risk-informed approach to resolving Generic Safety Issue (GSI)-191 (Reference 1), Enclosures 2-1 through 2-4 provide STP Nuclear Operating Company (STPNOC) requests for exemptions under 10 CFR 50.12 from certain requirements in 10 CFR 50.46 and 10 CFR Part 50 Appendix A General Design Criteria (GDC). The exemption requests support approval of a license amendment request (LAR) provided in Enclosure 3 to this letter, proposing changes to the South Texas Project (STP) Units 1 and 2 Updated Final Safety Analysis Report (UFSAR) based on NRC acceptance of the risk-informed method and results.

Specific exemption requests, pertaining to requirements that concern Emergency Core Cooling System (ECCS) and Containment Spray System (CSS) system functions for core cooling, and containment heat removal and atmosphere cleanup following a postulated loss of cooling accident (LOCA), are provided as follows:

- Enclosure 2-1, Request for Exemption from 10 CFR 50.46(b)(5)
- Enclosure 2-2, Request for Exemption from GDC 35
- Enclosure 2-3, Request for Exemption from GDC 38
- Enclosure 2-4, Request for Exemption from GDC 41

Each separate Enclosure 2-1 through 2-4 identifies the applicable rule from which exemption is requested, the regulatory requirements involved, the purpose of the request, and the technical basis and justification for the exemption request, including the presence of special circumstances pursuant to 10 CFR 50.12(a). The requested exemptions are part of a risk-informed approach to resolve GSI-191 issues. The risk-informed approach is designed to be consistent with the guidance in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (Reference 2).

The STP risk-informed approach addresses the five key principles in RG 1.174 for risk-informed decision-making. The resulting risk metrics, i.e. changes in Core Damage Frequency (CDF) and Large Early Release Frequency (LERF), associated with GSI-191 concerns are used to determine whether plant modifications are warranted to ensure acceptable sump performance. The requested exemptions support this approach. A generic methodology for the STP approach is provided in Enclosure 1 to this letter.

The approach is intended to be a pilot for other licensees that are pursuing a risk-informed approach to resolving GSI-191. The STP approach is the risk-informed part of an overall graded approach that is based on the amount of fibrous insulation in the plant, as discussed in SECY-12-0093, "Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance," (Reference 3). STP Units 1 and 2 contain large amounts of fibrous-debris material such as insulation and coatings in the containment buildings and are expected to have higher risk of containment sump strainer

blockage and in-vessel core blockage as a result of potential debris-generating postulated loss of coolant accidents (LOCAs) than plants with relatively less fiber loading.

Based on the results for STP Units 1 and 2 showing that the risk for GSI-191 is far less than the threshold for Region III, "Very Small Changes," of RG 1.174, no physical changes to the facility or changes to the operation of the facility are proposed. However, the approach provides the capability to use risk-informed insights to prioritize plant modifications, such as targeting problematic fibrous insulation in containment, if required.

Background and Overview

GSI-191 concerns the possibility that debris generated during a LOCA could clog the containment sump strainers in pressurized-water reactors (PWRs) and result in loss of net positive suction head (NPSH) for the ECCS and CSS pumps, impeding the flow of water from the sump. Generic Letter (GL) 2004-02 (Reference 4) requested licensees to address GSI-191 issues focused on demonstrating compliance with the § 50.46 ECCS acceptance criteria. GL 2004-02 requested licensees to perform new, more realistic analyses using an NRC-approved methodology and to confirm the functionality of the ECCS and CSS during design basis accidents that require containment sump recirculation. As stated in GL 2004-02:

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

STP Units 1 and 2 have implemented compensatory and mitigative measures in response to Bulletin 2003-01 (Reference 5) and GL 2004-02 to address the potential for sump strainer clogging and other concerns associated with GSI-191. Larger containment sump strainers have been installed that greatly reduce the potential for loss of net positive suction head (NPSH). Additional compensatory measures such as operating procedures and instrumentation to monitor core level and temperature, and actions taken by operators if core blockage is indicated, were described in the STPNOC response (also refer to Enclosure 4-1).

The Commission issued Staff Requirements Memorandum (SRM)-SECY-10-0113 directing the staff to consider alternative options for resolving GSI-191 that are innovative and creative, as well as risk-informed and safety conscious (Reference 6). Subsequently, STPNOC, through interactions with the staff, developed a risk-informed approach for the resolution of GSI-191

using the methods described in RG 1.174. By Reference 7, STPNOC submitted to the NRC the preliminary results and notified the NRC of the intent to seek exemption from certain requirements of § 50.46.

SECY-12-0093 (Reference 3) described the staff plans to use STP as a pilot for other licensees choosing to use this approach, and the STP approach referred to as risk-informed Option 2. This approach requires an exemption request in accordance with 10 CFR 50.12 from certain requirements of 10 CFR 50.46, based in part on meeting the guidance in RG 1.174. Because the residual risk of GSI-191 concerns meets RG 1.174 acceptance guidelines, the approach allows fibrous insulation and other contributors to GSI-191 concerns to remain in containment.

The STP risk-informed approach to resolving GSI-191 is consistent with the NRC staff safety evaluation of NEI 04-07 (Reference 8) that discussed the modeling of sump performance as follows:

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

A description of the STP approach with respect to the requirements of 10 CFR 50.46 is provided in Enclosure 2-1. Descriptions of the STP risk-informed method including comparisons to previously approved guidance are included in Enclosures 4-1 and 4-3, and in the Attachment to this letter.

Summary of the STP Risk-Informed Approach

The proposed risk-informed method applies more recent information and knowledge, gained in support of studies and testing related to GSI-191 issues, to demonstrate that the ECCS would protect the reactor during a LOCA, specifically for meeting performance criterion 50.46(b)(5). The engineering analysis and evaluations, supporting calculations and codes are designed to model the plant as accurately as possible. Where uncertainty exists, it is identified and conservatism is applied. The result is a risk-informed PRA model that is capable of predicting ECCS performance using realistic evaluations of sump performance. The results show that there is a high probability that the ECCS acceptance criteria will be met for long term cooling.

The STP risk-informed method uses the STP PRA to quantify the residual risk from those issues related to GSI-191 concerns which have not been resolved using other methods. The supporting engineering analysis, including evaluation of defense-in-depth and safety margin, has been developed to conform to RG 1.174 guidance, acceptance guidelines, and documentation requirements. The STP approach models the physical characteristics of debris generation and transport over a full range of plausible conditions in order to provide inputs to the STP PRA.

RG 1.174 provides technical guidance for licensees who request NRC approval for changes in the licensing basis using a risk-informed approach. This guidance establishes five principles that should be considered for risk-informed changes to the licensing basis. The exemption requests are part of a risk-informed approach that addresses the principles stated in RG 1.174. Additional discussion related to implementation and monitoring the proposed change is included in Enclosure 4-1, and in the LAR in Enclosure 3 which is supported by this exemption request.

Enclosures 4-1 through 4-3 provide a summary of the STP PRA, risk assessment methodology, and engineering analysis, including modeling of physical plant properties and treatment of uncertainties, and references to other supporting information. The PRA is used to calculate the risk (CDF and LERF) associated with GSI-191 for the as-built, as-operating plant, to quantify residual risk associated with GSI-191 for the as-built, as-operated plant. The residual risk associated with GSI-191 for the as-built, as-operated plant represents those issues not previously resolved using the deterministic methods. The results demonstrate that the risk for STP Units 1 and 2 is far less than the threshold in Region III, "Very Small Changes," of RG 1.174, and provides a basis for approval:

- Change in CDF is ~ 1.1E-8/yr
- Change in LERF is ~ 8.6E-12/yr

The results are consistent with the Commission's Safety Goals for public health and safety. As such, no further physical modifications to the plant to address GSI-191 concerns are proposed.

References

1) Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance"

2) Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2 (ML100910006)

3) Commission SECY Paper, "Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance," SECY-12-0093, dated July 9, 2012 (ML121320270)

4) Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," dated September 13, 2004 (ML042360586)

5) Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors," dated June 9, 2003 (ML031600259)

6) Staff Requirements Memorandum (SRM)-SECY-10-0113, "Closure Options for Generic Safety Issue [GSI] - 191, Assessment of Debris Accumulation on Pressurized Water Reactor [PWR] Sump Performance," dated December 23, 2010 (ML103570354)

7) Letter, J. W. Crenshaw, STPNOC, to NRC Document Control Desk, "Status of the South Texas Project Risk-Informed (RI) Approach to Resolve Generic Safety Issue (GSI)-191," NOC-AE-11002775, dated December 14, 2011 (ML11354A386)

8) GSI-191 Safety Evaluation Report, Rev. 0, "Evaluation of NEI Guidance on PWR Sump Performance," dated December 6, 2004 (ML043280007)

ENCLOSURE 2-1

Request for Exemption from

10 CFR 50.46(b)(5)

Request for Exemption from Certain Requirements of 10 CFR 50.46(b)(5)

1. Exemption Request

Pursuant to 10 CFR 50.12, STP Nuclear Operating Company (STPNOC) is submitting this request for exemption from certain requirements of 10 CFR 50.46(b)(5), "*Long-term cooling*," as specified in § 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." 10 CFR 50.46(b)(5) states:

> *After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.*

Licensees are required to demonstrate this capability while assuming the most conservative (and worst) single failure. This regulation has been interpreted as requiring the use of a bounding calculation or other deterministic method to model sump performance, as discussed in Generic Letter 2004-02 (Reference 1) and Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance" (Reference 2). STPNOC requests an exemption from that requirement in order to enable the use of a risk-informed method to demonstrate acceptable sump performance and to validate assumptions in the Emergency Core Cooling System (ECCS) evaluation model.

Approval of this exemption will allow use of a risk-informed method to account for the probabilities and uncertainties associated with reliable containment emergency sump performance in support of the operation of the ECCS following postulated LOCAs. The method evaluates the effects on strainer blockage and core blockage resulting from debris concerns raised by GSI-191. In order to confirm acceptable sump design, the risk associated with GSI-191 is evaluated to include the failure mechanisms associated with loss of core cooling and strainer blockage.

This exemption request is in support of the accompanying License Amendment Request (LAR) (Enclosure 3) seeking NRC approval of the changes to the South Texas Project Electric Generating Station (STPEGS) Updated Final Safety Analysis Report (UFSAR), to reconstitute the long term cooling licensing basis based on acceptable design of the containment sump. The risk-informed method provides assurance, with high probability, for acceptable sump performance during ECCS operation in recirculation mode as calculated by the ECCS evaluation model.

2. Regulatory Requirements Involved

STPNOC seeks exemption to the extent that 10 CFR 50.46(b)(5) requires deterministic calculations or other analyses to address the concerns raised by GSI-191 related to acceptable plant performance during the recirculation mode in containment following a LOCA. The proposed changes to the licensing basis, submitted for NRC approval with the LAR in Enclosure 3, provide closure to GSI-191 for STP Units 1 and 2 on the basis that the associated risk is shown to meet the acceptance guidelines in Regulatory Guide

(RG) 1.174 (Reference 3) and that, in conjunction with the existing licensing basis, adequate safety is demonstrated.

This exemption request is for the purpose of allowing the use of a risk-informed method to demonstrate acceptable performance of the containment emergency sump during the recirculation mode in containment following postulated loss of coolant accidents (LOCAs), and is not intended to be applicable to other criteria in § 50.46(b). The containment sump has been evaluated, using deterministic methods, to meet the current licensing basis assumptions for analyzing the effects of post-LOCA debris blockage; however, these evaluations have not been shown to fully resolve GSI-191 for the as-built, as-operated plant (Reference 4). The risk-informed approach evaluates the sump design as part of the assessment of the residual risk associated with GSI-191 concerns. Based on confirmation of acceptable ECCS design as determined by the resulting risk meeting the acceptance guidelines in RG 1.174, the licensing basis for ECCS compliance with § 50.46(b)(5) is reconstituted.

Other regulatory requirements associated with containment sumps in support of ECCS and Containment Spray System (CSS) recirculation modes following postulated LOCAs include GDC 35 for core cooling, GDC 38 for containment heat removal, and GDC 41 for containment atmosphere cleanup. These requirements are addressed as part of separate exemption requests.

2.1 Evaluation of Impacts on the Balance of § 50.46 and Appendix K to Part 50

The exemption request to support closure for GSI-191 is intended to address ECCS acceptance criterion for long-term cooling as presented in § 50.46(b)(5) and is not applicable to the other acceptance criteria of § 50.46 (peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation, and coolable geometry).

For the purposes of demonstrating the balance of the acceptance criteria of § 50.46(b), the design and licensing basis descriptions of accidents requiring ECCS operation, including analysis methods, assumptions, and results, which are provided in South Texas Project Electric Generating Station (STPEGS) Updated Final Safety Analysis Report (UFSAR) Chapters 6 and 15 remain unchanged. The performance evaluations for accidents requiring ECCS operation described in UFSAR Chapters 6 and 15, based on the Appendix K Large-Break Loss-of-Coolant Accident (LBLOCA) analysis, demonstrate that for breaks up to and including the double-ended severance of a reactor coolant pipe, the ECCS will limit the clad temperature to below the limit specified in § 50.46 and assure that the core will remain in place and substantially intact with its essential heat transfer geometry preserved.

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

The reference to "*acceptable evaluation model*" in § 50.46(d) is discussed in § 50.46(a)(1) and defined in § 50.46(c)(2). The purpose of the risk-informed approach is

to evaluate the ECCS sump performance to determine if the sumps are capable of supporting the ECCS function to provide enough flow to ensure the long term cooling criterion § 50.46(b)(5) is met. This aspect is discussed in GL 2004-02, and the NRC safety evaluation report on NEI 04-07 (Reference 5) which states:

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

The STP risk-informed approach uses the plant-specific PRA model to quantify the residual risk associated with GSI-191, i.e. those issues which have not been resolved using deterministic methods, and shows that it meets the acceptance guidelines defined in RG 1.174. The exemption request is specific to the requirement for demonstrating long-term core cooling capability as required by § 50.46(b)(5) as it pertains to the validation of assumptions made in the ECCS evaluation model, and is not intended to be applicable to other requirements provided in § 50.46 or Appendix K to Part 50.

As noted above, the NRC staff considers the modeling of sump performance as an input to the ECCS evaluation model, and therefore the requirements of § 50.46 are applicable. Consistent with this, the requirements and attributes for the proposed STP risk-informed method include:

- A full spectrum of postulated, double-ended guillotine breaks is evaluated, up to and including the largest piping in containment.
- Engineering analyses and evaluations used to risk-inform the PRA model consider a wide range of effects, including those addressed in NEI 04-07 (Reference 6) and related NRC guidance for evaluation of sump performance. A realistic model, based on the as-built and as-operating facility, is used to evaluate the risk. Reliability of mitigating systems is assessed by considering a broader set of potential challenges to safety, such as common cause failures and multiple failures that extend beyond those required for deterministic methods and by the GDC.
- Applicable experimental data is used to risk-inform the PRA model.
- Simplifying assumptions are reasonable, and the bases for these assumptions are clearly stated.
- Uncertainties in the inputs to the model are identified and assessed so that uncertainty in the results of the model can be estimated, and where appropriate, conservatism added.

The proposed exemption does not affect any of the 10 CFR 50.46(a)(1) or Appendix K requirements for an acceptable ECCS evaluation model, and does not change the ECCS acceptance criteria in 50.46(b) as it applies to the calculated results. Application of the exemption request allows use of a risk-informed approach to evaluate the sump performance for the concerns associated with GSI-191 that may be present in an acceptable evaluation model. The results of the risk-informed method demonstrate that the risk associated with GSI-191 meets the acceptance guidelines of RG 1.174. The current licensing basis for addressing the adequacy of ECCS to meet the criteria of § 50.46, including the Appendix K Large-Break LOCA analysis and the associated Chapter 15 accident analysis for LOCA, remain in place.

The proposed exemption does not affect the requirements of 10 CFR 50.46(d), which states:

> *The criteria set forth in paragraph (b), with cooling performance calculated in accordance with an acceptable evaluation model, are in implementation of the general requirements with respect to ECCS cooling performance design set forth in this part, including in particular Criterion 35 of appendix A.*

With respect to the reference to GDC 35, this requirement is addressed as part of a separate exemption request.

2.2 Evaluation of Impacts on 10 CFR 50.67 and GDC 19

The impact of the proposed exemption on the licensing basis analysis for demonstrating radiological consequences of the design basis LOCA meet the radiological dose guidelines specified in 10 CFR 50.67 and the dose limits specified in GDC 19 was evaluated. The risk-informed method provides confirmation of reliable ECCS and CSS performance as required for the licensing basis analyses that demonstrate the requirements of 10 CFR 50.67 and GDC 19. The method demonstrates that sump performance continues to support reliable plant design and operation and does not entail any exemption from 10 CFR 50.67 or GDC 19.

For STP Units 1 and 2, which have implemented the Alternative Source Term (AST), the design-basis LOCA radiological consequence LOCA analysis is a deterministic evaluation based on the assumption of a major rupture of the reactor coolant system piping and the assumption of the deterministic failure of the ECCS to provide adequate core cooling (Reference 9). This scenario results in a significant amount of core damage as specified in RG 1.183 (Reference 10), and does not represent any specific accident sequence, but is representative of a class of severe damage incidents that were evaluated in the development of the RG 1.183 source term characteristics. Such a scenario would be expected to require multiple failures of systems and equipment and lies beyond the likely incidents evaluated for design-basis transient analyses (Reference 9). Since deterministic failure of ECCS is assumed at the onset of the accident by the analysis, the reliability of the containment emergency sumps with respect to ECCS operation does not affect the analysis for dose consequences. Therefore, for the purposes of this exemption request, the current licensing basis analyses for 10 CFR 50.67 and GDC 19 are not impacted.

2.3 Evaluation of Impacts on other Regulatory Requirements

The STP risk-informed approach, as described in Enclosures 1 and 4, uses the PRA model to quantify the risk associated with GSI-191, thereby quantifying the residual risk from those issues which have not been resolved using other methods. A determination that this approach and its results meet the key principles and acceptance guidelines in RG 1.174 demonstrates acceptable sump performance in support of ECCS and CSS operation in recirculation mode following postulated LOCAs, and demonstrates that the Commission's safety goals and public health and safety are maintained.

The proposed exemption does not result in any physical changes to the facility or changes to the operation of the plant, and does not change any of the programmatic requirements. Based on demonstrating acceptable containment emergency sump and ECCS design for reconstituting the current licensing basis for compliance with § 50.46(b) as described above, compliance with other regulatory requirements that rely on acceptable design for these systems and components continue to be met in the current licensing basis.

3. Basis for the Exemption Request

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

The exemption request meets a key principle of RG 1.174, which states "*The proposed change meets the current regulations unless it is explicitly related to a requested exemption.*" This exemption request is provided in support of the proposed change provided in the License Amendment Request provide in Enclosure 3.

3.1 Justification for the Exemption Request

As required by 50.12(a)(2), the Commission will not consider granting an exemption unless special circumstances are present. Special circumstances are present whenever one of the listed items (i) through (vi) under 50.12(a)(2) are applicable. STPNOC has evaluated the proposed exemption against the conditions specified in 10 CFR 50.12(a) and determined that this proposed exemption meets the requirements for granting an exemption from the regulation, and that special circumstances are present. The information supporting the determination is provided below.

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

The exemption is authorized by law.

The NRC has authority under the Atomic Energy Act of 1954, as amended, to grant exemptions from its regulations if doing so would not violate the requirements of law. This exemption is authorized by law as is provided by 10 CFR 50.12 which provides the NRC authority to grant exemptions from Part 50 requirements with provision of proper

justification. Approval of the exemption from 10 CFR 50.46(b)(5) would not conflict with any provisions of the Atomic Energy Act of 1954, as amended, any of the Commission's regulations, or any other law.

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

The purpose of § 50.46 is to establish acceptance criteria for ECCS performance, and together with GDC 35, to provide a high confidence that the systems will perform the required functions. The proposed exemption does not involve any modifications to the plant that could introduce a new accident precursor or affect the probability of postulated accidents, and therefore the probability of postulated initiating events is not increased. The PRA and engineering analysis demonstrate that the calculated risk is small and consistent with the intent of the Commission's Safety Goal Policy Statement, which defines an acceptable level of risk that is a small fraction of other risks to which the public is exposed.

As discussed in previous § 50.46 rulemaking (Reference 7), the probability of a large break LOCA is sufficiently low that the application of a risk-informed approach to evaluate the ability of the ECCS to meet § 50.46(b)(5) with high probability and with low uncertainty, rather than using a calculational model using deterministic methods to achieve similar understanding, would have little effect on public risk. This is applicable to evaluating acceptable containment sump design in support of ECCS and CSS recirculation modes.

The proposed change is to apply a risk-informed method rather than a traditional deterministic method to quantify the risk associated with GSI-191 and to establish a high probability of success for performance of ECCS to satisfy long term cooling criterion § 50.46(b)(5). The risk-informed approach involves a complete evaluation of the spectrum of LOCA breaks, including double-ended guillotine breaks, up to and including the largest pipe in the reactor coolant system (see Enclosure 4-3, Volume 3 Section 5.3.6). The risk-informed approach analyzes LOCAs, regardless of break size, using the same methods, assumptions, and criteria in order to quantify the uncertainties and overall risk metrics. This ensures that large break LOCAs with relatively small contribution to CDF due to the low probability of such a break as well as smaller break LOCAs with higher probabilities of occurrence are considered in the results. Since the design basis requirement for consideration of a double-ended guillotine breaks of the largest pipe in the reactor coolant system is retained and since no physical changes to the facility or changes to the operation of the facility are being made, the existing defense-in-depth and safety margin established for the design of the facility is not reduced.

This exemption only affects § 50.46(b)(5), and does not impact the adequacy of the acceptance criteria for cladding performance that is important to maintain adequate safety margins.

The exemption is consistent with the common defense and security.

The exemption involves a change to the licensing basis for the plant that has no relation to the control of licensed material or any security requirements that apply to STP Units 1 and 2. Therefore the exemption is consistent with the common defense and security.

3.2 Special Circumstances

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

Such special circumstances are present in this instance to warrant exemption from the implicit requirement in § 50.46(b)(5) to use a deterministic calculational method as the design basis for acceptable sump performance to validate the results of the ECCS evaluation model demonstrating long-term cooling criterion is met. Approval of this exemption request would allow the use a risk-informed method to reconstitute the design basis for acceptable performance of the containment emergency sump, as a validation of inputs in the ECCS evaluation model, and in support of the existing licensing bases for compliance with 10 CFR 50.46.

Specifically, § 50.12(a)(2)(ii) applies:

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

The intent of 10 CFR 50.46(b)(5) is to ensure long term cooling in the recirculation mode following postulated LOCAs. This exemption request is consistent with that purpose in that use of the proposed risk-informed approach demonstrates that long term cooling is realistically available and supports a high probability of successful ECCS performance, based on the risk results meeting the acceptance guidelines of RG 1.174.

As discussed in the *Commission's Policy Statement on "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities"* (Reference 8), NRC regulations and their implementation are generally based on deterministic approaches that consider a set of challenges to safety and determine how those challenges should be mitigated. This request does not seek exemption from any explicit language in the regulatory requirements. Rather, the need is based on the implicit requirements in the regulations for using deterministic methods to demonstrate acceptable design. Regulatory requirements are largely based on a deterministic framework, and are established for design basis accidents, such as the LOCA, with specific acceptance criteria that must be satisfied. Licensed facilities must be provided with safety systems capable of preventing and mitigating the consequences of design basis accidents to protect public health and safety. The deterministic regulatory requirements were designed to ensure that these systems are highly reliable. The LOCA analysis and the General Design Criteria (GDC) were established as part of this deterministic regulatory framework.

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

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

Specifically, § 50.12(a)(2)(iii) applies:

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

In order to meet a deterministic threshold value for containment debris loading, the amount of debris generating contributors in the STP plant design would need to be significantly reduced. Estimates of radiological exposure for insulation modifications are significant and on the order of hundreds of person-Rem, depending on the scope of the modifications.

With respect to the presence of such special circumstances, dose estimates for removal of insulation from STP Units 1 and 2 are described below. These dose estimates are for additional modifications to insulation in containment that would be required to achieve full resolution of GSI-191 using the previous deterministic methods. The residual risk associated with GSI-191 concerns bounds the expected improvement to overall plant risk that could be achieved by implementing these modifications.

<u>Dose Considerations: STP Units 1 and 2 Plant Dose Estimates for GSI-191 Insulation Removal</u>

STPNOC estimated the occupational dose for STP Units 1 and 2 that would be expected to be expended if plant modifications were undertaken for GSI-191, including insulation replacement and other modifications. The scope of the estimate included the replacement of fiberglass insulation with reflective metal insulation (RMI) for reactor coolant pump (RCP) insulation and a portion of the steam generator (SG) insulation, and the banding of existing fiberglass insulation on piping in containment. SG insulation replacement considered whether locations were within the zone of influence (ZOI) or beyond, with ZOI extending to seventeen times the diameter of the piping (17D).

The total dose expected to be expended for STP Units 1 and 2 in support of insulation replacement for GSI-191 is estimated to be 158 to 176 rem.

STP experience with this type of work suggests the lower expected dose estimates may be appropriate. However, bounding dose estimates based on the estimated installation

hours including scaffold work and the average dose rates STP has historically experienced working within the bioshield are also provided below.

Insulation modifications, summary estimates:

Activities	Estimated Man-Hours	Estimated Dose per Unit (Expected)	Estimated Dose per Unit (Bounding)
Replace SG and RCP insulation with RMI	35,000	70 Rem	315 Rem
Install bands on insulated piping	9,000	9 to 18 Rem	81 Rem

Pipe insulation banding scope:

Piping Size (inches)	Insulation Thickness (inches)	Estimated Length (lineal feet)
31	3.5	120
29	4	68
27.5	3.5	76
16	4	160
12	2.5	430
8	2.5	168
6	2.5	94
4	2.5	592
2	2.5	228

SG fiberglass insulation replacement scope:

SG section	Volume per SG (cubic feet)	Total Volume – Four SGs (cubic feet)
Bottom end (<17D)	85	340
El 37' to El 52' (<17D)	197	788
El 52' to El 68' (<17D)	214	856
El 68' to transition (<17D)	17	68
SG transition (<17D)	134	536
Transition to El 83' (<17D)	53	212
Above El 83' (>17D)	305	1220
Top end (>17D)	150	600
Total within ZOI (<17D)	700	2800
Total beyond ZOI (>17D)	455	1820

RCP insulation replacement scope:

- All RCP fiberglass insulation (thickness 3.5 inches) to be replaced with RMI
- Fiberglass insulation volume per RCP: 56 cubic feet

- Total fiberglass insulation volume (4 RCPs): 224 cubic feet per Unit (448 cubic feet total)

For the above estimates, the highest dose contributor is personnel work hours in close proximity to high dose sources such as steam generators and primary coolant piping. The estimates did not include any replacement of reactor pressure vessel (RPV) insulation, which is RMI as originally designed for STP, therefore while the estimates may be indicative of a plant with high fiber loading, they do not necessarily account for activities that may be required for similar plants assuming 100-percent replacement of fibrous insulation in all areas that could be affected by a postulated LOCA. The dose estimates for STP Units 1 and 2, in addition to the actual insulation replacement, considered man-hours required to erect and remove scaffolding to support the insulation modifications and the dose associated with removal of insulation. The estimates did not consider dose associated with disposal of removed insulation or dose associated with potential hanger modifications for small bore piping insulation change to RMI.

The dose considerations discussed above demonstrate that compliance would result in substantial personnel exposure due to insulation modifications in the containment which is not commensurate with the expected safety benefit based on the results showing that the risk associated with GSI-191 concerns is far less than the threshold for Region III in RG 1.174. Consequently, the special circumstances described in § 50.12(a)(2)(iii) apply.

In conclusion, special circumstances in § 50.12(a)(2)(ii) and § 50.12(a)(2)(iii) are present as required by § 50.12(a)(2) for consideration of the request for exemption.

4. Technical Justification for the Exemption

The supporting information that describes the STP risk-informed approach is provided in the enclosures to this letter. The generic methodology for the risk-informed method is described in Enclosure 1. Technical justification for the risk-informed method is provided in Enclosures 4-1 through 4-3, and in the LAR (Enclosure 3), which includes descriptions of the ECCS and containment emergency sump designs and performance criteria.

The proposed risk-informed approach meets the key principles in RG 1.174 in that it is consistent with the defense-in-depth philosophy, maintains sufficient safety margins, results in small increase in risk, and is monitored using performance measurement strategies. Detailed descriptions of the PRA and supporting engineering analyses are provided in Enclosures 4-2 and 4-3 to this letter. This proposed exemption to allow use of the risk-informed method is consistent with the key principle in RG 1.174 that requires the proposed change to meet current regulations unless explicated related to a requested exemption.

The results show that the risk associated with GSI-191 concerns is far less than the threshold in Region III, "Very Small Changes," of RG 1.174, and therefore are consistent with the Commission's Safety Goals for public health and safety.

5. Environmental Consideration

Pursuant to the requirements of § 51.41 and § 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments," the following information is provided. As demonstrated below, this exemption qualifies for a categorical exclusion in 10 CFR 51.22. However, if the NRC determines that an environmental assessment is necessary, this information will support a finding of no significant impact.

Identification of the Proposed Action

The proposed exemption is to allow for use of a risk-informed approach to evaluate the residual risk associated with GSI-191, i.e. those concerns that have not been fully addressed using deterministic methods, for the purpose of reconstituting the design basis for acceptable performance of the containment emergency sumps during recirculation mode following postulated LOCAs. Approval of the proposed exemption would allow for approval of a change to the UFSAR, as provided in Enclosure 3 to this letter, for implementation of the risk-informed method for STP Units 1 and 2.

Need for the Proposed Action

In the Commission's Policy Statement on "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities" (Reference 8), the Commission stated that "*the use of PRA technology in NRC regulatory activities should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach*" and that is consistent with traditional defense-in-depth concepts.

The intent of the Commission's Policy Statement intent is to use the PRA to further understand the risk associated with a proposed change for the purpose of removing unnecessary conservatism associated with regulatory requirements in order to focus attention and allocation of resources to areas of true safety significance.

To implement the Commission Policy Statement, the NRC issued RG 1.174 to provide guidance on an acceptable approach to risk-informed decision-making, based on a set of five key principles. The proposed action is needed to allow STPNOC to use a risk-informed method to resolve the issues associated with GSI-191 concerns regarding the potential for insulation and other debris generated in the event of a postulated LOCA within the containment impacting acceptable recirculation operation for ECCS, and challenge the ability of ECCS to provide adequate long-term core cooling. This proposed exemption is consistent with the key principle in RG 1.174 which requires the proposed change to meet current regulations unless explicated related to a requested exemption.

Environmental Impacts Consideration

The proposed exemption has been evaluated and determined to result in no significant radiological environmental impacts associated with the implementation of the change. This conclusion is based on the following.

The proposed exemption is to allow a risk-informed method for demonstrating the design and licensing bases for the ECCS. No physical modifications or changes to operating requirements are proposed for the site or facility, including any systems, structures and components relied upon to mitigate the consequences of a LOCA. The intent of the proposed change is to quantify the risk associated with GSI-191 concerns. This quantification, provided in the form of risk metrics using the guidance in RG 1.174, demonstrates that the risk is far less than the threshold for Region III, "Very Small Changes," in RG 1.174. Therefore, the proposed exemption supports a change that represents a very small change in Large Early Release Frequency (LERF) that corresponds to an insignificant impact on the environment.

Based on the results of the risk-informed method demonstrating that the increases in risk are very small, the proposed exemption has a negligible effect on accident probability, and adequate assurance of public health and safety is maintained. The proposed exemption does not involve any changes to the facility or facility operations that could create a new or significantly affect a previously analyzed accident or release path, and therefore would result in no significant changes in the types or quantities of radiological effluents that may be released offsite. The proposed change does not affect the generation of any radioactive effluents, and does not affect any of the permitted effluent release paths.

The proposed exemption has no impact on requirements related to the integrity of the reactor coolant system piping or any other aspect related to the initiation of a LOCA. No physical modifications or changes to operating requirements are proposed for the facility, including any systems, structures and components relied upon to mitigate the consequences of a LOCA. Therefore, the proposed exemption does not affect the probability of an accident initiator.

The proposed exemption does not significantly impact a release of radiological effluents during and following a postulated LOCA. The design-basis LOCA radiological consequence analysis in the current licensing basis is a deterministic evaluation based on the assumption of a major rupture of the reactor coolant system piping and a significant amount of core damage as specified in RG 1.183 (Reference 9). The current licensing basis analysis shows the resulting doses to the public and to control room and technical support center personnel do not exceed the regulatory limits. The proposed change validates and does not change the input parameter value used in the radiological analysis. Therefore, the proposed exemption does not affect the amount of radiation exposure resulting from a postulated LOCA.

The proposed exemption does not involve any changes to the site property, physical changes to the facility, or changes to the operation of the facility. Therefore there are no irreversible and irretrievable commitments of resources which would be involved in the proposed action should it be implemented. The risk-informed method requires a

determination that the risk associated with the proposed change meets the Commission's safety goals. Therefore the proposed action would not result in a significant increase in any radiological hazard beyond those events previously analyzed in the UFSAR. There will be no change to radioactive effluents that affect radiation exposures to plant workers and members of the public. Therefore, no significant changes or different types of radiological impacts are expected as a result of the proposed action. The proposed exemption does not change the input parameter value used in the radiological analysis. Therefore, the proposed change would not significantly increase the probability or consequences of an accident, and there will be no significant offsite impact to the public from approval of the proposed exemption.

No physical modifications or changes to operating requirements are proposed for the facility, including any systems, structures and components relied upon to mitigate the consequences of a LOCA. Therefore, the proposed exemption does not result in a significant increase in individual or cumulative occupational radiation exposure, and will not cause radiological exposure in excess of the dose criteria for restricted and unrestricted access specified in 10 CFR Part 20.

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

Since implementation of the exemption request, if approved, would result in no physical changes to the facility, there is no possibility of irreversible or irretrievable commitments of resources. Similarly, the proposed exemption does not involve the use of any resources not previously considered by the NRC in its past environmental statements for issuance of the facility operating licenses or other licensing actions for the facility. As a result, the proposed exemption does not involve any unresolved conflicts concerning alternative uses of available resources.

Alternatives

The alternative to this exemption is compliance with the existing provisions in 10 CFR 50.46(b)(5). Compliance with 10 CFR 50.46(b)(5) would entail removal and disposal of significant amounts of insulation and installation of new insulation less likely to impact sump performance in the event of a LOCA. As discussed below, the alternative would not be environmentally preferable or cost justified.

The exemption entails a very small risk and, correspondingly, a very small environmental impact. In particular, the change in LERF is ~ 8.6E-12/yr. This change is so small that it is remote and speculative for environmental purposes.

Removal and reinstallation of insulation would entail appreciable radiation exposures to workers (estimated from 158 to 176 rem). This option results in extensive modifications

to the facility and significant occupational dose. As such, the alternative is not environmentally preferable. Additionally, the cost of the installation replacement would be approximately $55 million. This cost is not justified in light of the very small risk associated with the risk-informed exemption.

Categorical Exclusion Consideration

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

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

(i) *The exemption involves no significant hazards consideration.*

An evaluation of the three criteria set forth in 10 CFR 50.92(c) as applied to the exemption is provided below. The evaluation is consistent with the no significant hazards consideration determination provided in Enclosure 3 in support of the LAR.

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

Approval of the proposed exemption and accompanying license amendment request would allow the results of a risk-informed evaluation to be included in the UFSAR that concludes the ECCS and CSS systems will operate with a high probability following a LOCA when considering the impacts and effects of debris accumulation on containment emergency sump strainers in recirculation mode, as well as core flow blockage due to in-vessel effects, following loss of coolant accidents (LOCAs).

The proposed change does not implement any physical changes to the facility or any structures, systems and components (SSCs), and does not implement any changes in plant operation. The proposed change confirms that required SSCs supported by the containment sumps will perform their safety functions as required, and does not alter or prevent the ability of SSCs to perform their intended function to mitigate the consequences of an accident previously evaluated within the acceptance limits. The safety analysis acceptance criteria in the UFSAR continue to be met for the proposed change. The proposed change does not affect initiating events. The proposed change does not significantly affect the operation of the containment systems needs to ensure that there is a large margin between the temperature and pressure conditions

reached in the containment and those that would lead to failure so that there is a high degree of confidence that damage of the containment cannot occur.

The calculated risk associated with the proposed change is very small and far less than the threshold for Region III as defined by RG 1.174, for both CDF and LERF. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any the accident previously evaluated in the UFSAR.

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

The proposed change neither installs nor removes any plant equipment, nor alters the design, physical configuration, or mode of operation of any plant structure, system or component. The proposed change does not introduce any new failure mechanisms or malfunctions that can initiate an accident. The proposed change does not introduce failure modes, accident initiators, or equipment malfunctions that would cause a new or different kind of accident. Therefore, the proposed change does not create the possibility for a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not involve a change in any functional requirements, the configuration, or method of performing functions of plant SSCs. The effects from a full spectrum of LOCAs, including double-ended guillotine breaks for all piping sizes up to and including the largest pipe in the reactor coolant system, are analyzed. Appropriate redundancy and consideration of loss of offsite power and worst case single failure are retained, such that defense-in-depth is maintained.

The risk-informed method demonstrates the containment sumps will continue to support the ability of safety related components to perform their design functions. The proposed change does not alter the manner in which safety limits are determined or acceptance criteria associated with a safety limit. The proposed change does not implement any changes to plant operation, and does not significantly affect SSCs that respond to safely shutdown the plant and to maintain the plant in a safe shutdown condition. The proposed change does not significantly affect the existing safety margins in the barriers for the release of radioactivity. There are no changes to any of the safety analyses in the UFSAR. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

No physical modifications or changes to operating requirements are proposed for the facility, including any systems, structures and components relied upon to mitigate the

consequences of a LOCA. Approval of the exemption requires the calculated risk associated with GSI-191 to meet the acceptance guidelines in RG 1.174, thereby maintaining public health and safety. Therefore there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

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

No physical modifications or changes to operating requirements are proposed for the facility, including any systems, structures and components relied upon to mitigate the consequences of a LOCA. Therefore, with respect to installation or use of a facility component located within the restricted area there is no significant increase in individual or cumulative occupational radiation exposure as a result of granting the exemption request.

Based on the above, STPNOC concludes that the proposed exemption meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Additional technical justification for this conclusion is provided on the basis that the guidance and acceptance criteria provided in RG 1.174 are satisfied as described in Enclosure 4-1.

6. Conclusion

Approval of an exemption to allow the use of the risk-informed approach is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security as required by § 50.12(a)(1). Furthermore, special circumstances required by § 50.12(a)(2) are present for item § 50.12(a)(2)(ii) in that application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule.

Based on the determination that the risk of the exemption meets the acceptance guidelines of RG 1.174, the results demonstrate there is reasonable assurance that the ECCS will function in the recirculation mode and that the public health and safety will be protected.

7. Implementation

To support the completion of work and the resolution schedule for closure of GSI-191 as described in SECY-12-0093, STPNOC requests that this exemption request be approved for implementation by December 2014.

8. References

1) Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," September 13, 2004 (ML042360586)

2) Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance"

3) Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2 (ML100910006)

4) "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Nos. 183 and 170 to Facility Operating License Nos. NPF-76 and NPF-80 STP Nuclear Operating Company, et al., South Texas Project, Units 1 and 2, Docket Nos. 50-498 and 50-499," March 25, 2008 (ML080360321)

5) GSI-191 Safety Evaluation Report, Rev. 0, "Evaluation of NEI Guidance on PWR Sump Performance," December 6, 2004 (ML043280007)

6) NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Volume 1 "Pressurized Water Reactor Sump Performance Evaluation Methodology," Revision 0, December 2004 (ML050550138)

7) Federal Register Notice, Vol. 53, No. 180 (53 FR 35996), Emergency Core Cooling Systems; Revisions to Acceptance Criteria, September 16, 1988

8) Commission's Policy Statement on "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," August 16, 1995 (60 FR 42622)

9) South Texas Project, Units 1 and 2 - Issuance of Amendments Re: Adoption of Alternate Radiological Source Term in Assessment of Design-Basis Accident Dose Consequences (TAC Nos. MD4996 and MD4997), March 6, 2008 (ML080160013)

10) Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000

ENCLOSURE 2-2

Request for Exemption from

General Design Criterion 35

Request for Exemption from Certain Requirements of General Design Criterion 35

1. Exemption Request

Pursuant to 10 CFR 50.12, STPNOC is submitting this request for exemption from certain requirements of 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 35, which states:

> *Criterion 35 — Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.*
>
> *Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.*

Licensees are required to demonstrate this capability while assuming the most conservative (and worst) single failure. This regulation has been interpreted as requiring the use of a bounding calculation or other deterministic method, for the purpose of addressing containment emergency sump performance as discussed in Generic Letter 2004-02 (Reference 1) and Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance" (Reference 2). STPNOC requests an exemption from that requirement in order to enable the use of a risk-informed method to demonstrate acceptable sump design.

Approval of this exemption will allow use of a risk-informed method to account for the probabilities and uncertainties associated with the containment emergency sump design in support of the operation of the Emergency Core Cooling System (ECCS) following postulated LOCAs. The method evaluates the effects on strainer blockage and core blockage resulting from debris concerns raised by GSI-191. In order to confirm acceptable sump design, the risk associated with GSI-191 is evaluated to include the failure mechanisms associated with loss of core cooling and strainer blockage.

This exemption request is in support of the accompanying License Amendment Request (LAR) (Enclosure 3) seeking NRC approval of the changes to the South Texas Project Electric Generating Station (STPEGS) Updated Final Safety Analysis Report (UFSAR), to reconstitute the licensing basis based on acceptable design of the containment sump. The risk-informed method provides assurance, with high probability, for an acceptable sump design that complies with GDC 35 and resolves GSI-191.

2. Regulatory Requirements Involved

STPNOC seeks exemption to the extent that GDC 35 requires deterministic calculations or other analyses to address the concerns raised by GSI-191 related to acceptable plant performance during the recirculation mode in containment following a LOCA. The proposed changes to the licensing basis, submitted for NRC approval with the LAR in Enclosure 3, provide closure to GSI-191 for STP Units 1 and 2 on the basis that the associated risk is shown to meet the acceptance guidelines in Regulatory Guide (RG) 1.174 (Reference 3) and that, in conjunction with the existing licensing basis, adequate safety is demonstrated.

This exemption request is for the purpose of allowing the use of a risk-informed method to demonstrate acceptable design of the containment emergency sump in support of the recirculation mode in containment following postulated loss of coolant accidents (LOCAs). The containment sump has been evaluated, using deterministic methods, to meet the current licensing basis assumptions for analyzing the effects of post-LOCA debris blockage; however, these evaluations have not been shown to fully resolve GSI-191 for the as-built, as-operated plant (Reference 4). The risk-informed approach evaluates the sump design as part of the assessment of the residual risk associated with GSI-191 concerns. Based on confirmation of acceptable ECCS design as determined by the resulting risk meeting the acceptance guidelines in RG 1.174, the licensing basis for ECCS compliance with GDC 35 is reconstituted.

Other regulatory requirements associated with containment sumps in support of ECCS and Containment Spray System (CSS) recirculation modes following postulated LOCAs include 10 CFR 50.46(b)(5) for core cooling, GDC 38 for containment heat removal, and GDC 41 for containment atmosphere cleanup. These requirements are addressed as part of separate exemption requests.

2.1 Evaluation of Impacts on § 50.46 and Appendix K to 10 CFR Part 50

This exemption request is intended to address ECCS design requirements for meeting GDC 35 design requirements, and is not applicable to § 50.46 or Appendix K.

A separate exemption request provided in Enclosure 2-1 addresses exemption from § 50.46(b)(5) for long term cooling and discusses impacts on § 50.46 and Appendix K.

2.2 Evaluation of Impacts on 10 CFR 50.67 and GDC 19

The impact of the proposed exemption on the licensing basis analysis for demonstrating radiological consequences of the design basis LOCA meet the radiological dose guidelines specified in 10 CFR 50.67 and the dose limits specified in GDC 19 was evaluated. The risk-informed method provides confirmation of reliable ECCS and CSS performance as required for the licensing basis analyses that demonstrate the requirements of 10 CFR 50.67 and GDC 19. The method demonstrates that sump performance continues to support reliable plant design and operation and does not entail any exemption from 10 CFR 50.67 or GDC 19.

For STP Units 1 and 2, which have implemented the Alternative Source Term (AST), the design-basis LOCA radiological consequence LOCA analysis is a deterministic evaluation based on the assumption of a major rupture of the reactor coolant system piping and the assumption of the deterministic failure of the ECCS to provide adequate core cooling (Reference 9). This scenario results in a significant amount of core damage as specified in RG 1.183 (Reference 10), and does not represent any specific accident sequence, but is representative of a class of severe damage incidents that were evaluated in the development of the RG 1.183 source term characteristics. Such a scenario would be expected to require multiple failures of systems and equipment and lies beyond the likely incidents evaluated for design-basis transient analyses (Reference 9). Since deterministic failure of ECCS is assumed at the onset of the accident by the analysis, the reliability of the containment emergency sumps with respect to ECCS operation does not affect the analysis for dose consequences. Therefore, for the purposes of this exemption request, the current licensing basis analyses for 10 CFR 50.67 and GDC-19 are not impacted.

2.3 Evaluation of Impacts on other Regulatory Requirements

The STP risk-informed approach, as described in Enclosures 1 and 4, uses the PRA model to quantify the risk associated with GSI-191, thereby quantifying the residual risk from those issues which have not been resolved using other methods. A determination that this approach and its results meet the key principles and acceptance guidelines in RG 1.174 demonstrates acceptable sump performance in support of ECCS and CSS operation in recirculation mode following postulated LOCAs, and demonstrates that the Commission's safety goals and public health and safety are maintained.

The proposed exemption does not result in any physical changes to the facility or changes to the operation of the plant, and does not change any of the programmatic requirements. Based on demonstrating acceptable containment emergency sump and ECCS design for reconstituting the current licensing basis for compliance with GDC 35 as described above, compliance with other regulatory requirements that rely on acceptable design for these systems and components continue to be met in the current licensing basis.

3. Basis for the Exemption Request

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

The exemption request meets a key principle of RG 1.174, which states "*The proposed change meets the current regulations unless it is explicitly related to a requested exemption.*" This exemption request is provided in support of the proposed change provided in the License Amendment Request provide in Enclosure 3.

3.1 Justification for the Exemption Request

As required by 50.12(a)(2), the Commission will not consider granting an exemption unless special circumstances are present. Special circumstances are present whenever one of the listed items (i) through (vi) under 50.12(a)(2) are applicable. STPNOC has evaluated the proposed exemption against the conditions specified in 10 CFR 50.12(a) and determined that this proposed exemption meets the requirements for granting an exemption from the regulation, and that special circumstances are present. The information supporting the determination is provided below.

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

The exemption is authorized by law.

The NRC has authority under the Atomic Energy Act of 1954, as amended, to grant exemptions from its regulations if doing so would not violate the requirements of law. This exemption is authorized by law as is provided by 10 CFR 50.12 which provides the NRC authority to grant exemptions from Part 50 requirements with provision of proper justification. Approval of the exemption would not conflict with any provisions of the Atomic Energy Act of 1954, as amended, the Commission's regulations, or any other law

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

The proposed change is to apply a risk-informed method rather than a traditional deterministic method in order to quantify the residual risk associated with GSI-191 and to establish a high confidence of acceptable ECCS design. The purpose of GDC 35 is to establish acceptable design for ECCS, and together with the acceptance criteria of § 50.46, to provide a high probability that the systems will perform the required functions. The proposed exemption does not involve any modifications to the plant that could introduce a new accident precursor or affect the probability of postulated accidents, and therefore the probability of postulated initiating events is not increased. The PRA and engineering analysis demonstrate that the calculated risk is small and consistent with the intent of the Commission's Safety Goal Policy Statement, which defines an acceptable level of risk that is a small fraction of other risks to which the public is exposed.

As discussed in previous § 50.46 rulemaking (Reference 7), the probability of a large break LOCA is sufficiently low that the application of a risk-informed approach to evaluate the ability of the ECCS to meet § 50.46(b)(5) with high probability and with low uncertainty, rather than using a calculational model using deterministic methods to achieve similar understanding, would have little effect on public risk. This is applicable to evaluating acceptable containment sump design in support of ECCS and CSS recirculation modes.

The risk-informed approach involves a complete evaluation of the spectrum of LOCA breaks, including double-ended guillotine break, up to and including the largest pipe in the reactor coolant system (see Enclosure 4-3, Volume 3 Section 5.3.6). The risk-informed approach analyzes LOCAs, regardless of break size, using the same methods,

assumptions, and criteria in order to quantify the uncertainties and overall risk metrics. This ensures that large break LOCAs with relatively small contribution to CDF due to the low probability of such a break as well as smaller break LOCAs with higher probabilities of occurrence are considered in the results. Since the design basis requirement for consideration of a double-ended guillotine break of the largest pipe in the reactor coolant system is retained and since no physical changes to the facility or changes to the operation of the facility are being made, the existing defense-in-depth and safety margin established for the design of the facility is not reduced.

The exemption is consistent with the common defense and security.

The exemption involves a change to the licensing basis for the plant that has no relation to the possession of licensed material or any security requirements that apply to STP Units 1 and 2. Therefore the exemption is consistent with the common defense and security.

3.2 Special Circumstances

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

Such special circumstances are present in this instance to warrant exemption from the implicit requirement in GDC 35 to use a deterministic method to evaluate for acceptable containment emergency sump design. Approval of the exemption request would allow use of a risk-informed method to reconstitute the design basis for acceptable containment sump design in support of ECCS design for compliance with GDC 35. Specifically, § 50.12(a)(2)(ii) applies:

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

The intent of GDC 35 is to ensure ECCS design provides abundant core cooling to mitigate fuel and clad damage and clad metal-water reaction following any loss of reactor coolant. GDC 35 sets forth the general ECCS cooling performance design requirements, which are in addition to the requirements of 10 CFR 50.46. This exemption request is consistent with that purpose in that use of the proposed risk-informed approach demonstrates a high probability of successful ECCS performance, which includes realistically available long term cooling, based on the risk results meeting the acceptance guidelines of RG 1.174. The risk-informed approach assesses ECCS design for a full spectrum of breaks, and assesses equipment failures that include loss of offsite power and worst case single failure, consistent with the GDC 35 requirements.

Since the proposed exemption does not involve any physical changes to the plant, there is no affect on the GDC 35 requirements for ECCS design for redundancy in components and features, interconnections, leak detection, isolation, and containment

capabilities. The current licensing basis evaluations for ECCS compliance with GDC 35 for these aspects continue to be met.

As discussed in the *Commission's Policy Statement on "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities"* (Reference 8), NRC regulations and their implementation are generally based on deterministic approaches that consider a set of challenges to safety and determine how those challenges should be mitigated. This request does not seek exemption from any explicit language in the regulatory requirements. Rather, the need is based on the implicit requirements in the regulations for using deterministic methods to demonstrate acceptable design. Regulatory requirements are largely based on a deterministic framework, and are established for design basis accidents, such as the LOCA, with specific acceptance criteria that must be satisfied. Licensed facilities must be provided with safety systems capable of preventing and mitigating the consequences of design basis accidents to protect public health and safety. The deterministic regulatory requirements were designed to ensure that these systems are highly reliable. The LOCA analysis and the General Design Criteria (GDC) were established as part of this deterministic regulatory framework.

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

An objective of GDC 35 is to maintain low risk to the public health and safety through a reliable ECCS, as supported by the containment sump. The supporting analysis demonstrates that a risk-informed approach to sump performance is consistent with the Commission's Safety Goals for nuclear power plants, and supports ECCS operation with a high degree of reliability. Consequently, the special circumstances described in § 50.12(a)(2)(ii) apply.

Specifically, § 50.12(a)(2)(iii) applies:

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

In order to meet a deterministic threshold value for containment debris loading, the amount of debris generating contributors in the STP plant design would need to be significantly reduced. Estimates of radiological exposure for insulation modifications are significant and on the order of hundreds of person-Rem, depending on the scope of the modifications.

With respect to the presence of such special circumstances, dose estimates for removal of insulation from STP Units 1 and 2 are described below. These dose estimates are for additional modifications to insulation in containment that would be required to achieve full resolution of GSI-191 using the previous deterministic methods. The residual risk associated with GSI-191 concerns bounds the expected improvement to overall plant risk that could be achieved by implementing these modifications.

Dose Considerations: STP Units 1 and 2 Plant Dose Estimates for GSI-191 Insulation Removal

STPNOC estimated the occupational dose for STP Units 1 and 2 that would be expected to be expended if plant modifications were undertaken for GSI-191, including insulation replacement and other modifications. The scope of the estimate included the replacement of fiberglass insulation with reflective metal insulation (RMI) for reactor coolant pump (RCP) insulation and a portion of the steam generator (SG) insulation, and the banding of existing fiberglass insulation on piping in containment. SG insulation replacement considered whether locations were within the zone of influence (ZOI) or beyond, with ZOI extending to seventeen times the diameter of the piping (17D).

The total dose expected to be expended for STP Units 1 and 2 in support of insulation replacement for GSI-191 is estimated to be 158 to 176 rem.

STP experience with this type of work suggests the lower expected dose estimates may be appropriate. However, bounding dose estimates based on the estimated installation hours including scaffold work and the average dose rates STP has historically experienced working within the bioshield are also provided below.

Insulation modifications, summary estimates:

Activities	Estimated Man-Hours	Estimated Dose per Unit (Expected)	Estimated Dose per Unit (Bounding)
Replace SG and RCP insulation with RMI	35,000	70 Rem	315 Rem
Install bands on insulated piping	9,000	9 to 18 Rem	81 Rem

Pipe insulation banding scope:

Piping Size (inches)	Insulation Thickness (inches)	Estimated Length (lineal feet)
31	3.5	120
29	4	68
27.5	3.5	76
16	4	160
12	2.5	430
8	2.5	168
6	2.5	94
4	2.5	592
2	2.5	228

SG fiberglass insulation replacement scope:

SG section	Volume per SG (cubic feet)	Total Volume – Four SGs (cubic feet)
Bottom end (<17D)	85	340
El 37' to El 52' (<17D)	197	788
El 52' to El 68' (<17D)	214	856
El 68' to transition (<17D)	17	68
SG transition (<17D)	134	536
Transition to El 83' (<17D)	53	212
Above El 83' (>17D)	305	1220
Top end (>17D)	150	600
Total within ZOI (<17D)	700	2800
Total beyond ZOI (>17D)	455	1820

RCP insulation replacement scope:

- All RCP fiberglass insulation (thickness 3.5 inches) to be replaced with RMI
- Fiberglass insulation volume per RCP: 56 cubic feet
- Total fiberglass insulation volume (4 RCPs): 224 cubic feet per Unit (448 cubic feet total)

For the above estimates, the highest dose contributor is personnel work hours in close proximity to high dose sources such as steam generators and primary coolant piping. The estimates did not include any replacement of reactor pressure vessel (RPV) insulation, which is RMI as originally designed for STP, therefore while the estimates may be indicative of a plant with high fiber loading, they do not necessarily account for activities that may be required for similar plants assuming 100-percent replacement of fibrous insulation in all areas that could be affected by a postulated LOCA. The dose estimates for STP Units 1 and 2, in addition to the actual insulation replacement, considered man-hours required to erect and remove scaffolding to support the insulation modifications and the dose associated with removal of insulation. The estimates did not consider dose associated with disposal of removed insulation or dose associated with potential hanger modifications for small bore piping insulation change to RMI.

The dose considerations discussed above demonstrate that compliance would result in substantial personnel exposure due to insulation modifications in the containment which is not commensurate with the expected safety benefit based on the results showing that the risk associated with GSI-191 concerns is far less than the threshold for Region III in RG 1.174. Consequently, the special circumstances described in § 50.12(a)(2)(iii) apply.

In conclusion, special circumstances in § 50.12(a)(2)(ii) and § 50.12(a)(2)(iii) are present as required by § 50.12(a)(2) for consideration of the request for exemption.

4. Technical Justification for the Exemption

The supporting information that describes the STP risk-informed approach is provided in the enclosures to this letter. The generic methodology for the risk-informed method is described in Enclosure 1. Technical justification for the risk-informed method is provided in Enclosures 4-1 through 4-3, and in the LAR (Enclosure 3), which includes descriptions of the ECCS and containment emergency sump designs and performance criteria.

The proposed risk-informed approach meets the key principles in RG 1.174 in that it is consistent with the defense-in-depth philosophy, maintains sufficient safety margins, results in small increase in risk, and is monitored using performance measurement strategies. Detailed descriptions of the PRA and supporting engineering analyses are provided in Enclosures 4-2 and 4-3 to this letter. This proposed exemption to allow use of the risk-informed method is consistent with the key principle in RG 1.174 that requires the proposed change to meet current regulations unless explicated related to a requested exemption.

The results show that the risk associated with GSI-191 concerns is far less than the threshold in Region III, "Very Small Changes," of RG 1.174, and therefore are consistent with the Commission's Safety Goals for public health and safety.

5. Environmental Consideration

Pursuant to the requirements of § 51.41 and § 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments," the following information is provided. As demonstrated below, this exemption qualifies for a categorical exclusion in 10 CFR 51.22. However, if the NRC determines that an environmental assessment is necessary, this information will support a finding of no significant impact.

Identification of the Proposed Action

The proposed exemption is to allow for use of a risk-informed approach to evaluate the residual risk associated with GSI-191, i.e. those concerns that have not been fully addressed using deterministic methods, for the purpose of reconstituting the design basis for acceptable performance of the containment emergency sumps during recirculation mode following postulated LOCAs. Approval of the proposed exemption would allow for approval of a change to the UFSAR, as provided in Enclosure 3 to this letter, for implementation of the risk-informed method for STP Units 1 and 2.

Need for the Proposed Action

In the Commission's Policy Statement on "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities" (Reference 8), the Commission stated that "*the use of PRA technology in NRC regulatory activities should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach*" and that is consistent with traditional defense-in-depth concepts.

The intent of the Commission's Policy Statement intent is to use the PRA to further understand the risk associated with a proposed change for the purpose of removing unnecessary conservatism associated with regulatory requirements in order to focus attention and allocation of resources to areas of true safety significance.

To implement the Commission Policy Statement, the NRC issued RG 1.174 to provide guidance on an acceptable approach to risk-informed decision-making, based on a set of five key principles. The proposed action is needed to allow STPNOC to use a risk-informed method to resolve the issues associated with GSI-191 concerns regarding the potential for insulation and other debris generated in the event of a postulated LOCA within the containment impacting acceptable recirculation operation for ECCS, and challenge the ability of ECCS to provide adequate long-term core cooling. This proposed exemption is consistent with the key principle which requires the proposed change to meet current regulations unless explicated related to a requested exemption.

Environmental Impacts Consideration

The proposed exemption has been evaluated and determined to result in no significant radiological environmental impacts associated with the implementation of the change. This conclusion is based on the following.

The proposed exemption is to allow a risk-informed method for demonstrating the design and licensing bases for the ECCS. No physical modifications or changes to operating requirements are proposed for the site or facility, including any systems, structures and components relied upon to mitigate the consequences of a LOCA. The intent of the proposed change is to quantify the risk associated with GSI-191 concerns. This quantification, provided in the form of risk metrics using the guidance in RG 1.174, demonstrates that the risk is far less than the threshold for Region III, "Very Small Changes," in RG 1.174. Therefore, the proposed exemption supports a change that represents a very small change in Large Early Release Frequency (LERF) that corresponds to an insignificant impact on the environment.

Based on the results of the risk-informed method demonstrating that the increases in risk are very small, the proposed exemption has a negligible effect on accident probability, and adequate assurance of public health and safety is maintained. The proposed exemption does not involve any changes to the facility or facility operations that could create a new or significantly affect a previously analyzed accident or release path, and therefore would result in no significant changes in the types or quantities of radiological effluents that may be released offsite. The proposed change does not affect the generation of any radioactive effluents, and does not affect any of the permitted effluent release paths.

The proposed exemption has no impact on requirements related to the integrity of the reactor coolant system piping or any other aspect related to the initiation of a LOCA. No physical modifications or changes to operating requirements are proposed for the facility, including any systems, structures and components relied upon to mitigate the consequences of a LOCA. Therefore, the proposed exemption does not affect the probability of an accident initiator.

The proposed exemption does not significantly impact a release of radiological effluents during and following a postulated LOCA. The design-basis LOCA radiological consequence analysis in the current licensing basis is a deterministic evaluation based on the assumption of a major rupture of the reactor coolant system piping and a significant amount of core damage as specified in RG 1.183 (Reference 9). The current licensing basis analysis shows the resulting doses to the public and to control room and technical support center personnel do not exceed the regulatory limits. The proposed change validates and does not change the input parameter value used in the radiological analysis. Therefore, the proposed exemption does not affect the amount of radiation exposure resulting from a postulated LOCA.

The proposed exemption does not involve any changes to the site property, physical changes to the facility, or changes to the operation of the facility. Therefore there are no irreversible and irretrievable commitments of resources which would be involved in the proposed action should it be implemented. The risk-informed method requires a determination that the risk associated with the proposed change meets the Commission's safety goals. Therefore the proposed action would not result in a significant increase in any radiological hazard beyond those events previously analyzed in the UFSAR. There will be no change to radioactive effluents that affect radiation exposures to plant workers and members of the public. Therefore, no significant changes or different types of radiological impacts are expected as a result of the proposed action. The proposed exemption does not change the input parameter value used in the radiological analysis. Therefore, the proposed change would not significantly increase the probability or consequences of an accident, and there will be no significant offsite impact to the public from approval of the proposed exemption.

No physical modifications or changes to operating requirements are proposed for the facility, including any systems, structures and components relied upon to mitigate the consequences of a LOCA. Therefore, the proposed exemption does not result in a significant increase in individual or cumulative occupational radiation exposure, and will not cause radiological exposure in excess of the dose criteria for restricted and unrestricted access specified in 10 CFR Part 20.

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

Since implementation of the exemption request, if approved, would result in no physical changes to the facility, there is no possibility of irreversible or irretrievable commitments of resources. Similarly, the proposed exemption does not involve the use of any resources not previously considered by the NRC in its past environmental statements for issuance of the facility operating licenses or other licensing actions for the facility. As a result, the proposed exemption does not involve any unresolved conflicts concerning alternative uses of available resources.

Alternatives

The alternative to this exemption is compliance with the existing provisions in GDC 35. Compliance with GDC 35 would entail removal and disposal of significant amounts of insulation and installation of new insulation less likely to impact sump performance in the event of a LOCA. As discussed below, the alternative would not be environmentally preferable or cost justified.

The exemption entails a very small risk and, correspondingly, a very small environmental impact. In particular, the change in LERF is ~ 8.6E-12/yr. This change is so small that it is remote and speculative for environmental purposes.

Removal and reinstallation of insulation would entail appreciable radiation exposures to workers (estimated from 158 to 176 rem). This option results in extensive modifications to the facility and significant occupational dose. As such, the alternative is not environmentally preferable. Additionally, the cost of the installation replacement would be approximately $55 million. This cost is not justified in light of the very small risk associated with the risk-informed exemption.

Categorical Exclusion Consideration

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

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

(i) *The exemption involves no significant hazards consideration.*

An evaluation of the three criteria set forth in 10 CFR 50.92(c) as applied to the exemption is provided below. The evaluation is consistent with the no significant hazards consideration determination provided in Enclosure 3 in support of the LAR.

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

Approval of the proposed exemption and accompanying license amendment request would allow the results of a risk-informed evaluation to be included in the UFSAR that concludes the ECCS and CSS systems will operate with a high probability following a LOCA when considering the impacts and effects of debris accumulation on containment emergency sump strainers in recirculation

mode, as well as core flow blockage due to in-vessel effects, following loss of coolant accidents (LOCAs).

The proposed change does not implement any physical changes to the facility or any structures, systems and components (SSCs), and does not implement any changes in plant operation. The proposed change confirms that required SSCs supported by the containment sumps will perform their safety functions as required, and does not alter or prevent the ability of SSCs to perform their intended function to mitigate the consequences of an accident previously evaluated within the acceptance limits. The safety analysis acceptance criteria in the UFSAR continue to be met for the proposed change. The proposed change does not affect initiating events. The proposed change does not significantly affect the operation of the containment systems needs to ensure that there is a large margin between the temperature and pressure conditions reached in the containment and those that would lead to failure so that there is a high degree of confidence that damage of the containment cannot occur.

The calculated risk associated with the proposed change is very small and far less than the threshold for Region III as defined by RG 1.174, for both CDF and LERF. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any the accident previously evaluated in the UFSAR.

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

The proposed change neither installs nor removes any plant equipment, nor alters the design, physical configuration, or mode of operation of any plant structure, system or component. The proposed change does not introduce any new failure mechanisms or malfunctions that can initiate an accident. The proposed change does not introduce failure modes, accident initiators, or equipment malfunctions that would cause a new or different kind of accident. Therefore, the proposed change does not create the possibility for a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not involve a change in any functional requirements, the configuration, or method of performing functions of plant SSCs. The effects from a full spectrum of LOCAs, including double-ended guillotine breaks for all piping sizes up to and including the largest pipe in the reactor coolant system, are analyzed. Appropriate redundancy and consideration of loss of offsite power and worst case single failure are retained, such that defense-in-depth is maintained.

The risk-informed method demonstrates the containment sumps will continue to support the ability of safety related components to perform their design functions. The proposed change does not alter the manner in which safety

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

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

No physical modifications or changes to operating requirements are proposed for the facility, including any systems, structures and components relied upon to mitigate the consequences of a LOCA. Approval of the exemption requires the calculated risk associated with GSI-191 to meet the acceptance guidelines in RG 1.174, thereby maintaining public health and safety. Therefore there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

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

No physical modifications or changes to operating requirements are proposed for the facility, including any systems, structures and components relied upon to mitigate the consequences of a LOCA. Therefore, with respect to installation or use of a facility component located within the restricted area there is no significant increase in individual or cumulative occupational radiation exposure as a result of granting the exemption request.

Based on the above, STPNOC concludes that the proposed exemption meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Additional technical justification for this conclusion is provided on the basis that the guidance and acceptance criteria provided in RG 1.174 are satisfied as described in Enclosure 4-1.

6. Conclusion

Approval of an exemption to allow the use of the risk-informed approach is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security as required by § 50.12(a)(1). Furthermore, special circumstances required by § 50.12(a)(2) are present for item § 50.12(a)(2)(ii) in that application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule.

Based on the determination that the risk of the exemption meets the acceptance guidelines of RG 1.174, the results demonstrate there is reasonable assurance that the ECCS will function in the recirculation mode and that the public health and safety will be protected.

7. Implementation

To support the completion of work and the resolution schedule for closure of GSI-191 as described in SECY-12-0093, STPNOC requests that this exemption request be approved for implementation by December 2014.

8. References

1) Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," September 13, 2004 (ML042360586)

2) Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance"

3) Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2 (ML100910006)

4) "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Nos. 183 and 170 to Facility Operating License Nos. NPF-76 and NPF-80 STP Nuclear Operating Company, et al., South Texas Project, Units 1 and 2, Docket Nos. 50-498 and 50-499," March 25, 2008 (ML080360321)

5) GSI-191 Safety Evaluation Report, Rev. 0, "Evaluation of NEI Guidance on PWR Sump Performance," December 6, 2004 (ML043280007)

6) NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Volume 1 "Pressurized Water Reactor Sump Performance Evaluation Methodology," Revision 0, December 2004 (ML050550138)

7) Federal Register Notice, Vol. 53, No. 180 (53 FR 35996), Emergency Core Cooling Systems; Revisions to Acceptance Criteria, September 16, 1988

8) Commission's Policy Statement on "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," August 16, 1995 (60 FR 42622)

9) South Texas Project, Units 1 and 2 - Issuance of Amendments Re: Adoption of Alternate Radiological Source Term in Assessment of Design-Basis Accident Dose Consequences (TAC Nos. MD4996 and MD4997), March 6, 2008 (ML080160013)

10) Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000

ENCLOSURE 2-3

Request for Exemption from

General Design Criterion 38

Request for Exemption from Certain Requirements of General Design Criterion 38

1. Exemption Request

Pursuant to 10 CFR 50.12, STPNOC is submitting this request for exemption from certain requirements of 10 CFR Part 50 Appendix A, General Design Criteria (GDC) 38.

> *Criterion 38— Containment heat removal. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.*
>
> *Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.*

Licensees are required to demonstrate this capability while assuming the most conservative (and worst) single failure. This regulation has been interpreted as requiring the use of a bounding calculation or other deterministic method, for the purpose of addressing containment emergency sump performance as discussed in Generic Letter 2004-02 (Reference 1) and Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance" (Reference 2). STPNOC requests an exemption from that requirement in order to enable the use of a risk-informed method to demonstrate acceptable sump design. This exemption request does not apply to other SSCs which may be required to demonstrate compliance with GDC 38.

Approval of this exemption will allow use of a risk-informed method to account for the probabilities and uncertainties associated with the containment emergency sump design in support of the operation of the Containment Spray System (CSS) following postulated LOCAs. The method evaluates the effects on containment emergency sump strainer blockage resulting from debris concerns raised by GSI-191. In order to confirm acceptable sump design, the risk associated with GSI-191 is evaluated to include the failure mechanisms associated with strainer blockage affecting CSS recirculation mode.

This exemption request is in support of the accompanying License Amendment Request (LAR) (Enclosure 3) seeking NRC approval of the changes to the South Texas Project Electric Generating Station (STPEGS) Updated Final Safety Analysis Report (UFSAR), to reconstitute the licensing basis based on acceptable design of the containment sump. The risk-informed method provides assurance, with high probability, for an acceptable sump design that complies with GDC 38 and resolves GSI-191.

2. Regulatory Requirements Involved

STPNOC seeks exemption to the extent that GDC 38 requires deterministic calculations or other analyses to address the concerns raised by GSI-191 related to acceptable plant performance during the recirculation mode in containment following a LOCA. The proposed changes to the licensing basis, submitted for NRC approval with the LAR in Enclosure 3, provide closure to GSI-191 for STP Units 1 and 2 on the basis that the associated risk is shown to meet the acceptance guidelines in Regulatory Guide (RG) 1.174 (Reference 3) and that, in conjunction with the existing licensing basis, adequate safety is demonstrated.

This exemption request is for the purpose of allowing the use of a risk-informed method to demonstrate acceptable design of the containment emergency sump in support of the recirculation mode in containment following postulated loss of coolant accidents (LOCAs). The containment sump has been evaluated, using deterministic methods, to meet the current licensing basis assumptions for analyzing the effects of post-LOCA debris blockage; however, these evaluations have not been shown to fully resolve GSI-191 for the as-built, as-operated plant (Reference 4). The risk-informed approach evaluates the sump design as part of the assessment of the residual risk associated with GSI-191 concerns. Based on confirmation of acceptable containment emergency sump and CSS design as determined by the resulting risk meeting the acceptance guidelines in RG 1.174, the licensing basis for CSS compliance with GDC 38 is reconstituted.

Other regulatory requirements associated with containment sumps in support of Emergency Core Cooling System (ECCS) and CSS recirculation modes following postulated LOCAs include 10 CFR 50.46(b)(5) for core cooling, GDC 35 for core cooling, and GDC 41 for containment atmosphere cleanup. These requirements are addressed as part of separate exemption requests.

2.1 Evaluation of Impacts on 10 CFR 50.67 and GDC 19

The impact of the proposed exemption on the licensing basis analysis for demonstrating radiological consequences of the design basis LOCA meet the radiological dose guidelines specified in 10 CFR 50.67 and the dose limits specified in GDC 19 was evaluated. The risk-informed method provides confirmation of reliable ECCS and CSS performance as required for the licensing basis analyses that demonstrate the requirements of 10 CFR 50.67 and GDC 19. The method demonstrates that sump performance continues to support reliable plant design and operation and does not entail any exemption from 10 CFR 50.67 or GDC 19.

For STP Units 1 and 2, which have implemented the Alternative Source Term (AST), the design-basis LOCA radiological consequence LOCA analysis is a deterministic evaluation based on the assumption of a major rupture of the reactor coolant system piping and the assumption of the deterministic failure of the ECCS to provide adequate core cooling (Reference 9). This scenario results in a significant amount of core damage as specified in RG 1.183 (Reference 10), and does not represent any specific accident sequence, but is representative of a class of severe damage incidents that were evaluated in the development of the RG 1.183 source term characteristics. Such a scenario would be expected to require multiple failures of systems and equipment and

lies beyond the likely incidents evaluated for design-basis transient analyses (Reference 9). Since deterministic failure of ECCS is assumed at the onset of the accident by the analysis, the reliability of the containment emergency sumps with respect to ECCS operation does not affect the analysis for dose consequences. The risk-informed method confirms reliable CSS operation as an input to the AST analysis. Therefore, for the purposes of this exemption request, the current licensing basis analyses for 10 CFR 50.67 and GDC 19 are not impacted.

2.2 Evaluation of Impacts on other Regulatory Requirements

The STP risk-informed approach, as described in Enclosures 1 and 4, uses the PRA model to quantify the risk associated with GSI-191, thereby quantifying the residual risk from those issues which have not been resolved using other methods. A determination that this approach and its results meet the key principles and acceptance guidelines in RG 1.174 demonstrates acceptable sump performance in support of ECCS and CSS operation in recirculation mode following postulated LOCAs, and demonstrates that the Commission's safety goals and public health and safety are maintained.

The proposed exemption does not result in any physical changes to the facility or changes to the operation of the plant, and does not change any of the programmatic requirements. Based on demonstrating acceptable containment emergency sump and CSS design for reconstituting the current licensing basis for compliance with GDC 38 as described above, compliance with other regulatory requirements that rely on acceptable design for these systems and components continue to be met in the current licensing basis.

3. Basis for the Exemption Request

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

3.1 Justification for the Exemption Request

As required by 50.12(a)(2), the Commission will not consider granting an exemption unless special circumstances are present. Special circumstances are present whenever one of the listed items (i) through (vi) under 50.12(a)(2) are applicable. STPNOC has evaluated the proposed exemption against the conditions specified in 10 CFR 50.12(a) and determined that this proposed exemption meets the requirements for granting an exemption from the regulation, and that special circumstances are present. The information supporting the determination is provided below.

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

The exemption is authorized by law.

The NRC has authority under the Atomic Energy Act of 1954, as amended, to grant exemptions from its regulations if doing so would not violate the requirements of law. This exemption is authorized by law as is provided by 10 CFR 50.12 which provides the NRC authority to grant exemptions from Part 50 requirements with provision of proper justification. Approval of the exemption would not conflict with any provisions of the Atomic Energy Act of 1954, as amended, the Commission's regulations, or any other law

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

The proposed change is to apply a risk-informed method rather than a traditional deterministic method in order to quantify the residual risk associated with GSI-191 and to establish a high confidence of acceptable containment sump design. The purpose of GDC 38 is to establish acceptable design for the containment heat removal system, which includes the CSS, to provide a high probability that the systems will perform the required functions. The proposed exemption does not involve any modifications to the plant that could introduce a new accident precursor or affect the probability of postulated accidents, and therefore the probability of postulated initiating events is not increased. The PRA and engineering analysis demonstrate that the calculated risk is small and consistent with the intent of the Commission's Safety Goal Policy Statement, which defines an acceptable level of risk that is a small fraction of other risks to which the public is exposed.

As discussed in previous § 50.46 rulemaking (Reference 7), the probability of a large break LOCA is sufficiently low that the application of a risk-informed approach to evaluate the ability of the ECCS to meet § 50.46(b)(5) with high probability and with low uncertainty, rather than using a calculational model using deterministic methods to achieve similar understanding, would have little effect on public risk. This is applicable to evaluating acceptable containment sump design in support of ECCS and CSS recirculation modes.

The risk-informed approach involves a complete evaluation of the spectrum of LOCA breaks, including double-ended guillotine break, up to and including the largest pipe in the reactor coolant system (see Enclosure 4-3, Volume 3 Section 5.3.6). The risk-informed approach analyzes LOCAs, regardless of break size, using the same methods, assumptions, and criteria in order to quantify the uncertainties and overall risk metrics. This ensures that large break LOCAs with relatively small contribution to CDF due to the low probability of such a break as well as smaller break LOCAs with higher probabilities of occurrence are considered in the results. Since the design basis requirement for consideration of a double-ended guillotine break of the largest pipe in the reactor coolant system is retained and since no physical changes to the facility or changes to the operation of the facility are being made, the existing defense-in-depth and safety margin established for the design of the facility is not reduced.

The exemption is consistent with the common defense and security.

The exemption involves a change to the licensing basis for the plant that has no relation to the possession of licensed material or any security requirements that apply to STP Units 1 and 2. Therefore the exemption is consistent with the common defense and security.

3.2 Special Circumstances

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

Such special circumstances are present in this instance to warrant exemption from the implicit requirement in GDC 38 to use a deterministic method to evaluate for acceptable containment emergency sump design. Approval of the exemption request would allow use of a risk-informed method to reconstitute the design basis for acceptable containment sump design in support of CSS design for compliance with GDC 38. Specifically, § 50.12(a)(2)(ii) applies:

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

The intent of GDC 38 is to ensure a containment heat removal system is provided to rapidly reduce containment pressure and temperature following any LOCA and maintain them at acceptably low levels. This exemption request is consistent with that purpose in that use of the proposed risk-informed approach demonstrates a high probability of successful CSS performance, which includes realistically available recirculation flow, based on the risk results meeting the acceptance guidelines of RG 1.174. The risk-informed approach assesses the design for a full spectrum of breaks, and assesses equipment failures that include loss of offsite power and worst case single failure, consistent with the GDC 38 requirements.

Since the proposed exemption does not involve any physical changes to the plant, there is no affect on the GDC 38 design requirements for redundancy in components and features, interconnections, leak detection, isolation, and containment capabilities. The current licensing basis evaluations for CSS compliance with GDC 38 for these aspects continue to be met.

As discussed in the *Commission's Policy Statement on "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities"* (Reference 8), NRC regulations and their implementation are generally based on deterministic approaches that consider a set of challenges to safety and determine how those challenges should be mitigated. This request does not seek exemption from any explicit language in the regulatory requirements. Rather, the need is based on the implicit requirements in the regulations for using deterministic methods to demonstrate acceptable design. Regulatory requirements are largely based on a deterministic framework, and are established for design basis accidents, such as the LOCA, with specific acceptance criteria that must be satisfied. Licensed facilities must be provided with safety systems capable of preventing and mitigating the consequences of design basis accidents to protect public health and safety. The deterministic regulatory requirements were designed to ensure that these systems are highly reliable. The LOCA analysis and the General Design Criteria (GDC) were established as part of this deterministic regulatory framework.

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

An objective of GDC 38 is to maintain low risk to the public health and safety through a reliable CSS, as supported by the containment sump. The supporting analysis demonstrates that a risk-informed approach to sump performance is consistent with the Commission's Safety Goals for nuclear power plants, and supports CSS operation with a high degree of reliability. Consequently, the special circumstances described in § 50.12(a)(2)(ii) apply.

Specifically, § 50.12(a)(2)(iii) applies:

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

In order to meet a deterministic threshold value for containment debris loading, the amount of debris generating contributors in the STP plant design would need to be significantly reduced. Estimates of radiological exposure for insulation modifications are significant and on the order of hundreds of person-Rem, depending on the scope of the modifications.

With respect to the presence of such special circumstances, dose estimates for removal of insulation from STP Units 1 and 2 are described below. These dose estimates are for additional modifications to insulation in containment that would be required to achieve full resolution of GSI-191 using the previous deterministic methods. The residual risk associated with GSI-191 concerns bounds the expected improvement to overall plant risk that could be achieved by implementing these modifications.

<u>Dose Considerations: STP Units 1 and 2 Plant Dose Estimates for GSI-191 Insulation Removal</u>

STPNOC estimated the occupational dose for STP Units 1 and 2 that would be expected to be expended if plant modifications were undertaken for GSI-191, including insulation replacement and other modifications. The scope of the estimate included the replacement of fiberglass insulation with reflective metal insulation (RMI) for reactor coolant pump (RCP) insulation and a portion of the steam generator (SG) insulation, and the banding of existing fiberglass insulation on piping in containment. SG insulation replacement considered whether locations were within the zone of influence (ZOI) or beyond, with ZOI extending to seventeen times the diameter of the piping (17D).

The total dose expected to be expended for STP Units 1 and 2 in support of insulation replacement for GSI-191 is estimated to be 158 to 176 rem.

STP experience with this type of work suggests the lower expected dose estimates may be appropriate. However, bounding dose estimates based on the estimated installation

hours including scaffold work and the average dose rates STP has historically experienced working within the bioshield are also provided below.

Insulation modifications, summary estimates:

Activities	Estimated Man-Hours	Estimated Dose per Unit (Expected)	Estimated Dose per Unit (Bounding)
Replace SG and RCP insulation with RMI	35,000	70 Rem	315 Rem
Install bands on insulated piping	9,000	9 to 18 Rem	81 Rem

Pipe insulation banding scope:

Piping Size (inches)	Insulation Thickness (inches)	Estimated Length (lineal feet)
31	3.5	120
29	4	68
27.5	3.5	76
16	4	160
12	2.5	430
8	2.5	168
6	2.5	94
4	2.5	592
2	2.5	228

SG fiberglass insulation replacement scope:

SG section	Volume per SG (cubic feet)	Total Volume – Four SGs (cubic feet)
Bottom end (<17D)	85	340
El 37' to El 52' (<17D)	197	788
El 52' to El 68' (<17D)	214	856
El 68' to transition (<17D)	17	68
SG transition (<17D)	134	536
Transition to El 83' (<17D)	53	212
Above El 83' (>17D)	305	1220
Top end (>17D)	150	600
Total within ZOI (<17D)	700	2800
Total beyond ZOI (>17D)	455	1820

RCP insulation replacement scope:

- All RCP fiberglass insulation (thickness 3.5 inches) to be replaced with RMI
- Fiberglass insulation volume per RCP: 56 cubic feet

- Total fiberglass insulation volume (4 RCPs): 224 cubic feet per Unit (448 cubic feet total)

For the above estimates, the highest dose contributor is personnel work hours in close proximity to high dose sources such as steam generators and primary coolant piping. The estimates did not include any replacement of reactor pressure vessel (RPV) insulation, which is RMI as originally designed for STP, therefore while the estimates may be indicative of a plant with high fiber loading, they do not necessarily account for activities that may be required for similar plants assuming 100-percent replacement of fibrous insulation in all areas that could be affected by a postulated LOCA. The dose estimates for STP Units 1 and 2, in addition to the actual insulation replacement, considered man-hours required to erect and remove scaffolding to support the insulation modifications and the dose associated with removal of insulation. The estimates did not consider dose associated with disposal of removed insulation or dose associated with potential hanger modifications for small bore piping insulation change to RMI.

The dose considerations discussed above demonstrate that compliance would result in substantial personnel exposure due to insulation modifications in the containment which is not commensurate with the expected safety benefit based on the results showing that the risk associated with GSI-191 concerns is far less than the threshold for Region III in RG 1.174. Consequently, the special circumstances described in § 50.12(a)(2)(iii) apply.

In conclusion, special circumstances in § 50.12(a)(2)(ii) and § 50.12(a)(2)(iii) are present as required by § 50.12(a)(2) for consideration of the request for exemption.

4. Technical Justification for the Exemption

The supporting information that describes the STP risk-informed approach is provided in the enclosures to this letter. The generic methodology for the risk-informed method is described in Enclosure 1. Technical justification for the risk-informed method is provided in Enclosures 4-1 through 4-3, and in the LAR (Enclosure 3), which includes descriptions of the CSS and containment emergency sump designs and performance criteria.

The proposed risk-informed approach meets the key principles in RG 1.174 in that it is consistent with the defense-in-depth philosophy, maintains sufficient safety margins, results in small increase in risk, and is monitored using performance measurement strategies. Detailed descriptions of the PRA and supporting engineering analyses are provided in Enclosures 4-2 and 4-3 to this letter. This proposed exemption to allow use of the risk-informed method is consistent with the key principle in RG 1.174 that requires the proposed change to meet current regulations unless explicated related to a requested exemption.

The results show that the risk associated with GSI-191 concerns is far less than the threshold in Region III, "Very Small Changes," of RG 1.174, and therefore are consistent with the Commission's Safety Goals for public health and safety

5. Environmental Consideration

Pursuant to the requirements of § 51.41 and § 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments," the following information is provided. As demonstrated below, this exemption qualifies for a categorical exclusion in 10 CFR 51.22. However, if the NRC determines that an environmental assessment is necessary, this information will support a finding of no significant impact.

Identification of the Proposed Action

The proposed exemption is to allow for use of a risk-informed approach to evaluate the residual risk associated with GSI-191, i.e. those concerns that have not been fully addressed using deterministic methods, for the purpose of reconstituting the design basis for acceptable performance of the containment emergency sumps during recirculation mode following postulated LOCAs. Approval of the proposed exemption would allow for approval of a change to the UFSAR, as provided in Enclosure 3 to this letter, for implementation of the risk-informed method for STP Units 1 and 2.

Need for the Proposed Action

In the Commission's Policy Statement on "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities" (Reference 8), the Commission stated that "*the use of PRA technology in NRC regulatory activities should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach*" and that is consistent with traditional defense-in-depth concepts.

The intent of the Commission's Policy Statement intent is to use the PRA to further understand the risk associated with a proposed change for the purpose of removing unnecessary conservatism associated with regulatory requirements in order to focus attention and allocation of resources to areas of true safety significance.

To implement the Commission Policy Statement, the NRC issued RG 1.174 to provide guidance on an acceptable approach to risk-informed decision-making, based on a set of five key principles. The proposed action is needed to allow STPNOC to use a risk-informed method to resolve the issues associated with GSI-191 concerns regarding the potential for insulation and other debris generated in the event of a postulated LOCA within the containment impacting acceptable recirculation operation for the CSS, and challenge its ability to provide adequate containment heat removal. This proposed exemption is consistent with the key principle in RG 1.174 which requires the proposed change to meet current regulations unless explicated related to a requested exemption.

Environmental Impacts Consideration

The proposed exemption has been evaluated and determined to result in no significant radiological environmental impacts associated with the implementation of the change. This conclusion is based on the following.

The proposed exemption is to allow a risk-informed method for demonstrating the design and licensing bases for the CSS. No physical modifications or changes to operating requirements are proposed for the site or facility, including any systems, structures and components relied upon to mitigate the consequences of a LOCA. The intent of the proposed change is to quantify the risk associated with GSI-191 concerns. This quantification, provided in the form of risk metrics using the guidance in RG 1.174, demonstrates that the risk is far less than the threshold for Region III, "Very Small Changes," in RG 1.174. Therefore, the proposed exemption supports a change that represents a very small change in Large Early Release Frequency (LERF) that corresponds to an insignificant impact on the environment.

Based on the results of the risk-informed method demonstrating that the increases in risk are very small, the proposed exemption has a negligible effect on accident probability, and adequate assurance of public health and safety is maintained. The proposed exemption does not involve any changes to the facility or facility operations that could create a new or significantly affect a previously analyzed accident or release path, and therefore would result in no significant changes in the types or quantities of radiological effluents that may be released offsite. The proposed change does not affect the generation of any radioactive effluents, and does not affect any of the permitted effluent release paths.

The proposed exemption has no impact on requirements related to the integrity of the reactor coolant system piping or any other aspect related to the initiation of a LOCA. No physical modifications or changes to operating requirements are proposed for the facility, including any systems, structures and components relied upon to mitigate the consequences of a LOCA. Therefore, the proposed exemption does not affect the probability of an accident initiator.

The proposed exemption does not significantly impact a release of radiological effluents during and following a postulated LOCA. The design-basis LOCA radiological consequence analysis in the current licensing basis is a deterministic evaluation based on the assumption of a major rupture of the reactor coolant system piping and a significant amount of core damage as specified in RG 1.183 (Reference 9). The current licensing basis analysis shows the resulting doses to the public and to control room and technical support center personnel do not exceed the regulatory limits. The proposed change validates and does not change the input parameter value used in the radiological analysis. Therefore, the proposed exemption does not affect the amount of radiation exposure resulting from a postulated LOCA.

The proposed exemption does not involve any changes to the site property, physical changes to the facility, or changes to the operation of the facility. Therefore there are no irreversible and irretrievable commitments of resources which would be involved in the proposed action should it be implemented. The risk-informed method requires a determination that the risk associated with the proposed change meets the Commission's safety goals. Therefore the proposed action would not result in a significant increase in any radiological hazard beyond those events previously analyzed in the UFSAR. There will be no change to radioactive effluents that affect radiation exposures to plant workers and members of the public. Therefore, no significant

changes or different types of radiological impacts are expected as a result of the proposed action. The proposed exemption does not change the input parameter value used in the radiological analysis. Therefore, the proposed change would not significantly increase the probability or consequences of an accident, and there will be no significant offsite impact to the public from approval of the proposed exemption.

No physical modifications or changes to operating requirements are proposed for the facility, including any systems, structures and components relied upon to mitigate the consequences of a LOCA. Therefore, the proposed exemption does not result in a significant increase in individual or cumulative occupational radiation exposure, and will not cause radiological exposure in excess of the dose criteria for restricted and unrestricted access specified in 10 CFR Part 20.

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

Since implementation of the exemption request, if approved, would result in no physical changes to the facility, there is no possibility of irreversible or irretrievable commitments of resources. Similarly, the proposed exemption does not involve the use of any resources not previously considered by the NRC in its past environmental statements for issuance of the facility operating licenses or other licensing actions for the facility. As a result, the proposed exemption does not involve any unresolved conflicts concerning alternative uses of available resources.

Alternatives

The alternative to this exemption is compliance with the existing provisions in GDC 38. Compliance with GDC 38 would entail removal and disposal of significant amounts of insulation and installation of new insulation less likely to impact sump performance in the event of a LOCA. As discussed below, the alternative would not be environmentally preferable or cost justified.

The exemption entails a very small risk and, correspondingly, a very small environmental impact. In particular, the change in LERF is ~ 8.6E-12/yr. This change is so small that it is remote and speculative for environmental purposes.

Removal and reinstallation of insulation would entail appreciable radiation exposures to workers (estimated from 158 to 176 rem). This option results in extensive modifications to the facility and significant occupational dose. As such, the alternative is not environmentally preferable. Additionally, the cost of the installation replacement would be approximately $55 million. This cost is not justified in light of the very small risk associated with the risk-informed exemption.

Categorical Exclusion Consideration

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

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

(i) *The exemption involves no significant hazards consideration.*

An evaluation of the three criteria set forth in 10 CFR 50.92(c) as applied to the exemption is provided below. The evaluation is consistent with the no significant hazards consideration determination provided in Enclosure 3 in support of the LAR.

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

Approval of the proposed exemption and accompanying license amendment request would allow the results of a risk-informed evaluation to be included in the UFSAR that concludes the ECCS and CSS will operate with a high probability following a LOCA when considering the impacts and effects of debris accumulation on containment emergency sump strainers in recirculation mode, as well as core flow blockage due to in-vessel effects, following loss of coolant accidents (LOCAs).

The proposed change does not implement any physical changes to the facility or any SSCs, and does not implement any changes in plant operation. The proposed change confirms that required SSCs supported by the containment sumps will perform their safety functions as required, and does not alter or prevent the ability of SSCs to perform their intended function to mitigate the consequences of an accident previously evaluated within the acceptance limits. The safety analysis acceptance criteria in the UFSAR continue to be met for the proposed change. The proposed change does not affect initiating events. The proposed change does not significantly affect the operation of the containment systems needs to ensure that there is a large margin between the temperature and pressure conditions reached in the containment and those that would lead to failure so that there is a high degree of confidence that damage of the containment cannot occur.

The calculated risk associated with the proposed change is very small and far less than the threshold for Region III as defined by RG 1.174, for both CDF and

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

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

The proposed change neither installs nor removes any plant equipment, nor alters the design, physical configuration, or mode of operation of any plant structure, system or component. The proposed change does not introduce any new failure mechanisms or malfunctions that can initiate an accident. The proposed change does not introduce failure modes, accident initiators, or equipment malfunctions that would cause a new or different kind of accident. Therefore, the proposed change does not create the possibility for a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not involve a change in any functional requirements, the configuration, or method of performing functions of plant SSCs. The effects from a full spectrum of LOCAs, including double-ended guillotine breaks for all piping sizes up to and including the largest pipe in the reactor coolant system, are analyzed. Appropriate redundancy and consideration of loss of offsite power and worst case single failure are retained, such that defense-in-depth is maintained.

The risk-informed method demonstrates the containment sumps will continue to support the ability of safety related components to perform their design functions. The proposed change does not alter the manner in which safety limits are determined or acceptance criteria associated with a safety limit. The proposed change does not implement any changes to plant operation, and does not significantly affect SSCs that respond to safely shutdown the plant and to maintain the plant in a safe shutdown condition. The proposed change does not significantly affect the existing safety margins in the barriers for the release of radioactivity. There are no changes to any of the safety analyses in the UFSAR. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

No physical modifications or changes to operating requirements are proposed for the facility, including any systems, structures and components relied upon to mitigate the consequences of a LOCA. Approval of the exemption requires the calculated risk associated with GSI-191 to meet the acceptance guidelines in RG 1.174, thereby maintaining public health and safety. Therefore there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

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

No physical modifications or changes to operating requirements are proposed for the facility, including any systems, structures and components relied upon to mitigate the consequences of a LOCA. Therefore, with respect to installation or use of a facility component located within the restricted area there is no significant increase in individual or cumulative occupational radiation exposure as a result of granting the exemption request.

Based on the above, STPNOC concludes that the proposed exemption meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Additional technical justification for this conclusion is provided on the basis that the guidance and acceptance criteria provided in RG 1.174 are satisfied as described in Enclosure 4-1.

6. Conclusion

Approval of an exemption to allow the use of the risk-informed approach is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security as required by § 50.12(a)(1). Furthermore, special circumstances required by § 50.12(a)(2) are present for item § 50.12(a)(2)(ii) in that application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule.

Based on the determination that the risk of the exemption meets the acceptance guidelines of RG 1.174, the results demonstrate there is reasonable assurance that the CSS will function in the recirculation mode and that the public health and safety will be protected.

7. Implementation

To support the completion of work and the resolution schedule for closure of GSI-191 as described in SECY-12-0093, STPNOC requests that this exemption request be approved for implementation by December 2014.

8. References

1) Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," September 13, 2004 (ML042360586)

2) Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance"

3) Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2 (ML100910006)

4) "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Nos. 183 and 170 to Facility Operating License Nos. NPF-76 and NPF-80 STP Nuclear Operating Company, et al., South Texas Project, Units 1 and 2, Docket Nos. 50-498 and 50-499," March 25, 2008 (ML080360321)

5) GSI-191 Safety Evaluation Report, Rev. 0, "Evaluation of NEI Guidance on PWR Sump Performance," December 6, 2004 (ML043280007)

6) NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Volume 1 "Pressurized Water Reactor Sump Performance Evaluation Methodology," Revision 0, December 2004 (ML050550138)

7) Federal Register Notice, Vol. 53, No. 180 (53 FR 35996), Emergency Core Cooling Systems; Revisions to Acceptance Criteria, September 16, 1988

8) Commission's Policy Statement on "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," August 16, 1995 (60 FR 42622)

9) South Texas Project, Units 1 and 2 - Issuance of Amendments Re: Adoption of Alternate Radiological Source Term in Assessment of Design-Basis Accident Dose Consequences (TAC Nos. MD4996 and MD4997), March 6, 2008 (ML080160013)

10) Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000

ENCLOSURE 2-4

Request for Exemption from

General Design Criterion 41

Request for Exemption from Certain Requirements of General Design Criterion 41

1. Exemption Request

Pursuant to 10 CFR 50.12, STPNOC is submitting this request for exemption from certain requirements of 10 CFR Part 50 Appendix A, General Design Criteria (GDC) 41.

> *Criterion 41— Containment atmosphere cleanup. Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.*
>
> *Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.*

Licensees are required to demonstrate this capability while assuming the most conservative (and worst) single failure. This regulation has been interpreted as requiring the use of a bounding calculation or other deterministic method, for the purpose of addressing containment emergency sump performance as discussed in Generic Letter 2004-02 (Reference 1) and Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance" (Reference 2). STPNOC requests an exemption from that requirement in order to enable the use of a risk-informed method to demonstrate acceptable sump design in support of the operation of the Containment Spray System (CSS) in recirculation mode following postulated LOCAs, as required to show compliance with GDC 41. This exemption request does not apply to other structures, systems and components (SSCs) which may be required to demonstrate compliance with GDC 41.

Approval of this exemption will allow use of a risk-informed method to account for the probabilities and uncertainties associated with the containment emergency sump design in support of the operation of the CSS following postulated LOCAs. The method evaluates the effects on containment emergency sump strainer blockage resulting from debris concerns raised by GSI-191. In order to confirm acceptable sump design, the risk associated with GSI-191 is evaluated to include the failure mechanisms associated with strainer blockage affecting CSS recirculation mode.

This exemption request is in support of the accompanying License Amendment Request (LAR) (Enclosure 3) seeking NRC approval of the changes to the South Texas Project Electric Generating Station (STPEGS) Updated Final Safety Analysis Report (UFSAR), to reconstitute the licensing basis based on acceptable design of the containment sump. The risk-informed method provides assurance, with high probability, for an acceptable

sump design that complies with GDC 41 and resolves GSI-191.

2. Regulatory Requirements Involved

STPNOC seeks exemption to the extent that GDC 41 requires deterministic calculations or other analyses to address the concerns raised by GSI-191 related to acceptable plant performance during the recirculation mode in containment following a LOCA. The proposed changes to the licensing basis, submitted for NRC approval with the LAR in Enclosure 3, provide closure to GSI-191 for STP Units 1 and 2 on the basis that the associated risk is shown to meet the acceptance guidelines in Regulatory Guide (RG) 1.174 (Reference 3) and that, in conjunction with the existing licensing basis, adequate safety is demonstrated.

This exemption request is for the purpose of allowing the use of a risk-informed method to demonstrate acceptable design of the containment emergency sump in support of the recirculation mode in containment following postulated loss of coolant accidents (LOCAs). The containment sump has been evaluated, using deterministic methods, to meet the current licensing basis assumptions for analyzing the effects of post-LOCA debris blockage; however, these evaluations have not been shown to fully resolve GSI-191 for the as-built, as-operated plant (Reference 4). The risk-informed method evaluates the sump design as part of the assessment of the residual risk associated with GSI-191 concerns. Based on confirmation of acceptable containment emergency sump and CSS design as determined by the resulting risk meeting the acceptance guidelines in RG 1.174, the licensing basis for CSS compliance with GDC 41 is reconstituted.

Other regulatory requirements associated with containment sumps in support of Emergency Core Cooling System (ECCS) and CSS recirculation modes following postulated LOCAs include 10 CFR 50.46(b)(5) for core cooling, GDC 35 for core cooling, and GDC 38 for containment heat removal. These requirements are addressed as part of separate exemption requests.

2.1 Evaluation of Impacts on 10 CFR 50.67 and GDC 19

The impact of the proposed exemption on the licensing basis analysis for demonstrating radiological consequences of the design basis LOCA meet the radiological dose guidelines specified in 10 CFR 50.67 and the dose limits specified in GDC 19 was evaluated. The risk-informed method provides confirmation of reliable ECCS and CSS performance as required for the licensing basis analyses that demonstrate the requirements of 10 CFR 50.67 and GDC 19. The method demonstrates that sump performance continues to support reliable plant design and operation and does not entail any exemption from 10 CFR 50.67 or GDC 19.

For STP Units 1 and 2, which have implemented the Alternative Source Term (AST), the design-basis LOCA radiological consequence LOCA analysis is a deterministic evaluation based on the assumption of a major rupture of the reactor coolant system piping and the assumption of the deterministic failure of the ECCS to provide adequate core cooling (Reference 9). This scenario results in a significant amount of core damage as specified in RG 1.183 (Reference 10), and does not represent any specific accident sequence, but is representative of a class of severe damage incidents that were evaluated in the development of the RG 1.183 source term characteristics. Such a

scenario would be expected to require multiple failures of systems and equipment and lies beyond the likely incidents evaluated for design-basis transient analyses (Reference 9). Since deterministic failure of ECCS is assumed at the onset of the accident by the analysis, the reliability of the containment emergency sumps with respect to ECCS operation does not affect the analysis for dose consequences. The risk-informed method confirms reliable CSS operation as an input to the AST analysis. Therefore, for the purposes of this exemption request, the current licensing basis analyses for 10 CFR 50.67 and GDC 19 are not impacted.

2.2 Evaluation of Impacts on other Regulatory Requirements

The STP risk-informed approach, as described in Enclosures 1 and 4, uses the PRA model to quantify the risk associated with GSI-191, thereby quantifying the residual risk from those issues which have not been resolved using other methods. A determination that this approach and its results meet the key principles and acceptance guidelines in RG 1.174 demonstrates acceptable sump performance in support of ECCS and CSS operation in recirculation mode following postulated LOCAs, and demonstrates that the Commission's safety goals and public health and safety are maintained.

The proposed exemption does not result in any physical changes to the facility or changes to the operation of the plant, and does not change any of the programmatic requirements. Based on demonstrating acceptable containment emergency sump and CSS design for reconstituting the current licensing basis for compliance with GDC 41 as described above, compliance with other regulatory requirements that rely on acceptable design for these systems and components continue to be met in the current licensing basis.

3. Basis for the Exemption Request

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

3.1 Justification for the Exemption Request

As required by 50.12(a)(2), the Commission will not consider granting an exemption unless special circumstances are present. Special circumstances are present whenever one of the listed items (i) through (vi) under 50.12(a)(2) are applicable. STPNOC has evaluated the proposed exemption against the conditions specified in 10 CFR 50.12(a) and determined that this proposed exemption meets the requirements for granting an exemption from the regulation, and that special circumstances are present. The information supporting the determination is provided below.

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

The exemption is authorized by law.

The NRC has authority under the Atomic Energy Act of 1954, as amended, to grant exemptions from its regulations if doing so would not violate the requirements of law. This exemption is authorized by law as is provided by 10 CFR 50.12 which provides the NRC authority to grant exemptions from Part 50 requirements with provision of proper justification. Approval of the exemption would not conflict with any provisions of the Atomic Energy Act of 1954, as amended, the Commission's regulations, or any other law

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

The proposed change is to apply a risk-informed method rather than a traditional deterministic method in order to quantify the residual risk associated with GSI-191 and to establish a high confidence of acceptable containment sump design. The purpose of GDC 41 is to establish acceptable design for the containment atmospheric cleanup system, which includes the CSS, to provide a high probability that the systems will perform the required functions. The proposed exemption does not involve any modifications to the plant that could introduce a new accident precursor or affect the probability of postulated accidents, and therefore the probability of postulated initiating events is not increased. The PRA and engineering analysis demonstrate that the calculated risk is small and consistent with the intent of the Commission's Safety Goal Policy Statement, which defines an acceptable level of risk that is a small fraction of other risks to which the public is exposed.

As discussed in previous § 50.46 rulemaking (Reference 7), the probability of a large break LOCA is sufficiently low that the application of a risk-informed approach to evaluate the ability of the ECCS to meet § 50.46(b)(5) with high probability and with low uncertainty, rather than using a calculational model using deterministic methods to achieve similar understanding, would have little effect on public risk. This is applicable to evaluating acceptable containment sump design in support of ECCS and CSS recirculation modes.

The risk-informed approach involves a complete evaluation of the spectrum of LOCA breaks, including double-ended guillotine break, up to and including the largest pipe in the reactor coolant system (see Enclosure 4-3, Volume 3 Section 5.3.6). The risk-informed approach analyzes LOCAs, regardless of break size, using the same methods, assumptions, and criteria in order to quantify the uncertainties and overall risk metrics. This ensures that large break LOCAs with relatively small contribution to CDF due to the low probability of such a break as well as smaller break LOCAs with higher probabilities of occurrence are considered in the results. Since the design basis requirement for consideration of a double-ended guillotine break of the largest pipe in the reactor coolant system is retained and since no physical changes to the facility or changes to the operation of the facility are being made, the existing defense-in-depth and safety margin established for the design of the facility is not reduced.

The exemption is consistent with the common defense and security.

The exemption involves a change to the licensing basis for the plant that has no relation to the possession of licensed material or any security requirements that apply to STP Units 1 and 2. Therefore the exemption is consistent with the common defense and security.

3.2 Special Circumstances

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

Such special circumstances are present in this instance to warrant exemption from the implicit requirement in GDC 41 to use a deterministic method to evaluate for acceptable containment emergency sump design. Approval of the exemption request would allow use of a risk-informed method to reconstitute the design basis for acceptable containment sump design in support of CSS design for compliance with GDC 41.

Specifically, § 50.12(a)(2)(ii) applies:

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

The intent of GDC 41 is to ensure systems in the plant design to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained. This exemption request is consistent with that purpose in that use of the proposed risk-informed approach demonstrates a high probability of successful containment emergency sump and CSS performance, which includes realistically available recirculation flow, based on the risk results meeting the acceptance guidelines of RG 1.174. The risk-informed approach assesses the design for a full spectrum of breaks, and assesses equipment failures that include loss of offsite power and worst case single failure, consistent with the GDC 41 requirements.

Since the proposed exemption does not involve any physical changes to the plant, there is no affect on the GDC 41 design requirements for redundancy in components and features, interconnections, leak detection, isolation, and containment capabilities. The current licensing basis evaluations for CSS compliance with GDC 41 for these aspects continue to be met.

As discussed in the *Commission's Policy Statement on "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities"* (Reference 8), NRC regulations and their implementation are generally based on deterministic approaches that consider a set of challenges to safety and determine how those challenges should be mitigated. This request does not seek exemption from any explicit language in the regulatory requirements. Rather, the need is based on the implicit requirements in the regulations for using deterministic methods to demonstrate acceptable design. Regulatory requirements are largely based on a deterministic framework, and are established for design basis accidents, such as the LOCA, with specific acceptance criteria that must be

satisfied. Licensed facilities must be provided with safety systems capable of preventing and mitigating the consequences of design basis accidents to protect public health and safety. The deterministic regulatory requirements were designed to ensure that these systems are highly reliable. The LOCA analysis and the General Design Criteria (GDC) were established as part of this deterministic regulatory framework.

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

An objective of GDC 41 is to maintain low risk to the public health and safety through a reliable CSS, as supported by the containment sump. The supporting analysis demonstrates that a risk-informed approach to sump performance is consistent with the Commission's Safety Goals for nuclear power plants, and supports CSS operation with a high degree of reliability. Consequently, the special circumstances described in § 50.12(a)(2)(ii) apply.

Specifically, § 50.12(a)(2)(iii) applies:

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

In order to meet a deterministic threshold value for containment debris loading, the amount of debris generating contributors in the STP plant design would need to be significantly reduced. Estimates of radiological exposure for insulation modifications are significant and on the order of hundreds of person-Rem, depending on the scope of the modifications.

With respect to the presence of such special circumstances, dose estimates for removal of insulation from STP Units 1 and 2 are described below. These dose estimates are for additional modifications to insulation in containment that would be required to achieve full resolution of GSI-191 using the previous deterministic methods. The residual risk associated with GSI-191 concerns bounds the expected improvement to overall plant risk that could be achieved by implementing these modifications.

<u>Dose Considerations: STP Units 1 and 2 Plant Dose Estimates for GSI-191 Insulation Removal</u>

STPNOC estimated the occupational dose for STP Units 1 and 2 that would be expected to be expended if plant modifications were undertaken for GSI-191, including insulation replacement and other modifications. The scope of the estimate included the replacement of fiberglass insulation with reflective metal insulation (RMI) for reactor coolant pump (RCP) insulation and a portion of the steam generator (SG) insulation, and the banding of existing fiberglass insulation on piping in containment. SG insulation replacement considered whether locations were within the zone of influence (ZOI) or beyond, with ZOI extending to seventeen times the diameter of the piping (17D).

The total dose expected to be expended for STP Units 1 and 2 in support of insulation replacement for GSI-191 is estimated to be 158 to 176 rem.

STP experience with this type of work suggests the lower expected dose estimates may be appropriate. However, bounding dose estimates based on the estimated installation hours including scaffold work and the average dose rates STP has historically experienced working within the bioshield are also provided below.

Insulation modifications, summary estimates:

Activities	Estimated Man-Hours	Estimated Dose per Unit (Expected)	Estimated Dose per Unit (Bounding)
Replace SG and RCP insulation with RMI	35,000	70 Rem	315 Rem
Install bands on insulated piping	9,000	9 to 18 Rem	81 Rem

Pipe insulation banding scope:

Piping Size (inches)	Insulation Thickness (inches)	Estimated Length (lineal feet)
31	3.5	120
29	4	68
27.5	3.5	76
16	4	160
12	2.5	430
8	2.5	168
6	2.5	94
4	2.5	592
2	2.5	228

SG fiberglass insulation replacement scope:

SG section	Volume per SG (cubic feet)	Total Volume – Four SGs (cubic feet)
Bottom end (<17D)	85	340
El 37' to El 52' (<17D)	197	788
El 52' to El 68' (<17D)	214	856
El 68' to transition (<17D)	17	68
SG transition (<17D)	134	536
Transition to El 83' (<17D)	53	212
Above El 83' (>17D)	305	1220
Top end (>17D)	150	600
Total within ZOI (<17D)	700	2800
Total beyond ZOI (>17D)	455	1820

RCP insulation replacement scope:

- All RCP fiberglass insulation (thickness 3.5 inches) to be replaced with RMI
- Fiberglass insulation volume per RCP: 56 cubic feet
- Total fiberglass insulation volume (4 RCPs): 224 cubic feet per Unit (448 cubic feet total)

For the above estimates, the highest dose contributor is personnel work hours in close proximity to high dose sources such as steam generators and primary coolant piping. The estimates did not include any replacement of reactor pressure vessel (RPV) insulation, which is RMI as originally designed for STP, therefore while the estimates may be indicative of a plant with high fiber loading, they do not necessarily account for activities that may be required for similar plants assuming 100-percent replacement of fibrous insulation in all areas that could be affected by a postulated LOCA. The dose estimates for STP Units 1 and 2, in addition to the actual insulation replacement, considered man-hours required to erect and remove scaffolding to support the insulation modifications and the dose associated with removal of insulation. The estimates did not consider dose associated with disposal of removed insulation or dose associated with potential hanger modifications for small bore piping insulation change to RMI.

The dose considerations discussed above demonstrate that compliance would result in substantial personnel exposure due to insulation modifications in the containment which is not commensurate with the expected safety benefit based on the results showing that the risk associated with GSI-191 concerns is far less than the threshold for Region III in RG 1.174. Consequently, the special circumstances described in § 50.12(a)(2)(iii) apply.

In conclusion, special circumstances in § 50.12(a)(2)(ii) and § 50.12(a)(2)(iii) are present as required by § 50.12(a)(2) for consideration of the request for exemption.

4. Technical Justification for the Exemption

The supporting information that describes the STP risk-informed approach is provided in the enclosures to this letter. The generic methodology for the risk-informed method is described in Enclosure 1. Technical justification for the risk-informed method is provided in Enclosures 4-1 through 4-3, and in the LAR (Enclosure 3), which includes descriptions of the CSS and containment emergency sump designs and performance criteria.

The proposed risk-informed approach meets the key principles in RG 1.174 in that it is consistent with the defense-in-depth philosophy, maintains sufficient safety margins, results in small increase in risk, and is monitored using performance measurement strategies. Detailed descriptions of the PRA and supporting engineering analyses are provided in Enclosures 4-2 and 4-3 to this letter. This proposed exemption to allow use of the risk-informed method is consistent with the key principle in RG 1.174 that requires the proposed change to meet current regulations unless explicated related to a requested exemption.

The results show that the risk associated with GSI-191 concerns is far less than the threshold in Region III, "Very Small Changes," of RG 1.174, and therefore are consistent with the Commission's Safety Goals for public health and safety.

5. Environmental Consideration

Pursuant to the requirements of § 51.41 and § 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments," the following information is provided. As demonstrated below, this exemption qualifies for a categorical exclusion in 10 CFR 51.22. However, if the NRC determines that an environmental assessment is necessary, this information will support a finding of no significant impact.

Identification of the Proposed Action

The proposed exemption is to allow for use of a risk-informed approach to evaluate the residual risk associated with GSI-191, i.e. those concerns that have not been fully addressed using deterministic methods, for the purpose of reconstituting the design basis for acceptable performance of the containment emergency sumps during recirculation mode following postulated LOCAs. Approval of the proposed exemption would allow for approval of a change to the UFSAR, as provided in Enclosure 3 to this letter, for implementation of the risk-informed method for STP Units 1 and 2.

Need for the Proposed Action

In the Commission's Policy Statement on "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities" (Reference 8), the Commission stated that "*the use of PRA technology in NRC regulatory activities should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach*" and that is consistent with traditional defense-in-depth concepts.

The intent of the Commission's Policy Statement intent is to use the PRA to further understand the risk associated with a proposed change for the purpose of removing unnecessary conservatism associated with regulatory requirements in order to focus attention and allocation of resources to areas of true safety significance.

To implement the Commission Policy Statement, the NRC issued RG 1.174 to provide guidance on an acceptable approach to risk-informed decision-making, based on a set of five key principles. The proposed action is needed to allow STPNOC to use a risk-informed method to resolve the issues associated with GSI-191 concerns regarding the potential for insulation and other debris generated in the event of a postulated LOCA within the containment impacting acceptable recirculation operation for the CSS, and challenge its ability to provide adequate containment atmosphere cleanup. This proposed exemption is consistent with the key principle in RG 1.174 which requires the proposed change to meet current regulations unless explicated related to a requested exemption.

Environmental Impacts Consideration

The proposed exemption has been evaluated and determined to result in no significant radiological environmental impacts associated with the implementation of the change. This conclusion is based on the following.

The proposed exemption is to allow a risk-informed method for demonstrating the design and licensing bases for the CSS. No physical modifications or changes to operating requirements are proposed for the site or facility, including any systems, structures and components relied upon to mitigate the consequences of a LOCA. The intent of the proposed change is to quantify the risk associated with GSI-191 concerns. This quantification, provided in the form of risk metrics using the guidance in RG 1.174, demonstrates that the risk is far less than the threshold for Region III, "Very Small Changes," in RG 1.174. Therefore, the proposed exemption supports a change that represents a very small change in Large Early Release Frequency (LERF) that corresponds to an insignificant impact on the environment.

Based on the results of the risk-informed method demonstrating that the increases in risk are very small, the proposed exemption has a negligible effect on accident probability, and adequate assurance of public health and safety is maintained. The proposed exemption does not involve any changes to the facility or facility operations that could create a new or significantly affect a previously analyzed accident or release path, and therefore would result in no significant changes in the types or quantities of radiological effluents that may be released offsite. The proposed change does not affect the generation of any radioactive effluents, and does not affect any of the permitted effluent release paths.

The proposed exemption has no impact on requirements related to the integrity of the reactor coolant system piping or any other aspect related to the initiation of a LOCA. No physical modifications or changes to operating requirements are proposed for the facility, including any systems, structures and components relied upon to mitigate the consequences of a LOCA. Therefore, the proposed exemption does not affect the probability of an accident initiator.

The proposed exemption does not significantly impact a release of radiological effluents during and following a postulated LOCA. The design-basis LOCA radiological consequence analysis in the current licensing basis is a deterministic evaluation based on the assumption of a major rupture of the reactor coolant system piping and a significant amount of core damage as specified in RG 1.183 (Reference 9). The current licensing basis analysis shows the resulting doses to the public and to control room and technical support center personnel do not exceed the regulatory limits. The proposed change validates and does not change the input parameter value used in the radiological analysis. Therefore, the proposed exemption does not affect the amount of radiation exposure resulting from a postulated LOCA.

The proposed exemption does not involve any changes to the site property, physical changes to the facility, or changes to the operation of the facility. Therefore there are no irreversible and irretrievable commitments of resources which would be involved in the proposed action should it be implemented. The risk-informed method requires a determination that the risk associated with the proposed change meets the Commission's safety goals. Therefore the proposed action would not result in a significant increase in any radiological hazard beyond those events previously analyzed in the UFSAR. There will be no change to radioactive effluents that affect radiation exposures to plant workers and members of the public. Therefore, no significant changes or different types of radiological impacts are expected as a result of the

proposed action. The proposed exemption does not change the input parameter value used in the radiological analysis. Therefore, the proposed change would not significantly increase the probability or consequences of an accident, and there will be no significant offsite impact to the public from approval of the proposed exemption.

No physical modifications or changes to operating requirements are proposed for the facility, including any systems, structures and components relied upon to mitigate the consequences of a LOCA. Therefore, the proposed exemption does not result in a significant increase in individual or cumulative occupational radiation exposure, and will not cause radiological exposure in excess of the dose criteria for restricted and unrestricted access specified in 10 CFR Part 20.

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

Since implementation of the exemption request, if approved, would result in no physical changes to the facility, there is no possibility of irreversible or irretrievable commitments of resources. Similarly, the proposed exemption does not involve the use of any resources not previously considered by the NRC in its past environmental statements for issuance of the facility operating licenses or other licensing actions for the facility. As a result, the proposed exemption does not involve any unresolved conflicts concerning alternative uses of available resources.

Alternatives

The alternative to this exemption is compliance with the existing provisions in GDC 41. Compliance with GDC 41 would entail removal and disposal of significant amounts of insulation and installation of new insulation less likely to impact sump performance in the event of a LOCA. As discussed below, the alternative would not be environmentally preferable or cost justified.

The exemption entails a very small risk and, correspondingly, a very small environmental impact. In particular, the change in LERF is ~ 8.6E-12/yr. This change is so small that it is remote and speculative for environmental purposes.

Removal and reinstallation of insulation would entail appreciable radiation exposures to workers (estimated from 158 to 176 rem). This option results in extensive modifications to the facility and significant occupational dose. As such, the alternative is not environmentally preferable. Additionally, the cost of the installation replacement would be approximately $55 million. This cost is not justified in light of the very small risk associated with the risk-informed exemption.

Categorical Exclusion Consideration

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

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

(i) *The exemption involves no significant hazards consideration.*

An evaluation of the three criteria set forth in 10 CFR 50.92(c) as applied to the exemption is provided below. The evaluation is consistent with the no significant hazards consideration determination provided in Enclosure 3 in support of the LAR.

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

Approval of the proposed exemption and accompanying license amendment request would allow the results of a risk-informed evaluation to be included in the UFSAR that concludes the ECCS and CSS will operate with a high probability following a LOCA when considering the impacts and effects of debris accumulation on containment emergency sump strainers in recirculation mode, as well as core flow blockage due to in-vessel effects, following loss of coolant accidents (LOCAs).

The proposed change does not implement any physical changes to the facility or any SSCs, and does not implement any changes in plant operation. The proposed change confirms that required SSCs supported by the containment sumps will perform their safety functions as required, and does not alter or prevent the ability of SSCs to perform their intended function to mitigate the consequences of an accident previously evaluated within the acceptance limits. The safety analysis acceptance criteria in the UFSAR continue to be met for the proposed change. The proposed change does not affect initiating events. The proposed change does not significantly affect the operation of the containment systems needs to ensure that there is a large margin between the temperature and pressure conditions reached in the containment and those that would lead to failure so that there is a high degree of confidence that damage of the containment cannot occur.

The calculated risk associated with the proposed change is very small and far less than the threshold for Region III as defined by RG 1.174, for both CDF and LERF. Therefore, the proposed change does not involve a significant increase

in the probability or consequences of any the accident previously evaluated in the UFSAR.

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

The proposed change neither installs nor removes any plant equipment, nor alters the design, physical configuration, or mode of operation of any plant structure, system or component. The proposed change does not introduce any new failure mechanisms or malfunctions that can initiate an accident. The proposed change does not introduce failure modes, accident initiators, or equipment malfunctions that would cause a new or different kind of accident. Therefore, the proposed change does not create the possibility for a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not involve a change in any functional requirements, the configuration, or method of performing functions of plant SSCs. The effects from a full spectrum of LOCAs, including double-ended guillotine breaks for all piping sizes up to and including the largest pipe in the reactor coolant system, are analyzed. Appropriate redundancy and consideration of loss of offsite power and worst case single failure are retained, such that defense-in-depth is maintained.

The risk-informed method demonstrates the containment sumps will continue to support the ability of safety related components to perform their design functions. The proposed change does not alter the manner in which safety limits are determined or acceptance criteria associated with a safety limit. The proposed change does not implement any changes to plant operation, and does not significantly affect SSCs that respond to safely shutdown the plant and to maintain the plant in a safe shutdown condition. The proposed change does not significantly affect the existing safety margins in the barriers for the release of radioactivity. There are no changes to any of the safety analyses in the UFSAR. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

No physical modifications or changes to operating requirements are proposed for the facility, including any systems, structures and components relied upon to mitigate the consequences of a LOCA. Approval of the exemption requires the calculated risk associated with GSI-191 to meet the acceptance guidelines in RG 1.174, thereby maintaining public health and safety. Therefore there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

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

No physical modifications or changes to operating requirements are proposed for the facility, including any systems, structures and components relied upon to mitigate the consequences of a LOCA. Therefore, with respect to installation or use of a facility component located within the restricted area there is no significant increase in individual or cumulative occupational radiation exposure as a result of granting the exemption request.

Based on the above, STPNOC concludes that the proposed exemption meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Additional technical justification for this conclusion is provided on the basis that the guidance and acceptance criteria provided in RG 1.174 are satisfied as described in Enclosure 4-1.

6. Conclusion

Approval of an exemption to allow the use of the risk-informed approach is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security as required by § 50.12(a)(1). Furthermore, special circumstances required by § 50.12(a)(2) are present for item § 50.12(a)(2)(ii) in that application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule.

Based on the determination that the risk of the exemption meets the acceptance guidelines of RG 1.174, the results demonstrate there is reasonable assurance that the CSS will function in the recirculation mode and that the public health and safety will be protected.

7. Implementation

To support the completion of work and the resolution schedule for closure of GSI-191 as described in SECY-12-0093, STPNOC requests that this exemption request be approved for implementation by December 2014.

8. References

1) Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," September 13, 2004 (ML042360586)

2) Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance"

3) Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2 (ML100910006)

4) "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Nos. 183 and 170 to Facility Operating License Nos. NPF-76 and NPF-80 STP Nuclear Operating Company, et al., South Texas Project, Units 1 and 2, Docket Nos. 50-498 and 50-499," March 25, 2008 (ML080360321)

5) GSI-191 Safety Evaluation Report, Rev. 0, "Evaluation of NEI Guidance on PWR Sump Performance," December 6, 2004 (ML043280007)

6) NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Volume 1 "Pressurized Water Reactor Sump Performance Evaluation Methodology," Revision 0, December 2004 (ML050550138)

7) Federal Register Notice, Vol. 53, No. 180 (53 FR 35996), Emergency Core Cooling Systems; Revisions to Acceptance Criteria, September 16, 1988

8) Commission's Policy Statement on "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," August 16, 1995 (60 FR 42622)

9) South Texas Project, Units 1 and 2 - Issuance of Amendments Re: Adoption of Alternate Radiological Source Term in Assessment of Design-Basis Accident Dose Consequences (TAC Nos. MD4996 and MD4997), March 6, 2008 (ML080160013)

10) Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000

ENCLOSURE 3

License Amendment Request for STP Piloted Risk-Informed Approach to Closure for GSI-191

License Amendment Request for STP Piloted Risk-Informed Approach to Closure for GSI-191

1. Summary Description

Pursuant to 10 CFR 50.90, STP Nuclear Operating Company (STPNOC) requests to amend Operating Licenses NPF-79 and NPF-80 for South Texas Project (STP) Units 1 and 2. The proposed amendment will revise the South Texas Project Electric Generating Station (STPEGS) Updated Final Safety Analysis Report (UFSAR) using a risk-informed approach to address safety issues discussed in Generic Safety Issue (GSI) -191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance" (Reference 1).

GSI-191 concerns include the effects on long-term cooling due to debris accumulation on Emergency Core Cooling System (ECCS) and Containment Spray System (CSS) sump strainers in recirculation mode, as well as core flow blockage due to in-vessel effects, following loss of coolant accidents (LOCAs). Since sump performance and in-vessel effects are important considerations in the safety analysis and are not explicitly described in the STP current licensing basis, the proposed amendment incorporates a risk-informed method used to demonstrate that sump performance and in-vessel effects are acceptable so that the current licensing basis safety analysis conclusions remain valid.

The risk-informed approach is consistent with the guidance of NRC Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 2).

The proposed change is to use a Probabilistic Risk Assessment (PRA) model that is risk-informed by an engineering analysis (CASA Grande) to evaluate the containment emergency sump performance. The analysis covers a full spectrum of postulated LOCAs, including double-ended guillotine breaks (DEGBs), for all pipe sizes up to and including the design basis accident (DBA) LOCA, to provide assurance that the most severe postulated loss-of-coolant accidents are evaluated. The evaluation includes single failure criteria as well as multiple failures of systems and equipment. Success criteria are based on maintaining available net positive suction head (NPSH) and strainer structural integrity in support of ECCS and CSS operation in recirculation mode following postulated LOCAs and adequate in-vessel core flow, considering the debris loading associated with GSI-191 concerns. The risk-informed method is used to generate risk metrics for GSI-191 for comparison to the acceptance guidelines in RG 1.174.

The proposed change to the UFSAR implements a risk-informed rather than a deterministic method to demonstrate compliance. STPNOC is also requesting exemptions from certain regulatory requirements as provided in Enclosures 2-1 through 2-4 to this letter. Approval of the proposed amendment is contingent on approval of the exemption requests.

2. Detailed Description

The proposed amendment will add Appendix 6A, "Risk-Informed Approach to Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents," to the STP UFSAR. This appendix describes the evaluations performed using a risk-informed

approach to address GSI-191 concerns including the effects on long-term cooling due to debris accumulation on containment sump strainers for ECCS and CSS in recirculation mode, as well as core flow blockage due to in-vessel effects, following loss of coolant accidents (LOCAs).

Modifications Previously Implemented to Address GSI-191

Appendix 6A discusses the ECCS sump performance evaluations performed to address GSI-191. These evaluations account for previously implemented hardware modifications and plant procedures and processes to provide high confidence that the sump design supports long-term core cooling following a design basis loss of coolant accident (LOCA). Those design modifications and procedure changes were implemented in accordance with 10 CFR 50.59 and are not part of this license amendment request.

Installation of New Sump Strainer Assemblies

To address concerns associated with GSI-191, new ECCS containment sump strainer assemblies were installed in STP Unit 1 in October 2006 and in STP Unit 2 in April 2007. The strainer upgrade was achieved at a cost on the order of $6.3M. The surface area of the strainers was increased from approximately 155 square feet per sump to approximately 1818 square feet per sump. The screen-hole size of the strainers was reduced from 0.25 inches diameter to 0.095 inches diameter. Small particles in water entering the suction pipe from the sump cannot clog the containment spray nozzles (3/8-inch orifice diameter). Installation of the new strainers did not affect the independence and redundancy of the sumps.

The sump strainer design implemented by these modifications meets the current design basis requirements with respect to net positive suction head (NPSH) and ECCS performance. The sumps are designed according to RG 1.82 proposed Revision 1, dated May 1983 (Reference 3), which recommends a calculation of sump screen head loss due to debris blockage. Utilizing the current licensing basis methodology (Reference 3), the pump NPSH is sufficient to accommodate this head loss. The STP sumps meet the function to preclude passage of debris particles large enough to damage downstream components in the ECCS and CSS. Detailed evaluations and descriptions of the design are provided in References 4 and 5. The sump strainer design has been evaluated to meet the current design basis assumptions for analyzing the effects of post-accident debris blockage and for compliance with 10 CFR 50.46 for long term cooling, General Design Criterion (GDC) 35 for emergency core cooling, GDC 36 for inspection of ECCS, GDC 38 for containment heat removal, GDC 39 for inspection of containment heat removal system, and GDC 41 for containment atmosphere cleanup (References 4, 5 and 6).

The strainer upgrade increased the capacity for fibrous debris loading per unit area by nearly a factor of 12. Due to the larger surface area, vulnerability to debris penetration for small debris loads that do not fully cover the surface of the new design was increased. However, these modifications resulted in a net safety improvement that eliminated more than an estimated 99% of the number of cases where the ECCS was predicted to fail due to debris loading on the sump screens. The estimated 99% reduction is based on the number of cases predicted to fail decreasing by a factor on the order of 100 to 800 times, and represents a substantial safety improvement. This safety improvement contributed to the

desired shift from RG 1.174 Region II, "Small Changes," to Region III, "Very Small Changes," for the results from the risk-informed method discussed in the next section.

The current licensing basis for the sumps is based on a deterministic methodology that was used to analyze susceptibility of the ECCS and CSS recirculation functions for adverse effects of post-accident debris (Reference 7). The methodology was largely in accordance with Nuclear Energy Institute report NEI 04-07 (Reference 8) and the associated NRC Safety Evaluation Report (SER) (Reference 9). In addition, evaluations were performed in accordance with WCAP-16406-P-A, "Evaluation of Downstream Sump Debris Effects in Support of GSI- 191," Revision 1 (Reference 10) and WCAP-16793-NP," Evaluation of Long-Term Cooling Considering Particulate and Chemical Debris in the Recirculating Fluid" (Reference 11) to consider the effects that debris carried downstream of the containment sump screen and into the reactor vessel has on core cooling, including fuel and vessel blockage. STPNOC also evaluated the type and expected quantity of chemical products that would be expected to form in the recirculation fluid specifically for STP using the methodology developed in WCAP-16530-NP, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," (Reference 16). To support the sump performance evaluation, STPNOC performed containment walkdowns using the guidance of NEI 02-01 (Reference 12).

However, these deterministic evaluations have not been shown to demonstrate that the new strainer design fully resolves GSI-191 concerns for the as-built, as-operated plant.

Following the installations of the new sump strainers, protective gratings were installed in front of the strainers to preclude inadvertent damage to these components. The framing structure for the protective gratings consists of vertical grating panels attached to metal columns that are welded to base plates that are anchored into the concrete floor, and is qualified for Seismic II/I loading to ensure the maximum stresses are below the allowable limits. The structure is made of carbon steel and has a qualified coating applied.

Replacement of Fibrous Insulation

Per the original plant design, the reactor vessel nozzles were insulated with a non-crush insulation material composed of calcium silicate (brand name Marinite). Based on concerns related to the potential for calcium silicate to combine with the tri-sodium phosphate (TSP) pH buffer during post-LOCA conditions to form calcium phosphate precipitates which could block the strainers, the adequacy of the design and validation testing of the emergency sump strainers was questioned. Marinite insulation was identified as a significant contributor to the debris loading associated with one of the worst case LOCA scenarios for strainer head loss based on previous evaluations. As a result, all of the Marinite insulation was replaced with NUKON fiberglass insulation. These modifications were accomplished during the Fall 2009 refueling outage for STP Unit 1 and the Spring 2010 refueling outage for Unit 2.

Additional replacement of the fibrous insulation in containment required to fully resolve GSI-191 based on the deterministic evaluations was estimated to cost approximately $55M with a dose estimate of approximately 88 Rem per unit. Enclosure 2 to this letter provides details and discussion on the estimated dose considerations for these insulation modifications. Given the very low risk associated with the impact of this insulation on sump performance,

STPNOC decided to pursue a risk-informed resolution to GSI-191 rather than incur the economic and radiological costs associated with replacement of the fibrous insulation.

Use of a Risk-Informed Approach to Resolving GSI-191

The risk associated with GSI-191 issues has been quantified and is far less than the threshold for "Very Small Changes" in Region III as defined in RG 1.174. Appendix 6A describes the risk-informed approach used to confirm that the ECCS and CSS will operate with a high probability following a LOCA when considering the impacts and effects described by GSI-191. Therefore no further physical modifications to STP Units 1 and 2 are proposed as part of this license amendment request to implement the risk-informed approach.

Attachment 1 to this Enclosure provides the Licensee Commitment to implement the proposed amendment following approval and to revise affected sections of the UFSAR as indicated by the proposed changes in Attachment 2 to this Enclosure. Upon approval of the proposed amendment, applicable UFSAR safety system and design bases descriptions that take credit for the evaluation described in Appendix 6A will be revised. In addition, conforming changes to the Technical Specification (TS) Bases are provided in Attachment 3 to this Enclosure for information only, to be implemented following NRC approval of the LAR.

The design and licensing basis descriptions of accidents requiring ECCS operation, including analysis methods, assumptions, and results provided in UFSAR 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 far less than the Region III acceptance guidelines defined by RG 1.174.

The performance evaluations for accidents requiring ECCS operation described in Chapters 6 and 15, based on the South Texas Project Units 1 and 2 Appendix K Large-Break Loss-of-Coolant Accident (LBLOCA) analysis, demonstrate that for breaks up to and including the double-ended guillotine break of a reactor coolant pipe, the ECCS will limit the clad temperature to below the limit specified in § 50.46, and assure that the core will remain in place and substantially intact with its essential heat transfer geometry preserved.

The proposed amendment does not require a change to the STP TS. The basis for this conclusion is found in the Regulatory Evaluation Section of this LAR. Specific values and limits contained in the TS and the response times for the safety system assumed in the accident analyses are not changed.

System redundancy, independence, and diversity features are not changed for those safety systems credited in the accident analyses. No new programmatic compensatory activities or reliance on manual operator actions is required to implement this change.

3.0 Technical Evaluation

3.1 Current System Descriptions

Emergency Core Cooling System

The Emergency Core Cooling System (ECCS) is designed to cool the reactor core and provide shutdown capability subsequent to the following accident conditions:

1. Pipe breaks in the Reactor Coolant System (RCS) which cause a discharge larger than that which can be made up by the normal makeup system, up to and including the instantaneous circumferential rupture of the largest pipe in the RCS.

2. Rupture of a control rod drive mechanism (CRDM) causing a rod cluster control assembly (RCCA) ejection accident.

3. Pipe breaks in the steam system, up to and including the instantaneous circumferential rupture of the largest pipe in the steam system.

4. A steam generator tube rupture.

The primary function of the ECCS is to remove the stored and fission product decay heat from the reactor core and to provide shutdown capability during accident conditions.

The ECCS provides shutdown capability for the accidents above by means of boron injection. It is designed to tolerate a single active failure in the short term or a single active or passive failure in the long term. The system meets its minimum required performance level with onsite or offsite electrical power.

The ECCS consists of the high head safety injection (HHSI) and low head safety injection (LHSI) pumps, Safety Injection System (SIS) accumulators, residual heat removal (RHR) heat exchangers (HXs), the refueling water storage tank (RWST) along with the associated piping, valves, instrumentation, and other related equipment.

The design bases for selecting the functional requirements of the ECCS, such as peak fuel cladding temperature, etc., are derived from Appendix K limits as delineated in 10CFR50.46. The subsystem functional parameters are integrated so that the Appendix K requirements are met over the range of anticipated accidents and single failure assumptions.

Reliability of the ECCS has been considered in selection of the functional requirements, selections of the particular components, and location of components and connected piping. Redundant components are provided where the loss of one component would impair reliability. Valves are provided in series where isolation is desired. Redundant sources of the ECCS actuation signal are available so that the proper and timely operation of the ECCS is not inhibited. Sufficient instrumentation is available so that a failure of an instrument does not impair readiness of the system. The active components of the ECCS are powered from separate buses which are energized from offsite power supplies.

In addition, the standby diesel generators assure that adequate redundant sources of auxiliary onsite power are available to meet all ECCS power requirements. Each diesel is capable of driving all pumps, valves, and necessary instruments associated with one train of the ECCS.

The elevated temperature of the sump solution during recirculation is well within the design temperature of all ECCS components. In addition, consideration has been given to the potential for corrosion of various types of metals exposed to the fluid conditions prevalent immediately after the accident or during long-term recirculation operations.

10 CFR 50.46(b) provides the following criteria to judge the adequacy of the ECCS.

1. Peak clad temperature calculated shall not exceed 2,200°F.

2. The calculated total oxidation of the clad shall nowhere exceed 0.17 times the total clad thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the clad with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the clad cylinders surrounding the fuel, excluding the clad around the plenum volume, were to react.

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by long-lived radioactivity remaining in the core.

Containment Heat Removal System (CHRS)

The CHRS consist of the Reactor Containment Fan Cooler (RCFC) Subsystem, which is a part of the Reactor Containment HVAC System, and the Containment Spray System (CSS). The ECCS assists the CHRS by transferring heat from the reactor core to the Containment sump. The Residual Heat Removal (RHR) heat exchangers, in conjunction with the ECCS low-head Safety Injection (SI) pumps, are used to transfer heat from the Containment sumps to the Component Cooling Water System (CCWS). The RCFCs are also cooled by the CCWS following an SI signal. The CCWS rejects this heat to the ultimate heat sink via the Essential Cooling Water System (ECWS).

The CSS transfers heat from the Containment atmosphere to the Containment sump; it also removes iodine from the Containment atmosphere.

The CHRS meets the following design bases:

1. The CHRS is capable of removing sufficient energy to limit the peak Containment pressure and to limit the Containment pressure to a low value at the end of 24 hours after a DBA.

2. In order to ensure the satisfactory operation of the systems after a DBA, each active component is testable during reactor power operation.

3. The system is divided into three trains, with each train receiving power from a separate Emergency Safety Features (ESF) power supply.

The CSS is:

1. Designed such that it will tolerate a single active failure.

2. Designed to accommodate the operating basis earthquake (OBE) within stress limits of applicable codes and to withstand the safe shutdown earthquake (SSE) without loss of function.

3. Designed to assist in reducing offsite exposures resulting from a design basis accident (DBA) to less than the limits of 10 CFR 100 by rapidly reducing the airborne elemental iodine and particulate concentrations in the Containment following a DBA.

Containment Emergency Sump

The Containment emergency sump meets the following design bases:

1. Sufficient capacity and redundancy to satisfy the single-failure criteria. To achieve this, each CSS/ECCS train draws water from a separate Containment emergency sump.

2. Capable of satisfying the flow and net positive suction head (NPSH) requirements of the ECCS and the CSS under the most adverse combination of credible occurrences. This includes minimizing the possibility of vortexing in the sump.

3. Minimizes entry of high-density particles (specific gravity of 1.05 or more) or floating debris into the sump and recirculating lines.

4. Sumps are designed in accordance with RG 1.82, proposed revision 1, May 1983.

Three independent sumps serve as reservoirs to the ECCS and CSS pumps during the recirculation phase post-DBA. Each sump is stainless steel lined, contains a Vortex Suppressor, and is provided with four stainless steel strainer assemblies. The strainer assemblies for each sump consist of two 5-module assemblies, one 4-module assembly, and one 6-module assembly with each module made up of eleven strainer discs. The strainer screen consists of perforated plate with nominal 0.095 inch diameter openings. Flow leaving the strainer enters a four inlet plenum box (one inlet for each strainer assembly). The plenum box collects the flow from the strainer assemblies and directs the flow vertically downward directly into the sump pit. An access cover is provided on the plenum box for internal inspections of the sump structures, vortex suppressor, and the strainer assemblies.

The sumps are located at the -11 feet-3 inch level of Reactor Containment Building (RCB). The sumps are physically separated from each other with no high-energy piping in the area. The floor around the emergency sumps slopes away from them and toward normal sumps

located in the area. The drains from the upper levels of the RCB do not terminate in the immediate area of the sumps.

The sump structures are designed to withstand the SSE without loss of structural integrity. Water entering the suction pipe from the sump may contain a small amount of particulate and fibrous debris (less than 0.095-inch in diameter). This debris cannot clog the spray nozzles (3/8-in. orifice diameter) which are the limiting restrictions in the CSS system served by the sump.

At the beginning of the recirculation phase, the minimum water level above the RCB floor is adequate to provide the required NPSH for the ECCS and CSS pumps.

The sumps are designed to NRC RG 1.82, proposed revision 1, May 1983. The sump structures are designed to limit approach flow velocities to less than 0.009 feet/second permitting high-density particles to settle out on the floor and minimize the possibility of clogging the strainers. The sump structures are designed to withstand the maximum expected differential pressure imposed by the accumulation of debris.

Most potential sources of debris are remote from the emergency sumps and are separated by shield walls or other partitions. Expected debris is pieces of insulation and paint. The possibility of paint chips peeling off has been reduced by requiring proper surface preparation and by painting large surface components (such as: the Containment liner, RCS supports, floors, and structural steel) with coatings which have been qualified under DBA conditions.

The major insulation types used in the RCB are stainless steel reflective, blanket fiberglass, and cellular glass. The stainless steel reflective insulation is used on the major NSSS components. The blanket fiberglass type is used on the hot piping and equipment. Cellular glass insulation is used on cold piping for antisweat purposes. Microtherm is also used for piping in the wall penetrations.

Containment emergency sumps are inspected periodically as delineated in the Technical Specifications, as described in Section 4.1.3 below.

The current design for the containment sumps and strainers was assessed in response to NRC Generic Letter 2004-02 (Reference 7), and STPNOC provided specific information regarding the deterministic methodology for demonstrating compliance, applying industry and NRC guidance. However, since this methodology has not been demonstrated to fully resolve GSI-191 without the need for additional changes to the plant design, such as extensive modifications to insulation in the containment, a risk-informed approach is applied to evaluate acceptable sump design using the guidance in RG 1.174.

3.2 Background

GSI-191 concerns the possibility that debris generated during a LOCA could clog the containment sump strainers in pressurized-water reactors (PWRs) and result in loss of net positive suction head (NPSH) for the ECCS and CSS pumps, impeding the flow of water from the sump. GL 2004-02 requested licensees to address GSI-191 issues, focused on demonstrating compliance with the ECCS acceptance criteria in § 50.46. GL 2004-02

requested licensees to perform new, more realistic analyses using an NRC-approved methodology and to confirm the functionality of the ECCS and CSS during design basis accidents that require containment sump recirculation.

The STP piloted risk-informed approach maintains the defense-in-depth measures in place to mitigate the residual risk of strainer or in-vessel issues to provide closure for GSI-191. These measures include those implemented in response to NRC Bulletin 2003-01 and GL 2004-02 to address the potential for sump strainer clogging and other concerns associated with GSI-191. Additional measures such as operating procedures and instrumentation to monitor core level and temperature, and actions taken by operators if core blockage is indicated, have been implemented. These actions have been implemented pursuant to 10 CFR 50.59 and are not the subject of this license amendment request. Detailed discussion regarding defense-in-depth is provided in Enclosure 4-1 (Section 2 and Appendix C) to this letter. These measures are part of the defense-in-depth for STP and remain in place.

In addition, larger containment sump strainers have been installed that greatly reduce the potential for loss of net positive suction head (NPSH).

The Commission issued Staff Requirements Memorandum (SRM)-SECY-10-0113, "Closure Options for Generic Safety Issue (GSI) - 191, Assessment of Debris Accumulation on Pressurized Water Reactor (PWR) Sump Performance," directing the staff to consider alternative options for resolving GSI-191 that are innovative and creative, as well as risk-informed and safety conscious. Subsequently, STPNOC, through interactions with the staff, developed a risk-informed approach for the resolution of GSI-191 based on the guidance in RG 1.174. By Reference 13, STPNOC submitted to the NRC the preliminary results showing that the risks, Core Damage Frequency (CDF) and Large Early Release Frequency (LERF), associated with GSI-191 concerns are far less than the threshold for Region III, "Very Small Changes," of RG 1.174, and notified the NRC of the intent to seek exemption from certain requirements of § 50.46. SECY-12-0093, "Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance," described the staff plans to use STPNOC as a pilot for other licensees choosing to use this approach (Reference 14). The exemption request from certain requirements of § 50.46 including impacted General Design Criteria of Appendix A of Part 50 is provided in Enclosure 2 to this letter.

3.3 Evaluation

3.3.1 Engineering Analysis Overview

The design and licensing basis descriptions of accidents requiring ECCS and CSS operation, including analysis methods, assumptions, and results provided in UFSAR 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 far less than the Region III acceptance guidelines defined by RG 1.174.

The performance evaluations for accidents requiring ECCS operation described in Chapters 6 and 15, based on the South Texas Project Units 1 and 2 Appendix K Large-Break Loss-of-Coolant Accident (LBLOCA) analysis, demonstrate that for breaks up to and including the

double-ended guillotine break of a reactor coolant pipe, the ECCS will limit the clad temperature to below the limit specified in § 50.46, and assure that the core will remain in place and substantially intact with its essential heat transfer geometry preserved.

The plant-specific Probabilistic Risk Assessment (PRA) model was used to calculate the difference in risk (delta risk) between the actual plant configuration subject to the concerns raised by GSI-191 and a hypothetical plant configuration not subject to GSI-191, but otherwise identical. The difference in risk is a quantification of the risk associated with GSI-191 concerns. This risk includes the effects on long-term cooling due to debris accumulation on the ECCS and CSS containment sump strainers and the in-vessel effects following LOCAs that require recirculation flow from the containment sump to mitigate the event. The quantification of the risk associated with GSI-191 concerns conservatively defines the change to be evaluated, as discussed in RG 1.174,

The methodology for calculating the risk associated with GSI-191 concerns using the probabilistic risk assessment (PRA) model considered a full spectrum of postulated LOCAs, including double-ended guillotine breaks (DEGBs), for all pipe sizes up to and including the design basis accident (DBA) LOCA. The physical processes are modeled as realistically as possible, using results from industry and plant-specific testing, and applying conservatism, where appropriate. The risk due to GSI-191 concerns is then shown to meet RG 1.174 acceptance guidelines for changes to Core Damage Frequency (CDF) and Large Early Release Frequency (LERF).

RG 1.200 Revision 2 provides new guidance for certain external events but contains the same guidance as RG 1.200 Revision 1 for internal events. STPNOC's PRA is compliant with RG 1.200 Revision 1 for internal events, and therefore is compliant with RG 1.200 Revision 2 for assessing the risk of internal events associated with GSI-191. For purposes of GSI-191, only internal events are of concern because GSI-191 involves LOCAs. Enclosure 4-2 to this letter describes the STP PRA model used for the risk-informed approach and provides additional justifications for the applicability of the STP PRA model to this approach.

Using a risk-informed approach to address the concerns of GSI-191, the probability that the long-term cooling criterion would not be exceeded for the calculated ECCS cooling performance is quantified. The method of analysis for the risk-informed approach uses an integrative approach to explicitly provide the probabilities for post-LOCA events. This is accomplished by modeling the underlying physical phenomena of the basic events and by propagating uncertainties in the physical models.

To determine the risk associated with GSI-191 concerns, under the framework of RG 1.174, the STP piloted risk-informed approach to closure for GSI-191 applies the plant-specific PRA model to calculate the difference in risk (delta risk) for two cases:

- Case 1: the actual plant configuration, risk informed to model the failure mechanism associated with the concerns raised by GSI-191, and
- Case 2: a hypothetical plant assuming no failure mechanisms associated with the concerns raised by GSI-191, otherwise identical to the actual plant.

The risk associated with GSI-191 concerns is the difference in risk (CDF and LERF) between Case 1 and Case 2 for comparison with the acceptance guidelines in RG 1.174, Section 2.4.

The inputs to the risk model encompass the concerns raised in GSI-191, including the major topical areas discussed in NEI 04-07 (Reference 8), as appropriate:

- pipe break characterization
- debris generation/zone of influence (ZOI), including latent debris
- debris transport
- chemical effects
- strainer head loss, including structural margin
- air intrusion
- debris penetration
- ex-vessel downstream effects
- in-vessel downstream effects
- boron precipitation

For each input to the risk model, any differences between the methods to be used in the model and NRC-approved methods (e.g. Reference 9) are defined. To apply the inputs, the demand recirculation failure probability in the plant-specific PRA model is replaced with the following basic events:

- Pressure drop due to debris build-up on the sump strainers with chemical effects resulting in loss of net positive suction head (NPSH) margin for pumps.
- Strainer mechanical collapse, where P-buckle is the strainer structural design limit for the differential pressure (DP) across the ECCS strainers at which they are analyzed to be within code design allowable stresses.
- Air ingestion through the sump strainers, where F-void is the vapor fraction of the liquid just downstream of the ECCS strainers.
- Core blockage with chemical effects.
- Boron precipitation in the core.

Failure modes leading to core damage are explicitly modeled, excluding those that were previously addressed for the plant using deterministic evaluations.

Failure probabilities and associated uncertainties determined in the supporting engineering analysis provided inputs to the three new top events added to the PRA to accommodate composite GSI-191 failure processes, and the outcome of a full spectrum of LOCA events was tested against appropriate performance thresholds for the top events, as shown below. The bases for these limits are described in Enclosure 4-1 and Enclosure 4-3.

New Top Events	Performance Thresholds
1. Failure at sump strainers	1. Strainer DP ≥ NPSH margin 2. Strainer DP ≥ P-buckle 3. Strainer F-void ≥ 0.02
2. Boron precipitation in the core	4. Core fiber load ≥ cold leg break fiber limit for boron precipitation 5. Core fiber load ≥ hot leg break fiber limit for boron precipitation
3. Core flow blockage	6. Core fiber load ≥ cold leg break fiber limit for flow blockage 7. Core fiber load ≥ hot leg break fiber limit for flow blockage

The results of the risk-informed approach are as follows (from Enclosure 4-2, Volume 2 Section 4):

- Change in CDF is ~ 1.1E-8/yr
- Change in LERF is ~ 8.6E-12/yr

The changes in CDF and LERF are derived by comparing the results shown in Table 4-2, "Comparison of Core Damage Frequency and Large Early Release Frequency (mean values/year)," from Enclosure 4-2 Volume 2. The GSI-191 PRA base case (without GSI-191 phenomena) for CDF is 9.20E-6/yr and for LERF is 5.78E-07/yr, and the GSI-191 PRA (with GSI-191 phenomena) for CDF is 9.21E-6/yr and for LERF is 5.78E-07/yr. The GSI-191 PRA base case (without GSI-191 phenomena) represents a hypothetical STP plant with all fibrous insulation removed.

The results demonstrate that the calculated risk associated with GSI-191 concerns for STP Units 1 and 2 are very small and are far less than the threshold for Region III in RG 1.174. The functionality of the ECCS and CSS during design basis accidents is confirmed.

3.3.2 Evaluation of Defense-in Depth and Safety Margin

Defense-in-Depth (DID) Analysis

The proposed change is consistent with the DID philosophy in that the following aspects of the facility design and operation are unaffected:

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

The proposed amendment does not involve a change in any functional requirements or the configuration of plant structures, systems and components (SSCs).

The proposed amendment uses a risk-informed approach that analyzes a full spectrum of LOCAs, including double-ended guillotine breaks for all piping sizes up to and including the largest pipe in the reactor coolant system (RCS). By requiring that mitigative capability be maintained in a realistic and risk-informed evaluation of GSI-191 for a full spectrum of LOCAs, the approach ensures that defense-in-depth is maintained.

Appendix C of Enclosure 4-1 provides a more detailed description of the defense-in-depth measures that address potential sump blockage and in-core effects, including the means available to operators for detecting and mitigating inadequate recirculation flow and inadequate core cooling flow. The proposed change does not involve a change in any functional requirements, the configuration, or method of performing functions of plant SSCs. The effects from a full spectrum of LOCAs, including double-ended guillotine breaks for all piping sizes up to and including the largest pipe in the reactor coolant system, are analyzed. Appropriate redundancy and consideration of loss of offsite power and worst case single failure are retained. This approach ensures that DID is maintained.

The analysis shows that DID is maintained. The SSCs supporting DID are unchanged. The analysis shows there is no appreciable risk to containment integrity associated with the concerns raised in GSI-191. The as-built, as-operated containment design remains adequate to prevent a significant release into the environment. In quantification of the risk, no credit is taken for additional operator actions or programmatic activities beyond the existing as-built, as-operated plant.

Safety Margin Analysis

Approval of the proposed change would add the results of a risk-informed evaluation to the UFSAR that concludes the containment sumps will operate with a high probability in support of ECCS and CSS recirculation modes following a LOCA when considering the impacts and effects of debris on sump strainers, as well as core flow blockage due to in-vessel effects.

The proposed change does not involve a change in any functional requirements, the configuration, or method of performing functions of plant SSCs. The effects from a full spectrum of LOCAs, including double-ended guillotine breaks for all piping sizes up to and including the largest pipe in the reactor coolant system, are analyzed.

The proposed change does not result in any changes to the safety analyses demonstrating safety margin for the barriers to the release of radioactivity as described in the UFSAR. Appendix C of Enclosure 4-1 provides a more detailed discussion on how sufficient safety margins associated with the design are maintained by the proposed change.

3.3.3 Technical Adequacy of the PRA

The STPNOC PRA is a full-scope integrated Level I and Level II PRA that includes internal and external events with the focus of the GSI-191 concerns related to LOCA. Enclosure 4-2 to this letter provides a description of the PRA model. The STPNOC PRA and the engineering analysis supporting the GSI-191 analysis is representative of the as-built, as-

operated plant. The STPNOC PRA is reviewed for compliance/ adherence with the plant design and plant data review every 36 months as a UFSAR Chapter 13.7 commitment required for PRA applications.

The PRA is not significantly changed to specifically address the GSI-191 concerns. Instead, a detailed engineering analysis is performed in an uncertainty quantification framework that evaluates the required failure modes of ECCS and core cooling (in-vessel effects). Significant detail is included in the engineering analysis used to develop the required new basic events and top events. Details include physical models and mechanisms known to lead to failure, and the analyses account for experimental evidence used to support particular areas of concern.

RG 1.200 Revision 2 provides new guidance for certain external events. STPNOC's PRA is compliant with RG 1.200, Revision 1 for internal events, and therefore is compliant with Regulatory Guide 1.200, Revision 2 for assessing the risk of internal events associated with GSI-191. The methodologies, applications, and results derived from the STPNOC PRA are reviewed by peers in benchmarking and other activities and are also regularly published in the open literature and symposia.

3.3.4 Treatment of Uncertainties

The PRA used in the proposed amendment does not rely solely on numerical results for change in risk. The uncertainties in the models and the in models' parameters have been accounted for in the results.

Qualitative conservatisms are included along with the quantifiable uncertainties to increase confidence in the adequacy of the results. The STPNOC PRA analysis includes uncertainties that have been postulated in deterministic analyses for the following concerns related to GSI-191:

- Zone-of-influence (ZOI);
- Chemical effects;
- Debris transport;
- Head loss;
- Boric acid precipitation; and
- Air ingestion to ECCS pumps.

In some cases, the uncertainties have been addressed through well-known conservative approximations. In other cases, specific experimentation has been performed to analyze the impact of the phenomena on plant performance in response to LOCA. The treatment of uncertainties and the methods of analysis are described in more detail in the Attachment to this letter with references to appropriate sections in Enclosure 4-3.

Types of uncertainties and methods of analysis

Both aleatory and epistemic uncertainties have been included in the STPNOC PRA. Uncertainties have also been addressed using conservative assumptions where appropriate or where large uncertainties are seen. For example, the assumption of a larger ZOI results in scenarios that are conservative.

The engineering analysis used to develop the basic event failure probabilities for the PRA uses an approach that optionally integrates all uncertainty or preserves the quantiles of selected input distributions. The LOCA frequency, for example, has a large uncertainty envelope that has been preserved in this manner. Another large uncertainty envelope that could be preserved in this way is the ECCS strainer differential pressure. By preserving the uncertainty quantiles for selected sources, their effect can be explicitly observed in the resultant basic event distributions.

The LOCA epistemic uncertainty on failure probability is quantified separately for each of the five ECCS pump combinations considered in the analysis. As a result, the failure probabilities resulting from GSI-191 phenomena for the five pump combination cases are correlated with
the correct initiating event frequency associated with the combination.

The software used for the quantification is specifically designed to appropriately correlate elements from a group to which the same parameter value applies. This is accomplished using the "Big Loop Monte Carlo" option. For each trial of the "Big Loop Monte Carlo" option, a random set of values is selected from all input variables in the PRA model. These sample values are then used to re-evaluate all PRA model elements; that is, basic event probabilities, split fraction failure probabilities, initiator frequencies, and sequence frequencies that are then summed to give the CDF and LERF. Importantly, the option is also selected for the uncertainty quantification of the difference in the PRA metrics for change in CDF and change in LERF so that the uncertainty of the difference is correctly calculated.

The one exception to this correlation of input parameters among PRA model elements pertains to the CASA Grande analysis. By necessity, the PRA is quantified using failure probability distributions developed in the CASA Grande analysis which are functions of many data variables. The GSI-191 failure probabilities are quantified separately for each of the five ECCS pump state combinations considered in the STPNOC Pilot Project analysis. In this way, the key parameter of the PRA sequence models (that is sump flow rates) is effectively correlated in software with the CASA Grande analysis.

In the CASA Grande analysis, failure probabilities associated with engineering models of LOCA phenomena are also evaluated separately for five percentiles of the LOCA frequency uncertainty analysis. These five sets of results are the basis for the five-bin uncertainty distributions on each of the GSI-191 phenomena failure probabilities.

The correlation between the uncertainties in the relative break sizes used in the CASA Grande analysis and the absolute LOCA frequencies used in the PRA sequences models is not significant and therefore not modeled.

Parameter uncertainty

Parameter uncertainties are addressed in the STPNOC PRA analysis. For the physical models addressing the concerns of GSI-191, input parameters were derived from historical data and physical limits (for example, total contained volume in a tank).

The uncertainty associated with all important parameters has been included and sampling of the parameter distributions was done in Latin Hypercube Sampling schemes to accurately preserve the distribution. Human error probabilities are included in the STPNOC PRA however, for the most severe accident scenarios (that is large-break LOCA), there is very little opportunity for human actions to cause increases in the failure likelihood. In these cases, automatic actuation of the ECCS will occur prior to operator intervention.

Model Uncertainty

Models are developed to be accurate representations of the plant including parameter uncertainties. For the STPNOC analysis, the most important contribution to CDF is the model of chemical effects, both on the strainer and in the core, due to the use of reasonably conservative "bump up" factors, described in detail in Enclosures 4-1 and 4-3. Although chemical effects in STP post-LOCA fluid conditions are benign compared to the conditions assumed for the experiments performed in WCAP 16793-NP (Reference 17), the STPNOC analysis assumes that adverse chemical effects can occur, both at the strainer and in the core. Bounding values are used for strainer differential pressure, that is, higher differential pressures than observed in experiments representative of STP conditions. The model is less sensitive to strainer differential pressure than core failure loading which is chosen at one half the 15 grams per fuel assembly (g/FA) limit found in WCAP 16793-NP (Reference 17) as a threshold for the potential of boron precipitation.

In the STPNOC analysis, several new predictive models are applied. These include the debris penetration/ filtration model that was benchmarked to test data, and the time-dependent, debris circulation model that addresses coolant bypass around the reactor core. Additional conservatism accounts for the uncertainty associated with the predictions of first-order models. Additional conservatism was incorporated in the treatment of both conventional and chemical-induced differential head-loss estimations.

The application of an overall inflation factor that is distributed in magnitude according to the best interpretation of available data represents the extent of model uncertainty.

Completeness Uncertainty

The STPNOC analysis has extended the completeness of uncertainties associated with the concerns raised in GSI-191 by including phenomena expected to occur in the recirculation mode of ECCS operation where traditional analyses end. The STPNOC analysis uses realistic or prototypical conditions to model anticipated post-LOCA phenomena during all LOCA phases. Uncertainties are quantified based on distributions that encompass plant conditions and equipment operating states that, although important to long-term cooling, are not considered in traditional (e.g., UFSAR Chapter 15) analyses.

Examples of completeness uncertainties that have been considered and excluded from the current analysis are listed below:

- Multiple simultaneous RCS pipe breaks would result in reduced damage due to the very rapid depressurizaton of the RCS. Although more damage zones would be involved, less damage would be likely at each location.
- Physical security events that cause a LOCA. Such events would contribute equally to both the "ideal" plant and the as-built, as-operated plant. The STPNOC security force undergoes continuous evaluation and improvements are made in processes and procedures that would help preclude such events.
- Events occurring during shutdown modes of operation (includes lifting and transport of Heavy Loads). Heavy loads are not being moved during Mode 3. During the time heavy loads are being moved, the plant is cooled down and depressurized.
- Structural failures (containment building, interior containment walls or partitions that could be postulated to induce a LOCA). These beyond design basis events would contribute equally to both the "ideal plant" and the as-built, as-designed plant. In both cases, it would be assumed that core damage and large early release (in the case of containment failure leading to LOCA) would occur.
- Organizational decision making and safety culture.

The plant operating states of de-fueled conditions (No Mode), Refueling (Mode 6), and Cold Shutdown (Mode 5) are not evaluated. The basis for this is that operating pressures and temperatures are sufficiently low so that piping failure mechanisms typically associated with LOCA events cannot reasonably be expected to occur. Modes 1, 2, 3 and 4 are bounded by the at-power model.

Summary

The uncertainty quantification in the PRA analysis to address GSI-191 is a significant improvement in the understanding of RCS and containment building behavior under LOCA conditions. Uncertainties, not explicitly quantified, are either bounded by other uncertainties associated with more dominant contributors or are sources of uncertainty outside the scope and boundary of GSI-191 safety issues.

3.3.5 Comparison with Acceptance Guidelines

The STPNOC analysis shows that the risk associated with the concerns raised in GSI-191 is very small and far less than the threshold for Region III, "Very Small Changes," of RG 1.174.

In the STPNOC PRA analysis, the mean values used to perform the evaluation against the acceptance criteria are probability distributions that come from the propagation of the uncertainties of the input parameters and those model uncertainties explicitly represented in the model. The STPNOC PRA analysis uses a formal propagation of the uncertainty to account for any state-of-knowledge uncertainties that arise from the use of the same parameter values for several basic event probability models.

Where epistemic uncertainties have been identified in the STPNOC analysis, they have been either reduced through experimental evidence or bounded through assumption. The STPNOC PRA margin to the acceptance criteria guidelines is significant, providing confidence that any contributor to risk that may have been missed or otherwise not modeled would not make a significant change to the risk determined in the STPNOC analysis.

In the STPNOC analysis, reliance on importance measures is unnecessary and is not used. The focus of the analysis is to understand the risk associated with the concerns raised in GSI-191 and importance measures, while useful in evaluations concerned with other applications, are not useful in the STPNOC analysis.

The STPNOC PRA is an integrated-level model that includes all internal and external events (referring to Level I and Level II analysis) related to the GSI-191 post-LOCA concerns. Care has been taken in the STPNOC GSI-191 PRA to ensure that all concerns associated with GSI-191 have been addressed in the analysis.

3.3.6 Integrated Decision Making

There are many qualitative insights that form the basis for the conclusion in the STP PRA analysis that there is a very small risk for the concerns associated with GSI-191. The analysis has been enhanced to include parameter uncertainties.

In the analysis, uncertainties of input parameters in the traditional engineering models are propagated through the uncertainty quantification of basic events and aggregated (with uncertainty distributions) for use in PRA basic events or top events. By integrating qualitative insights, bounding uncertainties, and quantifying the uncertainties inherent in engineering models, the STPNOC PRA analysis is a robust, integrated analysis that can be relied on to accurately evaluate the risk associated with the concerns raised in GSI-191. Although the STPNOC PRA analysis relies on a full scope PRA, the analysis is specifically focused on the concerns raised in GSI-191. In particular, only the LOCA initiating events are of concern and the physical models are directed at long-term cooling.

3.3.7 Implementation and Monitoring Program

Design modifications addressing GSI-191 concerns, including installation of new sump strainers and replacement of problematic insulation, have been previously implemented using the STP design change process.

STPNOC has implemented procedures and programs for monitoring, controlling and assessing changes to the plant that have a potential impact on plant performance related to GSI-191 concerns. These provide the capability to monitor the performance of the sump strainers and the ability to assess impacts to the inputs and assumptions used in the PRA and the associated engineering analysis that support the proposed change. Programmatic requirements ensure that the potential for debris loading on the sump does not materially increase. These include:

- <u>Programs and procedures have been implemented to evaluate and control potential sources of debris in containment</u>:

- Technical Specification Surveillance Requirements implemented by STP procedures require visual inspections of all accessible areas of the containment to check for loose debris, and each containment sump to check for debris, as described in Section 4.1.3.

- The STP Design Change Package procedure includes provisions for managing potential debris sources such as insulation, qualified coatings, addition of aluminum or zinc, and potential effects of post-LOCA debris on recirculation flow paths and downstream components. The procedure has been augmented to explicitly require changes that involve any work or activity inside the containment be evaluated for the potential to affect the following:

    - Reactor coolant pressure boundary integrity
    - Accident or post-accident equipment inside containment
    - Quantity of metal inside containment
    - Quantity or type of coatings inside containment
    - Thermal insulation changed or added
    - Post-LOCA recirculation flow paths to the emergency sumps
    - Post-LOCA recirculation debris impact on internals of fluid components
    - Addition or deletion of cable

  A 10 CFR 50.59 screening or evaluation is required to be completed for all design changes. This process ensures that new insulation material that may differ from the initial design is evaluated for GSI-191 concerns.

- Programs to ensure that Service Level 1 protective coatings used inside containment are procured, applied, and maintained in compliance with applicable regulatory requirements. Additional details are discussed in the STP response to Generic Letter 98-04 (Reference 15). In addition, procedures have been implemented to govern the use of signs and labels inside containment.

- <u>As a necessary and required support function for ECCS and CSS, the sump strainers are within the STP 10 CFR 50.65 Maintenance Rule program</u>:

    - As part of the STP Corrective Action Program, condition reports written due to adverse conditions identified during the containment inspections or containment emergency sumps and strainers surveillances are reviewed for impact on Maintenance Rule scoped systems, as appropriate.

    - The STP Maintenance Rule program includes performance monitoring of functions associated with ECCS and CSS, including sump recirculation The inclusion of the ECCS and CSS into the Maintenance Rule program and the assessment of acceptable system performance provide continued assurance of the availability for performance of the required functions.

- PRA Updates: For the purpose of monitoring future facility changes or other conditions that may impact the PRA results associated with GSI-191, appropriate changes to the as-built, as-operated plant are reflected in updates to the STP at-power PRA reference model. The STP PRA Program is a living program and, as such, is subject to periodic review and updates. These PRA model periodic updates are performed in accordance with STP procedures. The effect of changes to the at-power PRA Reference model are assessed to ensure impacts to GSI-191 PRA results are evaluated against the criteria in RG 1.174, and any significant findings or conclusions are identified, documented, and entered into the Corrective Action Program as appropriate.

- Licensed Operator Training: Licensed Operators are trained on indications of and actions in response to sump blockage issues related to GSI-191, and performance is evaluated during training scenarios designed to simulate plant responses.

  Operator actions required to respond to emergency sump clogging are currently trained on a biennial basis in the Licensed Operator Requalification program. Simulator training objectives are trained every two years on the topics of transfer to cold leg recirculation, transfer to hot leg recirculation, and total loss of Emergency Sump recirculation capability. Indications of sump blockage are included as part of the Licensed Operator training administered for Emergency Operating Procedure (EOP) performance of switchover activities in addition to general familiarization with the indications of loss of pump suction. Licensed Operator Training includes the monitoring of operating ECCS and CSS pumps during the evolution for transfer to cold leg recirculation (0POP05-EO-ES13, "Transfer To Cold Leg Recirculation") and hot leg recirculation (0POP05-EO-ES14, "Transfer To Hot Leg Recirculation"). Operator training also includes actions required on a total loss of Emergency Sump recirculation capability (0POP05-EO-EC11, "Loss of Emergency Coolant Recirculation"). 0POP05-EO-EC11 is trained on a biennial basis in the Licensed Operator Requalification program (LOR), and simulator scenarios are utilized for this training.

- Quality Assurance (QA): The STP QA program is implemented and controlled in accordance with the Operations Quality Assurance Plan (OQAP) and is applicable to SSCs to an extent consistent with their importance to safety, and complies with the requirements of 10CFR50, Appendix B and other program commitments as appropriate.

  The QA Program is implemented with documented instructions, procedures, and drawings which include appropriate quantitative and qualitative acceptance criteria for determining that prescribed activities have been satisfactorily accomplished. Procedures control the sequence of required inspections, tests, and other operations when important to quality. To change these controls, the individual procedure must be changed and a similar level of review and approval given to the original procedure is required. Such instructions, procedures, and drawings are reviewed and approved for compliance with requirements appropriate to their safety significance.

  QA program controls are applied to safety-related SSCs to provide a high degree of confidence that they perform safely and activities are performed as expected. The rigorous controls imposed by the QA program provide adequate quality control elements to ensure system component reliability for the required functions.

- STPNOC has adopted other programs that help provide early detection and mitigation of leakage in other applications. The proposed change does not involve any changes to ASME Section XI inspection programs or mitigation strategies that have been shown effective in early detection and mitigation of weld and material degradation in Class I piping applications.

3.3.8 Technical Evaluation Conclusion

The design and licensing basis descriptions of accidents requiring ECCS operation, including analysis methods, assumptions, and results provided in UFSAR 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 far less than the Region III acceptance guidelines defined by RG 1.174.

The performance evaluations for accidents requiring ECCS operation described in Chapters 6 and 15, based on the South Texas Project Units 1 and 2 Appendix K Large-Break Loss-of-Coolant Accident (LBLOCA) analysis, demonstrate that for breaks up to and including the double-ended guillotine break of a reactor coolant pipe, the ECCS will limit the clad temperature to below the limit specified in § 50.46, and assure that the core will remain in place and substantially intact with its essential heat transfer geometry preserved.

The effects on long-term cooling due to debris accumulation on the ECCS and CSS containment sump strainers and the in-vessel effects following LOCAs that require recirculation flow from the containment sump to mitigate the event are the primary safety concerns of GSI-191. These concerns are addressed using a risk-informed analytical approach.

The STPNOC PRA and the engineering analysis supporting the GSI-191 analysis are representative of the as-built, as-operated plant. The STP PRA is compliant with RG 1.200 Revision 1 for internal events, and therefore, for the purpose of assessing the risk of internal events associated with GSI-191, is compliant with Regulatory Guide 1.200 Revision 2 for assessing the risk of internal events associated with LOCAs.

The results demonstrate that the calculated risk associated with GSI-191 concerns for STP Units 1 and 2 are very small and are far less than the threshold for Region III defined by RG 1.174. Acceptable containment sump design in support of ECCS and CSS during design basis accidents is confirmed.

4.0 Regulatory Evaluation

4.1 Applicable Regulatory Requirements/Criteria

4.1.1 Regulatory Requirements

The following regulations apply to the proposed amendment. Approval of the proposed amendment is contingent upon approval of the requests for exemptions from these regulations as provided and justified in Enclosures 2-1 through 2-4.

- § 50.46(b)(5), "*Long-term cooling*," states that after any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

- General Design Criteria (GDC) 35, "*Emergency core cooling*," states that a system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

  Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure."

- GDC 38, "*Containment heat removal*," states that a system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

  Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

- GDC 41, "*Containment atmosphere cleanup*," states that systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

4.1.2 Regulatory Guidance

NRC Regulatory Guide 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," provides the NRC staff's recommendations for using risk information in support of licensee-initiated Licensing Basis changes to a nuclear power plant that require NRC review and approval. This regulatory guide describes an acceptable approach for assessing the nature and impact of proposed Licensing Basis changes by considering engineering issues and applying risk insights.

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

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

   The exemption request in Enclosure 2 to this letter implements this principle.

2. *The proposed change is consistent with a defense-in-depth philosophy.*

   The proposed change is consistent with the defense-in-depth philosophy in that the following aspects of the facility design and operation are unaffected:

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

   The STP risk-informed approach analyzes a full spectrum of LOCAs, including double-ended guillotine breaks for all piping sizes up to and including the largest pipe in the reactor coolant system (RCS). By requiring that mitigative capability be maintained in a realistic and risk-informed evaluation of GSI-191 for a full spectrum of LOCAs, the approach ensures that defense-in-depth is maintained.

3. *The proposed change maintains sufficient safety margins.*

   The proposed change does not involve a change in any functional requirements or the configuration of plant SSCs. The safety analyses in the UFSAR are unchanged. Therefore, sufficient safety margins associated with the design will be maintained by the proposed change.

4. *When proposed changes result in an increase in CDF or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.*

   The proposed change is defined as the risk associated with GSI-191 concerns. Using engineering analysis and the PRA this risk has been calculated and shown to be far less than the threshold for Region III, "Very Small Changes," and is therefore consistent with the Commission's Safety Goal Policy Statement

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

   A description is provided in Section 3.3.7 of the STP programmatic requirements that ensure the potential for debris loading on the sump does not materially increase. As

noted in Section 3.3.7, STP has committed to estimating the effect of any change to or error in the PRA model used to evaluate the risk associated with GSI-191 and to report any significant change or error, in support of the LAR.

NRC Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," describes one acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for light-water reactors. The STPNOC PRA model used for the risk-informed approach for addressing GSI-191 concerns is in compliance with Revision 1 of RG 1.200.

4.1.3 Technical Specifications

As concluded by the evaluations described below, the proposed amendment does not require a change to the Technical Specifications (TS).

10 CFR 50.36 Evaluation

10 CFR 50.36(b) requires each license authorizing operation of a production or utilization facility of a type described in §50.21 or §50.22 will include technical specifications. The technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to §50.34. The Commission may include such additional technical specifications as the Commission finds appropriate.

10 CFR 50.36(c) states that the technical specifications will include items in the following categories:

(1) *Safety limits, limiting safety system settings, and limiting control settings.*

The proposed amendment does not propose any changes to the technical specifications safety limits, limiting safety system settings or limiting control settings, or propose any new safety limits, limiting safety system settings or limiting control settings.

As discussed in the NRC SER on NEI 04-07 (Reference 9):

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

The proposed change to the UFSAR does not involve any changes to the other acceptance criteria of 10 CFR 50.46 (peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation, and coolable geometry). The proposed change uses a risk-informed method, rather than a deterministic method, to reconstitute the design basis of the sumps and demonstrate acceptable sump performance, consistent with the requirements of 10 CFR 50.46:

- Accident sequences from a full spectrum of LOCAs are analyzed to provide assurance that the most severe postulated LOCAs are included in the model.
- The supporting engineering analysis for the risk-informed method includes justification for showing that the analyses realistically describe the behavior of the plant during a LOCA. Accident sequences are analyzed in a realistic time-dependent manner.
- The supporting analyses are informed by applicable experimental data where appropriate.
- Uncertainties in the analyses are identified and assessed so that uncertainty in the calculated results is estimated. For input into the PRA, uncertainty propagation is used to determine the probabilities of various failures potentially leading to core damage.

In the PRA model, the demand recirculation failure probability includes assessment of a loss of NPSH for pumps among the basic events as described for the STP risk-informed methodology, and accident sequences from a full spectrum of postulated LOCAs are analyzed. This approach demonstrates a high level of probability that adequate core cooling is maintained by maintaining a positive NPSH margin. As stated in (Ref NEI 04-07 SER), maintaining a positive NPSH margin to demonstrate adequate core cooling shows that the 10 CFR 50.46 acceptance criteria should not be challenged.

The results of the risk-informed method confirm that the ECCS cooling performance calculated by the Appendix K Evaluation Model demonstrates with a high level of probability that the long-term cooling criterion of 10 CFR 50.46(b)(5) would not be exceeded. The results also show acceptable CSS performance for containment heat removal and for maintaining the containment as an effective fission product barrier, as demonstrated by acceptable sump performance. These conclusions are based on the risk-informed approach meeting the key principles of RG 1.174, and the PRA results that show the residual risk associated with GSI-191 is far less than the threshold for "Very Small Changes" in Region III as defined in RG 1.174.

(2) *Limiting conditions for operation.*

Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. The proposed amendment does not propose any changes to the functional capability or performance levels of the Emergency Core Cooling System or the Containment Spray System.

The criteria for establishing a technical specification limiting condition for operation were evaluated.

(A) *Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.*

The proposed changes do not affect the instrumentation provided in the Technical Specifications that are used to detect degradation of the reactor coolant pressure boundary, including containment leak detection, containment temperature and pressure, and containment radiation levels.

The proposed changes do not affect the installed instrumentation used to detect inadequate recirculation flow and inadequate core flow conditions resulting from postulated LOCAs. As discussed in item (1) above, the proposed changes do not impact an initial condition of a design basis accident or transient analysis that either assumes the failure of, or presents a challenge to the integrity of a fission product barrier.

(B) *Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.*

The proposed changes do not impact any of the process variables, design features, or pose any operating restrictions that are an initial condition of a design basis accident or transient analysis that either assumes the failure of, or presents a challenge to the integrity of a fission product barrier.

The proposed changes reconstitute the design basis for the sumps to validate the assumptions in the Appendix K ECCS Evaluation model that demonstrates acceptable ECCS performance for postulated LOCAs. The existing LCO's for ECCS control the aspects of the design and plant conditions required to satisfy 10 CFR 50.46. The risk-informed method shows that the sump design supports the conclusion that there is a high probability that the long term cooing criterion would not be exceeded. The results also show that the existing LCO's are adequate for controlling the specified safety functions associated with CSS.

(C) *Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.*

The proposed amendment proposes a revision to the UFSAR to describe evaluations performed to address GSI -191 concerns. The evaluation concludes that the ECCS systems will operate with a high probability following a LOCA when considering the impacts and effects described by GSI-191. As discussed in item (1) above, the proposed changes reconstitute the design basis for acceptable sump performance during recirculation mode for ECCS and CSS to mitigate postulated LOCAs. No changes are proposed to the limiting condition for operation of the ECCS or CSS. The proposed amendment does not propose any change to how

the ECCS or CSS functions or actuates to mitigate a design basis accident or transient.

(D) *Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.*

The proposed amendment does not introduce the need for any new structure, system or component that is required to support the conclusions of the safety analysis. The PRA results show that the risk associated with the change is very small and does not warrant any technical specification under Criterion 4.

The existing LCO action statement completion times were also evaluated and determined to remain acceptable based on the very small risk contribution associated with the proposed change.

(3) *Surveillance requirements.* Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

The technical specification surveillance requirements (TS SRs) in place will continue to demonstrate that the safety significant measurable functions are satisfied to assure the operability of the ECCS and Containment Spray System, and provide assurance that the containment sump will perform its design function.

To demonstrate operability for each ECCS subsystem, TS SR 4.5.2.c specifies visual inspections of the containment for latent debris to be performed for establishing and maintaining Containment Integrity:

> *By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:*
>
> *1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and*
>
> *2) Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.*

TS SR 4.5.2.c is implemented by STP procedures 0PSP03-XC-0002, "Initial Containment Inspection to Establish Integrity," and 0PSP03-XC-0002A, "Partial Containment Inspection (Containment Integrity Established), Visual Inspection of Containment for Loose Debris."

- Containment inspections are performed prior to entering MODE 4 during plant startup, and include an elevation-by-elevation check to confirm the absence of loose debris that could clog the sumps, that all temporary storage

box lids are in place and secured, and that all tool cabinet doors are closed and secured.

- Walk-downs are performed by station management and Operations and a final acceptance walk-down is performed by Operations to confirm all requirements for Containment Integrity are met and to assure the containment building is free of loose debris.

Of particular relevance to GSI-191, TS SR 4.5.2.d. was previously revised to be consistent with the installation of the new advanced design sump strainers, and requires visual inspections of the sumps during each refueling outage:

> *At least once per 18 months by a visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components show no evidence of structural distress or abnormal corrosion.*

TS SR 4.5.2.d is implemented by STP procedure 0PSP04-XC-0001, "Inspection of Containment Emergency Sumps and Strainers Unit #1 1-A, 1-B, 1-C Unit #2 2-A, 2-B, 2-C." Visual inspections of each sump, the entire exterior and the interior of the strainers, and the vortex suppressor are performed. The sump inspection procedure includes specific criteria to assure the following:

- No external evidence of structural distress or abnormal corrosion
- No pathways that would allow foreign objects or debris to enter the sump
- No structural joints with gaps larger than 0.095"
- No gaps in the strainer modules or associated piping fit-up connections
- No foreign materials remain on or lodged into the gaps of the strainer modules.
- No foreign material inside the strainer core tubes, including the two strainer modules connected on a 45 degree angle on sumps "A" and "B".
- No restrictions in the sump suction inlets.
- No foreign objects, debris, or boron crystal build-up in the sumps, and that the sumps are dry.

The TS SR for visual inspections of the containment and emergency sumps support the specified safety functions for CSS in the recirculation mode also supported by the emergency sumps. Therefore, additional surveillance requirements are not warranted.

(4) <u>Design features</u>. Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in items (1), (2), and (3) above.

The proposed amendment does not propose any change to design features of the plant. As discussed above, the risk associated with GSI-191 issues for STP is very low and

does not warrant the discussion of any additional design features in the technical specifications.

(5) *Administrative controls. Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.*

The administrative controls in the technical specifications provide assurance that the facility will be operated in a safe manner. The proposed amendment proposes a revision to the UFSAR the risk-informed evaluation that concludes the ECCS and CSS will operate with a high probability following a LOCA when considering the impacts and effects described by GSI-191. No changes are proposed to any processes and procedures regarding operation of the ECCS or CSS.

Definition of Operable/Operability

Although a separate limiting condition for operation (LCO) is not being established, considerations regarding sump performance and its affect on existing LCOs (for example TS 3.5.2, 3.5.3.1 and 3.6.2.1) are already addressed by the TS definition of Operability.

STP TS Definition 1.20 states:

> *A system, subsystem, train, component or device shall be OPERABLE or have OPERABLITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).*

Operability for ECCS and CSS is based on the capability of each subsystem to perform its specified safety functions. Operability for ECCS is supported by acceptable performance of the sump strainers to provide the necessary support functions required for sufficient core cooling during the recirculation mode for all postulated break sizes ranging up to the double-ended guillotine break of the largest RCS cold leg pipe. Operability for CSS is similarly supported by the acceptable performance of the sump strainers during recirculation mode to support the specified safety functions.

Conditions that could potentially affect the ability of the containment emergency sump to support the LCO requirements of the ECCS or CSS would be evaluated and addressed by existing station procedures and processes and in accordance with the guidance of NRC Inspection Manual, Part 9900 (as discussed in Regulatory Issue Summary (RIS) 2005-20) for degraded and non-conforming conditions. Evaluations consider a change in condition inside the containment related to GSI-191 concerns, such as changes or additions to containment coatings and thermal insulation in the containment. Evaluations also consider how the condition may affect the inputs and assumptions to the supporting engineering analysis for the PRA model, as described in detail in Volume 3.

Operability determinations would be required to address certain conditions in containment that may affect debris loading of the strainers. Likely conditions include discovery of more

fibrous insulation in containment than expected or discovery of latent or miscellaneous fibrous debris. Applicable operability determinations should consider the specific design inputs used in the CASA Grande analysis discussed in Volume 3 Section 2.2 (Enclosure 4-3). The table below identifies possible contributors to debris loading that may be discovered in containment during power operation. The table provides the values used as input parameters in the risk-informed analysis and minimum margin represented in the values based on the analysis and engineering judgment, and may be used as guidance to immediately assess the potential impact due to unexpected material discovered in containment that may contribute to debris loading on the strainers.

Debris Type	**Input Parameter Value**	**Minimum Margin**	**Volume 3 Section**	**Notes**
Latent debris, consisting of: • Dirt/dust • Fiber, e.g. fibrous insulation	200 lbm (Total) 170 lbm (1.0 ft^3) 30 lbm (12.5 ft^3)	40 lbm (Total) 34 lbm (0.2 ft^3) 6 lbm (2.5 ft^3)	2.2.13	1, 2, 4
Miscellaneous debris, including but not limited to unqualified tags and labels	100 ft^2	10 ft^2	2.2.14	2, 4
Unqualified coatings	UFSAR Table 6.1-4	100 ft^2	2.2.11	1, 3
Aluminum metal	5000 ft^2 (Note 5)	100 ft^2	2.2.20	4, 5

Notes:
1. Material properties of debris are provided in Volume 3 Table 2.2.23 (Enclosure 4-3).
2. Conservatisms in the reported values are discussed in Reference 7.
3. Reported value includes allowance for future additions as discussed in Reference 15.
4. Guidelines for determining loose debris are provided in procedure 0PSP03-XC-0002, "Initial Containment Inspection to Establish Integrity."
5. Value based on guidance for aluminum scaffolding allowed inside containment as provided in procedure 0PGP03-ZM-0028, "Erection and Use of Temporary Scaffolding."

For immediate operability determinations, reasonable expectation that the SSCs are operable may be based on comparing a discovered condition with the expected debris loading in the containment, such as shown in the table above. A discovered condition that may exceed the values shown in the table does not preclude reasonable expectation of operability. Due to the complexity of the analysis, the potential exists for conditions to be discovered which are not represented by the values in the table, and for which follow-up evaluations will be required.

When warranted, an immediate operability determination will be followed by a prompt operability determination that will apply additional information and supporting analyses to confirm the immediate operability determination. Evaluations may consider additional information provided in the inputs to the CASA Grande analysis as well as the identified conservatisms associated with the categories of major assumptions in the CASA Grande analysis, Section 3 of Volume 3 (Enclosure 4-3):

1. General assumptions
2. Equipment failure assumptions (prior to start of recirculation)
3. LOCA frequency assumptions
4. Debris generation assumptions
5. Chemical effects assumptions
6. Debris transport assumptions
7. Head loss assumptions
8. Degasification assumptions
9. Penetration assumptions
10. Core blockage assumptions
11. Boron precipitation assumptions
12. Acceptance criteria assumptions

Conservative assumptions that are related to a discovered condition may be considered in support of the operability determinations.

For a discovered condition that potentially affects debris quantities in containment, the applicable CASA Grande input parameters and assumptions provide a means for immediate operability determinations and follow-up determinations, as warranted, to evaluate the impact on containment sump performance. If such a condition were determined to affect the ability of the ECCS or CSS to perform its specified function such that there is not reasonable expectation of operability, then the existing LCOs and action statements provide sufficient requirements to restore the system to operable or place the unit in a safe condition.

Conclusion

Based on the above evaluations of the 10 CFR 50.36(c) categories, the existing TS SR that support the proposed change, and the adequacy of the TS definition of Operable/Operability to address the proposed change as it relates to the required support function of the ECCS containment emergency sumps and strainers for the specified safety functions of the ECCS and CSS, no changes are needed to the operability requirements for ECCS and CSS, and no changes to the existing TS Action Statements are needed.

Changes to the Technical Specifications Bases

Although the above evaluations conclude that no changes are required to the TS, approval of the proposed changes requires conforming changes to the STP TS Bases to reflect the reconstituted design basis for the sumps and sump strainers. Changes to the TS Bases are provided for information only in Attachment 3 to this Enclosure, to be implemented following NRC approval of the LAR.

4.2 Precedents

The NRC plans to use STP Units 1 and 2 as a pilot for other licensees choosing to a risk-informed approach for closure of GSI-191 (Reference 14). The STP piloted risk-informed approach is expected to result in substantial benefit to both the NRC and industry in support of the development and implementation of risk-informed resolution of GSI-191.

The proposed amendment and accompanying exemption requests provide an approach for other licensees to revise their Licensing Basis in order to close GSI-191.

4.3 No Significant Hazards Consideration Determination

STPNOC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change adds the results of a risk-informed evaluation to the UFSAR that concludes the ECCS and CSS will operate with a high probability following a LOCA when considering the impacts and effects of debris accumulation on containment emergency sump strainers in recirculation mode, as well as core flow blockage due to in-vessel effects, following loss of coolant accidents (LOCAs).

   The proposed change does not implement any physical changes to the facility or any SSCs, and does not implement any changes in plant operation. The proposed change confirms that required SSCs supported by the containment sumps will perform their safety functions as required, and does not alter or prevent the ability of SSCs to perform their intended function to mitigate the consequences of an accident previously evaluated within the acceptance limits. The safety analysis acceptance criteria in the UFSAR continue to be met for the proposed change. The proposed change does not affect initiating events. The proposed change does not significantly affect the operation of the containment systems needs to ensure that there is a large margin between the temperature and pressure conditions reached in the containment and those that would lead to failure so that there is a high degree of confidence that damage of the containment cannot occur.

   The calculated risk associated with the proposed change is very small and far less than the threshold for Region III as defined by RG 1.174, for both CDF and LERF. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any the accident previously evaluated in the UFSAR.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

The proposed change neither installs nor removes any plant equipment, nor alters the design, physical configuration, or mode of operation of any plant structure, system or component. The proposed change does not introduce any new failure mechanisms or malfunctions that can initiate an accident. The proposed change does not introduce failure modes, accident initiators, or equipment malfunctions that would cause a new or different kind of accident. Therefore, the proposed change does not create the possibility for a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

Response: No.

The proposed change does not involve a change in any functional requirements, the configuration, or method of performing functions of plant SSCs. The effects from a full spectrum of LOCAs, including double-ended guillotine breaks for all piping sizes up to and including the largest pipe in the reactor coolant system, are analyzed. Appropriate redundancy and consideration of loss of offsite power and worst case single failure are retained, such that defense-in-depth is maintained.

The risk-informed method demonstrates the containment sumps will continue to support the ability of safety related components to perform their design functions. The proposed change does not alter the manner in which safety limits are determined or acceptance criteria associated with a safety limit. The proposed change does not implement any changes to plant operation, and does not significantly affect SSCs that respond to safely shutdown the plant and to maintain the plant in a safe shutdown condition. The proposed change does not significantly affect the existing safety margins in the barriers for the release of radioactivity. There are no changes to any of the safety analyses in the UFSAR. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

4.4 Conclusion

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations contingent upon approval of the exemption requested in Enclosure 2 to this letter, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 Environmental Consideration

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20. However, the proposed amendment does not involve (i) a significant

hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 References

1. Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance"

2. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2 (ML100910006)

3. Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," Proposed Revision 1, May 1983

4. STPNOC letter, David W. Rencurrel to NRC Document Control Desk, "Proposed Change to Surveillance Requirement 4.5.2.d," May 21, 2007, NOC-AE-07002156 (ML0715605)

5. STPNOC letter, David W. Rencurrel to NRC Document Control Desk, "Response to NRC Request for Additional Information on Proposed Change to Surveillance Requirement 4.5.2.d (TAC Nos. MD5705, MD5706)," November 26, 2007, NOC-AE-07002225 (ML073380340)

6. "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Nos. 183 and 170 to Facility Operating License Nos. NPF-76 and NPF-80 STP Nuclear Operating Company, et al., South Texas Project, Units 1 and 2, Docket Nos. 50-498 and 50-499," March 25, 2008 (ML080360321)

7. STPNOC letter, David W. Rencurrel to NRC Document Control Desk, "Supplement 4 to the Response to Generic Letter 2004-02 (TAC Nos. MC4719 and MC4720)," December 11, 2008, NOC-AE-08002372 (ML083520326)

8. NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Volume 1 "Pressurized Water Reactor Sump Performance Evaluation Methodology," Revision 0, December 2004 (ML050550138)

9. NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Volume 2 "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02, Revision 0, December 6, 2004," Revision 0, December 2004 (ML050550156)

10. WCAP-16406-P-A, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," Revision 1, March 2008 (ML081000025)

11. WCAP-16793-NP," Evaluation of Long-Term Cooling Considering Particulate and Chemical Debris in the Recirculating Fluid," Revision 0, May 2007 (ML071580139)

12. NEI 02-01, "Condition Assessment Guidelines: Debris Sources Inside PWR Containments," Revision 1, September 2002 (ML030420318)

13. STPNOC letter, J. W. Crenshaw, STPNOC, to NRC Document Control Desk, "Status of the South Texas Project Risk-Informed (RI) Approach to Resolve Generic Safety Issue (GSI)-191," NOC-AE-11002775, December 14, 2011 (ML11354A386)

14. Commission SECY Paper, "Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance," SECY-12-0093, July 9, 2012 (ML121320270)

15. STPNOC letter, T. H. Cloninger to NRC Document Control Desk, "Response to Generic Letter 98-04, 'Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment,'" November 11, 1998, NOC-AE-000350 (Legacy No. 9811230146)

16. WCAP-16530-NP, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," Revision 0, February 2006 (ML060890509)

17. WCAP-16793-NP," Evaluation of Long-Term Cooling Considering Particulate and Chemical Debris in the Recirculating Fluid," Revision 2, October 2011 (ML11292A020)

18. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities"

Attachment 1 to Enclosure 3

List of Commitments

List of Commitments

The following table identifies the actions to which STP Nuclear Operating Company (STPNOC) has committed. Statements in the submittal with the exception of those in the table below are provided for information purposes and are not considered regulatory commitments.

Commitment	Tracking Number	Scheduled Completion Date
The STP UFSAR will be revised to include the changes provided in Attachment 2 to Enclosure 3, "Licensing Amendment Request for STP Piloted Risk-Informed Approach to Closure for GSI-191."	CR 11-4249-9	90 days following approval of LAR

Attachment 2 to Enclosure 3

STPEGS UFSAR Page Markups

STPEGS UFSAR Page Markups

The changes to the South Texas Project Electric Generating Station (STPEGS) Updated Final Safety Analysis Report (UFSAR) are provided for NRC review and approval for the purpose of resolving Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance." The risk-informed approach following the guidance of RG 1.174 provided as justification for the changes.

The design and licensing basis descriptions of accidents requiring ECCS operation, including analysis methods, assumptions, and results provided in UFSAR Chapters 6 and 15 remain unchanged. The performance evaluations for accidents requiring ECCS operation described in Chapters 6 and 15, based on the South Texas Project Units 1 and 2 Appendix K Large-Break Loss-of-Coolant Accident (LBLOCA) analysis, demonstrate that for breaks up to and including the double-ended guillotine break of a reactor coolant pipe, the ECCS will limit the clad temperature to below the limit specified in § 50.46, and assure that the core will remain in place and substantially intact with its essential heat transfer geometry preserved.

The results of the risk-informed method determine acceptable containment sump design and performance and reconstitute the licensing bases for the supported ECCS and CSS specified functions required during recirculation mode following postulated LOCAs, for the purpose of resolving GSI-191.

Changes to the UFSAR are shown on the following pages in gray highlight.

3.1.2.4.6.1 Evaluation Against Criterion 35 – The ECCS is provided to cope with any LOCA in the plant design basis. Abundant cooling water is available in an emergency to transfer heat from the core at a rate sufficient to maintain the core in a coolable geometry and to assure that clad metal/water reaction is limited to less than 1 percent. Adequate design provisions are made to assure performance of the required safety functions even with a single failure.

Details of the capability of the systems are included in Section 6.3. An evaluation of the adequacy of the system functions is included in Chapter 15. Performance evaluations have been conducted in accordance with 10CFR50.46 and 10CFR50 Appendix K.

For resolution of GSI-191, a risk-informed approach meeting the guidance in RG 1.174 was used to confirm acceptable containment sump performance for ECCS and CSS in recirculation mode following postulated LOCAs, and to validate assumptions used for the ECCS performance evaluations to assure long-term cooling performance. The risk-informed approach described in Appendix 6A provided justification for an exemption from the implicit GDC 35 requirement to demonstrate acceptable performance using a deterministic evaluation.

3.1.2.4.9.1 Evaluation Against Criterion 38 – The CHRS consists of the CSS, the Reactor Containment Fan Cooler (RCFC) Subsystem and the residual heat removal (RHR) heat exchangers. The CHRS acts in conjunction with the Safety Injection System to remove heat from the Containment. The CHRS is designed to accomplish the following functions in the unlikely event of a LOCA: to rapidly condense the steam within the Containment in order to prevent over-pressurization during blowdown of the RCS; and to provide long-term continuous heat removal from the Containment.

Initially, the CSS and the high-and low-head safety injection (HHSI and LHSI) pumps take suction from the refueling water storage tank (RWST). During the recirculation phase, the CSS and the HHSI and LHSI pumps take suction from the Containment sumps and emergency sumps.

The CHRS is divided into three trains. Each train is sized to remove 50 percent of the system design heat load at the start of recirculation. Each train of the CHRS is supplied power from a separate independent Class 1E bus. The redundancy and capability of the Offsite and Emergency Power Systems are presented in the evaluation against Criterion 17. Redundant system trains and emergency diesel power supplies provide assurance that system safety functions can be accomplished.

For further discussion, see the following sections of the UFSAR:

Residual Heat Removal System	5.4.7
Containment Systems	6.2
Engineered Safety Features Actuation System	7.3
Onsite Power System	8.3
Accident Analysis	15.0

For resolution of GSI-191, a risk-informed approach meeting the guidance in RG 1.174 was used to confirm acceptable containment sump performance for ECCS and CSS in recirculation mode following postulated LOCAs, and to validate assumptions used for the ECCS performance evaluations to assure long-term cooling performance. The risk-informed approach described in Appendix 6A provided justification for an exemption from the implicit GDC 38 requirement to demonstrate acceptable performance using a deterministic evaluation.

3.1.2.4.12.1 Evaluation Against Criterion 41 – The CSS is provided to reduce the concentration and quantity of fission products in the Containment atmosphere following a LOCA. Per 10CFR50.44, hydrogen recombiners are no longer required for design basis accidents.

The equilibrium sump pH is maintained by trisodium phosphate (TSP) contained in baskets on the containment floor. The initial CSS water and spilled RCS water dissolves the TSP into the containment sump allowing recirculation of the alkaline fluid. Each unit is equipped with three 50-percent spray trains taking suction from the Containment sump. Each Containment spray train is supplied power from a separate bus. Each bus is connected to both the Offsite and the Standby Power Supply Systems. This assures that for Onsite or for Offsite Electrical Power System failure, their safety function can be accomplished, assuming a single failure.

Post-accident combustible gas control is assured by the use of the Supplementary Containment Purge Subsystem.

For further discussion, see the following sections of the UFSAR:

Containment Systems	6.2
Containment Spray System – Iodine Removal	6.5.2
Containment Hydrogen Sampling System	7.6.5
Containment HVAC System	9.4.5
Accident Analysis	15.0

For resolution of GSI-191, a risk-informed approach meeting the guidance in RG 1.174 was used to confirm acceptable containment sump performance for ECCS and CSS in recirculation mode following postulated LOCAs, and to validate assumptions used for the ECCS performance evaluations to assure long-term cooling performance. The risk-informed approach described in Appendix 6A provided justification for an exemption from the implicit GDC 41 requirement to demonstrate acceptable performance using a deterministic evaluation.

TABLE 3.12-1
REGULATORY GUIDE MATRIX

ABBREVIATIONS:

A Conform to guide

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
1.82	Sumps for Emergency Core Cooling and Containment Spray Systems	6.2.2.1.2 6.2.2.2.3 6.3.4.1	Proposed Rev 1 (5/83)	A See Note 103

NOTES

103 NRC Generic Letter 2004-02 (GL 04-02) "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," required licensees to evaluate the ECCS and CSS recirculation functions based on the potential susceptibility of sump screens to debris blockage during design basis accidents. Refer to Section 6.2.2.1.2.

6.2.2.1.2 Containment Emergency Sump Design Bases:

The Containment emergency sump meets the following design bases:

1. Sufficient capacity and redundancy to satisfy the single-failure criteria. To achieve this, each CSS/ECCS train draws water from a separate Containment emergency sump.

2. Capable of satisfying the flow and net positive suction head (NPSH) requirements of the ECCS and the CSS under the most adverse combination of credible occurrences. This includes minimizing the possibility of vortexing in the sump.

3. Minimizes entry of high-density particles (specific gravity of 1.05 or more) or floating debris into the sump and recirculating lines.

4. Sumps are designed in accordance with RG 1.82, proposed revision 1, May 1983 and with Generic Letter 2004-02 as described in ~~NOC-AE-08002372~~ Appendix 6A.

6.2.2.2.3 Containment Emergency Sump Description:

At the beginning of the recirculation phase, the minimum water level above the Containment floor is adequate to provide the required NPSH for the ECCS and CSS pumps. The sumps are designed to RG 1.82, proposed revision 1, May 1983 and to the requirements of Generic Letter 2004-02 as described in ~~NOC-AE-08002372~~ Appendix 6A. The sump structures are designed to limit approach flow velocities to less than 0.009 ft/sec permitting high-density particles to settle out on the floor and minimize the possibility of clogging the strainers. The sump structures are designed to withstand the maximum expected differential pressure imposed by the accumulation of debris.

6.2.2.3.5 Pump Net Positive Suction Head Requirements:

The minimum available net positive suction head (NPSH) for the CSS pumps is such that an adequate margin is maintained between the required and the available NPSH for both the injection and recirculation phase, ensuring the proper operation of the CSS as discussed in Appendix 6A. Recirculation operation gives the limiting NPSH requirements for the CSS pumps and is discussed in Appendix 6A.

The Westinghouse CSS pump design provides for the NPSH requirement to be met by the inherent design of the pump. CSS pumps are vertical motor-driven pumps, each sitting in an individual barrel. The design calls for a distance of 15 ft in this barrel between the suction nozzle centerline and the pump first-stage impeller. The 15-ft liquid-head in the pump barrel is thus expected to inherently satisfy the 15-ft NPSH requirement.

The analysis of available NPSH to the CSS pumps concerns itself with the NPSH at the pump suction nozzle, located at the top of the barrel. Since the pump barrels provide the required NPSH at the first-stage impeller, the piping layout need provide only sufficient NPSH at the pump suction nozzle to prevent flashing in the barrel.

Two modes of operation have been analyzed for the CSS pumps:

1. Pump taking suction from the RWST and delivering spray to the Containment

2. Pump taking suction from the Containment sump and delivering spray to the Containment

Case 2 represents the "worst case" since it gives the minimum available NPSH.

The assumptions and conservatisms used in the analysis are listed below. No exceptions are taken to RG 1.1.

1. Containment pressure equals the vapor pressure of the sump water.

2. The runout flows of each pump are used to account for maximum friction loses.

The minimum flood level in Containment is determined by considering the quantities of water trapped by the refueling cavity.

The results of the analysis show the available NPSH at the first-stage impeller of the CSS pumps to be greater than the required NPSH and show that the fluid at the suction flange is subcooled as qualified by the description in Appendix 6A for acceptable sump design and CSS pump performance in the recirculation mode. There is sufficient NPSH at the suction nozzle to prevent flashing in the barrel, and the analysis meets the guideline of RG 1.1. The NPSH parameters are listed in Table 6.2.2-4.

NPSH for the ECCS pumps is addressed in Section 6.3.

TABLE 6.2.2-4

CSS PUMP NPSH PARAMETERS

Required NPSH at Max Flow Rate, ft (max)	16.4
Available NPSH, ft (from RWST)	56.1
(From RCB Emergency Sump)	>17.6 (Note 1)

Note 1: The value is the result of a deterministic analysis used as an indication of pump performance. Acceptable pump performance in the recirculation mode is evaluated using a risk-informed approach described in Appendix 6A.

6.3.2.2 Equipment and Component Descriptions.

Net Positive Suction Head

Available and required net positive suction head (NPSH) for ECCS pumps are shown in Table 6.3-1. The safety intent of Regulatory Guide (RG) 1.1 is met by the design of the ECCS such that adequate NPSH is provided to system pumps.

The NPSH available for the injection mode is determined from the elevation head and the vapor pressure (atmospheric) of the water in the RWST, and the pressure drop in the suction piping from the tanks to the pumps. The NPSH evaluation is based on all pumps operating at maximum flow rate with no credit taken for the elevation head in the tank and full penalty assumed for head loss in the suction lines.

In addition to considering the static head and suction line pressure drop, the calculation of available NPSH in the recirculation mode assumes that the vapor pressure of the liquid in the sump is equal to the Containment ambient pressure. This assures that the actual available NPSH is always greater than the calculated NPSH.

See Appendix 6A for discussion of the pump performance in the recirculation mode.

TABLE 6.3-1

EMERGENCY CORE COOLING SYSTEM
COMPONENT PARAMETERS

High Head Safety Injection Pumps	
Required NPSH at max. flow rate, ft (max)	16.1
Available NPSH, ft (From RWST)	55.8
(From RCB Emergency Sump)	> 17.8 (Note 1)
Low Head Safety Injection Pumps	
Required NPSH, ft (max)	16.5
Available NPSH, ft (From RWST)	55.1
(From RCB Emergency Sump)	> 18.0 (Note 1)

Note 1: The value is the result of a deterministic analysis used as an indication of pump performance. Acceptable pump performance in the recirculation mode is evaluated using a risk-informed approach described in Appendix 6A.

NOTE: UFSAR change for Appendix 6A shown below consists entirely of new content, therefore gray highlight is not used.

APPENDIX 6A

RISK-INFORMED APPROACH TO POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING DESIGN BASIS ACCIDENTS

INTRODUCTION AND SUMMARY

NRC Generic Letter 2004-02 (GL 2004-02) "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," required licensees to perform an evaluation of the ECCS and CSS recirculation functions, and the flowpaths necessary to support those functions, based on the potential susceptibility of sump screens to debris blockage during design basis accidents requiring recirculation operation of ECCS or CSS. This Generic Letter resulted from the Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance." As a result of the evaluation required by GL 2004-02 and to ensure system function, sump design modifications were implemented (refer to Section 6.2.2.2.3).

GL 2004-02 sump performance evaluation activities, documented in References 6A-1 and 6A-2, included the following:

- Containment walkdowns
- Debris generation and transport analysis
- Calculation of required and available net positive suction head (NPSH)
- Screen requirements
- Screen structural analyses
- Potential or planned design/operational/procedural modifications
- Downstream effects
- Upstream effects
- Chemical effects

A risk-informed evaluation implemented in response to issues identified in GSI-191 provides high confidence that the sump design supports long-term core cooling following a design basis loss of coolant accident. The evaluation meets the acceptance guidelines in Regulatory Guide 1.174 (Reference 6A-3).

Acceptable sump design, based on meeting the acceptance guidelines in RG 1.174, demonstrates high probability of successful ECCS and CSS operation in recirculation mode following postulated LOCAs, with consideration of debris and other effects associated with GSI-191. This result reconstitutes the long term cooling licensing basis for meeting 10 CFR 50.46(b)(5) and provides closure for GSI-191. Demonstration of acceptable sump design using the risk-inform approach also reconstitutes the licensing basis for compliance with GDC 35, GDC 38 and GDC 41.

The use of a risk-informed method, rather than exclusively deterministic methods, demonstrates acceptable containment sump performance in support of ECCS and CSS operation in recirculation mode following postulated LOCAs. Therefore exemptions to 10 CFR 50.46(b)(5), GDC 35, GDC 38, and GDC 41 have been granted pursuant to 10 CFR 50.12.

DISCUSSION

The plant licensing basis considers long-term core cooling following a LOCA as identified in 10 CFR 50.46. Long-term cooling is supported by the ECCS which includes the Containment Spray (CS), the High Head Safety Injection (HHSI), the Low Head Safety Injection (LHSI), and the Residual Heat Removal (RHR) systems. Using a risk-informed approach to address the concerns of GSI-191, the probability associated with the operation of the ECCS to maintain long-term cooling following a LOCA has been quantified. The results show that the risk associated with GSI-191 concerns is less than the threshold for "Very Small Changes" in Region III as defined in Regulatory Guide 1.174.

The risk-informed method of analysis using an integrative approach models the underlying physical phenomena of the basic events and propagates uncertainties in the physical models.

In particular, the STP plant-specific PRA models the demand recirculation failure probability with the following basic events:

- Pressure drop due to build up of debris on the sump strainers with chemical effects resulting in loss of NPSH margin for the ECCS pumps
- Strainer mechanical collapse
- Air ingestion through the sump strainers
- Core blockage with chemical effects
- Boron precipitation in the core

The accident sequences are analyzed in a realistic time-dependent manner with uncertainty propagation to determine the probabilities of various failures potentially leading to core damage from a spectrum of location-specific pipe breaks for input into the PRA. The specific failure modes considered are:

1. Strainer head loss exceeds the NPSH margin for the pumps causing some or all of the ECCS and CSS pumps to fail.
2. Strainer head loss exceeds the strainer structural margin causing the strainer to fail, which subsequently results in larger quantities and larger sizes of debris being ingested into the ECCS and CSS.
3. Air intrusion exceeds the limits of the ECCS and CSS pumps causing degraded pump performance or complete failure due to gas binding.
4. Debris penetration exceeds ex-vessel effects limits causing a variety of potential equipment and component failures due to wear or clogging.
5. Debris penetration exceeds in-vessel effects limits resulting in partial or full core blockage with insufficient flow to cool the core.
6. Buildup of oxides, crud, LOCA-generated debris, and chemical precipitates on fuel cladding exceeds the limits for heat transfer resulting in unacceptably high peak cladding temperatures.

7. Boron concentration in the core exceeds the solubility limit leading to boron precipitation and subsequently resulting in unacceptable flow blockage or impaired heat removal.

Failure Modes 4 and 6 were conservatively addressed as part of the previous deterministic evaluations for STP with no issues of concern and were therefore not explicitly modeled in the PRA analysis. The remaining failure modes are explicitly modeled.

Failure probabilities and associated uncertainties determined in the supporting engineering analysis are passed to the plant-wide PRA, which determines the incremental risk associated with GSI-191 failure modes. The PRA assessment model includes the following events to accommodate composite GSI-191 failure processes:

1. Failure at the sump strainer
2. Boron precipitation in the core
3. Blockage of the core

The engineering analysis supports the three composite failure probabilities needed for the PRA by testing the outcome of postulated break scenarios against seven performance thresholds:

1. Strainer DP ≥ NPSH margin
2. Strainer DP ≥ P-buckle
3. Strainer F-void ≥ 0.02
4. Core fiber load ≥ cold leg break fiber limit for boron precipitation
5. Core fiber load ≥ hot leg break fiber limit for boron precipitation
6. Core fiber load ≥ cold leg break fiber limit for flow blockage
7. Core fiber load ≥ hot leg break fiber limit for flow blockage

P-buckle is the strainer structural design limit for the differential pressure (DP) across the ECCS strainers at which they are analyzed to be within code design allowable stresses. F-void is the vapor fraction of the liquid just downstream of the ECCS strainers.

Using the inputs noted above, the PRA assessment model is informed with risk insights for the failure modes associated with GSI-191 concerns. The PRA analysis yields results that are less than the threshold for Region III, "Very Small Changes," as defined by RG 1.174 (i.e., the change in CDF is less than 1E-6/yr and the change in LERF is less than 1E-7/yr).

On the basis that sump performance is determined acceptable by the risk-informed method, the assumptions in the ECCS evaluation model are valid and the long term cooling licensing basis for compliance with 10 CFR 50.46 is reconstituted, thus resolving GSI-191. Acceptable sump performance in support of ECCS and CSS design requirements in recirculation mode following postulated LOCAs also reconstitutes the licensing basis to meet GDC 35, 38, and 41.

EVALUATING IMPACTS

For the purpose of monitoring future facility changes or other conditions that may impact the PRA results associated with GSI-191, appropriate changes to the as-built, as-operated plant are reflected in updates to the STP at-power PRA reference model. The STP PRA Program is a living program and, as such, is subject to periodic review and updates. These PRA model periodic updates are performed in accordance with STP procedures. The effect of changes to

the at-power PRA Reference model are assessed to ensure impacts to GSI-191 PRA results are evaluated against the criteria in RG 1.174, and any significant findings or conclusions are identified, documented, and entered into the Corrective Action Program as appropriate.
The table below identifies contributors to debris loading that may be discovered in containment during power operation. The table provides the values used as input parameters in the risk-informed engineering analysis and minimum margin represented in the values based on the analysis and engineering judgment (see Reference 6A-4).

The table provides guidance that may be used to immediately assess the potential impact due to unexpected material discovered in containment that may contribute to debris loading on the strainers. As discussed in Reference 6A-4, these values are not necessarily the limiting amount of each type as analyzed. Conservatisms in the reported values are also discussed in Reference 6A-2. Therefore, a condition that may exceed the values shown in the table does not preclude reasonable expectation of operability.

Debris Type	**Input Parameter Value**	**Minimum Margin**
Latent debris, consisting of: • Dirt and/or dust • Fiber, e.g., fibrous insulation	200 lbm (Total) 170 lbm (1.0 cubic ft) 30 lbm (12.5 cubic ft)	40 lbm (Total) 34 lbm (0.2 cubic ft) 6 lbm (2.5 cubic ft)
Miscellaneous debris, including but not limited to unqualified tags and labels	100 sq-ft	10 sq-ft
Unqualified coatings	Table 6.1-4	100 sq-ft
Aluminum metal	5000 sq-ft	100 sq-ft

Due to the complexity of the analysis, the potential exists for conditions to be discovered which may not be represented by the values in the table, and for which evaluations would be required to evaluate the impacts (see Reference 6A-4).

REFERENCES

<u>Appendix 6A</u>:

6A-1 Correspondence NOC-AE-05001922, dated August 31, 2005

6A-2 Correspondence NOC-AE-08002372, dated December 11, 2008

6A-3 Regulatory Guide 1.174, "An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (May 2011)

6A-4 Correspondence NOC-AE-13002986, dated June 19, 2013

Attachment 3 to Enclosure 3

Technical Specifications Bases Page Markups

(Information Only)

Technical Specifications Bases Page Markups (Information Only)

Add the following to the end of Bases Section 3/4.5.2 and 3/4.5.3 ECCS Subsystems:

The OPERABILITY of the ECCS Subsystems is assured by the capability of the containment emergency sump to limit entry of high-density particles or floating debris into the sump and recirculating lines. This capability ensures that the flow and net positive suction head requirements of ECCS are satisfied under the most adverse combination of credible occurrence. Assurance that containment debris will not block the sump and render the ECCS Subsystem inoperable on emergency recirculation during design basis accidents is provided by inspection and engineering evaluation. UFSAR Appendix 6A provides a risk-informed approach that addresses the potential of debris blockage concluding that long-term core cooling following a design basis loss of coolant accident is assured with high probability. UFSAR Appendix 6A also provides guidance for assessing the potential impact on Operability due to unexpected material such as loose debris discovered in containment that may contribute to debris loading on the strainers.

Add the following to the end of Bases Section 3/4.6.2.1 Containment Spray System:

The OPERABILITY of the Containment Spray System is assured by the capability of the containment emergency sump to limit entry of high-density particles or floating debris into the sump and recirculating lines. This capability ensures that the flow and net positive suction head requirements of Containment Spray System are satisfied under the most adverse combination of credible occurrence. Assurance that containment debris will not block the sump and render the Containment Spray System inoperable on emergency recirculation during design basis accidents is provided by inspection and engineering evaluation. UFSAR Appendix 6A provides a risk-informed approach that addresses the potential of debris blockage concluding that heat removal capability and atmospheric cleanup capability following a design basis loss of coolant accident are assured with high probability. UFSAR Appendix 6A also provides guidance for assessing the potential impact on Operability due to unexpected material such as loose debris discovered in containment that may contribute to debris loading on the strainers.

ENCLOSURE 4

Risk-Informed Closure of GSI-191

Supporting Engineering Analysis and PRA

Introduction and Overview

Risk-Informed Closure of GSI-191 Supporting Engineering Analysis and PRA – Introduction and Overview

Introduction

Enclosures 4-1, 4-2 and 4-3 provide the engineering evaluation, risk analysis, technical information and references in support of the proposed changes to the STP Units 1 and 2 licensing basis subject to the license amendment request (Enclosure 3) and requested exemptions (Enclosures 2-1 through 2-4).

The structure and content of the supporting information is designed to meet the requirements for risk-informed applications to changes to the licensing bases as specified in RG 1.174.

Overview

Enclosure 4-1, Volume 1 Project Summary

This volume addresses the required content of a RG 1.174 application using the same section numbering scheme as in RG 1.174. This volume summarizes the generic methodology and the plant-specific implementation of the methodology, and discusses how the risk metrics associated with the residual risk of GSI-191 are determined:

- The overall analysis approach describing how engineering analyses are used in a risk-informed framework to support the PRA.
- Identifies where the methods adopted for the STP approach involve deviations from those previously approved for deterministic methods to resolve GSI-191 concerns, primarily NEI 04-07.
- Quantifies the change in risk associated with the concerns raised in GSI-191 in the as-built, as-operated plant (such as fibrous debris beds, chemical effects, in-vessel fiber loads, etc.).

Enclosure 4-2, Volume 2 Probabilistic Risk Analysis

This volume describes the PRA treatment of GSI-191 safety issues and interfaces with deterministic analyses, provides quantification of CDF and LERF risk metrics for comparison to RG 1.174 acceptance guidelines, and provides PRA uncertainty quantification.

Enclosure 4-3, Volume 3 Engineering (CASA Grande) Analysis

Volume 3 is a technical evaluation that provides a high-level description of the phenomenological portion of the overall risk-informed GSI-191 evaluation. This volume provides a detailed summary of supporting engineering analyses (CASA Grande evaluation) including input parameters, assumptions, methodology, analysis and results, and the use of any test results in the analysis. The STP Units 1 & 2 implementation of the generic methodology of the risk-informed approach (Enclosure 1) for addressing the required inputs to the plant-specific PRA model is described.