ENCLOSURE 4-1

Risk-Informed Closure of GSI-191

Volume 1

Project Summary

Risk-Informed Closure of GSI-191
Volume 1.0
Project Summary

June 18, 2013

Document: STP-RIGSI191-V01
Revision: 1

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Abstract

The PRA analyses that provide the technical background in the project to close Generic Safety Issue 191 at the South Texas Project using a risk-informed approach are summarized. The overall methodology used in the PRA analyses is summarized. The elements of Regulatory Guide 1.174 required for a Risk-Informed license submittal are documented. Qualitative and quantitative results of the PRA analyses are presented. The results of the Independent Technical Oversight activities are summarized. The basic calculation flow of the engineering analysis supporting the PRA is summarized. The methodology used to sample and propagate uncertainties is described.

Acknowledgements

The Risk-Informed GSI-191 Closure Pilot Program is piloted by the STP Nuclear Operating Company and jointly funded with several other licensees. It is a collaboration of experts from industry, academia, and a national laboratory. In general, all products are developed jointly and reviewed in regularly scheduled (monthly) Technical Team Meetings and weekly teleconferences as well as in specific review cycles by Independent Oversight (technical evaluation of all materials), STP Nuclear Operating Company project management, and STP Nuclear Operating Company quality management. The business entities, the main areas of investigation, and the principal investigators of the Pilot Program are summarized below.

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Executive Summary

The main objective of the STPNOC Risk-Informed GSI-191 Closure Pilot Project [1, 2] is, "Through a risk-informed approach, establish a technical basis that would demonstrate that the STP as-built, as-operated plant design is sufficient to gain NRC approval to close the issues raised in GSI-191 by the end of 2013." In 2012, the STP approach has been referred to as Option 2b in the industry.

The results presented in this summary are the joint work of STPNOC Risk Management, Los Alamos National Laboratory, The University of Texas at Austin, Texas A&M University, Alion Science and Technology, ABS Consulting, The University of New Mexico, Soteria Consulting, and KNF Consulting, LLC. STPNOC has also collaborated with the PWROG and NEI in development of the Pilot Project.

In the risk-informed approach, STPNOC will seek NRC approval for closure of GSI-191 based on the associated risk, the defense-in-depth measures in place, and adequate safety margin. STP is committed to investigating plant modifications including insulation removal and other measures (such as selective insulation reinforcement or debris transport mitigation) to preserve sufficient margins for nuclear safety if the analysis shows excessive risk, inadequate defense-in-depth or safety margin.

The project is based on a two-phase approach that addresses all the concerns related to GSI-191. For the initial phase, in 2011, a quantification was performed to determine if a risk-informed approach would be feasible [3]. Since it was shown to be feasible, the project proceeded to a licensing action in 2012 and 2013.

In both the initial risk analysis in 2011 and the 2012 final quantification, the risk was analyzed to be very small with adequate defense-in-depth and safety margin. That is, the change in risk was shown to be less than 1×10^{-6} in core damage frequency and less than 1×10^{-7} for large-early release frequency. Although previous realistic testing [4] had shown that chemicals were unlikely to affect the head loss in STP debris beds (sump strainers and fuel assemblies), conservative head loss estimates due to the presence of chemical products were assumed for the initial phase. In 2012, experimental data, specific to the STP units, continued to demonstrate chemical effects are not likely to cause large increases in head loss in STP prototypical post-LOCA environments. Nevertheless, conservative estimates of chemical effects were included in the 2012 quantification. Both defense-in-depth and safety margin were evaluated to provide assurance of low risk.

The methodologies and results from the first phase were presented in the following documents: analysis of results from the physical process solver; uncertainty quantification and RELAP5 thermal-hydraulic analyses [5]; LOCA Frequency analysis [6]; uncertainty quantification methodologies and examples [7]; jet formation research [8]; and chemical effects research and experimental design [9].

For the second phase, the results of the 2012 quantification are documented in this report (Project Summary) and the references. This information is provided as the technical basis for the NRC review of the Pilot Project.

Introduction & Background

The purpose of this document is to summarize the PRA[3] quantification supporting the STPNOC[4] license submittal to resolve concerns raised in GSI–191[5] "Assessment of Debris Accumulation on PWR[6] Sump Performance" at the STP[7] plants. GSI–191 describes the NRC concerns with potential blockage of the ECCS[8]. Over several years of study, the scope of concern has come to include the possibility of effects in the RCS[9] including core blockage from debris and in 2012, linkage to boric acid precipitation in the core. All GSI–191 concerns are related to the LOCA[10] in high energy (Class 1) piping that would result in the release of fibrous material and other potential debris to the ECCS Emergency Sump.

The purpose of the PRA quantification is to understand the risk and uncertainty in the as-built, as-operated plant associated with having fibrous insulation and latent debris in the STP containment buildings. In particular, the PRA quantification forms the basis for what has come to be referred to as Option 2b, "Mitigative Measures and Alternative Methods Approach" identified as a GSI–191 closure path by the NRC Staff in 2012 [2]. The basic elements of the Option 2b submittal are shown in Figure 1, reproduced from RG1.174[11] [10]. The PRA licensing elements addressed in the analysis are highlighted in Figure 1.

STPNOC operates two identical four-loop Westinghouse-designed NSSS[12]. Each NSSS operated by STPNOC is licensed for 3853 MWth. The NSSS is contained in, and protected by, a large dry containment building with approximately 3,410,000 ft^3 of free volume. The primary elements of the ECCS are the HHSI[13], LHSI[14], CSS[15], and RCFC[16]. The three trains mentioned in the descriptions for the HHSI, LHSI, CSS, and RCFC systems are completely independent and piped into a single RCS loop. In addition, the HHSI and LHSI can be independently directed to their respective hot leg at their full (run out) flow rate.

Early in 2011, STPNOC began a project to develop risk-informed closure strategies that would meet the intent of the NRC memorandum promulgated by Vietti-Cook in late 2010, while preparing a site-specific licensing submittal. Several public meetings were conducted to inform the NRC staff of the modeling approach and to solicit feedback on the applicability and use of the approach for resolving GSI–191. These meetings included supporting material so that members of the public, and especially other plants, could be informed as well: [12], [13], [14, 15, 16, 17, 18, 19, 20, 21, 22, 23, 24, 25, 26, 27, 28, 29].

In the meetings referred to above, STPNOC described the additional physical models and necessary experimental studies required to support enhancement of the PRA to include

[3]Probabilistic Risk Assessment
[4]The STP Nuclear Operating Company
[5]Generic Safety Issue 191
[6]Pressurized Water Reactor
[7]South Texas Project Electric Generating Station
[8]Emergency Core Cooling System
[9]Reactor Coolant System
[10]Loss of Coolant Accident
[11]Regulatory Guide 1.174
[12]Nuclear Steam Supply System
[13]High Head Safety Injection
[14]Low Head Safety Injection
[15]Containment Spray System
[16]The Reactor Containment Fan Coolers

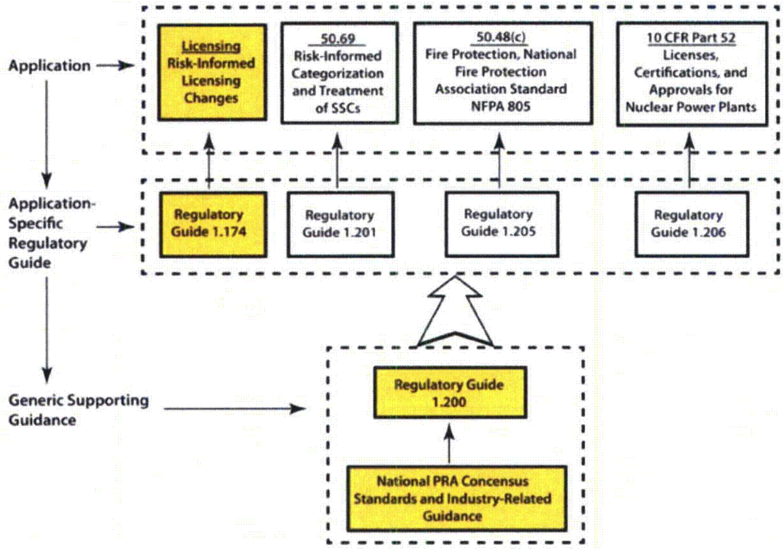

Figure 1: Reproduction of "Figure 1, Relationship of Regulatory Guide 1.174 to other risk-informed guidance" [10, Figure 1, Page 6] showing the elements used in the Option 2b analysis.

the phenomena associated with concerns raised in GSI–191. The overall approach that was adopted caused minimal impact to the PRA Model of Record [30].

The method of analysis uses an integrative approach to explicitly provide failure probabilities for a few post-LOCA basic events of the STPNOC plant-specific PRA (that is, Module 1 of Figure 2). These basic event probabilities are estimated in a separate module (that is, Module 2 of Figure 2) by modeling the underlying physical phenomena of the basic events and by propagating the uncertainties arising from the physical models. The analysis framework shown in Module 2 is called CASA Grande[17] and is explained in detail in Volume 3. The added basic event probabilities are shown as the dotted lines going from Module 2 to Module 1 in Figure 2. A conceptual outline of the uncertainty quantification process used in Module 2 of Figure 2 is illustrated in Figure 3. More details regarding the uncertainty quantification are available in Volume 3.

The added basic events that are related to the recirculation phase of LOCA and shown as the dotted lines coming from the engineering models in Figure 2 are solved outside the PRA in an uncertainty quantification framework. An illustration showing the typical process of uncertainty quantification is shown in Figure 3. As shown, the process models distributions developed in different contexts such as data measurement analysis and expert judgment. In the STPNOC risk-informed methodology, multivariate distributions, which might have complicated sampling and uncertainty propagation, have been avoided by assuming independence between parameters, where possible, and by enforcing explicit conditional dependencies, where appropriate.

[17]Containment Accident Stochastic Analysis (CASA) and Grande refers to the STPNOC large, dry containment

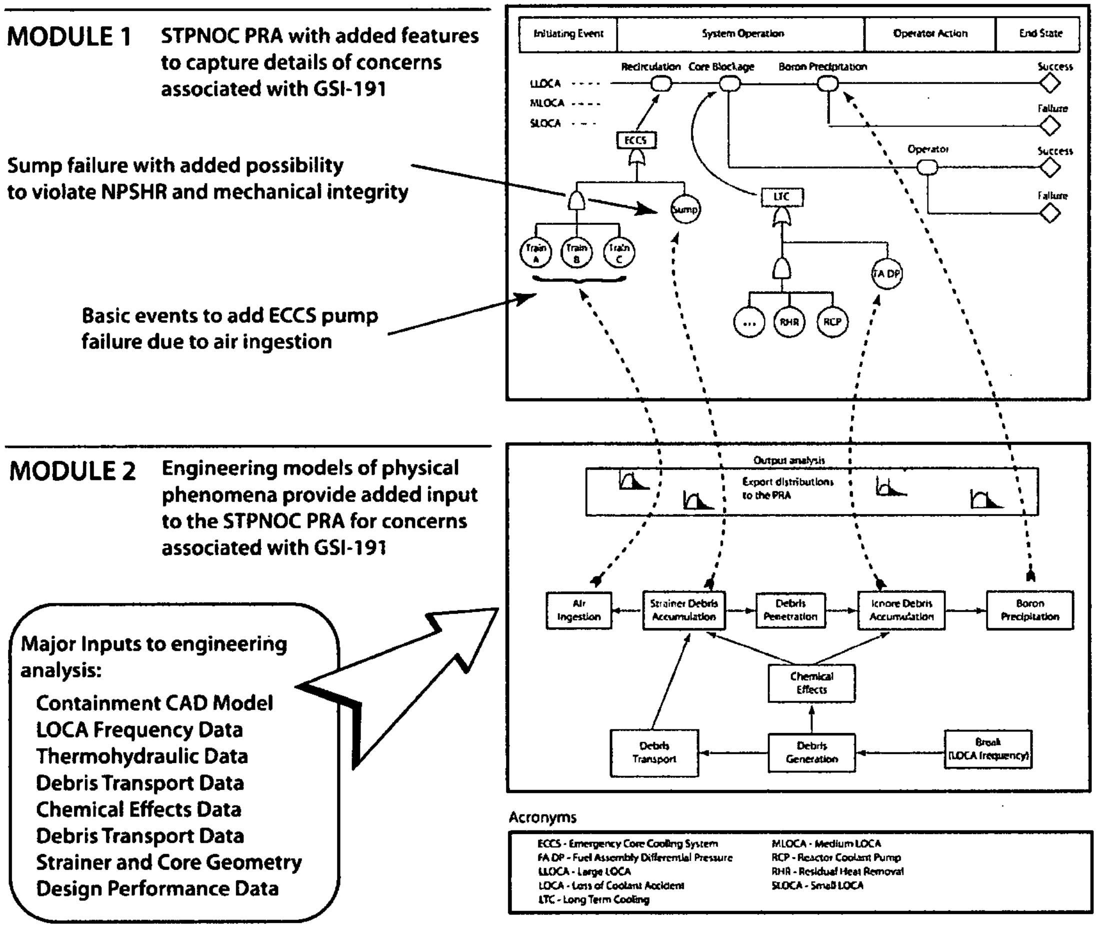

Figure 2: Illustration of the engineering model input to the PRA used in the Option 2b GSI-191 resolution.

In some cases, the distributions needed for the PRA involve relatively broad distributions which need to be carefully sampled so that the "tails" are properly accounted for. In general, NLHS[18] strategies have been developed to properly represent distributions with long tails, especially in LOCA frequencies.

A quality assurance plan was developed to include standard STPNOC practice for PRA assessments. Over the nearly two-year project duration, (nominally weekly) technical review teleconferences were conducted and supplemented at critical product development steps with on-site reviews. In addition, monthly face-to-face Technical Team meetings were held in 2012.

In general, the STP PRA analyst (STP Technical Team Lead) is responsible for review and verification of the PRA inputs developed. The STP PRA analyst review is supplemented by independent critical peer review intended to help disclose any overlooked technical gaps

[18]Nonuniform Latin Hypercube Sampling

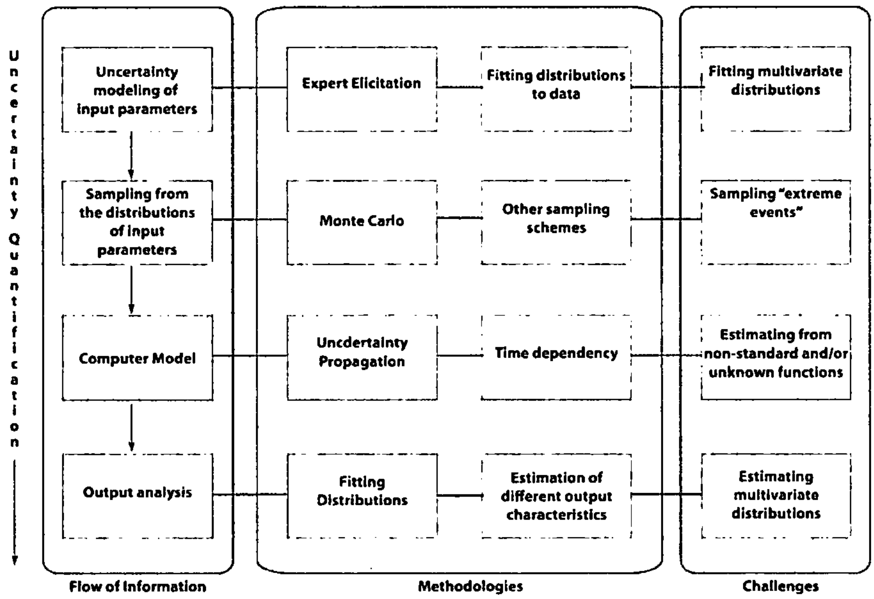

Figure 3: Conceptual illustration of the uncertainty quantification process used to add detail (basic events) to the STPNOC PRA analysis in the LOCA initiating event sequences for the Option 2b analysis.

that would compromise results and, although the analysis is developed for the industrial setting, also help ensure that the overall product is academically defensible. Independent Technical Oversight also helped to further focus the analysis efforts.

The overall quality assurance plan is illustrated in Figure 4 as a flow chart. Due to the diverse areas of investigation in the GSI-191 scope, the PRA inputs are developed by several experts. The CASA Grande integrating framework uses the inputs to generate the two main inputs to the PRA: the sump demand failure likelihood and the in-vessel cooling failure likelihood (for each category of LOCA and all possible equipment configurations). These elements are documented by the vendor and the normal STP vendor document review process is followed to ensure that those elements are suitable for input to the PRA. The overall STPNOC Pilot Project[19] quality assurance methodology is expected to be similar to most utilities' processes for PRA applications and is consistent with industry PRA standards, practices and procedures [see 31].

The technical and RG1.174 documentation that establishes a technical basis to close GSI–191 in an Option 2b approach consists of several volumes:

- Volume 1, Summary (this volume);
- Volume 2, The PRA analysis and quantification;

[19]STPNOC Risk-Informed GSI–191 Closure Pilot Project

- Volume 3, The engineering analysis supporting the added basic events and top events needed by the PRA to address the concerns raised in GSI–191;
- Volume 4, Quality assurance documentation, approach, and summary;
- Volume 5, Oversight (four volumes: 5.1, 5.2, 5.3, and 5.4); and
- Volume 6, Comment and Request for Additional Information Resolution.

Additional documentation (for example the PRA Model Revision 7 and support calculations) are also available through reference.

The remainder of this document is developed to reflect the RG1.174 outline. That is, starting with Part I (Proposed Change), through Part VI (Documentation), the section numbering and the names of its major parts are intended to correspond to the outline of RG1.174. A summary of the STPNOC Pilot Project Oversight activity is given in Part VIII Independent Technical Oversight. Many acronyms are used throughout the text. Each is expanded in a footnote when first used. In addition, Part IX provides both the complete name and a short description for most of the acronyms.

As mentioned earlier, the first Part I through Part VI, correspond to the RG1.174 outline. A checklist (Reg. Guide 1.174 Checklist) is provided as an additional resource for cross referencing RG1.174 items with the text in this document. Appendix B is provided to give an overview of the models implemented in the STPNOC Pilot Project and how they correspond to those recommended in NEI 04-07 [32]. Finally, Appendix C is a detailed summary of the STPNOC DID[20] measures that address the concerns raised in GSI–191.

[20]Defense–in–Depth

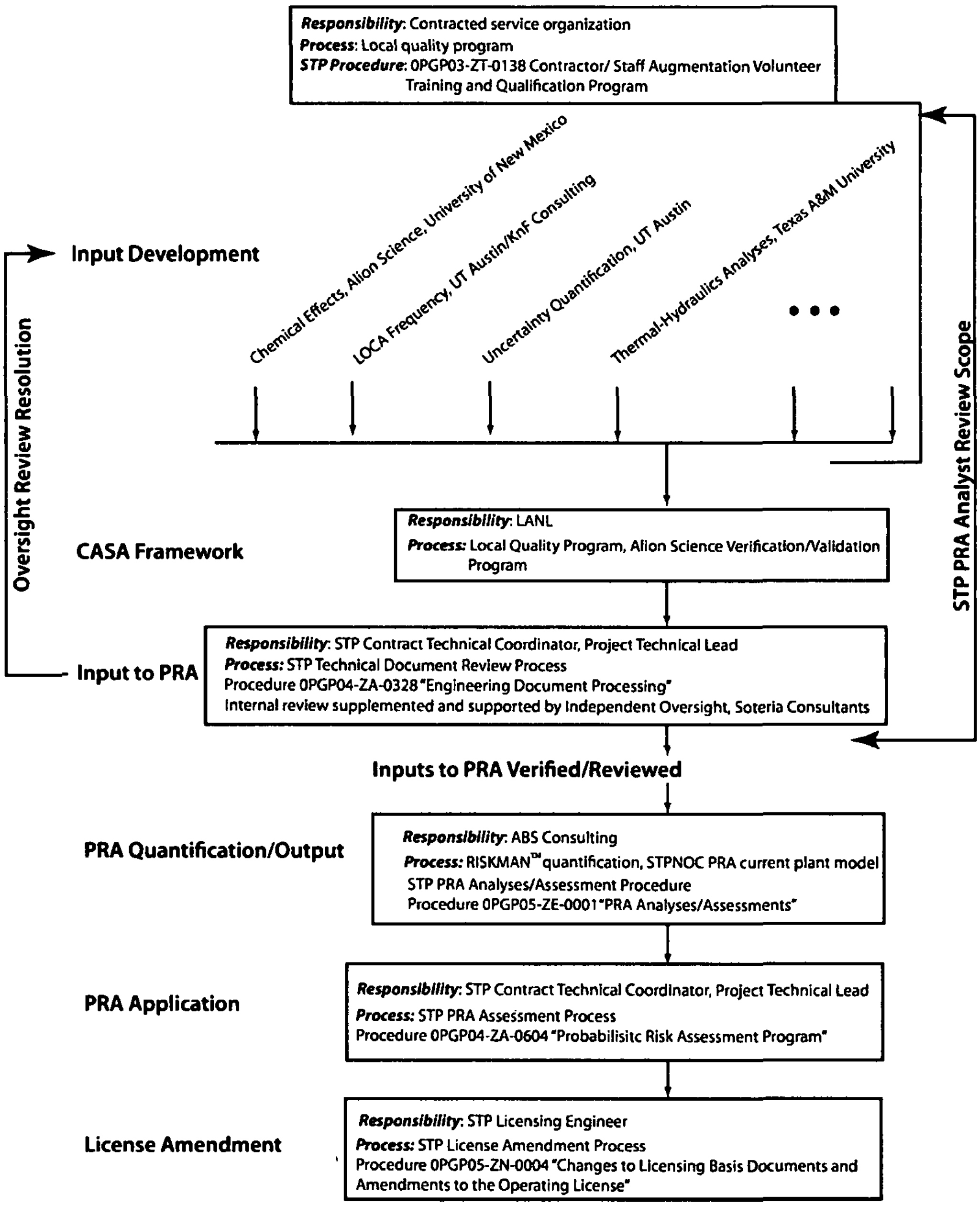

Figure 4: Illustration of the major elements of the STPNOC quality assurance process for risk-informed closure of GSI-191.

Part I

Proposed Change

Part of the STPNOC plant licensing basis change considers long-term core cooling as identified in 10 CFR §50.46 following a LOCA. Long-term cooling is supported by the ECCS which system includes the safety-related CSS, the HHSI, LHSI, and the RHR[21] system. The STPNOC licensing basis requires these particular systems to operate with high probability following a LOCA. In addition, the licensing basis requires evaluation of uncertainty associated with proper operation.

In this licensing basis change, STPNOC uses the guidance provided in RG1.174 to explicitly quantify the probability and uncertainty associated with the operation of the ECCS following a LOCA while verifying that DID measures are in place to prevent or mitigate any postulated GSI–191 events such that long-term core cooling is ensured with adequate safety margin. In the current license basis, neither the probability nor the uncertainty that long-term cooling will operate properly following LOCA is quantified. Therefore, the licensing basis change is to incorporate the probability and uncertainty associated with long-term cooling success of the as-built, as-operated plant (as required in the license basis change). This requires NRC approval where the cumulative risk is shown to be very small [10, Figures 4 and 5, page 16].

[21]Residual Heat Removal System

History of Defense in Depth and Safety Margin Activities

Since the inception of the GSI–191 issue, STP has made significant improvements to processes, programs, design, and operation that, in the unlikely event of a LBLOCA[22], would mitigate potential consequences. These improvements include design modifications to the plant hardware, operator training, and procedures. Appendix C is provided to help review the current status and show what is in place to address those concerns raised in GSI–191 before the the STPNOC Pilot Project started. In the following section, the primary activities from that history that were already in place are summarized.

Procedures and Activities in the Licensing Basis

Before the STPNOC Pilot Project started, STPNOC had already taken steps in STP design and operation to help eliminate, or greatly reduce, effects from the concerns raised in GSI–191 on long-term cooling at STP. Some of the steps taken include:

- installing very large, uniform-loading ECCS strainers having strainer flow area approximately 10 times that of the strainers originally installed;
- modifying the STP Emergency Operating Procedures to terminate containment spray early as a conditional action step to conserve RWST[23] inventory;

[22]Large Break Loss of Coolant Accident
[23]Refueling Water Storage Tank

- removing effectively all Marinite (calcium silicate) insulation from the containment building;
- reworking or replacing PWSCC[24]-susceptible welds in the steam generators and the pressurizer safe ends; and
- performing a comprehensive post-maintenance containment cleanup and inspection following refueling outages to help ensure the removal of material that would cause strainer blockage.

The following primary procedures and activities are implemented that directly or indirectly bear on mitigating or eliminating the concerns raised in GSI–191:

- "Condition Reporting Process," STPNOC plant procedure, 0PGP03-ZX-0002: The STPNOC process used to identify plant management, operations, and work control of any deficiencies or issues that may arise. This process requires identification and evaluation of the severity and required actions, to be taken as necessary for safe operation.
- "PRA Analyses/Assessments," STPNOC plant procedure, 0PGP05-ZE-0001: the STPNOC process used in PRA as the basis for applications and risk-based decision making.
- "Design Change Package," STPNOC plant procedure, 0PGP04-ZE-0309: the STPNOC engineering design change process governing all design changes. Section 4 of the design change checklist and the supporting descriptions specifically address maintaining the assumptions used for the engineering models in the STPNOC Pilot Project containment analysis.
- "Inspection of Containment Emergency Sumps and Strainers Unit #1 1-A, 1-B, 1-C Unit #2 2-A, 2-B, 2-C," STPNOC plant procedure, 0PSP04-XC-0001: the procedure satisfying Technical Specifications for ECCS sump operability. The specific procedure purpose is to provide instructions for cleanliness and structural inspection of containment emergency sumps and strainers 1-A, 1-B, 1-C or 2-A, 2-B, 2-C required by Technical Specifications 4.5.2.d and 4.5.3.1.1.
- "Initial Containment Inspection to Establish Integrity," STPNOC plant procedure, 0PSP03-XC-0002: the STPNOC process that ensures no loose debris which could be transported to the containment sump and cause restriction of pumps' suctions during LOCA conditions is present and is the procedure that satisfies Technical Specifications 4.5.2.c.1, 4.6.1.7.1, 4.6.1.7.4, and 3.6.1.7.b.
- "ISI[25]," STPNOC plant procedure, 0PSP11-RC-0015: This procedure ensures that the following requirements of Technical Specifications 4.0.5 /4.4.10 have been satisfied: completion of the ISI examinations of STP piping and component welds in accordance with the schedule requirements of the ASME Boiler and Pressure Vessel Code, Section XI (2004 Edition No Addenda); ISI of STP piping and equipment; component supports (excluding snubber assemblies [pin-to-pin]) in accordance with the schedule requirements of the Code; completion of the Inservice Service Inspections of the STP containment metal liner in accordance with the schedule requirements of the ASME Boiler and Pressure Vessel Code; and completion of the examinations of the

[24] Primary Water Stress Corrosion Cracking

[25] ASME Section XI Inservice Inspection

STP reactor coolant pump flywheels in accordance with the requirements of Regulatory Guide 1.14.

- "Transient Cycle Counting Limits," STPNOC plant procedure, 0PEP02-ZE-0001: The STPNOC process that provides for the monitoring of the number of primary and secondary plant operations that are explicitly considered as design transients for the NSSS primary system and components. This procedure includes the transients listed under the normal, upset, and test conditions in UFSAR Section 3.9, except for particular transients discussed in Step 1.2 of the procedure. This procedure is based on the recommendations of WCAP-12276.
- "Shielding," STPNOC plant procedure 0PRP07-ZR-0004: the STPNOC process for a consistent method of determining the need for, requesting, evaluating, installing, modifying, accounting for and removing shielding at STP. In particular, 0PRP07-ZR-0004 requires inspection for signs of wear such as cracking of the blanket material, damaged or corroded grommets, or other signs of physical damage. The inspection is performed prior to each removal and storage and thereby minimizes the possibility that transient lead can be introduced in the post-LOCA sump chemistry.

1 Method of Analysis

The method of analysis uses a RG1.174 approach to explicitly provide the probabilities for a few post-LOCA basic events of the STPNOC plant-specific PRA. This has been done by modeling the underlying physical phenomena of the basic events and by propagating uncertainties in the physical models. In particular, the simplistic demand recirculation failure probability is replaced with the following basic events:

- Air ingestion through the sump screen;
- Pressure drop due to buildup of debris on the sump screens with chemical effects, resulting in NPSHA[26] dropping below NPSHR[27] for the ECCS pumps;
- Boron precipitation;
- Core blockage with chemical effects; and
- Strainer mechanical collapse.

In order to assess the potential risk to long-term core cooling due to the issues raised in GSI–191, a theoretical "perfect plant" is hypothesized so its performance regarding the concerns raised in GSI–191 can be compared to the as-built, as-operated plant. Such a plant would not be subject to the postulated failure mechanisms that motivated GSI–191 and neither the as-built, as-operated plant nor the theoretically perfect plant would have any changes in commitments to current long-term cooling requirements or performance of the ECCS.

By adopting a RG1.174 approach that explicitly assesses the potential risk of the issues raised in GSI–191 to be very small but also ensuring DID and safety margin for any unlikely, but potentially severe scenarios, STPNOC would avoid significant cost and worker radiation exposure that would be incurred if using a so-called "bounding deterministic approach". Cost estimates for the two STPNOC units has been estimated at $50,000,000 to $60,000,000, consistent with other cost estimates in the industry. Compared to retaining the current design, radiation exposure to workers is also very high: 100REM to 200REM.

[26] Net Positive Suction Head Available
[27] Net Positive Suction Head Required

Part II
Engineering Analysis

Title 10 "Energy" of the Code of Federal Regulations (CFR) applies to all domestic commercial nuclear power stations. One of the several legal requirements defined in 10 CFR§50.46, "Acceptance criteria for ECCS for light-water nuclear power reactors," is that events leading to a loss of long-term core cooling must be mitigated with high probability. The main purpose of the ECCS is to mitigate hypothesized LOCA events by supplying cooling water to the reactor. LOCA events can be triggered by a valve failure or a structural failure and the ECCS is designed to mitigate the "worst case" of these failures with high probability.

Since 2001, GSI–191 has eluded resolution despite significant efforts by industry and the NRC. Although recent thought has been given to risk quantification [33, for example], and early recognition of the need for risk evaluation was identified [34, for example], serious investigation into risk quantification had not been undertaken. Instead, resolution had followed a classical deterministic approach. STPNOC's view, following an initial quantification [3], is that a risk-informed resolution path should be pursued in preference to a deterministic approach, thereby quantifying the safety margins and identifying any scenarios that pose significant risk in GSI–191.

The STPNOC PWR RCS operates at temperatures higher than 650°F. As a consequence, high efficiency insulation is used to prevent exceeding local and general environmental temperatures in the enclosed space of the reactor containment building. NUKON fiberglass insulation is specified for most high-temperature Class I piping and components in the STP containment buildings. The Reactor Vessel and Reactor Vessel Head are notable exceptions insulated with RMI[28].

In addition to the containment building application, insulation similar to NUKON is installed in high temperature steam cycle applications, piping, heaters, valves, etc. Because fiberglass insulation is in general usage, STPNOC has a great deal of experience in installing and removing it. Processes and procedures have been in place for many years and, as a result, the plant staff has significant experience with fiberglass insulation leading to maintenance efficiencies.

During the recirculation phase of a hypothesized LOCA, various materials (e.g., fibrous insulation ablated from piping and components insulated with NUKON, paint chips dislodged from painted surfaces, latent debris from inefficient containment building housekeeping, ablated concrete, and chemical precipitants) may cause high differential pressure on the ECCS strainers or reactor core fuel assemblies if the materials are transported to the containment emergency sump and then to the ECCS filter screens. If the conditions assumed in some of the more extremely pessimistic hypothesized cases were realized, the resulting ECCS filter screen differential pressure could be sufficient to cause core damage due to the loss of one or more trains of the ECCS. Filter inefficiency may lead to blockage of all the fuel assemblies which also may result in core damage. In addition to the concerns associated with differential pressures, boron precipitation could cause reduced heat transfer in the core.

The GSI–191 PRA shows the risk to core damage or large early release due to the concerns raised in GSI–191 in the as-built, as-operated design to be very small. In the analysis, the risk of core damage and/or large early release is quantified for a hypothetical plant designed and operated in the same

[28]Reflective Metal Insulation

manner as the STP plants except that it is not subject to the concerns raised in GSI–191. The STPNOC PRA meets the ASME/ANS PRA Standard as Capability Category II and has successfully provided the technical basis for several risk-informed applications at STPNOC, for example RMTS[29] [35, 36]. PRA is relied upon in this analysis to quantify the risk associated with the concerns raised in GSI–191.

The engineering analysis and experimental support for the proposed license basis change are both detailed and broad in scope, commensurate with the perceived complexity of the issues raised in GSI–191. The inherent uncertainty of the analysis is addressed through the sampling methodology in the uncertainty quantification and by adopting maximum or reasonably high bounds where the analyses or experimental data are incomplete. For example, NLHS is used in the uncertainty propagation methodology to emphasize random samples from the extreme tails of many uncertain parameters. In particular, when defining random-break scenarios, the methodology ensures that DEGB[30] conditions are included for every weld in the containment within the spectrum of random break sizes that are chosen. NLHS permits a more precise quantification of variability near the extreme conditions for the same number of random scenarios without biasing the propagation of uncertainty. Traditional engineering limits are used for equipment performance assessment. Examples include NPSHR for ECCS pumps, air entrainment in the ECCS supply lines, and cooling flow that is required to remove decay heat.

The findings of this analysis indicate that the risk associated with the issues raised in GSI–191 is very small and well within the Commission's safety goal. There are several reasons, that include adopting realistic uncertainty analysis and accounting for the evolution of processes over time, that contribute to a minimal risk result. However, one of the most important contributors to the risk's being small is that along the timeline of the issues motivating GSI–191, STPNOC took several steps in the design of the ECCS, containment maintenance, operation of the CSS, and insulation design that significantly increased the safety margin against the issues that were raised in GSI–191. The most significant change in design was the introduction of very large ECCS sump strainers that, under realistic assumptions of LOCA behavior, would prevent NPSHA from dropping below NPSHR for the ECCS pumps.

Some insulation types have shown increased head loss in fiber debris beds. STPNOC took steps to remove, or to preclude the installation of, insulation (such as Microtherm and calcium silicate) that could be responsible. To prevent introduction of a direct debris path due to strainer damage, the exposed strainer modules have an added protective fence. Taking these steps after the concerns were originally raised in GSI–191 and within the context of continuous performance improvement, has greatly improved the safety margin and assurance of DID in the as-built, as-operated STP plants.

1 Analysis in Module 2

The following description of information flow is intended to provide a summary of the CASA Grande analysis process which is closely aligned with the engineering intuition used to formulate the basic events supplied to the PRA. Introduced in the Introduction & Background discussion earlier, the notional setting for the engineering analysis is captured as Module 2 in Figure 2 showing certain basic events provided to the PRA using the uncertainty quantification process summarized in Figure 3. Reviewers familiar with

[29] Risk Managed Technical Specifications

[30] Double-Ended Guillotine Break

deterministic analyses of the post–LOCA accident progression often carry a mental list of information that is needed to fully calculate the outcome from a single complete accident scenario. This summary traces a single accident from start to finish and enumerates both the random variables that are sampled during the analysis and the primary performance metrics that are calculated from the outcome of the scenario.

It is often easier to understand statistical sampling strategies after a firm understanding of the basic event is established. In this case, the basic event consists of a single accident progression that is initiated by a broken pipe and continues for 30 days. The most basic statistical sampling approach consists of "brute force" repetition of this event under many, many random conditions that are introduced in proper proportion. This summary is not intended to provide a literal implementation guide for the CASA Grande framework because of the complexities inherent in the analysis implemented to achieve both numerical and statistical efficiency. Statistical sampling strategies are discussed below and further in Section 1.3.

This summary provides a structured context for conveniently referencing additional detail provided in Volumes 2–6. Volume 3 contains detailed descriptions of most physical models that are referenced here. The following outline focuses on principal physics equations that support quantification of time–dependent quantities like debris mass inventory, and differential pressure, but the high–level description of accident progression also provides a basis for understanding where specific topical concerns fit into the integrated analysis, and illustrates how prior dependencies in the accident conditions can affect the relevance of each concern.

By comparison to predictive physics models like RELAP that enumerate field equations and constitutive relationships, CASA Grande embodies only mass conservation in the form of a first–order rate equation to track debris fractions in the containment pool. Energy balance is addressed in principle by external calculations of pool temperature. In this respect, CASA Grande is primarily an uncertainty propagation tool, but the timeline of the accident progression is determined by tracking debris through the system circulation history. The timeline supports externally calculated parameters such as decay heat, pool temperature, operational configurations (EOP[31] response), chemical product formation, coatings degradation, and provides a basis for comparison to time–dependent performance metrics like NPSHA, and core debris loading relative to boron dilution strategies like switching to ECCS hot leg injection.

1.1 Structured Information Process Flow

1. Set plant failure state (number of trains, and specific pumps available). Failure state determines available flow rates through each train and guides operator action via EOPs.

2. Randomly select a weld type/case based on relative frequency of break occurrence. Relative frequencies reflect susceptibility to degradation (failure).

3. Randomly select a specific weld from this type/case [6] (equal probability among all welds of same type/case)[37]. Weld location defines $P(x, y, z)$, and HLB[32] or CLB[33] condition. Each weld location has a pre–calculated list of insulation targets that can be "seen" in every direction. Concrete walls are

[31] Emergency Operating Procedure
[32] Hot Leg Break
[33] Cold Leg Break

the only feature that can shield insulation from potential damage. We assume pipes and large equipment to have no effect on a ZOI[34].

4. Conditional upon having a break for this specific weld type/case, sample a break diameter that is consistent with NUREG–1829 [38]:

$$D_{break} \sim F_{D_{break}|\,weld\,case}. \quad (1)$$

Record break contribution to SLOCA[35], MLOCA[36], or LBLOCA category. The designation of SLOCA, MLOCA, or LBLOCA becomes an explicit correlation for many following physical variables, both user–specified input (like typical times for operator action, chemical head–loss increase, containment pool volume, etc.) and externally computed trends (like temperature histories).

5. Randomly select a complete temperature history $T(t)$ from appropriate correlations of thermal–hydraulic trends for SLOCA, MLOCA, or LBLOCA events. The temperature history drives water properties, assumed arrival of chemical products, and $NPSH_{margin}$.

6. Calculate radii $R_{i,j,k}$ of the three damage zones indexed by $i = 1, 2, 3$, debris sizes (fines, small pieces, large pieces, or intact blankets) indexed by $j = 1, 2, 3, 4$, and target type indexed k, where $k \in \mathcal{K}$ indexes insulation products in containment. We distinguish three sets indexed by k: $\mathcal{K}$ denotes insulation products, $\mathcal{F}$ denotes fiber–based insulation, and $\mathcal{L}$ denotes all types of debris, including insulation and other debris such as unqualified coatings and crude particulate; so, $\mathcal{F} \subset \mathcal{K} \subset \mathcal{L}$. The $R_{i,j,k}$ damage zones for Nukon are scaled to the maximum damage radius for insulation k. Figure 5 is an illustration that shows the nomenclature of damage for a hypothetical break that has its damage radii truncated by a wall.

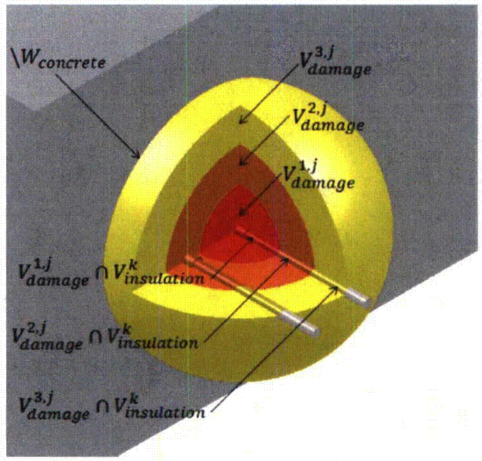

Figure 5: Illustration of a hypothetical spherical break (double–ended guillotine) damage zone truncated by a wall with the nomenclature of the damage characteristics; see eq. (2).

7. If $D_{break} < D_{pipe}$ then choose random direction perpendicular to pipe according to $\phi \sim U(0, 2\pi)$. Else, ϕ is assigned a flag that indicates a spherical ZOI.

8. Calculate intersection of damage zones with insulation targets and clip by concrete walls to obtain amount of debris in each damage radius and debris size (i, j, k), and convert volume to mass:

$$M_{i,j,k} = \rho_k \left| \left(V^{i,j}_{damage}(\phi) \cap V^{k}_{insulation} \right) \setminus W_{concrete} \right| . \quad (2)$$

Here, the "$\setminus W_{concrete}$" designates exclusion of those insulation targets not dam-

[34] Zone of Influence
[35] Small Break Loss of Coolant Accident
[36] Medium Break Loss of Coolant Accident

aged due to structural concrete blocking the break blast.

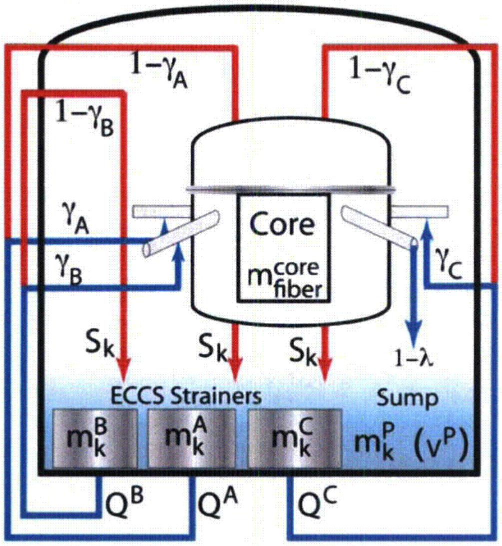

Figure 6: Illustration of the processes local to the ECCS screen that contribute to direct pressure drop on the screen that lead to decreased $NPSHA$ and downstream effects such as fiber penetration contributing to m^{core}_{fiber} and bubble formation during the recirculation phase.

9. Apply transport logic diagram to obtain all ZOI–generated debris mass arriving at the pool. Complex transport logic is represented here via the operator $F_{transport}$:

$$m^P(0) = F_{transport} \otimes M. \qquad (3)$$

The transport logic captures, e.g., erosion of fibers from large pieces to fines, in transforming the vector M of $M_{i,j,k}$ to the vector $m^P(t)$ of $m^P_{i,j,k}(t)$ $t = 0$.

10. Introduce fixed quantities of non–ZOI debris types (those in $\mathcal{L}$ but not $\mathcal{K}$ and not addressed above) like crude particulate, latent debris, and unqualified coatings debris.

11. Apply fill up transport fraction, F^{ℓ}_{fill}, to train ℓ's strainer sump cavity. This mass of debris is initially resident on each strainer, in addition to all other debris constituents that arrive over time:

$$m^{\ell}_{i,j,k}(0) = F^{\ell}_{fill}\, m^P_{i,j,k}(0). \qquad (4)$$

12. At each time t, assume homogeneous mixing in the pool:

$$C^P_{i,j,k}(t) = m^P_{i,j,k}(t)/V^P(t). \qquad (5)$$

While this form is never used explicitly, it is helpful to think about debris mixing, transport and accumulation as concentration.

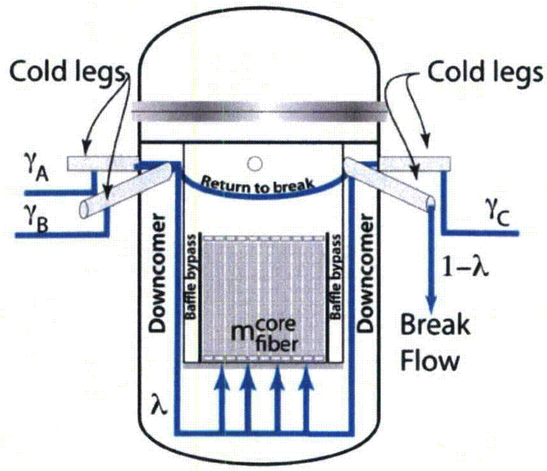

Figure 7: Illustration of the flow paths in the reactor vessel used to establish m^{core}_{fiber} accumulation and fiber bypass during the recirculation phase of ECCS operation in a medium or large cold leg break scenario.

13. Solve coupled differential equations for mass in pool, mass on strainer and mass on core (see Figures 6 to 7 for the

nomenclature setting):

$$\frac{d}{dt}m_k^P(t) = S_k(t) - \sum_{\ell=A,B,C} \frac{d}{dt}m_k^\ell(t) - \frac{d}{dt}m_k^{core}(t)\Big|_{k\in\mathcal{F}}, \ \forall k \in \mathcal{L} \qquad (6a)$$

$$\frac{d}{dt}m_k^\ell(t) = f\left(\sum_{k\in\mathcal{L}} m_k^\ell(t)\right)(Q^\ell(t)/V^P(t))m_k^P(t) - \eta\nu\, m_k^\ell(t), \ \forall k \in \mathcal{L} \qquad (6b)$$

$$\frac{d}{dt}m_k^{core}(t) = \lambda\eta \sum_{\ell=A,B,C} \gamma_\ell m_k^\ell(t), \ \forall k \in \mathcal{F}, \qquad (6c)$$

where sources $S_k(t)$ of debris type k can be time dependent, flow split λ is the fraction of ECCS injection that passes through the fuel, and flow split γ is the fraction of total strainer flow that is injected. The complement $(1-\gamma)$ is the fraction of total strainer flow passed to containment spray, and the complement $(1-\lambda)$ is the fraction of ECCS injection that bypasses the core. For CLB λ is determined based on the time–dependent boil–off rate. For HLB $\lambda = 1$. For simplicity in writing the equations here, we suppress additional subscripts and just index the masses by debris type $k \in \mathcal{L}$. That said, these other indices matter in implementation. For example, the last term in Equation 6a is only present when the k index indicates fiber, but it is also only present when the size index indicates fines. Constraint Equation 6c is only written for fiber, but is also only present when the size index is fines.

14. Given histories of fiber and particulate debris thickness, $\delta(t)$, on the strainer, compute time–dependent head loss across each strainer according to:

$$\Delta P^\ell(t) = H(m^\ell(t), Q^\ell(t))N(5,1)\Phi_{ch}(t) \quad (7)$$

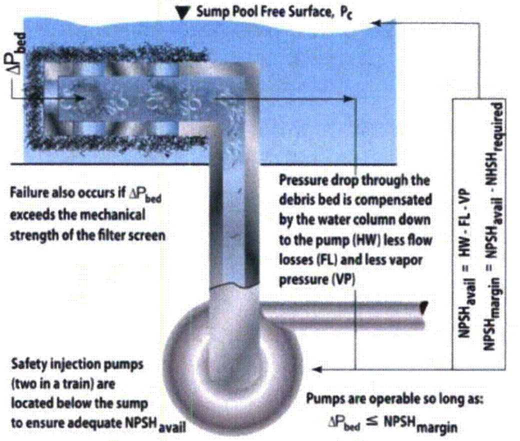

Figure 8: Illustration of the sump pool, screen, and pump annotated with the head losses to the SI pump suction. Also shown are the failure criteria associated with the pressure losses to the pump.

where, the function H is given by NUREG/CR–6224 [39, Appendix B] with arguments given by the vector $m^\ell(t)$ of $m_k^\ell(t)$ for all $k \in \mathcal{L}$ and velocity via the flow rate $Q^\ell(t)$ and where $N(5,1)$ is a truncated normal random variable with a mean of 5 and unit variance and where

$$\Phi_{ch}(t) = \begin{cases} 1, & \delta(t) < \frac{1}{16}'' \text{ or } T(t) > N(140,5) \\ \mathcal{E}, & \text{otherwise.} \end{cases}$$

Here, Φ_{ch} takes value 1 if the thickness is below 1/16-th of an inch or the temperature exceeds the specified normal random variable, centered on 140°F. Otherwise, Φ_{ch} takes the value of a shifted, and truncated, exponential random variable, which we denote by $\mathcal{E}$.

15. Compare time–dependent head loss to time–dependent NPSH and record the scenario as a failure if:

$$\max_{t,\ell}\left[\Delta P^\ell(t) - NPSH_{margin}(t)\right] > 0, \quad (8)$$

i.e., we record a failure for this scenario if anywhere along the 30–day time history the head loss exceeds the NPSH margin for any strainer $\ell = A, B, C$.

16. Compare time–dependent head loss to fixed mechanical collapse criterion and record the scenario as a failure if:

$$\max_{t,\ell} \Delta P^{\ell}(t) > \Delta P_{mech}, \tag{9}$$

where ΔP_{mech} is the design strainer mechanical strength inferred by the pressure drop across the strainer.

17. Given time–dependent head loss, calculate time–dependent gas evolution and record the scenario as a failure if:

$$\max_{t,\ell} F_{void}(\Delta P^{\ell}(t)) > 2\% \tag{10}$$

18. For CLB, compare the time–dependent fiber accumulation on the core against the assumed $7.5gm/FA$ threshold. Record a scenario failure if $\max_t m^{core}(t) > 7.5gm/FA$.

19. Given time–dependent fiber on the core, record scenario success for all HLB.

20. If any performance threshold is exceeded for the scenario then record a failure.

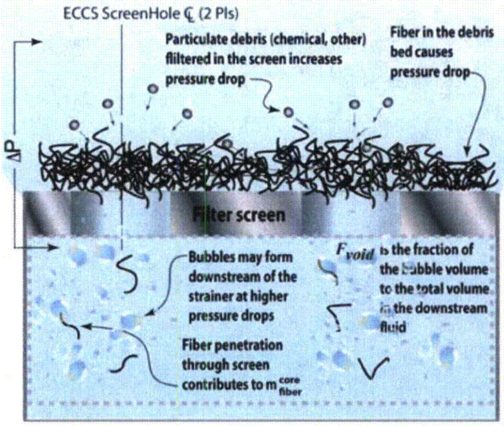

Figure 9: Illustration of the processes local to the ECCS screen that contribute to direct pressure drop on the screen that lead to decreased $NPSHA$ and downstream effects such as fiber penetration contributing to m^{core}_{fiber} and bubble formation during the recirculation phase.

Figure 9 is an illustration of the various processes listed above that need to be evaluated in GSI–191 for ECCS performance during the recirculation phase of operation.

Again, as we indicate above, these steps sketch the nature of what would be calculated within CASA Grande, if it were designed to run for a single scenario. That specific scenario includes numerous random realizations including: the selection of the specific weld location where the break occurs, the effective size of the break, the temperature profile, the direction of the break on the pipe, and more. Further, while not always made explicit in the above description, many of the steps outlined depend on the specifics of this scenario. To construct a Monte Carlo estimator of the failure probability, these steps would be replicated many times. However, we do not simply construct a so–called naive Monte Carlo estimator. Rather, we use techniques to reduce the variability of the estimator of the failure probability, and techniques to propagate uncertainties—such as the epistemic uncertainty in the initiating frequency—to the PRA, where these failure probabilities become branch fractions at the top event. Among our variance reduction techniques, we enumerate breaks at each weld location, and we employ a NLHS estimator, which ensures we sample low-probability large-break events. Both the stratification across weld locations and the NLHS estimator require us to use proper probabilistic weights associated with the specific scenario when constructing the estimator.

1.2 Method Comparisons with Prior Practice

Although the STPNOC Pilot Project adopted many of the commonly used engineering models developed for GSI–191, some of the engineering analyses differ from practices adopted for deterministic evaluations. The practices that differ were adopted to facilitate a risk–informed approach to evaluate the concerns raised in GSI–191. In this section, the differences between the risk–informed practice and previously–adopted (deterministic evaluations) practice are summarized or the risk–informed practice is summarized where describing differences breaks down. As mentioned previously, a summary table of the differences is also provided in Appendix Appendix B.

1.2.1 Unqualified Coatings

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.

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; and
- Although essentially all the unqualified coatings fines would transport to the strainer, the transport for the chips would be significantly reduced.

1.2.2 Blowdown Debris Capture

The STPNOC Pilot Project 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 [40]. 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 DDTS[37] [41] rather than a simple bounding value.

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; and

[37] Drywell Debris Transport Study

- Breaks outside secondary shield wall in the annulus.

1.2.3 Washdown Transport

The methodology used for the washdown transport 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). Note that neglecting the retention of small pieces on the concrete floors is a significant conservatism since the analysis documented in an appendix to the risk–informed debris transport calculation shows that the flow velocity would generally not be high enough to transport the debris[42].

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.

1.2.4 Debris Distribution at the Start of Recirculation

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 [43]. 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.

1.2.5 Time–Dependent Transport

Although some investigators have used time–dependent transport in their analysis, most deterministic analyses assume that debris transports instantaneously to the sump strainers at the start of recirculation. In the STPNOC Pilot Project analysis, different assumptions as listed below were adopted.

- 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 the 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.

1.2.6 Chemical Release and Precipitation Model

Several scenarios were investigated using the WCAP–16530 formula for chemical release. The scenarios used different combinations of liquid temperature, pH, water volume, and fiber quantity for several different break sizes up to DEGB. Results of the investigation indicate that little or no precipitates are formed for the majority of break sizes and conditions. However, for some extreme scenarios (when maximum temperature, and/or maximum fiber quantities are assumed), significant chemical precipitation is predicted to occur using the deterministically–based calculation (WCAP 16530).

1.2.7 Conventional Head Loss Model

The head loss model adopted in the STPNOC Pilot Project is nominally based on the NUREG/CR–6224 head loss correlation developed by the NRC in support of evaluation of the strainer clogging issue in BWRs. NUREG/CR–6224 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 the tests conducted in support of development, the NUREG/CR–6224 head loss correlation has bounded the experimental results. Limited testing was conducted in the STPNOC Pilot Project to ensure that the correlation provided a reasonable prediction of head loss under STP–specific conditions [44]. Nevertheless, based on historic experience with concerns raised by the NRC staff and the ACRS, head losses computed in the STPNOC Pilot Project were increased by a factor of 5 to help account for any uncertainties.

1.2.8 Chemical Effects Head Loss Model

As discussed in Section 1.2.6, using a deterministically–based model, there are a relatively limited number of scenarios where significant chemical effects would be observed. Because the deterministically–based model indicated that only a few extreme scenarios could be consequential, a simplistic chemical effects model was adopted. In the simplistic model, the magnitude of total head loss is no less than a factor of 1 greater than the conventional head loss, but could be as much as a factor of 24 times the conventional head loss discussed in Section 1.2.7. The model is implemented according to the description below:

The minimum factor (1 times the conventional head loss) is applied if the fiber quantity on a given strainer is less than 1/16 of an inch.

The minimum factor (1 times the conventional head loss) is applied if the sump temperature is above 140°F. Based on the sump temperature profiles implemented in the STPNOC Pilot Project, 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.

The probability distributions used in the simplistic chemical effects model were developed with a mean of approximately 2 for small breaks, 3 for medium breaks, and 3 for large breaks. The distribution extreme values are approximately 15 for small breaks, 18 for medium breaks, and 24 for large breaks. That is to say, at 5 hours after the start of a LBLOCA, the maximum value for head loss would be approximately 24 times the conventional head loss. Recall that the conventional head loss has a fixed increase of 5 times the NUREG/CR-6224 value giving a total of approximately 120 times the NUREG/CR-6224 value.

1.2.9 Fiber Penetration

The STPNOC Pilot Project has adopted two terms that relate to different bypass phenomena, penetration and bypass. Penetration is used with ECCS strainer performance and bypass is reserved for the in-vessel flow paths around the core.

Common practice for assessing debris penetration has been to weigh the total quantity of debris collected downstream of the strainer after several pool turnovers (after all penetration is completed). In the STPNOC Pilot Project, a time-dependent debris penetration model is adopted. The STPNOC Pilot Project model accounts for two mechanisms operative for penetration. 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

In order to implement the STPNOC Pilot Project strainer penetration model, specialized full-scale module tests were performed. Unlike common practice for debris penetration, the STPNOC Pilot Project debris was collected at many intervals during the test such that the time-dependent behavior could be empirically modeled.

1.2.10 Boric Acid Precipitation

The STPNOC Pilot Project used a simplistic approach to model boron precipitation. Previous deterministic analyses have shown that if hot leg switchover occurs within 7 hours following a CLB, boron would not precipitate. However, debris blockage could invalidate these analyses. Therefore, the STPNOC Pilot Project used a small amount of debris collection on the core (7.5 g/FA) as a threshold for failure.

1.2.11 In–Vessel Fiber Limits

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 STPNOC Pilot Project, thermal–hydraulic modeling that showed full blockage at the bottom of the core and core bypass would not result in core damage for any HLB, the acceptance criterion was set to essentially set to an infinite fiber quantity. An acceptance criterion of 15 g/FA was used for CLBs based on the conservative results of testing by the PWROG [45]. Note, however, that the core blockage acceptance criteria are bounded by the boron precipitation acceptance criteria. As discussed in Section 1.2.10, boron precipitation was not considered to be an issue for HLBs. For medium and large CLBs, the acceptance criterion for boron precipitation was assumed to be 7.5 g/FA of fiber debris on the core.

1.3 Uncertainty Quantification

CASA Grande uses numerous variables as detailed in Volume 3; see Figure 1.1 of Section 1 in [46] for an overview. Some of these input parameters are treated as deterministic parameters, while others are treated as random variables with specified probability distributions. The manner in which these probability distributions were determined depends on the nature of the information available regarding the specific parameter in question. To give an idea of the range of methods we use, we discuss how we determined probability distributions for LOCA frequency and fiberglass penetration. We provide a further discussion of modeling the joint distribution of multiple random parameters as implemented in CASA Grande.

1.3.1 LOCA Frequency

We use a probability distribution to model the LOCA frequency for breaks of different sizes at different locations within the plant. This probability distribution is specified in Section 2.2.3 of [46]. The assumptions we make in order to determine this distribution are given in Section 3 of [46] in Assumptions 3.a-3.f. The analysis that we use to develop the probability distribution, and the way in which the probability distribution is employed in the analysis using CASA Grande, is described in Section 5.3 of [46] with further details in [47]. Here we briefly summarize our approach.

Forming probability distributions for the frequencies of LOCA pipe breaks, particularly larger breaks, presents challenges because we have limited data from operating experience, due to the very low probabilities of these breaks occurring. The probability distribution for LOCA frequency that we construct is informed by two sources. First, we use NUREG/CR-1829 [38], which, among other scenarios, documents an expert elicitation of the percentiles (5th, 50th, and 95th) for breaks of six effective sizes for PWR plants without inclusion of contributions due to steam generator tube ruptures; namely, we use NUREG/CR-1829, Table 7.19 for the current-day fleet (25 year average fleet operation). Second, we use an STP-specific study

[48], which allows us to distribute an overall frequency associated with a particular break size across different weld locations in the plant, using a total of 45 categories of welds. This allows us to form a joint distribution across break size and weld location that distinguishes different weld types of the same size based on degradation mechanisms, while maintaining consistency with NUREG/CR-1829 for the fleet-wide quantiles.

While NUREG/CR-1829 uses six effective break sizes, we model a continuum of break sizes, using a linear interpolation between the neighboring break sizes for the NUREG/CR-1829 quantiles. This is equivalent to assuming a uniform distribution and governs the break size between, for example, the NUREG/CR-1829 sizes of a 7-inch and a 14-inch break. The steps used in determining the probability distribution and sampling that distribution in the CASA Grande implementation are summarized as follows [46, Section 5.3]:

1. Deterministically calculate the relative (conditional) probability of breaks for specific weld categories based on pipe size, weld type, applicable degradation mechanisms, etc.

2. Identify applicable weld category and spatial coordinates for each weld location.

3. Statistically fit the NUREG/CR-1829 frequencies (5th, 50th, and 95th) using a bounded Johnson distribution for each size category. These fits represent the epistemic uncertainty associated with LOCA frequencies.

4. Sample the epistemic uncertainty (e.g., the 62nd percentile) and determine the corresponding total frequency curve based on the bounded Johnson fits, assuming linear interpolation between size categories.

5. Distribute total LOCA frequency to each weld location based on the relative (conditional) probability from step 1.

6. Sample break sizes at each weld location and proceed with the GSI-191 analysis carrying the appropriate initiating event frequencies.

Step 1 amounts to forming conditional probabilities using the STP-specific study [48]. These give the probabilities that the break comes from each relevant weld category given that we have observed a break of a specific size. In step 3, we choose the parameters from the class of bounded Johnson distributions to minimize the sum of the squared deviations of the fit distributions from the NUREG/CR-1829 percentiles. To ensure, in step 4, that the tails of the break-size distribution are adequately sampled, we use a NLHS procedure [49] in the CASA Grande implementation.

1.3.2 Fiberglass Debris Penetration

We use a probabilistic model of the filtration function of the ECCS pump strainers, and the parameters of that probabilistic model are given in Section 2.2.33 of [46]. The assumptions regarding fiberglass penetration of the strainer are detailed in Section 3 of [46], Assumptions 9.a-9.d. The analysis that we use to develop the probabilistic model is based on a mass-transport theory described in Section 5.9 of [46] with the statistical fitting procedure detailed in [50]. Here we briefly summarize our approach.

Following a break in RCS piping, some of the fiberglass insulation debris from nearby piping and equipment would be transported to the ECCS sumps, where it would accumulate on the pump strainers. In addition, some of the fine debris would pass through, or penetrate, the strainer. Debris can pass through the strainer directly or via shedding from the accumulated fiber bed on the strainer. The filtration efficiency of a strainer increases towards one as mass accumulates on the strainer. Test data from prototype strainer module experiments performed at Alden Research Laboratory (ARL) in October 2012 provide measurements of mass that passed through the strainer with specified time resolution. A combination of 100% capture filter bags and isokinetic grab samples were used to gather data regarding the change in penetration as a function of time. We model the filtration efficiency of the strainer, as a function of the mass on the strainer, using the empirical filtration function in equation 14 in Section 2.2.33 of [46]. We estimate the parameters of this function using data from the ARL experiments. We further use the experimental data to estimate the shedding parameters of the mass-transport theory equations described in Section 5.9 of [46]. Here, we focus on the probabilistic model for the filtration efficiency function.

Rather than simply developing point estimates of the parameters of the filtration efficiency function (equation 14 [46, Section 2.2.33]) and using the resulting point estimate of the filtration function, we instead use the experimental data to form an empirical envelope for the filtration efficiency. Then, when executing a computer simulation in CASA Grande, using a uniform random variable, we repeatedly sample realizations of the filtration efficiency function from the empirical envelope, maintaining the same functional form of equation 14.

To construct the empirical envelope for the filtration function we carried out the following three steps:

1. We use data from each experiment at ARL to fit the parameters of the equations of the mass-transport theory described in Section 5.9 of [46]. These equations predict, as a function of time, the mass accumulated on the strainer, the mass that has passed through the strainer, and the mass remaining in the pool given the rate of flow, the flow fraction captured by the filters, and the masses of debris and the timing of their introduction. We find parameters of the mass-transport theory equations that most closely match the data, using a constrained weighted least-squares procedure detailed in [50].

2. We use the parameters obtained in step 1 to construct both filtration as a function of time and the mass-on-the-strainer as a function of time at discretized time steps. Then, we "eliminate time" to obtain what we will label a data series for each experiment, specifying filtration efficiency as a function of mass on the strainer.

3. Steps 1 and 2 are repeated for each of the experiments, and the results yield multiple data series indicating the variability seen across the experiments. Taking these data, we form an empirical envelope for filtration as a function of mass on the strainer by finding three functions: First, we use a least-squares fit to find a central fit to the multiple data series from step 2, optimizing the parameters of equation 14 [46, Section 2.2.33]. This yields the parameters in the "Center" row of Table 2.2.32 in [46]. The second function is also of the form of equation 14 but majorizes the data while having minimum area under the function. The third function is again of the form of equation 14 but minorizes the data and has maximum area under the function. These latter two functions correspond to the parameters in the "Upper" and "Lower" rows of Table 2.2.32 in [46].

To transform the parameters found for the experiments at ARL to parameters for plant conditions, the parameters must be appropriately scaled, as described in Section 2.2.33 in [46].

1.3.3 Modeling Dependencies

Multiple parameters are random in our analysis, and hence a joint distribution governs the associated random vector. This means we should describe the corresponding dependence structure. We have two main strategies for dealing with the challenge of handling multivariate uncertainties for CASA Grande input parameters, and these strategies involve: (i) appropriate dimension reduction by modeling "perfect correlations" and (ii) appropriate modeling of conditional independence.

As an example of dimension reduction, consider the uncertainty associated with LOCA exceedance frequencies for a 2-inch break and for a 6-inch break. Let Λ_2 denote the exceedance frequency for a 2-inch break with (cumulative) distribution function F_2, and let Λ_6 and F_6 denote the analogous quantities for a 6-inch break. Here, F_2 and F_6 are fit as we describe in Section 1.3.1 and describe in more detail in Section 5.3 of [46] and in [47].

We do not model the random variables Λ_2 and Λ_6 as being independent. (If one were to do so then it would be possible, in simulating observations from these distributions, that the 6-inch exceedance frequency would be greater than the 2-inch frequency.) Instead, we model dependence using dimension reduction as follows: Let $U \sim U(0,1)$ be a uniform random variable on the interval $(0,1)$. Then using the standard simulation technique called *inversion*, $\Lambda_2 = F_2^{-1}(U)$ has distribution F_2 and $\Lambda_6 = F_6^{-1}(U)$ has distribution F_6. We reduce the dimension by assuming a perfect correlation via the bivariate random vector $(F_2^{-1}(U), F_6^{-1}(U))$, where we use the *same* uniform random variable in both expressions. In this way, if the 2-inch frequency, Λ_2, is at the 62nd percentile (via $U = 0.62$) of the distribution F_2 then Λ_6 is at the 62nd percentile of F_6. This type of dimension reduction is employed for modeling break sizes in CASA Grande.

Appropriate modeling of conditional independence is our second main strategy for handling multivariate uncertainty, and this approach is used pervasively in our analysis. As a first example, the timing of key plant response actions are, strictly speaking, random variables. However, these are determined in a conditional manner as described in Section 2.2.1 of [46]. So for a break smaller than 2-inches, the accumulators would not inject, and the sprays would not be initiated. Similarly, the timing for switchover to recirculation depends on the volume of water in the RWST, the total ECCS and CSS flow rate, and the break size. Operating procedures are further conditioned on the number of operating CSS pumps. (Again, see Section 2.2.1 of [46].) The pool water level is discussed in Section 2.2.6 of [46] and this depends on the size of the break and on the elevation of the break. The pool temperature profiles depend on the size of the break, as described in Section 2.2.6 of [46].

2 Engineering Analysis

2.1 Defense–in–Depth and Safety Margin

No changes are proposed to DID or safety margin by this licensing basis change. Instead, the risk associated with the traditional design basis accident analysis is assessed and quantified. In keeping with the Commission's goal to increase the use of risk analysis in regulation, this analysis quantifies the risk and uncertainty incorporating the impact of steps taken to preserve high levels of nuclear safety against perceived risks, while balancing regulatory cost and the need for significant worker exposure to mitigate concerns where the risk to nuclear safety is significant. A detailed discussion of the DID in place at STP is provided in Appendix C.

2.1.1 Defense–in–Depth

The risk to reliable operation of the as–built, as–operated plant DID systems is analyzed to be very small. STP has three trains of safety injection and three trains of containment fan coolers. The containment fan coolers do not rely on the recirculation mode for cooling the sump water. Decay heat can also be removed by the steam generators using the auxiliary feed water system and the steam generator power operated relief valves.

The normal charging system is an alternate flow path that can be aligned to the RWST if the ECCS pumps become unavailable for any reason. The design provides for an entire volume of the RWST (approximately 500,000 gallons) to be refilled and injected into the containment. Normally, STP can refill the RWST in approximately 24 hours. When indicated by the EOP, the reactor coolant pumps can be operated to cool the core and prevent core damage.

The risk associated with the concerns raised in GSI–191 regarding the likelihood of radiation release from the as–built, as–operated plant, as evaluated by LERF[38], is effectively zero. The concerns raised in GSI–191 have no bearing on containment integrity or on the release of radiation.

2.1.1.1 General Design Criteria. Because the analysis evaluates the risk of the as–built, as–operated plant, the traditional engineering analysis that forms the basis for the design remains intact and is inherent in the analysis. That is, the design criteria ultimately result in certain performance standards for the ECCS, such as required flow rates, support system availability, and equipment failure combinations. Although all commitments to design criteria remain intact, they cannot guarantee that core damage or LERF are prevented for every postulated scenario. Therefore, as previously mentioned, the LB[39] change evaluates the significance of the (non–zero) risk associated with the as–built, as–operated plant. Because the design criteria are robust and because changes to the design have been made to address specific GSI–191 concerns, the risk produced by the analysis is very small. The analysis incorporates extreme effects of chemical phenomena on debris bed differential pressures as well as boron precipitation. Even with these extreme assumptions, the probability for core damage is found to be very small with no expectation for increased probability for LERF.

2.1.1.2 Defense–in–Depth Principle. The analysis shows that DID is maintained with high probability. The availability and reliability of the systems that support DID continue to be assured with high probability with consideration of uncertainty. The analysis shows that there is practically no risk to containment integrity associated with the concerns raised in GSI–191 and therefore, the license basis change would indicate that as–built, as–operated containment design remains adequate to prevent a significant release into the environment. In quantification of the risk, additional operator actions or programatic activities beyond the existing as–built, as–operated plant have not been included.

2.1.1.3 Uncertainties of Chemical Effects. As part of the analysis, experiments have been developed to investigate the significance of the concerns raised in GSI–191 for post–LOCA environments specific to the STP plants. These experiments

[38] Large Early Release Frequency

[39] Licensing Basis

examined conditions under which specific forms of chemical precipitates, particularly AlOOH, can be formed: in-situ over short time frames (on the order of hours or days) by, for example, direct injection of aluminum salts; and ex-situ (as in surrogate preparations developed elsewhere in the industry). The experiments also examined chemicals formed by actual corrosion sources (such as aluminum, zinc, or concrete) in prototypical post-LOCA environments.

Experiments have shown that using ex-situ methods of precipitate formation produced precipitate forms that are much more likely to result in head loss impacts in debris beds than those formed in-situ. Finally, and consistent with previous observations [4], the more recent experimental work performed for this analysis provides evidence that the chemical corrosion process that would take place in an actual post-LOCA environment is significantly more benign to debris bed head loss than would be suggested by any of the surrogate (in-situ or ex-situ) methods. The results of the chemical effects experimental program that are most similar to the actual post-LOCA sump conditions give confidence that experiments performed with surrogate preparations represent an upper bound for chemical effects on debris bed head loss.

2.1.1.4 Uncertainties of Head Loss.

The head loss associated with debris beds can be shown to be dependent not only on chemicals, but on the presence of particulates transported to the sump area. Such particulates have been hypothesized to result from failure of coatings unqualified for high radiation and post-LOCA fluid chemistry. The transport and failure extent of such particulates have been conservatively estimated in the STPNOC Pilot Project analysis so as to preserve their effect on the result. The failure extent and rate of failure used in the STPNOC Pilot Project are supported by experimental evidence.

Experiments have been conducted in a high-temperature vertical loop using expected post-LOCA fluid conditions (pH, boron and buffer chemical concentrations, and temperature) to examine the uncertainty of coefficients derived in correlations commonly used in the analysis of head loss concerns raised in GSI-191, for example, the NUREG/CR-6224 correlation. The experiments investigated a wide range of particulate size distribution and types (for example, different forms of silicon carbide and iron oxide) and showed that the NUREG/CR-6224 correlation bounds actual head loss in beds with post-LOCA fluid flow, chemistry, particulate, and bed formation prototypical of the STP plants. The experiments help in understanding the uncertainty and margin in the analysis of head loss from many hypothesized break sizes and locations with different debris loads.

Because current testing of STP conditions has only verified a few possible bed compositions, a multiplier has been applied to all debris-bed head loss calculations to compensate for residual uncertainties.

2.1.2 Safety Margin

In each scenario, the tails of extreme distributions are sampled and propagated through to the PRA. Where appropriate, the uncertainty distributions envelope attributes of both aleatory uncertainty and epistemic uncertainty. As will be explained later in this report, the only component of epistemic uncertainty that is explicitly preserved in the present analysis is the component attributable to the break-frequency size distributions taken from NUREG/CR-1829. All other sources of variability have been integrated into the estimates of failure probability reported for the composite failure modes used in the PRA. Composite failure modes applied in the PRA include (1) strainer failure by excessive differential pressure, excessive deaeration, and mechanical buckling; (2) core blockage; and (3) boron precipitation. Also, experimental results for chemical effects were obtained with existing amounts of aluminum exposed to post–LOCA fluids for 30 days, and they indicated very little to no precipitate formation in the bulk fluid.

Although such an extreme scenario would never be expected based on a realistic analysis of the LOCA response, thermal–hydraulic engineering evaluations of core flow blockage scenarios were conducted to understand safety margin in these scenarios. These evaluations [51] include assessments of extreme conditions of core blockage. It was shown that with complete blockage of the core inlet and all bypass paths, only a medium or large break cold leg LOCA would result in core damage. In addition, detailed modeling of the core and reactor vessel showed that only one fuel assembly flow passage needs to remain clear to prevent fuel overheating. The analyses included locating the open fuel assembly either at the core center or at an extreme periphery location. Multi–dimensional vessel and core simulations at the time of recirculation show that the core inflow is highly asymmetric indicating that it would be likely that several fuel assemblies would not be blocked by debris that might penetrate the ECCS sump screens.

Chemical effects testing conducted in the STPNOC Pilot Project has shown that significant amounts of chemical precipitation that would be expected to produce large head losses in typical debris beds are not present in solution in the STP post–LOCA environment. Where precipitation occurs, the test results suggest that the precipitates that actually form in solution have different morphology from the surrogate precipitates and are likely to have less impact on total head loss. Under extreme scenarios, chemical effects might be more significant than those observed in the STPNOC Pilot Project tests. To address this possibility, a chemical effects bump–up factor probability distribution with a tail including 15x, 18x, and 24x increases for small, medium, and large breaks, respectively, was included in the CASA Grande evaluation. The purpose of the extreme tail was to preserve a 10^{-5} probability of meeting or exceeding the stated limits, while preserving expectation values between 2 and 3 (factors of 2x to 3x) for each LOCA category. In addition, the contributions of chemical effects from bounding experiments with ex–situ prepared precipitates [52] are assumed in the core flow blockage success criteria, which success criteria was developed as a bounding value for all PWRs. Several other conservative assumptions leading to safety margin in the as–built, as–operated plant are detailed by NEI [53].

All STP large–bore piping PWSCC–susceptible welds (nozzle welds) have been replaced or otherwise mitigated, with the exception of the reactor vessel nozzle welds. The reactor vessel nozzle welds are less of a concern in the GSI–191 analysis than are other break locations because the reactor vessel is covered with RMI, and the primary shield wall would protect the majority of fiberglass insulation in the steam generator compartments. STPNOC is in compliance with ASME Section XI weld inspections.

The insulation, paint, and concrete damage choice of the ZOI used in the STPNOC engineering calculation is expanded to account for pipe whip. The calculations assumes piping constraints (especially on large–bore pipes) that would reduce the ZOI based on pipe whip restraint are not present. Finally, Ballew et al. [54] have shown that the ZOIs used in the GSI–191 risk analysis are significantly overestimated [32, Section 3.4.2].

2.2 Evaluation of Risk Impact

The risk assessment shows that any increases in CDF[40] and risk are very small and consistent with the intent of the NRC's Safety Goal Policy Statement. The expected change in CDF and LERF is very small in the analysis, which includes internal and external hazards in an at–power model that bounds risk contribution. An in–depth and comprehensive risk assessment using the STPNOC PRA was used to derive the quantified estimate of the total impact of the proposed change as opposed to a qualitative assessment using, for example, performance measures.

Because pressures and temperatures are greatly reduced in plant operating Modes 4, 5, and 6, the concerns raised in GSI–191 can not be realized in these shutdown modes of operation. For Mode 3, the at–power model is bounding and can be used as a surrogate for Mode 3 operation.

The quantitative risk metrics evaluated in the analysis are CDF and LERF. There may be risk metrics that are not reflected (or are inadequately reflected) by changes to CDF and LERF. Other risk metrics were considered, especially effects on containment integrity. However, no concerns related to GSI–191 that have a bearing on containment integrity following a LOCA have been identified in the analysis. Therefore, there is no effect on LERF, and therefore no impacts to offsite consequences.

The STPNOC PRA has been reviewed on multiple occasions by the NRC. The last independent peer review was for STPNOC PRA Revision 5 and assessed it as adequate for use in STPNOC PRA applications. Since Revision 5, there have not been any major changes to the PRA that require additional peer review. The PRA is currently at Revision 7, released late in 2012. The concerns raised in GSI–191 are isolated to long–term cooling in LOCAs. Other initiating events included in the PRA are therefore less important than the LOCA event trees. The STP baseline CDF and LERF are substantially below the Commission Safety Goal when the as–built, as–operated plant risk is evaluated with the concerns raised in GSI–191 included. Therefore, the concerns raised in GSI–191 would contribute negligible risk relative to the analyzed average plant risk.

[40] Core Damage Frequency

2.3 Technical Adequacy of the PRA Analysis

The STPNOC PRA is a full–scope integrated Level I and Level II PRA. Further details concerning the technical adequacy of the STPNOC PRA are found in Volume 4. However, the GSI–191 concerns pertain to LOCAs and in particular, the recirculation phase of a LOCA. The main concerns are with MLOCAs and LBLOCAs. As mentioned in Section 2.1.2, thermal–hydraulic response analysis shows that long–term core cooling is not challenged in SLOCA scenarios. The STPNOC PRA, like similar PRAs, included a very simplistic demand failure probability for recirculation failure. The GSI–191 risk analysis required a much better understanding of the failure probability and concomitant uncertainty for recirculation failure. In order to support a more informed basis for recirculation failure, the basic event likelihood and uncertainty needed engineering analysis support. A detailed uncertainty quantification was performed to solve the required engineering models and propagate their uncertainty to obtain a recirculation failure probability.

Similarly, the basic event failure likelihood and uncertainty for ECCS pump performance included only mechanical and electrical failures. However, the concerns raised in GSI–191 required an assessment of the likelihood for air ingestion and inadequate NPSHA when debris beds are hypothesized to form on the ECCS sump strainers. These added failure mechanisms were included in the PRA, with their failure probability and uncertainty determined through uncertainty propagation of appropriate physical models as described in detail in Volume 3. The failure thresholds for these kinds of events are from a standard engineering analysis of allowable air and NPSHA for the pumps during a worst case LOCA scenario.

Finally, downstream effects of core blockage and boron precipitation were included with the possibility of recirculation failure. Again, the added failure mechanisms were included in the PRA with their failure probability and uncertainty determined through uncertainty propagation of appropriate physical models.

2.3.1 Scope of the PRA

The scope of the STPNOC PRA is Level I and Level II, including external and internal hazards such as internal floods, seismic events, internal fires, high winds, and external flooding. This level of detail is actually not required because none of the LOCAs are evaluated in external events, which means that GSI–191 issues do not appear. The concerns raised in GSI–191 are related to LOCAs, and the at–power LBLOCA– and MLOCA–initiating events are the most important of the concerns. Being that the STPNOC PRA is an at–power PRA, no shutdown LOCA events are considered. The at–power scenarios bound the low power and shutdown LOCA events, not only because the decay heat load is significantly reduced, but because the energy available for debris generation is much less. Therefore, the STPNOC PRA overall scope is sufficient to address the concerns associated with GSI–191.

The STPNOC PRA Revision 7 initiating event frequency for LOCAs is taken from the most recent database used in PRA analyses [55]. Eide et al. refer to NUREG/CR–1829 [38] as the basis for LOCA initiating event frequencies. The frequencies used in the STPNOC PRA LOCA initiating event trees are preserved in the engineering analysis used to develop failure probabilities at locations throughout the Class 1 piping in the STP containment buildings. Also, the LOCA epistemic uncertainties used in the engineering analysis are taken from the same NUREG/CR–1829 table used by Eide et al.

2.3.2 Level of Detail

As previously mentioned, the PRA is not significantly changed to specifically address the concerns raised in GSI–191. 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 new basic events and top events required. Details include physical models and mechanisms known to lead to failure, and the analyses include experimental evidence used to support particular areas of concern.

2.3.3 Technical Adequacy

The safety issues associated with GSI–191 are within the scope of current PRAs that meet Regulatory Guide 1.200 [56, 57], Revision 1 or Revision 2. LOCAs are internal event initiators included in all versions of Regulatory Guide 1.200. The STPNOC PRA has been peer reviewed relative to internal events (including LOCA initiators). Since STPNOC's PRA is compliant with RG 1.200, Revision 1 for internal events, it is compliant with Regulatory Guide 1.200, Revision 2 for assessing the risk associated with GSI–191.

The PRA analysis is technically robust. The assumptions and/or actual modeling of the concerns raised in GSI–191 are bounded either in other work by experimental evidence or analysis, or by analysis and experimentation specifically performed for the STPNOC PRA evaluation. The STPNOC PRA is used in risk-informed applications extensively at STP.

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. Examples include [58, 59, 60, 61, 62, 36, 63, 64, 65, 66, 67, 68, 69, 70, 71]. In some cases, STPNOC has been the industry leader in PRA applications and application development, and in setting standards and practices. In the GSI-191 risk-informed resolution, STPNOC has followed the practices and methods known to be acceptable and consistent with industry PRA practices and standards.

2.3.4 Plant Representation

The STPNOC PRA and the engineering analysis supporting the GSI-191 analysis are representative of the as-built, as-operated plant. The STPNOC PRA is reviewed for compliance/adherence with the plant design and plant data every 36 months as an UFSAR Chapter 13.7 commitment required for PRA applications. Section 2.3.5 is a summary of the engineering analysis supporting the PRA analysis in the STPNOC Pilot Project. The STPNOC PRA configuration control is in accordance with STPNOC plant processes [72].

2.3.5 Model of the LOCA Processes, CASA Grande

One of the primary functions served by CASA Grande in the STPNOC Pilot Project is quantifying conditional failure probabilities related to GSI-191 phenomena for various plant modes and ECCS operating states. Failure probabilities are passed to the PRA to determine the decision metrics for acceptance. Three new top events are added to the PRA to accommodate composite GSI-191 failure processes:

- failure at the sump strainer;
- boron precipitation in the core; and
- blockage of the core.

These three composite failure probabilities are calculated by testing the outcome of every postulated break scenario against seven performance thresholds:

- (1) strainer $\Delta P \geq$ NPSHR Margin;
- (2) strainer $\Delta P \geq P_{buckle}$[41];
- (3) strainer F_{void}[42] ≥ 0.02;
- (4) core fiber load $\geq$ CLB fiber limit for boron precipitation;
- (5) core fiber load $\geq$ HLB fiber limit for boron precipitation;
- (6) core fiber load $\geq$ CLB fiber limit for flow blockage; and
- (7) core fiber load $\geq$ HLB fiber limit for flow blockage.

(1) through (3) above are counted as failures if any single operable strainer exceeds the performance thresholds at any time during the 36-hour calculation. (4) and (5) are assessed against the accumulated fiber penetration from all operable strainers and are counted as failures only if the performance threshold is exceeded before the time of hot-leg injection. The thresholds for (5) were set infinitely high so that only exceedance of the CLB boron precipitation loading (4) was recorded as failure. This approach is reasonable because the threshold for failure in (4) is

[41] Strainer structural design limit
[42] Void Fraction

substantially lower than for (5) through (7), and because (4) through (7) all depend on (1) through (3), and all the performance thresholds depend on the same internal flow distribution and fiber accumulation processes.

Violation of any of the seven performance thresholds is counted as an independent failure. Thus, it is possible that a single scenario can contribute to both a strainer–related failure tally and a core–fiber–load failure tally. After a suite of scenarios is performed, the sum of probability weights for failed scenarios within each LOCA category is divided by the sum of probability weights for all scenarios within each LOCA category to generate the conditional failure probabilities needed for the PRA. Table 1 reports the mean conditional failure probability associated with each composite failure mode for each of five plant operating states (cases). No failures were recorded for small– or medium–break events, and it transpired that only the higher range of large–break events contributed to failure. In addition to the composite PRA failure modes, total failure probability conditioned on the LOCA category is provided.

The Table 1 results indicate the following. Design–basis accident response with three trains operable (Case 1) is estimated to incur a total failure probability of 0.09% given that an LBLOCA occurs (that is, 9 failures in every 10,000 large–break events). If only one train is operable (Case 43), this estimate increases to 0.45%. The primary contributor to the increase is the additional head loss incurred at the single strainer by collecting all of the debris that was designed to flow in proportion across three strainers under Case 1. Conversely, failures incurred by exceeding the boron fiber load are reduced (compare first and last columns) because less cumulative fiber is penetrating the single, highly loaded strainer. Blockage failure is reported as zero probability because the thresholds were set very high, partly to avoid double counting blockage failures for events that first exceed the bounding low value for fiber–load thresholds related to boron precipitation in the core.

Conditional failure probabilities reported in Table 1 are described as "mean" or "expected" values because five point estimates associated with independent samples of the NUREG/CR–1829 break frequency envelope have been averaged for use in the PRA. The following discussion explains the origin and the mechanics of this averaging process.

The NUREG/CR–1829 tables [38] assign confidence levels to estimates of annual occurrence frequency as a function of break size. This assignment of confidence level defines an envelope of epistemic uncertainty that was fit using bounded Johnson probability density functions at each discrete break size for which percentiles of confidence were tabulated. The purpose of these fits was to enable interpolation of the confidence bands at any intermediate break size of interest. The relationship defined by NUREG/CR–1829 between annual occurrence frequency (events per year) and break size is presented in terms of a ccdf[43]. This format implies that the underlying pdf[44] has been integrated, and it is important to consider the form of the pdfs before selecting an interpolation scheme that will be applied to the ccdfs. Conversely, any presumption about interpolation of the ccdf would constrain the implied form of the pdf.

A pdf defined for break size must define the probability per unit size that a break occurs within the interval between the discrete sizes tabulated in NUREG/CR–1829. Without knowing the details of how fracture mechanics processes were treated during compilation of the NUREG/CR–1829 table, it is difficult to defend any assumption

[43] Complementary Cumulative Distribution Function

[44] A probability density function

Table 1: Mean LBLOCA conditional failure probabilities for five plant operating states. Failure probabilities shown are for strainer blockage, core fiber load exceeding flow blockage criteria, and sump differential pressure exceeding $P_{pbuckle}$. Each case refers to a plant operating state.

	Case 1	Case 9	Case 22	Case 26	Case 43
Blockage	0	0	0	0	0
Boron Fiber Load	6.94×10^{-04}	1.82×10^{-03}	7.51×10^{-05}	6.15×10^{-05}	3.42×10^{-06}
Sump Failure	2.45×10^{-04}	5.39×10^{-04}	1.32×10^{-03}	9.56×10^{-04}	4.45×10^{-03}
Total	9.38×10^{-04}	2.35×10^{-03}	1.40×10^{-03}	1.02×10^{-03}	4.45×10^{-03}

other than uniform probability density between the tabulated discrete sizes. Uniform probability density means that any break size within the interval is equally likely. Uniform (constant) break–size probability between two ccdf values is easily calculated as the positive difference between the complementary cumulative annual frequencies divided by the positive range of size across the interval divided by the total annual exceedance frequency for the smallest break size. The integral of a constant pdf, which is needed to form a ccdf, is a straight line, and this implies that linear–linear interpolation of the NUREG/CR–1829 table is the treatment most consistent with the assumption of constant underlying probability density.

Figure 10 uses log–log axes for plotting a linear–linear interpolation of the NUREG/CR–1829 table values, which causes the linear ccdf to appear as a periodically looping curve.

NLHS of break–frequency profiles from the Johnson pdf envelope are performed in exactly the same manner as for all other random variables. The nonuniform probability bins are predefined based on the desired number of samples and on the direction of

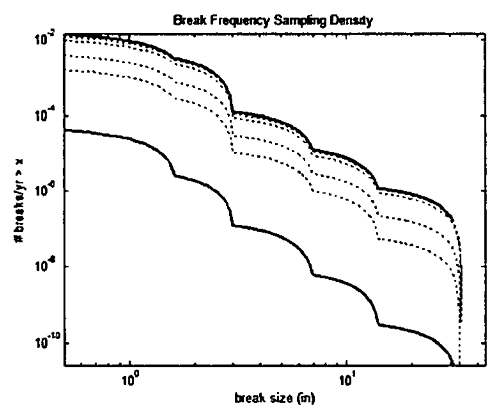

Figure 10: Linear–linear interpolation of bounded Johnson extrema (solid) with nonuniform stratified random break–size profiles (dashed).

presumed conservatism. Then, random percentiles are chosen from within each bin to represent, or "carry," the associated probability weights. For the STPNOC Pilot Project, five independent random samples were extracted from the Johnson envelope for each plant operating state, with an emphasis on upper percentiles of the break frequency uncertainty envelope. Given a sample of five percentiles, the Johnson fits are then inverted to find the corresponding annual frequencies. All Johnson fits are perfectly correlated by using the same fixed values of the sampled percentiles. Finally, the set of annual frequencies from each Johnson fit is linearly interpolated to create the break–frequency profiles shown as the dashed lines in Figure 10.

Each break frequency profile is fully analyzed in CASA Grande using a set of three batch replicates containing approximately 2,250 break scenarios each to obtain a point estimate of failure probability for the composite modes. Residual sampling imprecision of roughly 20% among the three replicates is typical of this scenario sampling size. Probability weights from stratified sampling of the Johnson envelope are then used to form the weighted conditional means reported in Table 1.

The resolution used in the STPNOC Pilot Project is 2,250 breaks for batch size, 3 replicates, and 5 break–frequency samples. Table 2 summarizes the five point estimates and their associated probability weights generated for the total failure probability under plant operating state Case 43 (one train operable). The weighted mean is formed simply by multiplying each point estimate by its probability weight and adding the products. Similar distributions were formed for all composite failure modes and for all plant operability states, but only the weighted means are presented in Table 1.

The cumulative distribution defined for total failure probability under Case 43 in Table 2 is plotted in Figure 11 to illustrate how epistemic quantiles could be preserved from the GSI–191 engineering analysis CASA Grande. This distribution reflects only the uncertainty inherent to the estimation of annual break frequency. All other random variability, including ranges on physical phenomena and decision criteria, has been integrated into each point estimate. As shown in Table 1 and Figure 10, typical variation in failure probability estimates spans a factor of 2 to 4 between the minimum and maximum values. This variation is caused solely by the shape of the randomly selected break–frequency profiles, which dictate the relative proportion of break frequency by size.

It is important to reemphasize that CASA Grande never directly uses the annual break frequency as a time–rate quantity. All analyses proceed conditioned on the assumption that a break has already occurred. Sample profiles taken from the break–frequency envelope then describe how to partition the relative occurrence of breaks by size. CASA Grande further redistributes the relative size probability across weld types in order to map the cumulative probability of a break as a function of size to discrete locations in the plant [37].

The PRA samples directly from the NUREG/CR–1829 Johnson pdf fits in each category to preserve the epistemic uncertainty in LOCA frequency. It is important for CASA Grande to use exactly the same representation of the epistemic uncertainty. The Johnson fits are evaluated analytically in CASA Grande to generate a table of empirical pdfs that are manually passed to the PRA (RISKMAN™ model) for repeated sampling in the risk quantification. Although the PRA generates thousands of samples from the Johnson pdf during quantification, CASA Grande sampling is relatively sparse

Table 2: Distribution of total conditional failures for LLOCA under Case 43 (one train operating).

Point Failure Probability	Johnson Probability Weight	Cumulative Probability
0.0	0.0	0.0
3.13×10^{-03}	8.22×10^{-01}	8.22×10^{-01}
7.49×10^{-03}	4.62×10^{-03}	8.27×10^{-01}
1.03×10^{-02}	1.46×10^{-01}	9.73×10^{-01}
1.15×10^{-02}	1.00×10^{-03}	9.74×10^{-01}
1.2×10^{-02}	2.60×10^{-02}	1.0
4.45×10^{-03}	weighted mean	

here. CASA Grande uses one quantification loop to generate point estimates of failure probability that are based on parameter variations and model uncertainties like chemical effects bump up, and an outer loop to preserve the epistemic quantiles of the break–frequency envelope (see Section 2.5.1). Sparse sampling of the epistemic envelope is a consequence of emphasizing aleatory uncertainties (inner loop) that drive the outcome of each break scenario and epistemic sampling relies on NLHS for generating unbiased estimates of the mean failure probability. Failure distributions similar to those shown in Figure 11 could alternatively be sampled by the PRA to generate distributions of incremental risk attributable to GSI–191 phenomena. A sampling scheme would necessarily preserve epistemic correlation in the distribution of failure probability that is generated by CASA Grande (Figure 11) and shared by the RISKMAN™ model.

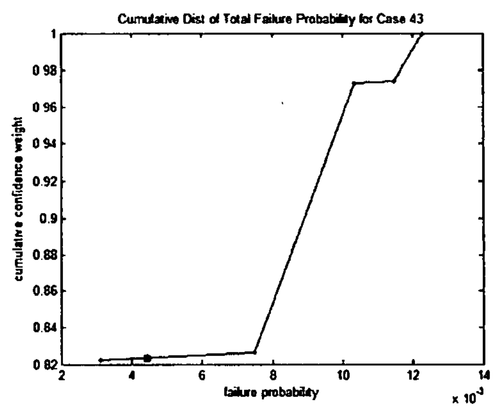

Figure 11: Empirical distribution of total failure probability for Case 43 (one train operating) based on five discrete samples of the NUREG/CR–1829 break–frequency uncertainty envelope. Weighted mean = 4.45×10^{-03} marked as bold dot.

Another key piece of information passed from CASA Grande to the PRA through the basic events supported is the conditional split fraction for cold leg breaks in each LOCA category. The total break size probability for a single NUREG/CR–1829 profile

Table 3: All cold–leg split fractions conditioned on LOCA categories small, medium, and large for Case 43. The fraction going to the hot leg is simply the complement of the cold leg fraction.

Total	Small	Medium	Large
4.2052034×10^{-01}	4.2962813×10^{-01}	4.8133459×10^{-01}	4.3059826×10^{-01}
4.2052034×10^{-01}	4.2962813×10^{-01}	4.8133459×10^{-01}	4.3059826×10^{-01}
4.2052034×10^{-01}	4.2962813×10^{-01}	4.8133459×10^{-01}	4.3059826×10^{-01}
4.2015626×10^{-01}	4.2933789×10^{-01}	4.8133521×10^{-01}	4.3048163×10^{-01}
4.2015626×10^{-01}	4.2933789×10^{-01}	4.8133521×10^{-01}	4.3048163×10^{-01}
4.2015626×10^{-01}	4.2933789×10^{-01}	4.8133521×10^{-01}	4.3048163×10^{-01}
4.2014556×10^{-01}	4.2932931×10^{-01}	4.8133576×10^{-01}	4.3044256×10^{-01}
4.2014556×10^{-01}	4.2932931×10^{-01}	4.8133576×10^{-01}	4.3044256×10^{-01}
4.2014556×10^{-01}	4.2932931×10^{-01}	4.8133576×10^{-01}	4.3044256×10^{-01}
4.2210420×10^{-01}	4.3087029×10^{-01}	4.8133228×10^{-01}	4.3115092×10^{-01}
4.2210420×10^{-01}	4.3087029×10^{-01}	4.8133228×10^{-01}	4.3115092×10^{-01}
4.2210420×10^{-01}	4.3087029×10^{-01}	4.8133228×10^{-01}	4.3115092×10^{-01}
4.3407111×10^{-01}	4.3931916×10^{-01}	4.8118954×10^{-01}	4.3960731×10^{-01}
4.3407111×10^{-01}	4.3931916×10^{-01}	4.8118954×10^{-01}	4.3960731×10^{-01}
4.3407111×10^{-01}	4.3931916×10^{-01}	4.8118954×10^{-01}	4.3960731×10^{-01}

is distributed across all welds in containment using the hybrid weighting scheme [37] to account for the contributions of small breaks on large pipes to the small and medium LOCA categories. Each break scenario sampled from this process carries a specific size and location and a fractional weight of the total break–size probability. Before any other physical parameters are considered, the distribution of probability weight can be partitioned into HLB and CLB events and by LOCA size.

Table 3 itemizes all cold–leg split fractions obtained for the fifteen batches associated with Case 43. These values were obtained by dividing the sum of probability weights for CLBs in each LOCA category by the sum of probability weights for all breaks in the LOCA category. HLB split fractions are simply the complement of any single entry in the table. Three replicates of 2,250 scenarios are evaluated for each of five break–frequency profiles for a total of $3\times2250\times5 = 33,750$ scenarios per plant operating state. CLB split fractions are mildly dependent on the break–frequency profile shape (note repetition in successive groups of three rows), but they are independent of the plant operating state. It is interesting to note that proportion of large CLBs is substantially smaller than the 50% proportion assumed in the 2011 [3] quantification.

Table 4 lists a sample of the specific welds, break sizes, and general containment zones that are associated with one or more failure modes in Case 43. This list includes only the first 34 of 1659 failed scenarios that were tallied during the analysis. The fact that no SLOCA or MLOCA events have been recorded as failure for any scenario evaluated in this quantification is a strong indication that there is a minimum size break below which insufficient debris can be formed to challenge the safety systems. The same consideration explains why most failure scenarios involve the DEGB assumption of spherical ZOI simply because more insulation volume can be involved in debris generation. The above illustration regarding Case 43 indicates the kinds of insights that can be realized in the STPNOC Pilot Project analysis approach.

2.4 Acceptance Guidelines

Regions are established on the phase planes defined by ΔCDF, CDF and $\Delta LERF$, $LERF$, as illustrated in Figure 12 and Figure 13. Acceptance guidelines are established for each region as discussed below. The figures show shading as the values increase on either axis. The shading indicates that greater scrutiny and support would be required for values that approach the region boundaries. Also illustrated, in the figures, is the desired trajectory for changes. That trajectory can be realized by using resources on projects that have the maximum risk benefit, a concept that is consistent with the Commission's direction to use risk insight to best achieve safety goals.

The comparison in the STPNOC GSI–191 analysis uses the full–scope (including internal and external hazards, at–power, low power, and shutdown) assessment of the change in risk metric and the baseline value of the risk metric (CDF and LERF). As noted above, the shutdown PRA analysis is bounded by the at–power model. In the STPNOC GSI–191 analysis, the maximum acceptable increase in CDF is 10^{-06} and the maximum acceptable increase in LERF is 10^{-07}.

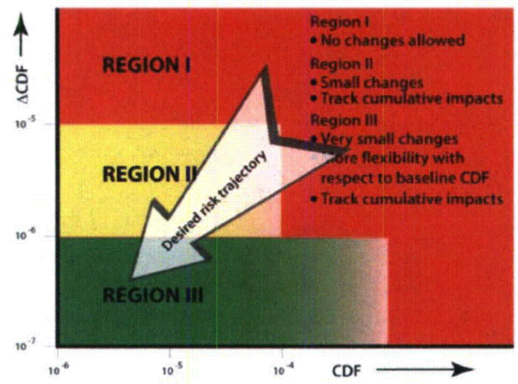

Figure 12: Reproduction of Figure 4 from Regulatory Guide 1.174, "Acceptance guidelines for core damage frequency", the ΔCDF, CDF phase plane.

2.5 Comparison of PRA Results with Acceptance Guidelines

The STPNOC Pilot Project PRA quantification is detailed in Volume 2. As mentioned previously, the quantification shows that the risk associated with the concerns raised in GSI–191 are very small when compared to the acceptance criteria of RG1.174.

The PRA used in the GSI–191 licensing basis change does not rely solely on numerical results for change in risk. Instead, the choice of models, solution methodology and incorporation of uncertainties provides a high level of confidence that the uncertainties in models' parameters has been properly accounted for in the results. The safety margin described in Section 2.1.2 associated with use of the methodology reflected in the license basis change analysis provides assurance that safe operation can be expected without reliance on numerical results alone.

As mentioned in Section 2.3.3, the STPNOC PRA is an integrated Level 1 model that includes all internal and external events, Level 1 and Level 2 analysis, the focus of the GSI–191 concerns are related to LOCA. The analysis of LOCA initiating event frequencies and local pipe failure probabilities included in development of the basic events for the scenarios that address the con-

Table 4: Sample attributes of break cases leading to failure for Case 43. In the table: Pipe is a text string defined in the inservice inspection program; System refers to STP System (all are RCS); Break Size is the size of the break in inches; LOCA size values of 1,2,3 denote small, medium, large LOCA events (all are large); DEGB, YES denotes the fully-severed pipe condition (failures dominated by DEGB); RCS Leg denotes break location (CLB or HLB); and Break Location denotes regions in the containment building related to debris transport fractions.

Pipe	System	Break Size	LOCA Size	DEGB	RCS Leg	Break Location
12RC-1112-BB1	RCS	10.126	3	YES	Hot	SG Compartment
12RC-1112-BB1	RCS	10.126	3	YES	Hot	SG Compartment
12RC-1112-BB1	RCS	10.126	3	YES	Hot	SG Compartment
12RC-1125-BB1	RCS	10.126	3	YES	Cold	SG Compartment
12RC-1125-BB1	RCS	10.126	3	YES	Cold	SG Compartment
12RC-1125-BB1	RCS	10.126	3	YES	Cold	SG Compartment
12RC-1125-BB1	RCS	10.126	3	YES	Cold	SG Compartment
12RC-1125-BB1	RCS	10.126	3	YES	Cold	SG Compartment
12RC-1125-BB1	RCS	10.126	3	YES	Cold	SG Compartment
12RC-1212-BB1	RCS	10.126	3	YES	Hot	SG Compartment
12RC-1212-BB1	RCS	10.126	3	YES	Hot	SG Compartment
12RC-1212-BB1	RCS	10.126	3	YES	Hot	SG Compartment
12RC-1212-BB1	RCS	10.126	3	YES	Hot	SG Compartment
12RC-1221-BB1	RCS	10.126	3	YES	Cold	SG Compartment
12RC-1221-BB1	RCS	10.126	3	YES	Cold	SG Compartment
12RC-1221-BB1	RCS	10.126	3	YES	Cold	SG Compartment
12RC-1221-BB1	RCS	10.126	3	YES	Cold	SG Compartment
12RC-1221-BB1	RCS	10.126	3	YES	Cold	SG Compartment
12RC-1221-BB1	RCS	10.126	3	YES	Cold	SG Compartment
12RC-1221-BB1	RCS	10.126	3	YES	Cold	SG Compartment
12RC-1312-BB1	RCS	10.126	3	YES	Hot	SG Compartment
12RC-1312-BB1	RCS	10.126	3	YES	Hot	SG Compartment
12RC-1312-BB1	RCS	10.126	3	YES	Hot	SG Compartment
12RC-1312-BB1	RCS	10.126	3	YES	Hot	SG Compartment
12RC-1312-BB1	RCS	10.126	3	YES	Hot	SG Compartment
12RC-1322-BB1	RCS	10.126	3	YES	Cold	SG Compartment
12RC-1322-BB1	RCS	10.126	3	YES	Cold	SG Compartment
12RC-1322-BB1	RCS	10.126	3	YES	Cold	SG Compartment
12RC-1322-BB1	RCS	10.126	3	YES	Cold	SG Compartment
16RC-1412-NSS	RCS	12.814	3	YES	Hot	SG Compartment
16RC-1412-NSS	RCS	12.036	3	NO	Hot	SG Compartment
16RC-1412-NSS	RCS	12.814	3	YES	Hot	SG Compartment
16RC-1412-NSS	RCS	11.273	3	NO	Hot	SG Compartment
16RC-1412-NSS	RCS	12.090	3	NO	Hot	SG Compartment
16RC-1412-NSS	RCS	12.118	3	NO	Hot	SG Compartment
16RC-1412-NSS	RCS	12.814	3	YES	Hot	SG Compartment
16RC-1412-NSS	RCS	12.814	3	YES	Hot	SG Compartment
16RC-1412-NSS	RCS	12.814	3	YES	Hot	SG Compartment
16RC-1412-NSS	RCS	12.814	3	YES	Hot	SG Compartment

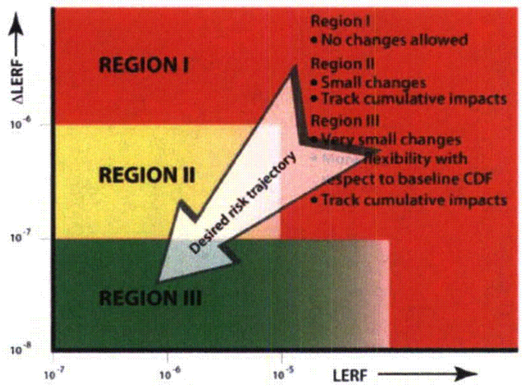

Figure 13: Reproduction of Figure 5 from Regulatory Guide 1.174, "Acceptance guidelines for large–early–release frequency", the $\Delta LERF$, $LERF$ phase plane.

cerns raised in GSI–191 include the full range of the epistemic uncertainty at each break size. Qualitative conservatisms that increase safety margin (as previously mentioned in Section 2.1.2) 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 concerns related to GSI–191:

- 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.

2.5.1 Types of Uncertainties and Aethods of Analysis

Both aleatory and epistemic uncertainties have been included in the STPNOC PRA. As mentioned in the previous section (Section 2.5), uncertainties have also been addressed using conservative assumptions where appropriate or where large uncertainties are seen. For example, assuming a larger ZOI will result in scenarios that are conservative.

2.5.1.1 Comments on Uncertainty Types

In the PRA community, the concept of "separate" types of probabilities or uncertainties is discussed frequently. In other communities, probability is simply probability and following quantification there is no distinction as to the source. (See Chapter 3 of [73] on PHSA[45] for a discussion including, "The panel concludes that, unless one accepts that all uncertainty is fundamentally epistemic, the classification of PHSA uncertainty as aleatory or epistemic is ambiguous.") So in an uncertainty quantification framework in which the goal is to obtain as output a point estimate or a probability distribution on a key performance measure by propagating the probability distributions associated with multiple sources of input uncertainty, there is typically no attempt to sort out the contribution due to each source of input uncertainty. That said, it is common practice to carry out a parametric analysis in which we effectively remove the probability distribution associated with an input parameter and simply vary the input parameter over a range of plausible values in order to assess the effect on the output for the key performance measure. Applying this

[45] Probabilistic Seismic Hazard Analysis

idea amounts to analyzing the output in a conditional manner, conditioned on the value of the corresponding input parameter. Such parametric analyses are usually done for one source of uncertainty at a time, as opposed to trying to simultaneously vary multiple input parameters.

Now, reconsider the probability distribution on the input parameter of focus. Output results for the key performance measure can be reported conditioned on the value of the input parameter, in turn, set to be specific quantiles from the input parameter's probability distribution. In this sense we can preserve the quantiles associated with a key input parameter when analyzing distributional output. The engineering analysis used to develop the basic event failure probabilities for the PRA uses an approach, likely new to PRA practitioners, that optionally integrates all uncertainty or preserves the quantiles of selected input distributions (which some may wish to label as being epistemic uncertainty). 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.

In the STPNOC Pilot Project quantification, the LOCA epistemic uncertainty on failure probability is quantified separately for each of the five ECCS pump combinations considered in the STPNOC Pilot Project 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 RISKMAN$^{\text{TM}}$ software used for the STPNOC Pilot Project 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 selected for the STPNOC Pilot Project quantification. 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 reevaluate 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 of ΔCDF and ΔLERF so that the uncertainty in the difference is calculated correctly.

The one exception to this correlation of input parameters among PRA model elements are those considered in CASA Grande. By necessity, the PRA is quantified using failure probability distributions developed in the CASA Grande analysis which are themselves functions of many data variables. In the STPNOC Pilot Project quantification, 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 RISKMAN$^{\text{TM}}$ 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 sparse sample of five bins on the distribution of failure probability is not an inherent limitation of the CASA Grande methodology, but was chosen only for the sake of current practicality. A more complete interrogation of the break–frequency uncertainty distribution can be made depending on the needs of the PRA. The initial presumption was that higher percentiles of the break–frequency distribution would lead to more conservative estimates of CDF and LERF, so more sampling resolution was placed in the upper tails of the envelope (see Fig. 5). The shape of each break–frequency profile defines the relative LOCA frequencies as a function of break size, as reflected in the variation between the five point estimates of failure probability. RISKMAN™ samples from the full uncertainty distribution, using 100 percentiles, for the absolute LOCA frequency and correlates each sample when evaluating the MLOCA and LBLOCA initiating event frequencies. 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 believed to be significant and therefore not modeled.

2.5.2 Parameter Uncertainty

Parameter uncertainties are addressed pervasively in the STPNOC PRA analysis. For the physical models addressing the concerns of GSI–191, input parameters were derived from both 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 LHS[46] schemes to accurately preserve the distribution. Human error probabilities are included in the STPNOC PRA however, for the most severe accident scenarios (that is LBLOCA), 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.

2.5.3 Model Uncertainty

As described on Page ix, the STPNOC PRA is supplied with failure probabilities resulting from GSI–191 phenomena developed from engineering models of the phenomena associated with the concerns raised in GSI–191. That is, in the PRA, the models are developed to be accurate representations of the plant including parameter uncertainties.

Over many years of study, the phenomena associated with the concerns raised in GSI–191 have been well characterized. However, the approach taken by most investigators in GSI–191 studies has been to demonstrate margin to performance limits by biasing inputs, not by studying uncertainty or actual performance in the as–built, as–operated plant. In the STPNOC PRA, investigators matched the phenomena to the performance of the as–built, as–operated plant.

In all cases, the difference between results of previous studies and results of the STPNOC GSI–191 studies can be explained by well–established analytical methods. The extensive body of work related to the issues raised in GSI–191 helps provide assurance that adequate models and methods are available to exploit.

Based on the STPNOC Pilot Project analysis performed, the most important contribution to CDF is the model of chemical effects, both on the strainer and in the core. Although (as mentioned previously in Section 2.1.2) chemical effects in STP post–LOCA fluid conditions are benign compared to the conditions assumed for the experiments performed in WCAP 16793–NP, the STPNOC Pilot Project assumes that adverse chemical effects can occur, both at the strainer and in the core. The STPNOC Pilot Project also uses bounding values for strainer differential pressure, that is,

[46] Latin Hypercube Sampling

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 15gm/FA limit found in WCAP 16793–NP as a threshold for the potential of boron precipitation.

In a classical interpretation, "model uncertainty" often refers to the degree of credibility held by one prediction of physical phenomena compared to that held by alternative predictions of the same phenomena. Formal methods have been developed to compare competing models that have been initialized with as near identical input as possible. Discrepancies between numerical predictions can then be used to quantify residual uncertainty in the prediction. These methods can even accommodate subjective measures of confidence that particular models (or none of the models) are more accurate than the others. Often, the primary difference between models lies in the degree of spatial resolution or physical fidelity, but sometimes, fully mature alternative methods are compared.

In the STPNOC Pilot Project, several new predictive models are being applied for the first time. 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. Relatively simple, first–order models are extremely useful for identifying trends, describing trade–offs between competing mechanisms, and prioritizing risk contributors; however, additional conservatism is warranted to explicitly acknowledge the uncertainty associated with the predictions of first–order models. For this reason, additional conservatism was incorporated in the treatment of both conventional and chemical–induced differential head-loss estimation. The practice of applying an overall inflation factor that is distributed in magnitude according to the best interpretation of available data represents the extent of model uncertainty that has been addressed in the STPNOC Pilot Project.

2.5.4 Completeness Uncertainty

Although prior investigations in GSI–191 have focused primarily on "test for success", they have nevertheless resulted in greater understanding and characterization of the post–LOCA behavior related to the concerns raised in GSI–191. In some cases, greater understanding has led to adoption of models that bound the experimental evidence simply because the space adopted is too large to fully explore experimentally. As a consequence, simplistic conservative approaches have been adopted where uncertainty is difficult to quantify [see 53]. On the other hand, STPNOC GSI–191 analysis has helped to extend 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 GSI–191 analysis uses realistic or prototypical conditions to model anticipated post–LOCA phenomena during all LOCA phases. Finally, where possible, 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 (UFSAR Chapter 15) analyses.

The confidence in completeness of the modeling scope for the concerns raised in GSI–191 is increased due to the number of years of study and work of independent investigators. In the STPNOC Pilot Project, all known physical models have been adopted and evaluated in the engineering analysis supporting the PRA.

As mentioned in Section 2.5.1, epistemic uncertainty has been considered in the STPNOC GSI–191 analysis. 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 possible 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. The STPNOC process for control of heavy loads [74] complies with Generic Letter 81–07, "Control of Heavy Loads," ANSI N14.6–1978 and NUREG 0612, and the TRM, Section 3/4.9.7.

- 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, for example see Mohaghegh [75]. STPNOC has a safety culture evaluation program that undergoes continuous improvement and examination.

With regard to plant operating states, some can be eliminated from further evaluation. These are under De–fueled conditions (No Mode), Refueling (Mode 6), and Cold Shutdown (Mode 5). 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.

The uncertainty quantification in the STPNOC GSI–191 PRA analysis 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.

2.5.5 Comparisons with Acceptance Guidelines

As mentioned in Part II, the STPNOC GSI–191 analysis shows that the risk associated with the concerns raised in GSI–191 is very small. Also, as defined byNuclear Regulatory Commission [10, Figures 4 and 5, Page 16] and previously mentioned in Part I, the STP average CDF and LERF are also very small. The estimates of ΔCDF and ΔLERF from the STPNOC GSI–191 analysis are far less than the Region III acceptance guidelines.

In the STPNOC GSI–191 PRA analysis, the mean values used to evaluate 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 GSI–191 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 GSI–191 analysis, they have been either reduced through experimental evidence or bounded through assumption as previously mentioned in Section 2.1.2. 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 GSI–191 analysis.

In the STPNOC Pilot Project analysis reliance on importance measures is not necessary nor 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 Pilot Project.

As discussed in Section 2.3, 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.

2.6 Integrated Decision Making

As discussed extensively in Section 2.1, there are many qualitative insights that form the basis for the conclusion in the STPNOC GSI–191 PRA analysis that there is a very small risk for the concerns associated with GSI–191. A significant effort has been expended to experimentally and analytically investigate the risk and uncertainties associated with the concerns raised in GSI–191.

Traditional engineering analysis, which generally ignores uncertainty, has been enhanced in the STPNOC GSI–191 PRA analysis by including parameter uncertainties. In as much of the analysis as possible, 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 GSI–191 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 GSI–191 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.

Part III Implementation and Monitoring

As stated in Part V, no changes are proposed to any programs, processes, or design with regard to the current as–built, as–operated plant that would result in a significant reduction to safety margin or DID. In particular, no changes are proposed to any ASME Section XI inspection programs [76, 77] or mitigation strategies that have been shown effective in early detection and mitigation of weld and material degradation in PWR Class I piping applications. STPNOC has adopted other programs that help provide early detection and mitigation of leakage in other applications [78]. Additionally, no changes are proposed to design modifications, processes, or programs that have resulted from addressing the concerns related to GSI–191 such as those mentioned in Section 2.1. In particular, design modifications that could affect any of these measures are specifically checked for in any design change [79, Checklist, Page 38].

Part IV
Submittal of Proposed Change

Proposed changes to the STP UFSAR, based on NRC approval of the STPNOC Pilot Project and LB change to resolve GSI–191, are submitted in the attachments to letter NOC–AE–13002954[80].

Part V
Quality Assurance

No design, operational, or performance changes are proposed to existing safety related systems, components, or structures in this analysis. Existing procedures and programs are unchanged by this license basis change. The STPNOC PRA analysis supporting the licensing basis change is performed using STPNOC PRA procedure as required for PRA analyses and assessments [31]. This is the STPNOC approved methodology for application evaluations using the PRA.

The support provided for the STPNOC PRA is performed by personnel qualified in their fields of expertise. All work performed in the licensing basis analysis is done following STPNOC procedures for contract personnel. An oversight program, Part VIII, is in effect for the duration of the entire project. All records and documentation are controlled under the STPNOC Document Control and Records Management systems. A detailed description of the Quality Assurance program supporting the STPNOC Pilot Project is provided in Volume 4.

Part VI
Documentation

1 Introduction

The total technical documentation consists of several volumes, Volume 1, Summary, Volume 2 PRA, Volume 3, the supporting engineering analysis, CASA Grande, Volume 4, Quality Assurance, and Volumes 5.1 through 5.4, Oversight. Additional documentation such as the PRA Model Revision 7 and support calculations are also made available through reference. In any case, all documentation is available in the STPNOC Records Management program.

2 Archival Documentation

Volumes 2 and 3 of the STPNOC GSI–191 license basis change submittal are detailed descriptions of the PRA and supporting engineering analyses conducted and results obtained. The analyses are primarily based on traditional engineering analyses that include experimental data obtained to specifically support the engineering models and analyses conducted as part of the licensing basis change. The full set of documentation created for this analysis are maintained as quality documents for the life of plant in the RMS[47] and can be retrieved using the following search fields and keywords:

- FSUG: D07090703,
- TYPE: VENDREC, and
- SUBTYPE: GSI191.

The STPNOC PRA model of record is also maintained in the RMS according to the normal PRA maintenance process and can be retrieved using the following search fields and keywords:

- FSUG: D6412,
- TYPE: DATA, and
- DOCUMENT NUMBER: 0PGP01ZA0305.

STPNOC PRA analyses are maintained in the STPNOC RMS. The PRA analysis performed for this work can be retrieved using the following search fields and keywords:

- FSUG: D64,
- TYPE: ANLYS, and
- DOCUMENT NUMBER: PRA13001.

Part VII
Submittal Documentation

The STPNOC proposed license basis change is consistent with the key principles of risk–informed regulation and NRC staff expectations based on the following points:

- The requirements for Long–Term Core Cooling summarized in 10 CFR§50.46 require the supporting systems to operate with a high level of probability including considerations of uncertainty. The licensing basis change requested quantifies the probability and uncertainty associated with long–term core cooling following the requirements as described in RG1.174. Based on the evaluation documented in the change request showing that the probability is very high that long–term core cooling will be satisfied, the impact to the licensing basis is insignificant.
- The proposed change has no impact on existing equipment performance requirements or performance assessment (equipment surveillance) requirements. For certain extremely low probability scenarios, when the extreme extent of the associated uncertainty is taken into account, the analysis shows that core damage could occur.
- No change to offsite dose or worker radiation dose is evaluated to occur. By implementing the proposed licensing basis change, a large worker radiation dose that would be incurred to mitigate a hypothesized event having insignificant likelihood is avoided.
- No change to existing DID is proposed. All equipment, as designed, is expected to be available and to continue to function with high probability.
- The proposed change is documented in the UFSAR, Chapter 6. No changes are proposed to any high–risk equipment.

In addition to the items listed above, the following also support consistency with the key principles of risk–informed regulation and NRC staff expectations:

[47] Records Management System

- The integrity of the Class 1 welds, piping, and components are maintained at a high level of reliability through the ASME Section XI inspection program;
- The materials stored in Containment, especially any transient lead, should be stored as required by Wire [81]. In addition, plant transients are monitored in the Transient Cycle Counting Limits Program [82];
- The structural integrity and cleanliness of the Containment Sump Strainers is monitored prior to leaving the containment [83, 84]. In particular, any condition noted that would result in direct passage of debris is evaluated through the Station Corrective Action Program [85] and repaired as necessary prior to Containment closeout. The PRA is maintained to reflect the as–built, as–operated plant as described in the STPNOC UFSAR, Section 13.7.2.3 to reflect the current plant design not to exceed every 36 months and to reflect the equipment performance (comprehensive data update) not to exceed 60 months. Unless major modifications are made to the containment design or insulation design, no changes should be required to the PRA analysis documented in this licensing submittal;
- Information to be provided as part of the plant LB (e.g., FSAR, technical specifications licensing condition);
- The GSI–191 PRA analysis is not used to enhance or modify safety–related functions of SSCs. The STPNOC GSI–191 PRA analysis is controlled under the existing STPNOC PRA application analysis and assessment process [31]; and
- There are no other changes to the existing requirements to any systems, structures or components as a consequence of this licensing basis change.

The program used to develop the results of the license basis change included an independent critical peer review oversight process requiring quarterly reporting and critical review question resolution. A summary of Independent Oversight activities and observations is addressed in Part VIII of this document. More details including Oversight comments and follow–up resolutions are available upon request (Independent Technical Oversight, Quarterly Reports [86, 87, 88, 89]).

As discussed on Page ix, minimal changes were made to the STPNOC PRA such that a new peer review would not be required. Although detailed models of post–LOCA behavior are included in the risk analysis, the models are not embedded in the PRA. Instead, detailed models of post–LOCA behavior are solved in an uncertainty quantification framework outside of the PRA and the results are supplied to the PRA as discrete probability distributions. In this way, contributions of specific issues raised in GSI–191 are encapsulated in familiar models and are therefore more easily scrutinized and understood, especially by investigators more familiar with the engineering models of behavior. Since much of the previous investigation into the issues raised in GSI–191 was not based on risk methodologies, the STPNOC GSI–191 analysis method is expected to be familiar to the majority of previous GSI–191 investigators.

STPNOC's PRA complies with Regulatory Guide 1.200, Revision 1, however; it does not comply with Regulatory Guide 1.200, Revision 2 with respect to Fire PRA and Seismic PRA requirements. Even though STPNOC's PRA contains both Fire and Seismic PRAs, they do not meet all the standards requirements in the current ASME/ANS RA–S–2009 PRA Standard, as endorsed by RG 1.200, Rev. 2, at a Capability Category II level. PRA model changes since the peer review are detailed in Volume 4, but are minimal. The Findings and Observations from the peer review are also reviewed in Volume 4.

STPNOC's PRA remains technically adequate to evaluate and quantify the risk associated with the concerns raised in GSI–191. GSI–191 is concerned with LOCA events and these events are explicitly modeled in the STPNOC PRA. STPNOC's PRA does meet Regulatory Guide 1.200, Revision 2 at Capability Category II for LOCA events. For the risk–informed GSI–191 methodology described in this study, the technical rigor provided to the PRA exceeds that performed in PRAs used today and is technically more than adequate to perform a risk–informed application meeting RG1.174 guidance.

Part VIII

Independent Technical Oversight

Since January 2012, Soteria Consultants, LLC (Soteria) has provided Independent Technical Oversight of the STPNOC STPNOC Pilot Project. STPNOC commissioned the oversight group to help ensure the quality and validity of the research and development undertaken. The main objective of Independent Technical Oversight has been to perform an in–depth scientific review of the phenomenological models and experiments developed and conducted for the STPNOC Pilot Project.

Soteria's approach included both "active" and "passive" oversight activities. Two members of Soteria Consultants (Dr. Zahra Mohaghegh[48] and Dr. Seyed Reihani[49]) interacted and collaborated with the technical teams to provide feedback and to offer active oversight services. Since the project involved new research, and because of its multidisciplinary and integrative nature, it required the oversight group to participate in meetings and to follow up on discussions and comments with the other team members. Specific areas of concerns and reviews were also discussed with Soteria's associate experts (that is, passive oversight members) including Dr. Ali Mosleh[50] and Dr. Reza Kazemi[51]

Soteria was involved in both "informal" and "formal" oversight activities for the STPNOC Pilot Project. Examples of informal activities were: (1) reviewing pre–meeting technical reports and documents related to NRC public meetings and providing comments; (2) providing technical support in developing ACRS presentations, and; (3) participating in brainstorming sessions on diverse technical topical areas with the required follow–up on the proposed ideas. Some of the formal Oversight activities included: (1) participating in weekly technical team teleconferences and providing feedback; (2) participating in monthly technical meetings and providing comments, and; (3) developing four Oversight Quarterly Reports [86, 87, 88, 89].

In order to make the review process more thorough and to enhance the effects and

[48]From Janary 2013, Assistant Professor in Nuclear Eng. Department at the University of Illinois at Urbana Champaign.

[49]From January 2013, Research Scientist in Nuclear Eng. Department at the University of Illinois at Urbana Champaign.

[50]Also, Professor of Mechanical Eng. Department at the University of Maryland, College Park.

[51]Also, Operations Research Analyst at the FDA (Individual's opinion and input to this project are his own personal views and do not reflect in any way that of the FDA).

efficiency of having an oversight function for the STPNOC Pilot Project, Soteria asked the technical team members to provide responses regarding each of Soteria's specific comments.

The main objectives of Oversight Quarterly Reports were to: (1) analyze the responses that Soteria had received from the members of the teams regarding oversight comments. The teams' responses were documented along with Soteria's responses, resolutions, and feedback on the unresolved issues; (2) provide an up–to–date report of Soteria activities during the quarter; (3) communicate additional comments based on the review of recent reports and participation in the technical meetings and teleconferences, and; (4) facilitate the interaction and collaboration of the oversight team with members of other technical teams. The Oversight Quarterly Reports contributed to the progress of the project by addressing critical peer review of the documents and by highlighting an up–to–date elaboration of areas of concern that required further investigation from the technical teams.

From Soteria's perspective, the STPNOC Pilot Project is an outstanding blend of advanced and conventional methods that not only contributes towards the closure of the GSI–191 issues, but also makes a significant contribution to the formal incorporation of underlying physical failure mechanisms of certain post–LOCA events into PRA. Soteria's oversight activities have concluded that the STPNOC Pilot Project, having a well–designed combination of probabilistic and deterministic methodologies, has made important contributions to the closure of GSI–191 issues. The detailed technical results of Soteria's critical reviews are available in the four Oversight Quarterly Reports [86, 87, 88, 89].

In addition to reviewing the various working documents and analyses in FY 2012, Soteria has been reviewing Volumes 1, 2, 3, and 4 of the submittals and their supporting documents. The members of technical teams (that is, PRA GSI–191 Analysis & Methodology Implementation; GAMI, Corrosion/Head Loss Experiments; CHLE, CASA Grande, Thermal Hydraulics; TH, Uncertainty Quantification; UQ, and Jet Formation; JF) have responded to and implemented the majority of Soteria's comments. Some specific comments (e.g., related to vertical head–loss tests and blender bed tests, etc.) have not yet been implemented, mainly due to time and budget constraints. The plan is to address these along with NRC's additional comments in FY 2013. The four Oversight Quarterly Reports [86, 87, 88, 89] include the resolution status of Soteria's comments.

Because of the large–scale nature of the STPNOC Pilot Project, Soteria believes that follow–up research, implementation, and experiments in FY 2013 would certainly improve the quality and validity of the project. For FY 2013, Soteria team members, who have joined the academic staff of the University of Illinois at Urbana Champaign, will continue the technical oversight function during ongoing technical work and the NRC review process.

Part IX
Acronyms & Notations

BAT Boric Acid Tank either one of two highly concentrated boric acid supply tanks that provide the ability to increase boron concentration in the RCS and connected systems.

CAD Computer Aided Design a computer aided design model STPNOC is using to represent the containment buildings that includes piping welds and insulation details in order to help accurately assess ablated materials following an hypothesized LOCA.

CASA Grande Containment Accident Stochastic Analysis (CASA) and Grande refers to the STPNOC large, dry containment is the framework used to perform the computerized uncertainty quantification (sampling of distributions, propagating uncertainties) to develop basic events that address the issues raised in GSI-191.

ccdf Complementary Cumulative Distribution Function: $\bar{F}(x) = 1 - \int_{-\infty}^{x} f(t)dt$, where $f(\cdot)$ is the pdf.

CCW Component Cooling Water System is a part of the STP Engineered Safety Systems and consists of three trains (Trains A, B, and C). CCW is cooled by the ECW.

CDF Core Damage Frequency is calculated at STPNOC using the STPNOC PRA.

cdf Cumulative Distribution Function: $F(x) = \int_{-\infty}^{x} f(t)dt$, where $f(\cdot)$ is the pdf.

CET Core Exit Thermocouple refers to one of the array of thermocouples arranged at the exit of the fuel assemblies in the STP core. The thermocouple data is used to help identify adverse trends in functions (for example, core cooling) and decision points in the CSFSTs to direct response actions..

CHRS Containment Heat Removal System is comprised includes the CSS and RCFCs. These systems mitigate the potential consequences of a LOCA or main steam line break.

CLB Cold Leg Break is a failure in the RCS piping between the steam generator cold leg nozzle and the reactor vessel cold leg nozzle.

CSFST Critical Safety Function Status Tree is one of several decision trees linked specific critical measurements used in the EOPs as necessary to direct decisions to restore required functions (for example, core cooling) in an event.

CSS Containment Spray System is a part of the STP Engineered Safety Systems and consists of three trains (Trains A, B, and C). Only two Containment Spray trains are required to meet the system's spray flow requirements. The STPNOC containment spray flow does not pass through the RHR heat exchanger.

CVCS Chemical Volume and Control System is the system that maintains the pressurizer level, RCS chemistry (chemical addition, ion control, filtering), and seal water flow during normal operation.

D_{break} the scenario-dependent break diameter. The break diameter is limited to the pipe diameter at the scenario-dependent break location. Any break diameter larger than the pipe diameter is assumed to be a double-ended guillotine break.

D_{pipe} the diameter of the pipe where a scenario-dependent break occurs.

DDTS Drywell Debris Transport Study is the NRC-sponsored Boiling Water Reactor study of the blowdown and washdown of debris to the suppression pool during LOCA.

DEGB Double-Ended Guillotine Break is a hypothetical condition that can be realized mathematically whereby a pipe instantaneously shears around its circumference and in the same instantaneous time, completely offsets such that the jets from each end of the shear plane can't interfere with each other.

DID Defense–in–Depth is the design concept that includes redundant and/or multiple barriers to a particular consequence.

ECCS Emergency Core Cooling System part of the STPNOC engineered safety features.

ECW Essential Cooling Water System a part of the STP Engineered Safety Systems and consists of three trains (Trains A, B, and C). The ECW takes and returns water through the Ultimate Heat Sink, a hardened cooling pond.

EOF Emergency Operations Facility is the support facility for the management of overall licensee emergency response (including coordination with Federal, State, and local officials), coordination of radiological and environmental assessments,.and determination of recommended public protective actions. The EOF also has technical data displays and plant records to assist in the diagnosis of plant conditions.

EOP Emergency Operating Procedure one of several plant procedures entered following reactor trip or SI that, in conjunction with other plant operating procedures, direct actions to avoid or mitigate any degraded plant state and help ensure the plant will arrive in a safe shut down condition following the trip or SI.

η is an empirically-derived constant related to the rate of fiber release through the strainer based on flow rate through the strainer.

$\mathcal{E}$ is a shifted exponential random variable used in computing the chemical bump-up factor, Φ_{ch}.

$f(\cdot)$ is the empirically–derived ECCS strainer filter efficiency as a function of the mass on the strainer.

$F_{transport}$ is an operator that applies transport logic to obtain the mass of all ZOI-generated debris arriving at the pool.

F^{ℓ}_{fill} is the fill up transport fraction to train ℓ's strainer sump cavity.

FHB is the fuel handling building, containing the high head safety injection system, low head safety injection system, and containment spray pumps.

$\mathcal{F}$ is an index set for all types of fiber-based insulation products in containment.

F_{void} Void Fraction is a function that computes the liquid vapor fraction just downstream of the ECCS strainer.

$F_{D_{break}|\,weld\,case}$ is the conditional distribution governing the random break diameter, D_{break}, given that a break occurs at a specified weld type/case.

ϕ is the scenario-dependent azimuthal angle of the break around the pipe.

$\Phi_{ch}(\cdot)$ is the time- and scenario-dependent chemical bump-up factor used in computing head loss across ECCS sump strainer.

γ_ℓ Fraction of total time-dependent ECCS flow ($Q^\ell(t)$) that passes through train ℓ's ECCS strainer and arrives in the RCS. The index $\ell = A, B, C$ refers to the associated train.

GL 2004-02 NRC Generic Letter 2004-02 was issued in response to the concerns raised in GSI-191 for PWRs.

GSI–191 Generic Safety Issue 191 the NRC Generic Safety Issue number 191.

$H(\cdot)$ Function based on NUREG-6224 used in computing head loss across ECCS sump strainer.

HHSI High Head Safety Injection a part of the ECCS. The STPNOC plants have three HHSI trains (Trains A, B, and C) that can provide ECCS flow at pressures up to around 1600 psi. The STPNOC HHSI flow does not pass through the RHR heat exchanger.

HLB Hot Leg Break is a failure in the RCS piping between the steam generator hot leg nozzle and the reactor vessel hot leg nozzle including the Pressurizer (D Loop).

ISI ASME Section XI Inservice Inspection is an ASME Section XI program that, among other things, verifies the weld integrity in critical piping.

i when used as a subscript refers to one of three damage zones in the ZOI $i = 1, 2, 3$.

j when used as a subscript refers to the size of debris generated within a damage zone (fines, small pieces, large pieces, and intact blankets) associated with the three assumed damaged radii for a scenario-dependent break.

k when used as a subscript refers to debris type which can come from $\mathcal{K}$ (all types of insulation products in containment), $\mathcal{F}$ (fiber-based insulation products), or $\mathcal{L}$ (all types of debris), where $\mathcal{F} \subset \mathcal{K} \subset \mathcal{L}$.

$\mathcal{K}$ is an index set for all types of insulation products in containment.

$\mathcal{L}$ is an index set for all types of debris in containment including insulation, crude particulate, unqualified coatings, and latent debris.

ℓ when used as a super- or subscript refers to ECCS sump strainers for train ℓ for $\ell = A, B, C$.

λ the fraction of the total ECCS flow arriving in the vessel.

LB Licensing Basis is the collection of commitments and requirements that licensee makes to the regulatory authority (in this case, the NRC) over the course of time.

LBB Leak before break is a proposed licensing approach that relies on the observation that prior to a catastrophic failure in large bore piping, a small, detectable flow initiates.

LERF Large Early Release Frequency STPNOC calculates large early release frequency using the STPNOC PRA.

LHS Latin Hypercube Sampling is a simulation-based procedure that generalizes the notion of stratified sampling to multiple dimensions and yields an unbiased point estimate, while attempting to reduce variance of the estimator over naive Monte Carlo sampling.

LHSI Low Head Safety Injection part of the ECCS. The STPNOC plants have three LHSI trains (Trains A, B, and C) that can provide ECCS flow at pressures up to around 400 psi. The LHSI train is the only ECCS train that uses the RHR heat exchangers for decay heat removal.

LBLOCA Large Break Loss of Coolant Accident a hypothetical instantaneous pressure boundary failure that is defined for STPNOC as greater than 6 inch equivalent diameter.

LOCA Loss of Coolant Accident a hypothetical instantaneous pressure boundary failure.

$M_{i,j,k}$ is the scenario-dependent mass of debris type k of size j originating from damage zone i.

$m^{\ell}_{i,j,k}(\cdot)$ is the scenario- and time-dependent mass of debris type k of size j originating from damage zone i at train ℓ's ECCS sump strainer.

$m^{P}_{i,j,k}(\cdot)$ is the scenario- and time-dependent mass of debris of type k of size j originating from damage zone i, in the containment sump pool.

$m^{core}_{k}(\cdot)$ is the scenario- and time-dependent mass accumulation of debris on the core (fuel assemblies). All debris that arrives at the core is assumed to deposit on the core. The subscript k is restricted to $k \in \mathcal{F}$, indicating that only fiber is transported to the vessel.

MLOCA Medium Break Loss of Coolant Accident a hypothetical instantaneous pressure boundary failure that is defined for STPNOC as greater than 2 inch equivalent diameter but less than 6 inch equivalent diameter.

ν is an empirically-fit parameter that expresses the fraction of all transportable debris that is mobile enough to be released via shedding from a strainer's accumulated bed of debris.

NLHS Nonuniform Latin Hypercube Sampling is a variant of the LHS scheme that allows the support of each marginal distribution to be partitioned into cells with non–equal probabilities.

NPSHA Net Positive Suction Head Available is the total pressure at the eye of the pump impeller. As long as the net positive suction head available is higher then the net positive suction required, the pump will have sufficient pressure at the impeller inlet to operate without cavitation.

NPSHR Net Positive Suction Head Required is the total pressure at the eye of the pump impeller required for the pump to operate properly, without excessive cavitation.

$NPSH_{margin}(\cdot)$ is the time-dependent NPSH margin; i.e., the difference between the NPSH available and the NPSH required.

NSSS Nuclear Steam Supply System the nuclear reactor, piping, pumps, steam generators, pressurizer, and auxiliary equipment associated with operation and control of the reactor system.

STPNOC Pilot Project STPNOC Risk-Informed GSI–191 Closure Pilot Project. The NRC works with licensees as they develop methods to address new regulatory approaches. STPNOC requested and was granted Pilot Project status for the methodology for closing GSI–191 using Option 2b

P_{buckle} Strainer structural design limit is the differential pressure across the ECCS strainers at which they are analyzed to be within code design allowable stresses. The limit is approximately 9.35 ftWC.

pdf A probability density function specifies the relative likelihood that a continuous random variable takes on a specific value. When integrated over a region, representing an event, the pdf yields the probability mass associated with the event, as in the probability of observing a break diameter between 2-inches and 5-inches.

PHSA Probabilistic Seismic Hazard Analysis is the probabilistic study of seismic events on systems, structures, and components to obtain failure likelihoods.

PRA Probabilistic Risk Assessment the STPNOC PRA is the platform for all quantitative risk assessment licensing activities at STPNOC. The current model (Model of Record) is Revision 7.

PWR Pressurized Water Reactor uses steam generators to isolate the Rankine steam cycle from the reactor coolant system. The STPNOC site consists of two, four loop, approximately 3850 MWth, Westinghouse Nuclear Steam Supply System reactors.

PWSCC Primary Water Stress Corrosion Cracking is a degradation mechanism for certain types of weld materials, especially Alloy 600.

$\Delta P^{\ell}(\cdot)$ time- and scenario-dependent head loss across train ℓ's strainer.

ΔP_{mech} fixed mechanical collapse criterion.

$Q^{\ell}(\cdot)$ time-dependent ECCS flow through strainer, where $\ell = A, B, C$ refers to the train.

QDPS Qualified Display Processing System is the STP computer system that displays critical information and controls certain critical functions in an event including information needed for CSFST monitoring.

ρ_k is the density of the k^{th} insulation material.

$R_{i,j,k}$ the scenario-dependent radius for the damage zone at the break location. The subscripts i, j, k refer to the damage zone, debris size, and insulation type.

RCB Reactor Containment Building is an additional barrier to release to the environment should both the fuel clad and RCS fail to contain radioactive fission products.

RCFC The Reactor Containment Fan Coolers a part of the STP Engineered Safety Systems and consist of three trains (Trains A, B, and C).

RCS Reactor Coolant System. The STPNOC reactor coolant system is a four loop Westinghouse design.

RG1.174 Regulatory Guide 1.174 is a regulatory guidance document that describes the overall methodology to quantify risk using the PRA together with deterministically-based criteria to evaluate the acceptability of a particular change. The quantitative risk measures are CDF and LERF. The risk is deemed to be "very small" when the change increases CDF less than 10^{-6} and the LERF less than 10^{-7}.

RHR Residual Heat Removal System a shutdown cooling system consisting of three independent trains. The RHR heat exchangers are shared with the LHSI train. If the LHSI train is using the heat exchanger for that train, the RHR train must be secured and vice versa.

RMI Reflective Metal Insulation is a fitted, rigid insulation that uses metal radiation heat shields and dead air space to reduce heat loss.

RMS Records Management System is the STPNOC document storage and retrieval system meeting the requirements of Regulatory Guide 1.33, Revision 2, Quality Assurance Program Requirements (Operation).

RMTS Risk Managed Technical Specifications changes the allowed outage time for risk significant equipment as derived from the configuration risk during the outage time.

RVWL Reactor Vessel Water Level is the STP (two trains) level measuring instruments in the reactor vessel that use heated junction thermocouples to detect the presence of liquid.

RWST Refueling Water Storage Tank the STPNOC reactor water storage tank holds approximately 500,000 gallons of water borated to the all rods out, xenon free boron concentration, approximately 2800 ppm.

$S_k(\cdot)$ time-dependent rate at which debris mass is added from each source, k, to the containment pool.

SI Safety Injection System is comprised of the valves, piping, pumps, and accumulators designed to deliver water the RCS following an actuation signal. The actuation signal for LOCA would be 2 out of 4 Pressurizer pressure signal (at the SI actuation set point).

SLOCA Small Break Loss of Coolant Accident is a hypothetical instantaneous pressure boundary failure that is defined for STPNOC as less than 2 inch equivalent diameter and greater than 1/2 inch equivalent diameter.

STP South Texas Project Electric Generating Station is the two commercial nuclear electric generating units located near Wadsworth, TX.

STPNOC The STP Nuclear Operating Company is the organization responsible for the safe and efficient operation of the South Texas Project electric generating station.

$T(t)$ is the scenario– and time–dependent temperature history correlated to thermal–hydraulic trends for SLOCA, MLOCA, or LBLOCA events.

TSC Technical Support Center is the onsite facility located in the STP power block (two identical facilities provided for each STP Unit) control room that provide plant management and technical support to the control room personnel located during emergency conditions.

VCT Volume Control Tank is a large surge volume provided in the CVCS to accommodate changes in water volume requirements in the RCS and connected systems while maintaining constant pressurizer level (for example).

$V^P(\cdot)$ is the scenario- and time-dependent volume of water in the containment pool.

$V^{i,j}_{damage}(\phi)$ is the scenario-dependent enclosed space of the damage zone indexed by i generating insulation debris size j. The argument ϕ refers to the angle associated with the hemispherical ZOI for non-guillotine breaks.

$V^k_{insulation}$ is the space of insulation of type $k \in \mathcal{K}$ within containment.

$W_{concrete}$ is the space that the concrete walls clip at the scenario-dependent break location.

ZOI Zone of Influence refers to the enclosed volume where damage to materials is hypothesized or assumed to occur. The damage assumed is from the energetic jet associated with the hypothesized instantaneous failure of Class 1 piping in the containment building.

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Appendices

Appendix A is a table with three columns, "Section", "Paragraph Summary", and "Where Addressed" developed to help ensure the requirements of RG1.174 have been addressed in the STPNOC Pilot Project. The first column, "Section", highlights the four elements identified in RG1.174. In an attempt to identify all sub-elements, items that clearly bear on the information needed were pulled out of the text and entered in the column "Paragraph Summary". The "Where Addressed" column primarily refers to the Section in this document (Volume 1) where the requirement is addressed. As mentioned in the Volume 1 Introduction & Background, the numbered sections of Volume 1 correspond to the numbered sections in RG1.174 which should also help in this regard.

Appendix B is a table with four columns, "Topical Area", "NRC-Approved Deterministic Methods", "STPNOC Pilot Project Methods for 2012 Quantification," and "Comments." The table is intended to help understand how the engineering analysis supporting the PRA used in the STPNOC Pilot Project relates to the NEI 04-07 recommended models. In particular, the collection of engineering models used in the CASA Grande analysis are itemized against the recommendations. NEI 04-07. "Topical Area" is the GSI–191 engineering model subject area. "NRC-Approved Deterministic Methods" is the methodology approved by the NRC for the particular topical area (not all topical areas had approved models at the time the STPNOC Pilot Project was completed). "STPNOC Pilot ProjectMethods for 2012 Quantification" is a quick description of the engineering model used in the STPNOC Pilot Project. "Comments" provides information about whether the model is the same (that is, "no difference") or a summary description of how the model adopted differs or in some cases is closely related to the NRC's model choice.

Appendix C is a detailed description of the DID and Safety Margin measures in place at STP as well as the measures STPNOC has taken in response to the GSI–191 issue.

A Checklist for Regulatory Guide 1.174 Inputs

Table 5: Checklist for Regulatory Guide 1.174

Section	Paragraph Summary	Where addressed
Element 1: Define the Proposed Change	Identify those aspects of the plants LB that may be affected by the proposed change, including but not limited to rules and regulations, FSAR, technical specifications, licensing conditions, and licensing commitments.	Part I, Page 3.
	Identify all structures, systems, and components (SSCs), procedures, and activities that are covered by the LB change being evaluated and should consider the original reasons for including each program requirement	Part I, Page 3.
	Identify all structures, systems, and components (SSCs), procedures, and activities that are covered by the LB change being evaluated and should consider the original reasons for including each program requirement	Prior changes and primary STPNOC processes bearing on this LB change are summarized in Part I, Page 3
	Identify regulatory requirements or commitments in its LB that it believes are overly restrictive or unnecessary to ensure safety at the plant.	GSI–191 and Generic Letter 2004-02 overly restrictive based on actual plant analysis.
	Identify design and operational aspects of the plant that should be enhanced consistent with an improved understanding of their safety significance. Such enhancements should be embodied in appropriate LB changes that reflect these enhancements.	No additional changes to the plant are recommended beyond the those already implemented. Part III

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Section	Paragraph Summary	Where addressed
	Identify available engineering studies, methods, codes, applicable plant-specific and industry data and operational experience, PRA findings, and research and analysis results relevant to the proposed LB change. With particular regard to the plant-specific PRA, the licensee should assess the capability to use, refine, augment, and update system models as needed to support a risk assessment of the proposed LB change.	Overview on Page ix, Figure 2. Further details provided in Volume 3. The PRA capability is described in Part II, Section 2.3 and further details are provided in Volumes 2 and 4.
	Describe the LB change and outline the method of analysis. The licensee should describe the proposed change and how it meets the objectives of the NRCs PRA Policy Statement (Ref. 1), including enhanced decision making, more efficient use of resources, and reduction of unnecessary burden.	Part II, Page 5
Combined Change Requests	Licensees may include several individual changes to the LB that have been evaluated and will be implemented in an integrated fashion.	This section is not applicable to the STPNOC Pilot Project.
Guidelines for Developing Combined Change Requests	The changes that make up a CCR should be related to one another.	This section is not applicable to the STPNOC Pilot Project.
Element 2: Perform Engineering Analysis	The scope, level of detail, and technical adequacy of the engineering analyses conducted to justify any proposed LB change should be appropriate for the nature and scope of the proposed change.	Part II. Defense-in-Depth is detailed in Appendix C. Detailed description is provided in Volume 3.
	Some proposed LB changes can be characterized as involving the categorization of SSCs according to safety significance.	Not applicable to this LB change.

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Section	Paragraph Summary	Where addressed
Evaluation of Defense-in-Depth Attributes and Safety Margins	Evaluate the proposed LB change with regard to the principles of maintaining adequate defense-in-depth, maintaining sufficient safety margins, and ensuring that proposed increases in CDF and risk are small and are consistent with the intent of the Commissions Safety Goal Policy Statement.	Part II, Section 2.1 summarizes Defense in Depth and Safety Margin. The risk is very small, (Part II, Page 7) and well within the Commissioners' safety goal.
	Show that the fundamental safety principles on which the plant design was based are not compromised by the proposed change.	No changes are proposed to plant design principles beyond those taken in response to the concerns raised in GSI-191. Part II, Paragraph 2.1.1.1.
	Evaluate whether the impact of the proposed LB change (individually and cumulatively) is consistent with the defense-in-depth philosophy.	Part II, Paragraph 2.1.1.2, Appendix C, Pages C1 to C3
	The evaluation should consider the intent of the general design criteria	Part II, Paragraph 2.1.1.1. Appendix C, Pages C8 to C9 has detailed descriptions of the affected GDC.

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Section	Paragraph Summary	Where addressed
	Assess whether the proposed LB change meets the defense-in-depth principle.	Paragraph 2.1.1.2 is a summary. Appendix C provides a detailed discussion of the defense-in-depth at regarding the concerns raised in GSI-191.
	Assess whether the impact of the proposed LB change is consistent with the principle that sufficient safety margins are maintained.	Section 2.1.2
Evaluation of Risk Impact, Including Treatment of Uncertainties	Risk assessment may be used to address the principle that proposed increases in CDF and risk are small and are consistent with the intent of the NRCs Safety Goal Policy Statement	Part II, Section 2.2.
	Impacts of the proposed change on aspects of risk not captured (or inadequately captured) by changes in CDF and LERF should be addressed. For example, changes affecting long-term containment performance would impact radionuclide releases from containment occurring after evacuation and could result in substantial changes to off- site consequences such as latent cancer fatalities.	Part II, Section 2.2.
Technical Adequacy of Probabilistic Risk Assessment Analysis	The scope, level of detail, and technical adequacy of the PRA are to be commensurate with the application for which it is intended and the role the PRA results play in the integrated decision process.	Part II, Section 2.3 and Section 2.3.1.
	Both aleatory and epistemic uncertainty should be evaluated. An understanding of the important contributors in the model should be developed.	Section 2.5.3.
Acceptance Guidelines	Regions are established in the two planes generated by a measure of the baseline risk metric (CDF or LERF) along the x-axis, and the change in those metrics (CDF or LERF) along the y-axis (Figures 4 and 5). Acceptance guidelines are established for each region.	Part II, Section 2.4.

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Section	Paragraph Summary	Where addressed
	It is recognized that many PRAs are not full scope and PRA information of less than full scope may be acceptable.	The scope and technical adequacy of the STPNOC PRA is also described in Part II, Section 2.3.3
	There are two sets of acceptance guidelines, one for CDF and one for LERF, and both sets should be used.	The STPNOC PRA evaluates both CDF and LERF. Both of these metrics are included in the STPNOC Pilot Project acceptance criteria (Part II, Section 2.2).
Comparison of PRA results with acceptance guidelines	In the context of integrated decision making, the acceptance guidelines should not be interpreted as being overly prescriptive. They are intended to provide an indication, in numerical terms, of what is considered acceptable.	Part II, Section 2.5.
	The assumptions made in response to these sources of model uncertainty and any conservatism introduced by the analysis approach can bias the results. This is of particular concern for the assessment of importance measures with respect to the combined risk assessment and the relative contributions of the hazard groups to the various risk metrics.	Importance measures are not relied on in the STPNOC Pilot Project (Page 39)

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Section	Paragraph Summary	Where addressed
	Comparison of the PRA results with the acceptance guidelines must be based on an understanding of the contributors to the PRA results and on the robustness of the assessment of those contributors and the impacts of the uncertainties, both those that are explicitly accounted for in the results and those that are not.	Section 2.5. Other contributors are captured in epistemic uncertainty as well as adoption of extreme thresholds for failure, especially in consideration of Boron Precipitation, ECCS strainer differential pressure and core blockage. Table 6. See Page 35.
	The analysis must be done to correlate the sample values for different PRA elements from a group to which the same parameter value applies.	Part II, Section 2.5.1, a description of uncertainty quantification is given in Part II, Section 1.3 and the modeling of dependencies in the engineering analysis is described in Part II, Section 1.3.3

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Section	Paragraph Summary	Where addressed
	It is important to develop an understanding of the impact of a specific assumption or choice of model on the predictions of the PRA. This is true even when the model uncertainty is treated probabilistically, since the probabilities, or weights, given to different models would be subjective. The impact of using alternative assumptions or models may be addressed by performing appropriate sensitivity studies or by using qualitative arguments, based on an understanding of the contributors to the results and how they are impacted by the change in assumptions or models.The impact of making specific modeling approximations may be explored in a similar manner.	Part II, Section 2.3.5 provides an example illustration of how the analysis provides understanding of engineering model impacts on the results.
	In many cases, the appropriateness of the models adopted is not questioned and these models have become, de facto, the consensus models to use.	Appendix B compares models used compared with industry de facto models. Part II, Section 2.5.1 and Section 2.3 also address model appropriateness.
Completeness Uncertainty	The issue of completeness of scope of a PRA can be addressed for those scope items for which methods are in principle available, and therefore some understanding of the contribution to risk exists, by supplementing the analysis with additional analysis to enlarge the scope,using more restrictive acceptance guidelines,or by providing arguments that, for the application of concern, the out-of-scope contributors are not significant.	Part II, Section 2.5.4
Comparisons with Acceptance Guidelines	Comparison with acceptance guidelines.	Part II, Section 2.5.5
Integrated decision making	In making a regulatory decision, risk insights are integrated with considerations of defense-in-depth and safety margins.	Part II, Section 2.6

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Section	Paragraph Summary	Where addressed
Element 3: Define Implementation and Monitoring Program	Careful consideration should be given to implementation of the proposed change and the associated performance-monitoring strategies. The primary goal of Element 3 is to ensure that no unexpected adverse safety degradation occurs due to the change(s) to the LB.	Part III
Element 4: Submit Proposed Change	Requests for proposed changes to the plants LB typically take the form of requests for license amendments (including changes to or removal of license conditions), technical specification changes, changes to or withdrawals of orders, and changes to programs under 10 CFR 50.54, "Conditions of Licenses" (e.g., quality assurance program changes under 10 CFR 50.54(a)).	Part IV
Documentation	To facilitate the NRC staffs review to ensure that the analyses conducted were sufficient to conclude that the key principles of risk-informed regulation have been met, documentation of the evaluation process and findings are to be maintained.	Part VI
	As part of evaluation of risk, licensees should understand the effects of the current application in light of past applications.	The STPNOC PRA is maintained current with the plant including application impacts as described in Part VII.

B NEI 04-07 Comparison

Table 6: Comparison of NEI 04–07 recommended engineering models with the models implemented in the STPNOC Pilot Project

Topical Area	**NRC–Approved Deterministic Methods**	**STPNOC Pilot Project Methods for 2012 Quantification**	**Comments**
Debris Generation	Use spherical or hemispherical ZOI	Use spherical or hemispherical ZOI	No difference.
	17D ZOI for Nukon and Thermal–Wrap	17D ZOI for Nukon and Thermal–Wrap	No difference.
	28.6D ZOI for Microtherm	28.6D ZOI for Microtherm	No difference.
	4D ZOI for qualified coatings	4D ZOI for qualified coatings	No difference.
	Truncate ZOI at walls	Truncate ZOI at walls	No difference.
	4–category size distribution for fiberglass debris including fines, small pieces, large pieces, and intact blankets	Alion proprietary 4–category size distribution methodology (consistent with guidance in SER appendices)	Alion 4 category size distribution methodology previously accepted by NRC for deterministic evaluations.
	100% fines for Microtherm debris	100% fines for Microtherm debris	No difference.
	100% fines (10μ) for qualified coatings debris	100% fines (10μ) for qualified coatings debris	No difference.
	100% failure for all unqualified coatings debris	Time–dependent failure of unqualified coatings based on available data.	New methodology documented in Volume 3.
	Unqualified coatings fail as 10μ particles if the strainer is fully covered or as chips if a fiber bed would not be formed.	Unqualified coatings fail in a size distribution based on coating type and available data.	Similar methods previously accepted by NRC for deterministic evaluations.

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Topical Area	NRC–Approved Deterministic Methods	STPNOC Pilot Project Methods for 2012 Quantification	Comments
	Plant–specific walkdowns required to determine latent debris quantity	STP–specific walkdown used to determine latent debris quantity	No difference.
	Latent debris consists of 85% dirt/dust and 15% fiber	Latent debris consists of 85% dirt/dust and 15% fiber	No difference.
Debris Transport	Logic tree approach to analyzing transport phases: blowdown, washdown, pool fill, recirculation, and erosion	Logic tree approach to analyzing transport phases: blowdown, washdown, pool fill, recirculation, and erosion	No difference.
	All large pieces and a portion of small pieces are captured when blowdown flow passes through grating.	Fines transport proportional to containment flow, grating and miscellaneous obstructions capture some small and large pieces.	Similar methods previously accepted by NRC for deterministic evaluations.
	100% washdown of fines, limited credit for hold–up of small pieces, and 0% washdown of large pieces through grating	100% washdown of fines. Credit for hold–up of some small piece debris on concrete floors and grating. 0% washdown of large pieces through grating.	Includes some new methodology documented in Volume 3.
	Pool fill transport to inactive cavities must be limited to 15% unless sufficient justification can be made	Pool fill transport to inactive cavities is less than 15%. Methodology is based on exponential equation with uniform mixing of fines.	Similar methods previously accepted by NRC for deterministic evaluations.

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Topical Area	NRC–Approved Deterministic Methods	STPNOC Pilot Project Methods for 2012 Quantification	Comments
	CFD refinements are appropriate for recirculation transport, but a blanket assumption that all debris is uniformly distributed is not appropriate.	Recirculation transport based on conservative CFD simulations developed for the deterministic STP debris transport calculation. All debris was not assumed to be uniformly distributed.	Methodology for CFD modeling and recirculation transport analysis previously accepted by NRC for deterministic evaluations.
	90% erosion should be used for non–transporting pieces of unjacketed fiberglass in the recirculation pool unless additional testing is performed to justify a lower fraction.	Probability distribution with a range of less than 10% erosion based on Alion testing.	Values are relatively close to the experimentally determined 10% erosion value previously accepted by the NRC for deterministic evaluations.
	1% erosion of small or large pieces of fiberglass held up in upper containment.	1% erosion of small or large pieces of fiberglass held up in upper containment.	No difference.
	Minimal previous analysis on time–dependent transport.	Time–dependent transport evaluated for pool fill, washdown, recirculation, and erosion.	Several aspects of the time–dependent transport are new engineering models documented in Volume 3.
Chemical Effects	Corrosion and dissolution of metals and insulation in containment is a function of temperature, pH, and water volume. Accepted model is WCAP–16530–NP.	WCAP 16530 NP model used to calculate corrosion for wide range of scenarios, and inform engineering judgment for chemical effects bump–up factors.	Overall chemical effects evaluation is a new approach as documented in Volume 3 CASA Grande Analysis.

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Topical Area	**NRC–Approved Deterministic Methods**	**STPNOC Pilot Project Methods for 2012 Quantification**	**Comments**
	100% of material in solution will precipitate.	Some material in solution may not precipitate depending on the temperature–dependent solubility limit of the precipitate.	Overall chemical effects evaluation is a new approach as in Volume 3 CASA Grande Analysis.
	Precipitates can be simulated using the surrogate recipe provided in WCAP–16530–NP.	Chemical products generally appear to be more benign than WCAP surrogate.	Overall chemical effects evaluation is a new approach as documented in Volume 3 CASA Grande Analysis.
Strainer Head Loss	Perform plant–specific head loss testing of the bounding scenario(s) with a prototype strainer module.	Use the NUREG/CR–6224 correlation so that head loss can be evaluated at the full range of scenarios.	Approach documented in Volume 3 CASA Grande Analysis.
	Address chemical effects head loss using WCAP–16530–NP surrogates in prototype strainer testing.	Address chemical effects head loss with bump–up factor conditional probability distributions.	Overall chemical effects evaluation is a new approach as documented in Volume 3 CASA Grande Analysis.
	Minimum fiber quantity equivalent to 1/16 inch debris bed on the strainers is required to form a thin bed.	Minimum fiber quantity equivalent to 1/16 inch debris bed on the strainers is required to form a thin bed.	No difference.
	Bounding strainer head loss compared to bounding NPSH margin and bounding structural margin to determine whether the pumps or strainer would fail.	Time–dependent strainer head loss compared to time–dependent NPSH margin and bounding structural margin to determine whether the pumps or strainer would fail.	Similar engineering model as documented in Volume 3.

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Topical Area	**NRC–Approved Deterministic Methods**	**STPNOC Pilot Project Methods for 2012 Quantification**	**Comments**
Air Intrusion	Release of air bubbles at the strainer calculated based on the water temperature, submergence, strainer head loss, and flow rate.	Release of air bubbles at the strainer calculated based on the water temperature, submergence, strainer head loss, and flow rate.	No difference.
	NPSH margin adjusted based on the void fraction at the pump inlet	NPSH margin adjusted based on the void fraction at the pump inlet	No difference.
	Void fraction at pumps compared to a steady–state void fraction of 2% to determine whether the pumps would fail.	Void fraction at pumps compared to a steady–state void fraction of 2% to determine whether the pumps would fail.	No difference.
Debris Penetration	Perform plant–specific fiber penetration testing of the bounding scenario(s) with a prototype strainer module.	Develop a fiber penetration correlation as a function of strainer flow rate and fiber accumulation based on a series of penetration tests.	New engineering model Documented in Volume 3.
	100% penetration of transportable particulate and chemical precipitates.	100% penetration of transportable particulate and chemical precipitates.	No difference.
Ex–Vessel Downstream Effects	Evaluate ex–vessel wear and clogging based on the methodology in WCAP–16406–P	Evaluate ex–vessel wear and clogging based on the methodology in WCAP–16406–P	No difference.

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Topical Area	NRC–Approved Deterministic Methods	STPNOC Pilot Project Methods for 2012 Quantification	Comments
In–Vessel Downstream Effects	Compare fiber quantity on core to bounding 15 g/FA limit based on WCAP–16793–NP.	Use RELAP5 simulations to show that cold leg small break LOCAs and all hot leg LOCAs would not go to core damage with full blockage at the base of the core. For medium and large cold leg breaks, use WCAP–16793–P for fiber limit on the core.	New approach documented in Volume 3 CASA Grande Analysis.
	Evaluate reduced heat transfer due to deposition on fuel rods using LOCADM software.	Evaluate reduced heat transfer due to deposition on fuel rods using LOCADM software.	No difference.
Boron Precipitation	No currently accepted methodology.	Evaluate fiber accumulation on the core for cold leg breaks during cold leg injection. Assume that 7.5 g/FA of fiber is sufficient to form a debris bed that would prevent natural mixing between the core and lower plenum. Assume failure due to boron precipitation if this quantity arrives prior to hot leg switchover.	New approach documented in Volume 3 CASA Grande Analysis.

C Defense-in-Depth and Safety Margin

C.1 Introduction

DID[1] for STP[2] Units 1 and 2 is based on the plant design, operating procedures, and administrative controls. The proposed change to the Updated Final Safety Analysis Report (UFSAR) reconstitutes the current licensing basis for acceptable containment emergency sump strainer design and performance in support of the recirculation modes for ECCS[3] and CSS[4] following postulated LOCAs[5], using a risk–informed approach to address GSI–191[6] [1].

GSI–191 addresses concerns that debris generated during a LOCA could clog the RCB[7] sump strainers in PWRs[8] and result in NPSHA[9] falling below NPSHR[10] for the ECCS pumps and CSS pumps. The NRC issued Bulletin 2003-01 [2] to address the potential for sump blockage. The NRC later issued Generic Letter (GL) 2004-02 [3], requesting that licensees address the issues raised by GSI–191 and focused on demonstrating compliance with 10 CFR 50.46.

In responses to Bulletin 2003-01 and GL 2004-02, STP described modifications to plant hardware (most notably new advanced design sump strainers), and operating procedures and administrative controls that were implemented to address GSI–191 concerns [4, 5, 6, 7, 8]. STP operating procedures have actions which prevent and mitigate strainer blockage and in–vessel core blockage based on indications available to operators such as instrumentation to monitor core water levels and temperatures. Actions include initiation of combined cold leg and hot leg injection, which provides an alternate flow path that bypasses core inlet blockage, and delaying the initiation of recirculation mode by delaying depletion of the RWST[11] including actions to refill the RWST. STP surveillance procedures implement Technical Specification requirements for cleanliness in accessible areas of the RCB to verify no loose debris (rags, trash, clothing, etc.) is present which could be transported to the RCB sump and cause restriction of the pump suctions during LOCA conditions, and for visual inspections of the RCB sumps to verify suction inlets are not restricted by debris and that the sump components show no evidence of structural distress or abnormal corrosion.

The new strainer design satisfies the current licensing basis for debris loading as described in the STP UFSAR. The new strainer design satisfies the current licensing basis for compliance with 10 CFR 50.46 and the regulatory requirements contained in GL 2004-02 including General Design Criteria (GDC) 35, 38, and 41, for the current licensing basis assumptions for analyzing the effects of post-accident debris blockage. This evaluation is documented as part of a previously approved license amendment request [9, 10, 11].

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[1] Defense–in–Depth
[2] South Texas Project Electric Generating Station
[3] Emergency Core Cooling System
[4] Containment Spray System
[5] Loss of Coolant Accidents
[6] Generic Safety Issue 191
[7] Reactor Containment Building
[8] Pressurized Water Reactors
[9] Net Positive Suction Head Available
[10] Net Positive Suction Head Required
[11] Refueling Water Storage Tank

The current licensing basis for the new sump strainers installed to address GSI–191 consists of the current assumptions, initial conditions and conclusions of GL 2004-02 related evaluations, including the current evaluations of design basis accident debris generation and transport, sump strainer performance, impact of chemical effects and downstream effects of debris. Substantial plant-specific testing that supports assumptions and corresponding conclusions contained in the GL 2004-02 evaluations for STP has been performed. This information supporting the previous deterministic methodology for demonstrating compliance is documented in supplemental information provided in response to GL 2004-02 [12]. However, the NRC has not fully accepted the evaluations to demonstrate complete resolution of GSI–191 for the as-built and as-operated plant design using the deterministic methodology. The risk-informed analyses associated with the proposed exemptions and license amendment along with the design, procedure and administrative controls already incorporated demonstrate that the RCB emergency sump strainers will perform their required functions.

To resolve GSI–191, STP has developed a risk-informed approach consistent with the guidance in RG1.174[12] to reconstitute the licensing basis for the strainer design for compliance with the regulatory requirements. The STP risk-informed approach follows RG1.174 [13], verifying DID and Safety Margin are maintained through design modifications, ongoing design modification controls, maintenance procedures including the ISI[13] program. The approach is comprehensive in nature, analyzing a full spectrum of LOCAs including DEGB[14] for all piping sizes up to and including the largest pipe in the RCS[15]. 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 DID is maintained. The risk-informed method meets the key principles of RG1.174 and demonstrates that the residual risk associated with GSI–191 concerns is far less than the threshold for Region III, "Very Small Changes," as defined by RG1.174 and therefore meets the Commissions Safety Goal.

The proposed change to the licensing basis is to use the methodology of a RG1.174 risk-informed approach to evaluate containment emergency sump strainer performance in support of ECCS and CSS recirculation modes following postulated LOCAs.

The proposed change to the licensing basis is consistent with maintaining DID in that the following aspects of the facility design and operation are maintained:

- Functional requirements and design configurations of systems
- Existing plant barriers to the release of fission products
- Design provisions for redundancy, diversity and independence
- Plant response to transients and other initiating events
- Preventative and mitigative capability of plant design features.

Based on the results of the risk-informed method and the hardware, operating procedures and administrative controls already implemented to address GSI–191 concerns, STP has

[12]Regulatory Guide 1.174

[13]ASME Section XI Inservice Inspection

[14]Double-Ended Guillotine Break

[15]Reactor Coolant System

high confidence that plant systems and operators would respond as required to mitigate postulated LOCAs. This confidence is bolstered by the DID features for STP described below.

C.2 Effectiveness of Defense-In-Depth Actions

The effectiveness of the DID actions is shown to be acceptable when considering the following:

- STP EOPs[16] are based on the approved industry standard Emergency Response Guidelines (ERGs). These symptom-based EOPs have generic or site–specific analyses that support them.
- STP Severe Accident Mitigation Guidelines (SAMGs) are based on approved industry standard guidance.
- The procedures are trained upon and evaluated as part of the classroom training.
- The DID actions are trained upon using the simulator to demonstrate effectiveness.
- The procedures that make the framework for the DID actions are evaluated during the STP station review and approval process.

C.3 Evaluations

STP DID measures that are associated with the concerns of GSI–191 are evaluated by applying regulatory guidance and industry guidance.

C.3.1 Guidance in RG1.174

STPNOC[17] proposes a licensing basis change to use a risk informed approach to address the concerns of GSI–191 with respect to maintaining long term cooling post–LOCA on the basis that the change meets the principles and acceptance guidelines of RG1.174. The DID elements given in Section 2.1.1 of RG1.174 discussed below have been evaluated to show that the proposed change is consistent with DID for STP Units 1 and 2. DID for STP is based on the hardware, operating procedures, and administrative controls and design modifications that have been implemented to address the concerns of GSI–191 and GL 2004-02. The proposed licensing basis change does not propose any additional DID measures.

A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.

STP Units 1 and 2 each have three trains of ECCS equipment for the prevention of core damage. Each train includes a SI[18] accumulator, HHSI[19] pump, LHSI[20] pump

[16] Emergency Operating Procedures

[17] The STP Nuclear Operating Company

[18] Safety Injection System

[19] High Head Safety Injection

[20] Low Head Safety Injection

that has its discharge routed through the RHR[21] heat exchanger for cooling by CCW[22]. There are three independent trains of equipment for containment heat removal to prevent containment failure. The heat removal equipment for each train includes a CSS pump and two RCFC[23] units per train that are cooled by safety–related CCW. Consequence mitigation is achieved using active equipment of the ECCS and CSS and by maintaining the containment building as an effective barrier to radioactive release.

The proposed change does not involve any equipment or design changes beyond the modifications that have been made in response to the concerns raised in GSI-191 nor does it involve any changes to the EOPs beyond the changes in place to address the concerns raised in GSI–191. As discussed further below, the proposed change does not significantly affect the containment integrity or the capability of the independent and safety–related RCFCs to remove post–LOCA decay heat from containment. There is no change to the strategies for the prevention of core damage, for prevention of containment failure, or for consequence mitigation. Thus the existing balance among these is preserved.

Over–reliance on programmatic activities as compensatory measures associated with the change in the licensing basis is avoided.

Programmatic activities associated with the proposed change include the ISI program, plant personnel training, RCS leak detection program, and containment cleanliness inspection activities.

The ISI program requires non-destructive examinations of the RCS components and piping. The inservice testing (IST) program requires testing of active components such as pumps and valves in the RCS, SI, and CSS systems. The proposed change does not rely heavily on programmatic activities as compensatory measures nor propose any new programmatic activities that could be heavily relied upon. The risk-informed approach does consider pipe break frequencies. STP has previously implemented a risk-informed ISI program that was approved by the NRC [14]. The ISI program is an effective element of DID that performs an important role in the prevention of pipe breaks. It is important to note that the risk-informed GSI-191 program and the risk-informed ISI program are complementary in that the risk insights from the stations plant specific PRA[24] are used in conjunction with deterministic information to improve the safety and effectiveness of the ISI program.

The leak detection program at STP is capable of early identification of RCS leakage to provide time for appropriate operator action before a flaw causing a leak would propagate to a break. This program is an important contributor to preventive DID.

Containment cleanliness inspection activities are performed prior to reactor startup following outages, as required by the Technical Specifications. The risk–

[21] Residual Heat Removal System

[22] Component Cooling Water System

[23] The Reactor Containment Fan Coolers

[24] Probabilistic Risk Assessment

informed approach uses an input for the assumed amount of latent debris inside containment after the cleanup activity is complete. However, this is the same amount as that given in the NEI 04-07 guidance for a deterministic approach [15, 16]. Thus, there is no over–reliance on STP programmatic activities to quantify or manage latent debris as compensatory measures for the risk–informed approach.

System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (for example, no risk outliers).

STP has three independent trains of ECCS equipment for the prevention of core damage. Each train includes a SI accumulator, HHSI pump, LHSI pump that has the discharge routed through the RHR heat exchanger for cooling by CCW. There are three independent trains of equipment for containment heat removal to prevent containment failure. The heat removal equipment for each train includes a CSS pump and two RCFC units that are cooled by CCW. Each train has an independent containment emergency sump with strainer to provide suction flow during the recirculation mode to the respective train's pumps (HHSI, LHSI, and CSS).

The proposed change does not require any design change to these systems. Thus system redundancy, independence, and diversity are preserved. The proposed licensing basis change also does not call for any changes to the system operating procedures. These systems have been fully analyzed relative to their contribution to nuclear safety through STPs plant–specific PRA. The STP PRA includes the risk contributions for the full spectrum of LOCA events and meets industry PRA standards for risk–informed applications. The treatment of uncertainties in the risk-informed model ensures results are obtained for realistic assessments, as discussed in detail in the supporting engineering analysis provided in Enclosure 4–3 (Volume 3). The uncertainties using the risk-informed approach methodology have been examined in the PRA and there are no risk outliers.

Defenses against potential common–cause failures are preserved, and the potential for the introduction of new common–cause failure mechanisms is assessed.

The proposed change does not change any defenses against common–cause failures. A potential common cause failure would be all of the sump strainers becoming clogged so that there would not be adequate flow to any of the SI and CSS pumps. The defenses that apply to potential strainer clogging (for example change in flow rate, conserving RWST inventory, use of alternate injection sources, and stopping/starting of pumps) are not changed by the use of the risk-informed methodology since there are no design changes to the equipment or changes to the EOPs.

The potential for new common–cause failure mechanisms has been assessed for the GSI–191 issue. The primary failure mechanisms of concern are recirculation sump strainer clogging and core clogging (that is, in-vessel effect). A new aspect of clogging is the consideration of chemical effects in addition to the fibrous particulate debris. However the defenses against chemically–induced clogging in either the ECCS sump strainers or in–vessel fuel blockage (which are discussed more in the next section) are effective, reasonable and acceptable operational measures to mitigate or ameliorate adverse strainer and core cooling performance. Additionally, these defenses do not change due to the proposed licensing basis change to use the RG1.174 risk–informed approach. Since the risk-informed approach does not involve any design changes to the equipment or changes to the operating procedures beyond those already taken in response to the concerns raised in GSI–191, it does not introduce any new common–cause failures or reduce the current plant defenses against common–cause failures

Independence of barriers is not degraded.

The three barriers to a radioactive release are the fuel cladding, the RCS piping and components, and the RCB. For the evaluation of a LOCA, the RCS barrier is postulated to be breached. The proposed licensing basis change does not involve any change to the design and analysis requirements for the fuel. Thus the fuel barrier independence is not degraded. Consequently, the risk–informed GSI–191 analysis approach focuses primarily on addressing the integrity of the fuel cladding by assuring the ECCS cooling function is maintained. STPs risk–informed evaluation includes both the ECCS cooling function and the containment function.

In the recirculation mode of accident mitigation, the post–LOCA fluid that collects on the containment floor is pumped by the HHSI, LHSI, and CSS pumps that are located in the Fuel Handling Building (FHB). Thus the recirculated fluid goes from the RCB to the FHB and back to the RCB. The barrier to release from the FHB is the SI and CSS piping and components in the recirculation flow path. The FHB HVAC system has filters to handle gaseous leakage that would come from any recirculating sump water leakage in the FHB. The proposed licensing basis change does not involve any change to the design and operating requirements for this equipment. Thus there is no change to the containment bypass path. The containment barrier is maintained.

The RCB is fully analyzed for not only design basis considerations but also from a Level 2 PRA perspective. Detailed analyses for severe accident phenomena, including LOCAs, have been evaluated for impact to containment building integrity; and these events do not challenge the overall capability of the containment to remain intact. Also, it should be noted that additional DID capability is available through the use of the RCFCs. The RCFCs have enough cooling capability to remove decay heat from the containment pool through containment atmosphere cooling during the ECCS recirculation phase thereby further reducing containment integrity challenges.

The proposed change does not involve any design change to these barriers (fuel, piping, building, HVAC filters). Thus the independence of the barriers is maintained and not degraded.

Defenses against human errors are preserved.

The proposed change does not involve any design change to the current equipment or for any change to operating procedures. Operator actions during the initial accident mitigation stage are focused on monitoring of the automatic mitigation actions including automatic ECCS and CSS responses to the event. Operators will secure one CSS train if all three trains are running at the initiation of the event to conserve RWST volume. Prior to depletion of the RWST, there is an automatic switchover of the ECCS and CSS pumps from taking suction from the RWST to taking suction from the containment emergency sumps. Operator action is needed at the end of the switchover sequence to close the RWST outlet valves. If RCS pressure is greater than the pumps shutoff head pressure, the operators are required to secure the ECCS pumps to prevent pump damage. After 5.5 hours, the switchover from cold leg injection to combined cold leg and hot leg injection is a manual action performed by the operator. The use of the methodology for the risk-informed approach does not change any of the EOPs that would be used or impose any additional operator actions or complexity. Thus the defenses that are already in place with respect to human errors are not impacted by the proposed licensing basis change.

The intent of the plants design criteria is maintained.

The proposed change does not involve any change to the design or design requirements of the current plant equipment associated with GSI–191. Based on the results of the proposed change showing that the risk-informed approach meets RG1.174 acceptance criteria, the proposed change reconstitutes the licensing basis for acceptable containment emergency sump strainer design and performance in support of ECCS and CSS operation in recirculation mode following postulated LOCAs. Therefore the intent of the plants design criteria is maintained.

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. The proposed change to the licensing basis continues to meet the intent of the GDC that apply to functions addressed by GSI–191. This conclusion is based on the results of the risk-informed approach that demonstrate that the calculated risk associated with GSI–191 concerns for STP Units 1 and 2 is very small and far less than the Region III acceptance guidelines defined by RG1.174. The functionality of the ECCS and CSS during design basis accidents is confirmed.

The performance evaluations for accidents requiring ECCS operation described in Chapters 6 and 15 are based on the STP Units 1 and 2 Appendix K Large–Break Loss–of–Coolant Accident (LBLOCA[25]) analysis. These evaluations 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 10 CFR 50.46, thus assuring that the core will remain in place and substantially intact with its essential heat transfer geometry preserved. The proposed change does not involve a change to the ECCS acceptance criteria specified in 10 CFR 50.46.

[25] Large Break Loss of Coolant Accident

C.4 General Design Criteria

GDC that apply to GSI–191 concerns are evaluated as follows.

C.4.1 Criterion 16–Containment Design

Containment and associated systems shall be provided to establish an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

For STP, the containment isolation system will limit leakage to small percentages by providing an essentially leak-tight barrier against radioactivity which may be released to the containment atmosphere in the unlikely event of an accident. Additional systems provided to prevent the uncontrolled release of radioactivity from the containment to the environment are the ECCS and CHRS[26] which includes the CSS and RCFCs. These systems mitigate the potential consequences of a LOCA or main steam line break. The containment and these associated engineered safety systems are designed to operate under all internal and external environmental conditions that may be postulated to occur during the life of the plant, including both short–and long–term effects following a LOCA.

C.4.2 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.

For STP, the ECCS is provided to cope with any LOCA up to and including the plant design basis DEGB of the RCS. 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 for any postulated LOCA 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. Additionally, the station's plant–specific PRA fully evaluates the risk of LOCAs and extends the analysis to beyond design basis events. Thus, additional DID considerations have been evaluated through the station's PRA to account for events such as multiple equipment failures, human errors, and external events including seismic events.

C.4.3 Criterion 38–Containment Heat Removal

A system to remove heat from the 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.

For STP, the CHRS consists of the CSS and the RCFC subsystem, and is assisted by the RHR heat exchangers acting in conjunction with the SI system to remove heat from

[26]Containment Heat Removal System

containment. The CHRS is designed to accomplish the following functions in the unlikely event of a LOCA:

- Rapidly condense the steam within containment in order to prevent over–pressurization during blowdown of the RCS; and
- Provide long-term continuous heat removal from containment.

Initially, the CSS and the HHSI and LHSI pumps take suction from the RWST. During the recirculation phase, the CSS and the HHSI and LHSI pumps take suction from the containment emergency sumps. The RCFC subsystem is also available as part of the CHRS to remove containment atmospheric heat and in so doing reduce containment temperature and pressure.

C.4.4 Criterion 41–Containment Atmosphere Cleanup

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in 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.

For STP, the CSS is provided to reduce the concentration and quantity of fission products in containment atmosphere following a LOCA. 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 fluid. Each unit is equipped with three 50 percent spray trains from a design basis perspective taking suction from the containment sump.

C.4.5 Criterion 50–Containment Design Bases

The reactor containment structure, including access openings, penetrations and the CHRS, shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. This margin shall reflect consideration of: (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators (SGs) and energy from metal/water and other chemical reactions that may result from degraded emergency core-cooling functioning; (2) the limited experience and experimental data available for defining accident phenomena and containment responses; and (3) the conservatism of the calculation model and input parameters.

For STP, the containment design basis is relevant to the risk-informed approach for that small fraction of events (typically involving beyond design basis failures) for which there is core damage that rely on containment for DID. The maximum temperature and pressure reached in the RCB during the worst-case design basis accident are well below the design temperature and pressure of this structure and there is substantial margin in the containment design to accommodate beyond design basis events. The proposed licensing basis change for the RG1.174 risk approach for GSI–191 does not change any of the design and testing requirements for the containment. The section below titled "Barriers for Release of Radioactivity" provides additional discussion.

C.5 NEI Guidance for Defense–in–Depth Measures in Support of GSI-191 Resolution

For the purposes of GSI–191 resolution, the primary regulatory objective is specified in 10 CFR 50.46(b)(5) as long-term cooling. A method for ensuring adequate DID is to maintain the capability for operators to detect and mitigate inadequate flow through recirculation strainers and inadequate flow through the reactor core due to the potential impacts of debris blockage. The following evaluation of the STP DID measures that support the STP application for a risk-informed approach to resolving GSI–191 is based on Nuclear Energy Institute (NEI) guidance [17] which includes additional justification for the measures discussed.

The STP Units 1 and 2 EOP framework has guidance for monitoring for the loss of emergency sump recirculation capabilities and actions to be taken if this condition occurs. These actions are as described in responses to Bulletin 2003–01 and GL 2004–02 [4, 5, 6, 7, 8], and remain in effect.

In summary, these actions include (1) reducing flow through the strainer(s) by stopping pumps, (2) monitoring for for proper pump operation, core exit thermocouples, and reactor water level indication, (3) refilling the RWST for injection flow, (4) using injection flow from alternate sources, and (5) transferring to combined hot leg/cold leg injection flow paths. STP EOPs that implement these actions include:

0POP05-EO-EO00 "Reactor Trip or Safety Injection"
0POP05-EO-EO10 "Loss of Reactor or Secondary Coolant"
0POP05-EO-EC11 "Loss of Emergency Coolant Recirculation"
0POP05-EO-ES13 "Transfer to Cold Leg Recirculation"
0POP05-EO-ES14 "Transfer to Hot Leg Recirculation"
0POP05-EO-FO02 "Core Cooling Critical Safety Function Status Tree"
0POP05-EO-FRC1 "Response to Inadequate Core Cooling"
0POP05-EO-FRC2 "Response to Degraded Core Cooling"
0POP05-EO-FRZ3 "Response to High Containment Radiation Level"

C.5.1 Strainer Blockage

Inadequate recirculation strainer flow refers to the condition where the head loss across the ECCS sump strainers develops to the condition where the NPSHA is less than the NPSHR of the HHSI, LHSI and CSS pumps taking suction on the strainers. This condition could result from the formation of chemical precipitates in the containment sump pool and their deposition on a debris bed on the sump strainers. The onset of inadequate sump strainer flow would not be expected to present itself as a problem until several hours into an event (following cooldown and the potential formation of chemical precipitates). This has been shown previously through generic and highly conservative (i.e., bounding) testing to significantly increase the strainer head loss above that which develops solely as a result of non-chemical debris. For STP, more recent testing for the realistic post–LOCA conditions has shown that impacts due to chemical effects are not deleterious to pump performance. In fact, the plant specific, prototypical (i.e., realistic) tests and experiments have shown that there has been an extremely small amount of precipitates formed in STP post–LOCA sump environments over a thirty day time period. However, the STP DID does include sump strainer contingencies due to debris-induced strainer clogging, as discussed above.

C.5.2 Prevention of Strainer Blockage

The primary means to delay or prevent this condition is to reduce the flow through the sump strainers by the following.

- STP has a continuous action step in the EOPs to remove the third CSS pump from service after conditions have been verified suitable. Upon the initiation of an event that would cause a CSS actuation, the STP EOPs secure one CSS pump if three CSS pumps are in service. The operator performs this at the onset of the event to conserve RWST volume. This will also reduce the flow demands on the associated emergency sump during emergency recirculation phase.
- The following additional pumps are removed as conditions allow: CSS pumps (with TSC[27] concurrence when containment pressure is less than 6.5 psig) and LHSI pumps (if RCS pressure is greater than 415 psig).
- For small to medium LOCAs, guidance to delay depletion of the RWST exists in procedure 0POP05-EO-ES12, "Post LOCA Cooldown and Depressurization". This procedure provides actions to cooldown and to depressurize the RCS to reduce the break flow, thereby reducing the injection flow necessary to maintain RCS subcooling and inventory. The operating HHSI pumps are sequentially stopped to reduce injection flow, based on pre-established criteria that maintain core cooling, resulting in less outflow from the RWST. If the break is not large enough to drop RCS pressure below 415 psig, then the three LHSI pumps would not be injecting into the RCS but would be on pump recirculation flow back to the RWST. This would greatly reduce the depletion of RWST volume since these are high volume pumps. The procedure would secure these pumps as long as RCS pressure is maintained above 415 psig.
- For smaller LOCAs, it is possible to cooldown and depressurize the RCS to cold shutdown conditions before the RWST is drained to the switchover level. Therefore cold leg recirculation is not required to be established for these breaks; and sump blockage is not an issue.

[27] Technical Support Center

Additional considerations for prevention of sump strainer blockage:

- For the Technical Specification Surveillance Requirement (TS SR) 4.5.2.c, STP has implemented procedures 0PSP03–XC–0002 "Initial Containment Inspection To Establish Integrity" and 0PSP03-XC-0002A "Partial Containment Inspection (Containment Integrity Established)," to visually inspect all accessible areas of the containment when Containment Integrity is established and maintained. The inspections ensure no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump. Walk–downs are performed by station management and Operations personnel and a final acceptance walk–down is performed by Operations to assure the containment building is free of loose debris prior to entering Mode 4 (Hot Shutdown). For subsequent entries, inspections of the travel path and work locations are required to assure the areas free of loose debris.
- For TS SR 4.5.2.d, STP has implemented procedures to verify by visual inspection that the suction inlets are not restricted by debris and that the sump components show no evidence of structural distress or abnormal corrosion. This TS SR is required every refueling outage.
- The RWST level is normally maintained at a nominal level from 490,000 to 500,000 gallons to ensure standby capacity is maintained above the Technical Specification minimum required volume of 458,000 gallons and the low level alarm setting of 473,000 gallons.
- Training has been provided to engineering personnel to raise their awareness of the more aggressive containment cleanliness requirements, the potential for sump blockage, and actions being taken to address sump blockage concerns.

C.5.3 Detection of Strainer Blockage

In a LOCA scenario, debris would be generated and could be transported to the emergency sump strainers. Following initiation of flow through the sump strainers in the recirculation mode, fiber and particulate debris could accumulate on the strainers resulting in increased head loss across the strainers.

If an excessive head loss condition were to develop, it would result in a condition of inadequate recirculation flow from the strainers to the pumps. This, in turn, could result in a condition where insufficient cooling is provided to cool the reactor core or insufficient flow is available for containment pressure control.

If a condition of inadequate recirculation strainer flow were to develop, it is important for the plant operators to be able to detect this condition in a timely manner. The primary methods for detection of this condition are:

Pump distress indications

STP has flow indication in the control room for all SI pumps (LHSI and HHSI) and for all CSS pumps. Instrumentation is available to provide the operator with indications of potential sump blockage. Indications of pump cavitation or pump suction pressure below NPSHR such as erratic flow or low discharge pressure can indicate a degradation in suction supply that could be caused by containment recirculation sump strainer clogging. Indications are provided for SI and CSS pump flows and SI pump discharge pressures that can be monitored for signs of degraded pump conditions, such as could be caused by containment sump clogging following establishment of recirculation flow.

Core cooling degrading

STP has core exit thermocouple (CET) indication and reactor vessel water level (RVWL) indication in the Control Room both on computer screens for the Integrated Computer System (ICS) and Qualified Display Parameter System (QDPS) to allow monitoring for any potential reduction in core cooling flow due to sump blockage. This indication is also displayed on the computer systems as part of the critical safety system status trees indicators. The Reactor Operators and the Shift Technical Advisor monitor these status tree indications. The status tree indicators provide change based on status tree logic to further enhance operator recognition of a distress condition developing.

C.5.4 Mitigation of Strainer Blockage

Multiple methods are available to mitigate an inadequate recirculation flow condition caused by the accumulation of debris on the sump strainer, including:

Reduction in flow demand on the emergency sump strainer

EOPs contain steps to reduce flow through the system up to and including stopping all pumps taking suction from the affected emergency sump strainer. In strainer head loss testing it has been observed that stopping all flow through a debris laden strainer has resulted in separation of portions of the debris bed from the strainer. The primary driver for this separation is gravity since the force that held the debris bed in place was the differential force (pressure) developed as a result of flow and head loss through the bed. Another contributor to the collapse of the debris bed is the reverse pressure wave that develops as a result of stopping the pumps and the consequential closure of discharge check valves. STP procedure 0POP05-EO-EC11 “Loss of Emergency Coolant Recirculation” minimizes the pumps required depending on plant conditions and directs shutting down all pumps as applicable.

The following actions would address degraded ECCS recirculation flow that may be caused by containment recirculation sump strainer clogging:

- stopping CSS pumps not needed for containment pressure control with adequate RCFCs used for containment heat removal, to conserve RWST.
- securing SI pumps to the RWST.
- aligning CCP to the VCT[28] and injecting into the RCS.

Alternation of Recirculation Trains

STP has a design configuration that allows independent operation of recirculation strainer trains. This enables the capability to operate the recirculation system in a sacrificial strainer arrangement. With two recirculation strainers put in service and

[28] Volume Control Tank

the third strainer in standby, the vast majority of the debris will collect on the first two strainers. This provides for a relatively clean, low head loss strainer that can be placed in service later if determined necessary due to blockage on the first two strainers.

The STP design has three independent trains each consisting of one HHSI pump, one LHSI pump, one CSS pump, one RHR heat exchanger, and one emergency sump strainer. The design does have the capability for operating only two trains at a time which would allow the third train to have relatively debris free strainer operation if called upon to be in service later in the accident mitigation scenario.

Emergency Sump Strainer Backwash

The STP plant design configuration can provide a gravity drain of water from the normal injection supply from the RWST backwards through the emergency sump strainer. This backflow from the RWST to the sump could occur only if the containment pressure is sufficiently low (below the RWST gravity head). The bottom of the RWST is at elevation (+) 10 feet and the sump strainers are at elevation (-) 11 feet. Gravity backwash removes accumulated debris blockage at the sump strainer. The TSC, using guidance provided in the SAMGs, would be expected to advise performance of this action, considering the plant conditions and available indications.

RWST Refill and Realignment for Injection Flow

The EOPs for STP contain steps to initiate makeup to the RWST following transfer to the recirculation mode of core cooling. In the event of strainer blockage, realignment to the direct injection flow path from the RWST would provide necessary cooling for an extended period of time. If aligned to the RWST, the operators would establish the minimum flow required for core decay heat removal depending on sub-cooling conditions. STP transfers from cold leg recirculation to hot leg recirculation at 5.5 hours after event initiation. Per the hot leg recirculation procedure requirements, Operators will transfer two trains to hot leg recirculation while maintaining one train in cold leg recirculation.

STP has the design capability to refill the RWST and to realign the SI pumps to use this injection water source in case the recirculation flow path was blocked by clogged sump strainers. STP has guidance in the SAMGs to inject more than one RWST volume, and the additional RWST volume to be added would be coordinated with the TSC.

Injection Flow from Alternate Sources

STP has the capability to use other sources of water to provide for core cooling. The STP plant design configuration can provide CVCS[29] Recycle holdup tanks as a water source to make up to the VCT allowing use of a charging pump to provide injection flow. STP has the capability to align the BAT[30] using the boric acid transfer pump to inject into the RCS.

[29] Chemical Volume and Control System
[30] Boric Acid Tank

C.5.5 Inadequate Reactor Core Flow

Inadequate reactor core flow refers to the condition where the normal core cooling flow path has become impeded (blocked) and is not allowing sufficient cooling water flow to reach the core. This condition could result from the formation of a flow limiting or blocking debris bed at the entrance to the core region from the lower plenum of the reactor vessel. The fiber bed that is developed is the result of fibers bypassing (flowing through) the emergency sump strainers and becoming trapped in the debris limiting openings at or near the bottom of the fuel assemblies. Tests have shown that the limiting conditions for fuel blockage require the combination of fibrous debris, particulates and chemical precipitates. Significantly higher fiber debris loads can be accommodated without flow reductions with the absence of or significant reduction in chemical precipitates or in a significantly increased particulate contribution. Similarly, with a significant reduction in fibrous debris, particulates and chemical precipitates can be accommodated without problems.

In a LOCA scenario, debris would be generated and deposited inside containment. Following initiation of flow through the emergency sump strainers during the recirculation mode, fibrous and particulate debris would be transported to the strainers. Some of the debris transported would pass through the strainers and enter the suction of the ECCS pumps (LHSI and HHSI) and be injected into the reactor. The ECCS recirculation flow will be directed initially to the cold legs of the RCS and flow through the reactor vessel to the lower plenum region and then up into the fuel assemblies. Depending on the break size and location, a portion of this flow can bypass the reactor core and flow out of the break location. Any fibrous debris that goes through the strainers and makes it to the reactor vessel will tend to collect on the bottom of the fuel to form a debris bed that would also capture particulate debris. As temperature in the containment pool reduces below the value associated with the development of chemical precipitates, these precipitates will interact with the debris bed formed at the fuel assemblies resulting in a further increase in head loss.

The STP plant design has combined hot leg and cold leg injection once the RWST is depleted and the pumps have been aligned during the recirculation mode. Initially the pumps are aligned for cold leg injection. At 5.5 hours after the initiating event, the switchover to combined hot/cold leg injection is made. Since core cooling flow in this configuration is directed from below and above the core, this design is less susceptible to the development of blockage conditions that would result in an inadequate reactor core flow condition for flow in only one direction.

C.5.6 Prevention of Inadequate Reactor Core Flow

Controlling (Reducing) Core Flow

The set of actions identified above for reducing or controlling flow through the emergency sump strainers during the recirculation mode can have a similar positive impact on reducing the potential for fuel blockage. Controlling flow to the reactor vessel to maintain fuel coverage and match decay heat has benefits through reduced head loss and delayed onset of any chemical precipitates.

Transfer to Combined Hot Leg / Cold Leg Injection Flow Paths

This step normally is performed at 5.5 hours following the design basis pipe break event. There are some factors to consider that establish the minimum time for normal transition to this mode of core cooling. These factors are the decay heat load versus the hot leg injection capability and the potential for steam binding of flow out of the core for certain hot leg break scenarios. For STP which has multiple hot leg injection flow paths, the safety injection flow rate is significantly greater than the core boil off rate. This ensures adequate flow to the core. Because debris beds observed in testing appear to be very unstable, transferring to hot leg injection also has the potential to disturb any debris collected on the bottom of the fuel. The STP EOPs call for switchover of two trains to hot leg injection while maintaining cold leg injection with the third train.

C.5.7 Detection of Inadequate Reactor Core Flow

Multiple methods exist for detection of a core blockage condition as manifested by an inadequate reactor coolant system (RCS) inventory or RCS and core heat removal condition. The primary methods include core exit thermocouples (CET) temperature indication and the RVWL monitoring system.

Monitoring is initiated early in the event in the EOPs through Critical Safety Function Status Trees (CSFST). The CSFSTs are performed continuously after completion of diagnosis of the event directed by the EOPs. The QDPS and ICS screens display the Critical Safety Function status at the bottom of the screen to allow the operating crew easy monitoring capability. In addition, the Shift Technical Advisor is responsible to provide an independent monitoring of the CSFST during an event and update the operating crew of any changes.

Emergency Response Personnel in the TSC or EOF[31] will also maintain oversight of plant status through review of information available in the TSC and EOF. An additional method for detection of a core blockage condition includes monitoring of containment radiation levels by the TSC or EOF staff and/or if an alarm setpoint is reached resulting in an alarm in the control room. Subcooling and containment radiation are monitored during an event by the operating crew and/or the Emergency Response Personnel and will be used to help determine if the event is escalating in severity and if one of the fission product barriers may be impacted.

Increasing core exit thermocouple (CET) temperature indication

CETs are monitored during EOP usage as well as for status tree functional restoration entries and the safety parameter display system (SPDS). As part of operator training, the operating crew must demonstrate the ability to detect increases in CET temperature indication and transition to the appropriate EOP for dealing with this condition.

Core exit temperature behavior is the primary indicator of adequate core cooling. If cold leg recirculation has been established with flow maintained into the RCS, core exit temperature should be stable or slowly lowering during the recovery. Increasing core exit temperatures while injection flow is maintained, regardless of reactor vessel water level behavior, is an unexpected condition that should be evaluated well before any CSFST temperature limits are approached. In this regard, when a core cooling concern is identified, STP's functional restoration procedure would attempt to establish injection flow. If unable to establish SI flow, then centrifugal charging pump (CCP) flow is established to allow maximum injection into the RCS utilizing the two CCPs.

[31] Emergency Operations Facility

Decreasing reactor water level indication

Reactor vessel water level is monitored throughout the EOPs. Through continuing training, operators demonstrate the ability to monitor and understand the implications of a decreasing reactor vessel water level and appropriately transition within the EOP framework to mitigate this condition.

STP uses a RVWL indicating system design consisting of heated and unheated junction thermocouple pairs which would indicate a lowering water level with lower core region flow blockage. The STP design uses eight pairs of heated and unheated junction thermocouples enclosed in a vertical shroud from the top of the core to the top of the reactor vessel head which provides discrete level indication (void/no void) at the elevation of each pair. This RVWL design does not rely on differential pressure sensors for indication.

Increasing containment or auxiliary building radiation levels

Increasing radiation levels would be indicated by alarms in the control room with specific procedural steps in both alarm response procedures and EOPs for addressing the condition. Radiation monitor indication in the auxiliary building may be indication of a LOCA outside containment or provide initial entry conditions into an Emergency plan (E-plan) due to increasing radiation levels. Abnormal containment radiation could require an escalation of the E-plan due to this being indication of fission product barrier degradation which is monitored by the control room. Abnormal containment radiation level is also one of the symptoms used to identify a LOCA inside containment in the EOPs. Due to the sensitivity of the monitors and the low alarm set points, identification of degrading core conditions is expected to be indicated well before a significant release of radioactivity to containment occurs.

C.5.8 Mitigation of Inadequate Reactor Core Flow

Multiple methods are available to mitigate an identified inadequate reactor core flow condition.

Upon identification of an inadequate RCS inventory or an inadequate core heat removal condition, the EOPs direct the operators to take actions to restore cooling flow to the RCS including:

- Increase SI flow to refill the reactor vessel by depressurizing the RCS.
- Depressurize the RCS to inject the accumulators.
- Attempt to start any available SI pumps not running.
- Secure SI pumps to prevent pump damage, as necessary.
- As necessary, secure SI flow to prevent pump damage.
- Refill the RWST.
- Provide injection flow from the VCT using the charging pumps on a loss of emergency recirculation.

- Provide injection flow using the positive displacement pump, if needed.
- Provide core cooling by steaming through the steam generators.
- Transfer to RHR if determined acceptable by the TSC.
- Transfer to hot leg recirculation.

The operators will also inform the TSC of indications of inadequate reactor core cooling. The TSC will evaluate the condition and recommend the following actions, as necessary, to the operators to restore core heat removal using SAMGs or mitigation procedure guidance.

- Throttle RCS injection flow rate to ensure long term minimum decay heat removal is met.
- Use the hot leg injection flow path.
- Gravity drain the RWST to backwash the containment emergency sumps.
- Establish alternate injection paths that include the VCT and BAT.
- Refill of the RWST from the CVCS or fire water system
- Restart Reactor Coolant Pumps (RCPs).
- Flood containment using the fire water system
- Transfer to combined hot leg / cold leg injection flow paths

At the time for switchover to hot leg injection, the containment sump inventory typically has been recirculated through the ECCS and RCS several times. Particulate and fibrous debris generated by the initial break and carried in the recirculating coolant is depleted by either capture on the sump strainer, fuel assemblies, or by settling out in the containment sump or in low flow locations of the ECCS reactor vessel flow path such as the reactor vessel lower plenum. Thus, the amount of particulates and fibrous debris in the recirculating flow at the time of initiation of hot leg recirculation is expected to be small. When considering chemical effects for STP, the results of tests using sump chemistry representative of post–LOCA conditions for STP indicate that significant impacts to strainer and fuel head loss due to chemical effects would not be expected.

STP's multiple hot leg injection flow paths provide a safety injection flow rate that is significantly greater than the core boil off rate. This ensures adequate flow to the core. Transferring to hot leg injection also has the potential to disturb any debris collected on the bottom of the fuel. The STP EOPs call for switchover of two trains to hot leg injection while maintaining cold leg injection with the third train.

Establishment of Alternate Flow Paths

If CET temperature indication reaches the established threshold, then alternate flow paths could be established to provide for core cooling. Some of the alternative flow paths considered are returning to the injection mode of core cooling through use of alternate water supplies. If unable to establish SI flow, then CCP flow is established to allow maximum injection into the RCS utilizing two CCPs. This alternate flow suction source can be aligned to either the RWST or VCT. As discussed previously, hot leg recirculation for STP using combined hot leg and cold leg injection flow paths has the potential to disturb the developed debris bed allowing for adequate core cooling.

Start a Reactor Coolant Pump

If CET temperature indication reaches the established threshold, then the operators could start an RCP. This action is expected to remove the material blocking the core and allow the normal recirculation injection flow paths to become effective at maintaining adequate core cooling.

Implementation of SAMGs or EDMGs

SAMG and Extensive Damage Mitigation Guidelines (EDMG) provide additional guidance and actions for addressing inadequate core flow conditions. Typically, SAMGs will be entered when directed by the EOPs and with the concurrence of the TSC. The SAMGs are used by the technical support staff in the TSC or EOF to evaluate alternative courses of action for a degrading condition. The SAMGs or the EDMGs will provide guidance for flooding containment above the reactor vessel hot and cold leg nozzles thus covering the break location to provide for convective circulation cooling of the reactor vessel.

C.5.9 Training Related to the Proposed Change

The proposed change does not result in changes to the symptom–based response procedures and guidelines beyond those already implemented in response to Bulletin 2003–01 and GL 2004–02. Initial training on sump blockage issues was completed as described in [4, 5, 6, 7, 8].

Licensed operator classroom and simulator training on indications of, and responses to, degraded pump flow indications which may be caused by containment sump clogging is provided during initial and requalification training.

Training has been conducted for Emergency Response Organization decision makers and evaluators in the TSC on indications of sump blockage and compensatory actions.

C.6 Barriers for Release of Radioactivity

The following evaluation demonstrates that the proposed change maintains sufficient safety margin for the current barriers for release of radioactivity which are the fuel cladding, the RCS boundary, the RCB, and the emergency plan (EP) actions. The evaluation concludes that the proposed licensing basis change:

- Does not affect or remove any of these levels of protection.
- Does not result in a significant increase in the existing challenges to the integrity of the barriers.
- Does not significantly change the failure probability of any individual barrier.

- Does not introduce new or additional failure dependencies among barriers that significantly increase the likelihood of failure when compared to the existing conditions.

- Does not change the overall redundancy and diversity features among the barriers that are sufficient to ensure compatibility with the risk acceptance guidelines.

C.6.1 Fuel Cladding

The fuel cladding barrier is maintained by the ECCS following a LOCA. After the initial phase of the accident mitigation, long term cooling is also maintained post–LOCA by the ECCS. The proposed licensing basis change for the change in methodology to use a RG1.174 risk-informed approach for GSI–191 does not make any change to the previous analyses and testing programs that demonstrate the acceptability of the ECCS for the initial phase of providing core cooling. The proposed licensing basis change shows that long term cooling is met for the additional accident mitigation and recovery phase. The proposed licensing basis change does not call for any equipment changes or design changes or for any changes to the plant operating and testing procedures beyond those already implemented in response to the concerns raised in GSI–191. There is no change to the design and analysis requirements for the fuel.

Emergency Core Cooling

To comply with GDC 35, "Emergency core cooling," STP has a system to provide abundant emergency core cooling. The system safety function is 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 are 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.

Long Term Cooling

To comply with 10 CFR 50.46(b)(5), "Long-term cooling," the STP RG1.174 risk-informed approach for post–LOCA sump performance shows that after the successful initial operation of the ECCS, the core temperature is maintained at an acceptable low value and decay heat is removed for the extended period of time required by the long-lived radioactivity remaining in the core.

C.6.2 Reactor Coolant System Pressure Boundary

The integrity of the RCS pressure boundary is postulated to be broken for the GSI–191 sump performance evaluation which is concerned with post–LOCA debris effects. However, the proposed change does not make any change to the previous analyses and testing programs that demonstrate the integrity of the RCS. Since the proposed licensing basis change does not impact any design or programmatic requirements for the reactor coolant pressure boundary, the likelihood of a LOCA is not affected.

Inservice Inspection Program

The ISI program performs an important role in the prevention of pipe breaks. The integrity of the Class 1 welds, piping, and components are maintained at a high level of reliability through the ASME Section XI inspection program. STP procedure, 0PSP11-RC-0015, for ASME Section XI Inservice Inspection, ensures that the following requirements of Technical Specifications 4.0.5 and 4.4.10 have been satisfied:

- Completion of the ISI program examinations of STP piping and component welds in accordance with the schedule requirements of the ASME Boiler and Pressure Vessel Code, Section XI (2004 Edition No Addenda).
- Completion of ISI of piping and equipment, and component supports (excluding snubber assemblies) in accordance with the schedule requirements of the Code.
- Completion of ISI containment metal liner in accordance with the schedule requirements of the ASME Boiler and Pressure Vessel Code.
- Completion of the examinations of the RCP flywheels in accordance with the requirements of RG1.174.

Reactor Vessel Nozzle Welds

All STP large bore piping welds (nozzle welds) susceptible to pressurized water stress corrosion cracking (PWSCC) have been replaced or otherwise mitigated with the exception of the Reactor Vessel nozzle welds. The reactor vessel nozzle welds are less of a concern in the GSI–191 analysis than other break locations because the reactor vessel is covered with reflective metal insulation (RMI), and the primary shield wall would protect the majority of fiberglass insulation in the steam generator compartments.

RCS leakage detection

The leak detection program at STP is capable of early identification of RCS leakage to provide time for appropriate operator action before a flaw causing a leak would propagate to a break. The effectiveness of this program is not reduced by the proposed licensing basis change to the risk–informed approach for GSI–191.

C.6.3 Containment Integrity

The evaluation of sump performance using a risk-informed approach is not a component of the analyses that demonstrate containment integrity. Previous analyses show that the containment structure can withstand the peak pressures calculated without loss of integrity. The containment remains a low leakage barrier against the release of fission products for the duration of the postulated LOCAs.

Containment Design Basis

The safety design basis for the containment is identified in GDC 50. The reactor containment structure, including access openings, penetrations, and containment heat removal systems, shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss of coolant accident.

Containment Heat Removal

The proposed change to the licensing basis does not involve any equipment changes beyond those modifications already made in response to the concerns raised in GSI–191. Thus there is no change to any of the containment heat removal components needed to maintain containment integrity. Therefore the proposed change does not significantly impact the structural capability and integrity of the RCB as an effective fission product barrier post–LOCA. The STP large, dry containments with safety–grade RCFCs are likely to survive a significant core damage event, even with a loss of the containment emergency sump

RCFCs are designed to operate independently in the post–LOCA environment and are not directly affected by the loss of the sump or containment sprays. This additional and independent capability to reject decay heat from containment ensures that the containment would not fail because of overpressure or overheating. Although core melt could be postulated, containment integrity would be maintained by operation of the RCFCs and the RCB would continue to be maintained as an effective fission product barrier [18].

Energy released to the containment atmosphere from the postulated accidents is removed by the CSS and RCFCs. STP has three groups of RCFCs with two fans and two heat exchangers in each group (total of six fans and heat exchangers). The RCFCs are designed to remove heat from the containment during both normal operation and accident conditions.

In the event of an accident, all RCFCs are automatically placed into operation on receipt of a safety injection signal. During normal operation, cooling water flow to the RCFCs is supplied by the non-safety grade chilled water system. Following an accident, cooling water flow to the fan coolers is supplied by the safety–grade CCW.

The RCFCs remove thermal energy from inside the containment to reduce the containment atmosphere pressure and temperature following loss of offsite power (LOOP) or a DBA. The operation of four of six RCFC units (two of three trains), or three of six RCFC units and two of three CSS trains are required to reduce the peak pressure and temperature of the RCB following a DBA.

Containment analyses consider operation of either two or three trains at the time of accident initiation. LOCAs for a DEGB pump suction break consider both maximum and minimum SI to assure coverage of all failure modes for the DBA. Minimum SI is based on single-failure of a standby diesel generator (SDG). This represents the most substantial loss of engineered safety features (ESF) equipment. ESF equipment lost with the SDG includes one train of SI, one train of CSS, one train of CCW to a RHR heat exchanger, and one train of RCFCs (two RCFC units).

The STP design calls for two trains of SI, two trains of CSS, and two trains of RCFCs to be used for accident mitigation to yield acceptable containment peak pressure results that are less than the containment design pressure of 56.5 psig. Analysis indicates that RCB failure takes place at more than 140 psig.

A study case has been performed to show that two LHSI pumps in the injection phase and one HHSI and one LHSI in the recirculation phase with zero CSS pumps and three RCFC trains gives acceptable results of containment pressure reaching 38.6 psig.

Another case study shows that one LHSI pump in the injection phase and one HHSI and one LHSI in the recirculation phase with zero SI pumps and one RCFC train results in a peak containment pressure of 62.0 psig.

Based on these study results, it is concluded that two trains of RCFCs are sufficient for containment heat removal if zero containment spray pumps are operating. Thus containment integrity is maintained if all the CSS pumps are secured.

Other industry studies have indicated the ability of the containment systems to survive challenges of 2.5 to 3 times the design levels. The Zion Probabilistic Safety Study showed that the containment ultimate capacity was 2.55 to 2.86 times the design capacity. Industry standard for large, dry containments is 2.5 to 3.0 times the design pressure limit [19].

Containment Testing

Technical Specification 6.8.3.j requires a Containment Leakage Rate Testing Program to be established to implement leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program is in accordance with the guidelines contained in RG 1.163 [20].

The proposed change does not impact the requirements for structural integrity and leak-tightness of the containment and does not involve any changes to the containment leakage testing requirements for demonstrating the effectiveness of the containment as a low leakage barrier is maintained. Testing requirements include RCB Integrated Leakage Rate Test (Type A), Containment Penetration Leakage Rate Test (Type B), and Containment Isolation Valve Leakage Rate Test (Type C) for compliance with Appendix A and Appendix J to 10 CFR Part 50.

C.6.4 Emergency Plan Actions

The proposed change to the licensing basis to use the methodology of a risk-informed approach does not involve any changes to the Emergency Plans. There is no change to the strategies for prevention of core damage, for prevention of containment failure, or for consequence mitigation. The use of the risk-informed approach does not impose any additional operator actions or complexity. Implementation of the proposed change would not result in any changes to the response requirements for Emergency Response Personnel during an accident. The STP DID approach includes the ability to detect, prevent, and mitigate post–LOCA strainer debris blockage and in–vessel debris blockage.

References

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