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

April 11, 2016

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

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 – REQUEST FOR ADDITIONAL INFORMATION RELATED TO REQUEST FOR EXEMPTIONS AND LICENSE AMENDMENT FOR USE OF A RISK-INFORMED APPROACH TO RESOLVE THE ISSUE OF POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING DESIGN-BASIS ACCIDENTS AT PRESSURIZED-WATER REACTORS (CAC NOS. MF2400 THROUGH MF2409)

Dear Mr. Koehl:

By letter dated June 19, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML131750250), as supplemented by letters dated October 3, October 31, November 13, November 21 and December 23, 2013 (two letters); and January 9, February 13, February 27, March 17, March 18, May 15 (two letters), May 22, June 25, and July 15, 2014; and March 10, March 25, and August 20, 2015 (ADAMS Accession Nos. ML13295A222, ML13323A673, ML13323A128, ML13338A165, ML14015A312, ML14015A311, ML14029A533, ML14052A110, ML14072A075, ML14086A383, ML14087A126, ML14149A353, ML14149A354, ML14149A439, ML14178A467, ML14202A045, ML15072A092, ML15091A440, and ML15246A125, respectively), STP Nuclear Operating Company (STPNOC) submitted exemption requests accompanied by license amendment requests (LARs) for a risk-informed approach to resolve Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR [Pressurized-Water Reactor] Sump Performance," at South Texas Project, Units 1 and 2.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the information provided in the documents above, and determined that additional information, as described in Enclosure 2 to this letter, is required to complete the review. It should be noted that this does not include risk-related questions; they will be issued under separate correspondence.

Draft copies of the enclosed request for additional information (RAI) questions were provided to Mr. Wayne Harrison of your staff via e-mails as shown on Table 1 of Enclosure 1 of this document. The RAI questions were discussed with your staff during public meetings held on the dates shown in Table 2 of Enclosure 1 of this document. It was agreed that STPNOC will provide responses to the requested information in accordance with the phased response Table 3 of Enclosure 1. If additional time is needed, STPNOC staff will discuss the additional time needed with the Division of Operating Reactor Licensing Project Manager to reach a mutually agreeable timeframe. Any additional delays to your responses may result in a delay to

the NRC staff's final decision on your requested actions. All responses are requested by July 31, 2016, to allow the staff to complete its review and make a decision by December 31, 2016.

If you have any questions, please contact me at 301-415-1906 or via e-mail at Lisa.Regner@nrc.gov.

Sincerely,

Lisa M. Regner, Senior Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosures:
1. RAI Phased Response Tables
2. RAI

cc w/encls: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION PHASED RESPONSE TABLES

EXEMPTION REQUESTS AND LICENSE AMENDMENT REQUESTS

RISK-INFORMED APPROACH TO RESOLVE GENERIC SAFETY ISSUE 191

STP NUCLEAR OPERATING COMPANY

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

Table 1. Table of Draft Questions Sent to Licensee and Captured in ADAMS

Branch(es)	E-mail sent	ADAMS Accession No.	Comments
SNPB	10/21/2015	ML16022A177	Initial T-H significant questions: QA and CLB change questions
SNPB	12/11/2015	ML16022A176	After November T-H audit, updated questions
SSIB DORL EPNB ESGB SCVB	1/14/2016	ML16022A008	Round 3 draft questions for listed branches
STSB	1/27/2016	ML16092A054	Technical Specifications change question
SNPB	2/4/2016	ML16040A069	Pre-audit draft T-H questions in preparation for audit on Feb. 24-26
EPNB	2/5/2016	ML16036A193	Updated EPNB-only questions following clarification call
SNPB	3/18/2016	ML16081A004	Round 3 draft T-H questions following audit on Feb. 24-26

NRR Branch Acronyms:
DORL – Division of Operating Reactor Licensing
EPNB – Component Performance, Non-destructive Examination and Testing Branch
ESGB – Steam Generator Tube Integrity and Chemical Engineering Branch
SCVB – Containment and Ventilation Branch
SNPB – Nuclear Performance and Code Review Branch
SSIB – Safety Issue Resolution Branch
STSB – Technical Specifications Branch

Table 2. Public Meetings Held on GSI-191 Since August 20, 2015

Public Meeting Date	Summary ADAMS Accession No.	Topics
10/1/2015	ML16011A061	Overview of methodology change, thermal-hydraulics, coatings
1/14/2016	ML16028A152	Debris generation and transport, thermal-hydraulics, boric acid precipitation, containment analysis
2/18/2016	ML16088A243	Thermal-hydraulics
3/3/2016	ML16092A044	Thermal-hydraulics, containment analysis, STPNOC's RAI Applicability Matrix, the risk audit planned for April 2016
3/17/2016	Ml16092A085	Thermal-hydraulics, boric acid precipitation, coatings, technical specification change, exemption to long-term core cooling, schedule

Table 3. Staggered Response for Round 3 Questions

Response Period (days from issuance of RAI)	Questions Due
30 days	Follow-up RAIs 18, 34, 38, and 44 SSIB-3-1 through 3, DORL-3-1, ENPB-3-1 through 3, ESGB-3-1, SNPB-3-1, -3, -5, -12, -14, -16, -19
60 days	Follow-up RAIs 19, 26, 33, 37 SSIB-3-7 through 9, SCVB-3-1, SNPB-3-2, -4, -8, -9, -10, -11, -13, -15, -18, -21, and SNPB-3-23 through 31
90 days	SSIB-3-4 through 6 STSB-3-1 SNPB-3-6, -7, -17, -20, -22, -32

REQUEST FOR ADDITIONAL INFORMATION

EXEMPTION REQUESTS AND LICENSE AMENDMENT REQUESTS

RISK-INFORMED APPROACH TO RESOLVE THE ISSUE OF POTENTIAL

IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING

DESIGN-BASIS ACCIDENTS AT PRESSURIZED-WATER REACTORS

STP NUCLEAR OPERATING COMPANY

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

By letter dated June 19, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML131750250), as supplemented by letters dated October 3, October 31, November 13, November 21 and December 23, 2013 (two letters); and January 9, February 13, February 27, March 17, March 18, May 15 (two letters), May 22, June 25, and July 15, 2014; and March 10, March 25, and August 20, 2015 (ADAMS Accession Nos. ML13295A222, ML13323A673, ML13323A128, ML13338A165, ML14015A312, ML14015A311, ML14029A533, ML14052A110, ML14072A075, ML14086A383, ML14087A126, ML14149A353, ML14149A354, ML14149A439, ML14178A467, ML14202A045, ML15072A092, ML15091A440, and ML15246A125, respectively), STP Nuclear Operating Company (STPNOC, the licensee) submitted exemption requests accompanied by license amendment requests (LARs) for a risk-informed approach to resolve Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR [Pressurized-Water Reactor] Sump Performance," at South Texas Project, Units 1 and 2 (STP).

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the information provided in your application, supplements, and responses to NRC staff requests for additional information (RAIs), and determined that additional information is needed, as described below.

NOTE: For the follow-up questions below, which reference additional information needed from previous NRC RAIs, the numbering system has been retained for consistency. New questions in this document have been assigned using a new numbering system for ease of use. The system consists of the Office of Nuclear Reactor Regulation (NRR) Branch acronym, the NRC RAI Round No., then a sequential number starting a 1.

NRR Branch Acronyms:
DORL – Division of Operating Reactor Licensing
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Enclosure 2

Follow-up Questions

Debris Transport

Follow-up RAI 18 – In the December 23, 2009, RAI (ADAMS Accession No. ML093410607), the NRC staff asked for a justification for an erosion value of 10 percent. Since the RAI was written, the NRC has accepted 10 percent, but STP uses 7 percent in its RoverD evaluation. Please justify the use of 7 percent as an erosion fraction for low density fiber glass (LDFG) in the pool for the deterministic portion of the evaluation. The referenced Alion test report concluded that 10 percent is a conservative erosion value for plants that can show that the test is applicable to their plant conditions. NRC acceptance of the Alion erosion report states that values less than 10 percent should not be used. There is no basis for acceptance of the 7 percent erosion value.

Follow-up RAI 19 – In the December 23, 2009, RAI, the NRC staff asked the licensee to estimate the quantity of fines that could erode from small pieces of fiberglass debris that were assumed to transport to the strainer, but settled during the head loss test. The NRC staff's position is that the small fibrous debris should be accounted for, either as settled in the pool and eroded, or by performing a test that confirms that any fiber that is calculated to transport to the strainer transports to the strainer during the test. The RoverD analysis does not explicitly account for small fibrous debris in that it only uses the fine debris amounts from the July 2008 test as a datum of comparison to the CASA Grande generated, transported, and eroded fine fiber quantities. Small fiber contributes to the RoverD fiber fines mass through erosion of small fiber retained in structures or settled in the pool. Previous approaches that may have been designed to overestimate the amount of small fiber accumulating on the strainer (in the context of now obsolete head loss computations), would result in underestimates of the amount of fiber fines in the RoverD analysis.

a) Please provide information on the amounts of small fiber assumed to transport to the strainer and assumed to settle in the pool (and potentially subject to erosion into fiber fines).

b) Please provide a sensitivity analysis of the RoverD results (e.g., set of critical welds and magnitude of the breaks causing the debris to exceed the tested amount of fiber fines, and changes on the delta core damage frequency) considering the assumptions and uncertainties on the amount of small fiber settling and retained on structures and transported to the strainer, as well as adequate values of erosion fraction (e.g., either 7 percent or 10 percent as discussed in the follow-up to RAI-18).

Follow-up RAI 26 – RAI 26 questioned the effects of the addition of 25 percent of the latent fiber to the test flume prior to starting the recirculation pump. The response to the question provided in the August 20, 2015, submittal references sinking metrics for stagnant water. The sump pool is not stagnant, but is significantly turbulent during pool fill-up. NRC staff disagrees with the statement that fiber would have mixed with particulate debris resulting in trapping or sediment of the fiber. Existing guidance states that all fine fiber should be considered to transport to the strainer. In addition, the 2008 head loss test is used to determine an acceptable

fiber limit for comparison. The comparison amount from the test should reflect the amount of fine fiber that was on the strainer at the test completion. Please provide a justification that placing 25 percent of the latent fiber into the test flume prior to starting the pump would result in the transportation of the fiber, or that the amount of fiber under consideration is insignificant.

Head Loss and Vortexing

Follow-up RAI 33 – In the December 23, 2009, RAI, the NRC staff asked the licensee to provide the margin to flashing across the strainer and debris bed, and the assumptions for the calculation. The licensee provided a calculation for the margin to flashing for large break loss-of-coolant accidents (LOCAs) at the start of recirculation as (in pounds per square inch [psi]):

Containment pressure + submergence – total strainer head loss – vapor pressure

= 43.1 + 0.3 - 1.5 - 39

= 2.3 psi

The licensee stated that post-LOCA containment over-pressure credit is needed to eliminate the potential for flashing. The licensee stated that the minimum strainer submergence was conservatively determined to be 0.5 inch for small break LOCA (SBLOCA), sump temperature and containment pressure would be lower for a SBLOCA than a large break LOCA (LBLOCA), strainer flow rate would also be lower, and debris transported to strainers would be much less such that there would be open strainer areas. Therefore, flashing is not expected to be an issue for SBLOCAs. The NRC guidance[1] is that sump temperature should be calculated conservatively high and containment pressure conservatively low to ensure no flashing will occur. It is acceptable to perform a time-based calculation taking viscosity, chemical timing, and strainer submergence into account. Most design basis calculations maximize containment pressure, which is non-conservative from a flashing perspective. Please explain in detail how sump temperature and containment pressure were calculated.

Follow-up RAI 34 – In the December 23, 2009, RAI, the NRC staff asked the licensee to provide an evaluation of the potential for deaeration of the fluid as it passes through the debris bed and strainer and whether any entrained gasses could reach the pump suction. The licensee stated that the net void fraction is 0 percent, and therefore, void fraction is not an issue for any of the pressure and temperature combinations associated with the post-LOCA fluid. The licensee explained that any void fraction that could occur at the strainer debris bed is minimal and that if any should occur, it is reversed before the strainer discharge water leaves the sump due to significant static head of water above the emergency core cooling system (ECCS) / containment spray system (CSS) pump suction inlets within the sump. The licensee concluded that the net void fraction is therefore zero and not problematic for any STP pressures and

[1] "Revised Content Guide for Generic Letter 2004-02 Supplemental Responses, November 2007" (ADAMS Accession No. ML073110278), and Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident."

temperatures from the strainer to the ECCS/CSS pump suction inlets within the sump. With respect to the RAI response:

a) Do the head loss values listed include clean strainer head loss?

b) How was the SBLOCA value determined?

c) Was containment spray head loss (CSHL) calculated separately?

d) Is the SBLOCA value less than the CSHL value alone due to the lower flow rate?

e) Since head loss is greater at lower temperatures and solubility is greater at lower temperatures, were any evaluations done at lower and higher temperatures to ensure the provided calculation is bounding?

f) As with the flashing evaluation, it is acceptable to use a time-based calculation that includes submergence, temperature, chemical timing, etc. If the values used in the deaeration evaluation are not considered to be design basis values, please provide information regarding how they were calculated. Please provide the design basis values for head loss, SBLOCA, CSHL, and the associated conditions.

g) The NRC staff understands that gas voids will decrease in volume as pressure increases. However, reabsorption of gasses into the fluid may not occur instantly. Please explain what is meant by the reversal of the void fraction due to the static head of the water.

Net Positive Suction Head

Follow-up RAI 37 – In the December 23, 2009, RAI, the NRC staff asked the licensee to provide net positive suction head (NPSH) margin results for low head safety injection (LHSI), high head safety injection (HHSI), and core spray (CS) pumps, for the LBLOCA and SBLOCA cases, under conditions of hot-leg recirculation. The licensee provided the NPSH margin for the LBLOCA for LHSI, HHSI, and CS. The licensee stated that its SBLOCA scenario will have little to no debris on the strainer to contribute to head loss. The licensee concluded that there is only clean strainer head loss for the SBLOCA case, which is much less than LBLOCA total strainer head loss. The lower flow for the SBLOCA would reduce clean strainer head loss compared to LBLOCA and the NPSH available would be slightly higher for SBLOCA since piping friction loss is less due to lower flow. Therefore, for SBLOCA compared to LBLOCA, NPSH margin would increase somewhat and total strainer head loss would be much less. Please provide additional detail for the basis that near-zero debris head loss is justified for the SBLOCA case.

Follow-up RAI 38 – In the December 23, 2009, RAI, the NRC staff asked the licensee to describe the methodology and assumptions used to compute the limiting pump flow rates for all pumps taking suctions from the ECCS sumps. The licensee stated that each of the three emergency sumps supplies water to the respective CS pump, LHSI pump, and HHSI pump for its associated train. The CS pumps discharge to common ring header piping arrangement and

the flow used for the NPSH evaluation is based on two CS pumps operating resulting in higher flow per pump than if all three were operating. Flow rates used for LHSI and HHSI are maximum values per the Technical Specifications (TSs). Please provide additional details on the methodology used to calculate the CS flows. For example, were they calculated by hand, or a hydraulic software package, or some other method?

Follow-up RAI 44 – In the December 23, 2009, RAI, the NRC staff asked the licensee to identify the volume of holdup assumed for the refueling canal and provide further information that justifies that the refueling canal drains cannot become fully or partially blocked such that additional holdup could occur, or the extent to which holdup could occur. The licensee provided a detailed explanation for why its refueling cavity drain lines will not become blocked. The licensee concluded that the refueling cavity drain lines are not assumed to become blocked and there is no water inventory holdup other than the water below the elevation of the drain lines. Please explain in detail the basis for the assertion that large pieces of debris will not reach the refueling cavity. If large pieces may transport to the refueling cavity, explain why they would not block the drain lines.

New Questions

SSIB-3-1

Nuclear Energy Institute (NEI) 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," December 2004 (ADAMS Accession No. ML050550138), recommends treating labels and tags either as intact and transportable to the strainer using 75 percent of the total area, or as 100 percent fine fiber if they may not remain intact. Alternately, it can be shown that they will remain intact under accident conditions and not transport to the strainer. Section 3b of the licensee's December 11, 2008, submittal (ADAMS Accession No. ML083520326), states that the miscellaneous debris term is bounded by 100 square feet. Section 3d states that the 100 square feet was implemented in the debris generation and transport analysis. Section 3f states that transport testing was completed that determined that the miscellaneous debris objects would not transport to the strainer so they were not included in the test. Please explain the effect that miscellaneous debris would have on the RoverD evaluation.

SSIB-3-2

Based on NRC staff's review of the RoverD submittal, it appears that large pieces of debris do not reach the sump pool since they are eroded at 1 percent, as would be expected for debris held up on gratings. Please confirm that large pieces of debris do not reach the pool, and provide details on what prevents any large pieces from falling into the pool.

SSIB-3-3

The NRC staff needs additional information to verify the RoverD computations. For all debris types and sizes, please provide a summary table of the fractions of:

a) all debris transported to the recirculation pool and the strainer,

b) the debris retained in structures, and

c) the debris settled in the pool.

SSIB-3-4

The NRC staff needs additional information to verify how each debris type and size is bounded by the July 2008 test or otherwise accounted for in RoverD.

a) Please provide an explicit comparison of RoverD transported debris amounts to the 2008 test amounts for each type and size of debris that could be produced in a LOCA event. Note that some qualified coating types listed as present in containment in Item 3.h, 1 of the RoverD submittal do not appear to be explicitly considered in the RoverD approach (e.g., qualified alkyds and baked enamel).

b) Please illustrate clearly how each type and size of debris that is predicted to transport to the strainer by RoverD is accounted for in the 2008 test. That is, show which test surrogate was assigned to each RoverD debris type.

c) The RoverD submittal states that the amount of Microtherm and unqualified epoxy included in the 2008 head loss test was deficient, but excess amounts of Marinite and inorganic zinc (IOZ) address the deficiency. Please explain in detail the basis for the assumption that the Marinite and IOZ account for the debris that was not included in the test. Provide the deficient amount of Microtherm and unqualified epoxy, and the amount and type of each test surrogate that is used to account for the deficient test amounts. The NRC staff has observed that Microtherm (and Min-K) has had a significantly larger impact on head loss than Marinite during strainer tests, and Marinite has been observed to have a larger effect on head loss than coatings.

d) Please show that each debris type predicted by RoverD to transport to the strainer is bounded by the surrogate included in the 2008 test, using appropriately conservative bounding transport assumptions for each debris size (e.g., smalls are maximally transported when considering the RoverD surrogate for smalls but are maximally eroded when considering the RoverD surrogate for fibers).

SSIB-3-5

Please justify the selection of fiber fines to define the sole failure criterion in RoverD. The 2008 test included fines and larger size fiber debris. Explain why only fine fiber amount should be

used as the failure criteria. Please address the 2008 test conditions and findings when performing this evaluation.

SSIB-3-6

In defining test amounts of epoxy coatings that need to be accounted for in the evaluation, it was assumed that coatings in the reactor cavity would not transport to the strainer. However, welds located inside the reactor cavity may allow unqualified coatings to fail and transport even if not within a zone of influence (ZOI).

a) Please provide more detail on the technical basis for test amounts, considering breaks inside the reactor cavity.

b) The Updated Final Safety Analysis Report (UFSAR) markup states that 100 percent of unqualified coatings fail as particulate. Table 16 in Attachment 1-2 states that 2008 pounds of unqualified epoxy were considered to fail as chips during the 2008 test. Please explain how this is accounted for in the evaluation.

c) Please provide a breakdown of the sizing of the unqualified epoxy outside the reactor cavity, including an explanation of why only 48 percent transports to the strainer.

d) Please evaluate whether amounts of epoxy particles in the July 2008 tests are truly bounding for cases of breaks inside the reactor cavity.

e) If the coating amounts from the 2008 test are not bounding, please provide an evaluation of the effects of excess coatings on the evaluation or provide a listing of the surrogates that were included in the test in excess that can be assigned to account for epoxy that may not have been adequately represented in the test.

SSIB-3-7

LOCA Containment Pressure, Sump Temperature, and Sump Level Response Analysis

The response to item 3.g.1 provides the sump temperature for the most recent LOCA containment pressure/temperature analysis performed by the licensee using the computer code GOTHIC. The UFSAR Section 6.2.1.1.3.1, "Containment Pressure and Temperature Analysis," states the LOCA containment pressure and temperature response analysis was performed using the CONTEMPT4/MOD5 computer code. The response to item 3.g.8 describes the methodology used to determine the post-LOCA sump level. In the current analysis in the UFSAR, the outputs of containment analysis for maximum sump temperature response should also include a sump level response.

a) Please explain if the containment pressure, sump temperature, and sump level reported in the August 20, 2015 supplement are consistent with the licensing basis results.

b) There is an STP license amendment request for an extension to the containment integrated leak rate testing from 10 years to 15 years currently under NRC review. Since this identifies a change from using CONTEMPT to GOTHIC for the licensing basis containment analysis, please explain what licensing basis methodology will be used going forward.

c) Please explain if the change in methodology from CONTEMPT to GOTHIC for the licensing basis LOCA containment analysis will maintain the methodology that results in maximizing the sump temperature and minimizing the sump level for the sump level and temperature response for determining available NPSH.

SSIB-3-8

FSAR NPSH Values

Tables 6.3-1 and 6.2.2-4 of Attachment 3-4 show UFSAR changes for ECCS component parameters and CSS pump NPSH parameters specifically for required and available NPSH.

The changes in Table 6.3-1 are as follows:

High Head Safety Injection Pump

Parameter	**Currently approved**	**For approval in the LAR**
Required NPSH at max. flow rate, ft (max)	16.1	1.1
Available NPSH, ft (From RWST [refueling water storage tank)	55.8	41.1
Available NPSH, ft (From RCB [reactor coolant building] Emergency Sump)	> 17.8	7.4

Low Head Safety Injection Pump

Parameter	**Currently approved**	**For approval in the LAR**
Required NPSH at max. flow rate, ft (max)	16.5	1.5
Available NPSH, ft (From RWST)	55.1	40.8
Available NPSH, ft (From RCB Emergency Sump)	> 18.0	7.5

The changes in Table 6.2.2-4 are as follows:

CSS Pump

Parameter	**Currently approved**	**For approval in the LAR**
Required NPSH at max. flow rate, ft (max)	16.4	1.4
Available NPSH, ft (From RWST)	56.1	41.4
Available NPSH, ft (From RCB Emergency Sump)	> 17.6	7.2

Please confirm that the UFSAR changes for the above parameters are not a result of changes to the NPSH licensing basis calculations, but instead a change in the reference point for the calculated values. Define what the reference point is for the currently approved values, for the values for approval in the LAR, and where this definition is described in licensing basis documents.

SSIB-3-9

The UFSAR markup in the RoverD submittal does not provide a clear value for NPSH margin. The acronym TSHL is not defined in the markup. It is not clear in the tables whether margin includes the strainer losses. Since TSHL changes with temperature, the listing of a single value can be misleading, especially when it is not the limiting value. Consider including a more robust description of TSHL and how it affects NPSH margins under varying conditions.

DORL-3-1

Please provide a specific list of all licensing basis changes, in the application, for which you are requesting NRC review and approval via Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR).

EPNB-3-1

In Attachment 1-2, page 4, of the August 20, 2015, submittal, the licensee stated that the large main steam and feedwater line breaks were not evaluated because recirculation is not required under the plant licensing basis for STP.

a) Please explain why sump recirculation is not required for main steam and feedwater line breaks inside the containment.

b) Please discuss whether any other American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Class 2 piping inside the containment, besides the main steam and feedwater lines, are evaluated for debris generation and sump recirculation. If none, explain why ASME Code Class 2 piping inside the containment are not evaluated.

EPNB-3-2

In Attachment 1-4, page 22, of the August 20, 2015, submittal, under the heading, *Reactor Coolant System Weld Mitigation*, the licensee stated that "...All STP large bore RCS [reactor coolant system] welds susceptible to pressurized water stress corrosion cracking (PWSCC) have been replaced with Alloy 690 material which is not susceptible to PWSCC (SG [steam generator] nozzles) or overlaid with non-susceptible Alloy 52/52M/152 material (pressurizer piping safe ends) with the exception of the reactor vessel nozzle welds..."

a) Please clarify whether "the reactor vessel nozzle welds" discussed in the above statement are the J-groove welds associated with the reactor vessel closure head penetration nozzles to house the control rod drive mechanisms (i.e., control

rod drive mechanism (CRDM) nozzles), or the full-penetration butt welds associated with the hot-leg nozzles that are attached to the reactor vessel shell.

b) Please discuss of what material the CRDM nozzles and the associated J-groove welds in both units are made. If the nozzles are composed of Alloy 600 material and the welds are Alloy 82/182, then discuss why these were not selected as break locations since these materials are susceptible to PWSCC.

c) Please identify: 1) the large bore RCS piping (e.g., hot leg, cold leg, or crossover piping) and other ASME Code Class 1 pipes (e.g., pressurizer surge line, pressurizer spray line, or safety injection piping) that contain either Alloy 690 weld material or are mitigated with Alloy 52/52M/152 material, and that are considered in the GSI-191 evaluation; and (2) all ASME Class 1 piping that is larger than 2 inches that contain Alloy 82/182 weld material, has not been mitigated with Alloy 52/52M/152 material, and are considered in the GSI-191 evaluation.

EPNB-3-3

Please explain why breaks from the pressurizer heater sleeves and reactor vessel bottom-mounted instrumentation nozzles were not considered as a source of debris generation.

STSB-3-1

Paragraph 10 CFR 50.36(c)(2)(i) states, in part, that:

> Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

In its second supplement to the LAR for a risk-informed approach to resolving GSI-191, STPNOC proposed the following new condition 'c.' for TS LCO 3/4.5.2, ECCS SUBSYSTEMS - TAVG GREATER THAN OR EQUAL TO 350°F:

c. With less than the required flow paths OPERABLE solely due to potential effects of LOCA generated and transported debris that exceeds analyzed amounts, perform the following:

   1. Immediately initiate action to implement compensatory actions,

      AND

   2. Within 90 days restore the affected flowpath(s) to OPERABLE status,

   OR

   Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

The NRC staff is concerned the 90-day proposed completion time would allow an inappropriate amount of time for certain scenarios. For instance, 90 days appears to be excessive for a scenario where gross blockage of the strainer is evident and it is clear that the ECCS would be incapable of performing its specified safety function (e.g., if tarps were inadvertently left covering the sump screens following an outage). In other cases (e.g., where an administrative fiber limited is inadvertently exceeded), the 90 days may be appropriate because of conservatisms in the licensing basis analysis.

Most TS completion times are of limited duration, making the consideration of the spectrum of scenarios that could render a structure, system, or component (SSC) inoperable unnecessary. For long duration completion times, however, such as the 90 days discussed above, the licensee needs to show that the completion time minimizes the level of risk to the public. The NRC staff requests the licensee to provide an explanation of the technical basis for the 90 days, including how any currently established programs, such as the configuration risk management program, would factor into its response to an unlikely, but severe scenario discussed above.

ESGB-3-1

Provide additional details and clarification with respect to the manner in which the total mass of unqualified coatings was calculated. Attachment 1-2, page 67 of 95 of the August 20, 2015, submittal states, in part, that "the weight of applied coatings are determined based on a theoretical coating spread rates (sq. ft per gallon @ 1 mil thick) instead of specific vendor coating spread rates." If a 1 mil (thousandths of an inch) thick coating was assumed for IOZ or epoxy coatings, the analysis may be significantly underestimating the amount of coating debris. Please describe the thickness used in the analysis for both epoxy and IOZ coatings since the mass of epoxy within the ZOI may be impacted and the mass of unqualified IOZ throughout containment may be impacted.

SCVB-3-1

The STP UFSAR Section 6.2.1.3, "Mass and Energy Release Analyses for Postulated Loss of Coolant Accidents," states the use of Westinghouse Electric Company, LLC (Westinghouse) WCAP-10325-P-A methodology for LOCA mass and energy release analysis. Westinghouse has issued Nuclear Safety Advisory Letters (NSALs)-06-6, -11-5, and -14-2, and InfoGram IG-14-1 reporting errors in this methodology. Also a new methodology (GOTHIC) is used for LOCA containment analysis for which the mass and energy input needs to be corrected based on the above NSALs and the InfoGram. Please submit the following revised licensing basis containment analysis for NRC review which should include the following: (a) changes, and justification of changes in inputs and assumptions from the current analysis, and (b) results.

a) LOCA containment mass and energy release analysis.

b) Pressure and temperature response analysis for containment integrity

c) Peak temperature analysis for equipment environmental qualification (EEQ)

d) Sump temperature and level response for NPSH analysis.

e) Minimum containment pressure response for ECCS analysis.

Thermal-Hydraulic Review Questions

Note 1: the draft SNPB questions sent to STPNOC by e-mail dated October 21, 2015 (ADAMS Accession No. ML16022A177), were subsumed by questions SNPB-3-3 and SNPB-3-20 below.

Note 2: The following SNPB questions are from the criteria set forth in the following two NRC staff guidance documents:

- *Safety Evaluation for the Westinghouse Topical Report WCAP-16793-NP, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," Revision 2. This will be identified as "SE for WCAP."*

- *NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 15.0.2. The subsection referenced will be provided with the question. This will be identified as "SRP."*

SNPB-3-1

Cladding Oxide					
Please demonstrate that the thickness of the cladding oxide and the deposits of material on the fuel do not exceed 0.050 inches in any fuel region.					
Criterion	0.1	**Reference**	SE for WCAP		

SNPB-3-2

Accident Scenario Progression					
Please provide a description of the accident progression of the accident scenarios being simulated using the long-term core cooling (LTCC) evaluation model (EM). This description should start at the initiation of the break, define each phase, and provide the important phenomena occurring in that phase in the various locations of the reactor coolant system (RCS) (e.g., core, reactor vessel, steam generators - both primary and secondary side, loops, pressurizer, pumps, containment).					
Criterion	1.2	**Reference**	SRP, III.3c		

SNPB-3-3

Clarification on Core Bypass Blockage					
During the audit, STPNOC considered performing the LTCC analysis with the core bypass open to allow flow in the axial direction. If STPNOC credits the use of the bypass, it should provide test data to demonstrate that the bypass will not block during the scenarios. This test data should bound the flow rates, flow areas, and debris loading expected in the RCS.					
Criterion	1.3	**Reference**	SRP, III.3c		

SNPB-3-4

Describe Important Phenomena					
Please provide a description of the important phenomena being modeled in the LTCC EM for each of the accident scenarios being simulated. These phenomena should include those important to obtaining the correct <u>initial conditions</u> for the long-term phase, and the <u>important phenomena</u> during the long-term phase.					
Criterion	1.3	**Reference**	SRP, III.3c		

SNPB-3-5

Debris at Grid Spacers					
Please describe how the LTCC EM accounts for potential blockages at the spacer grid in the core above the bottom grid.					
Criterion	1.3	**Reference**	SRP, III.3c		

SNPB-3-6

Initial and Boundary Conditions for each Accident Scenario					
Please demonstrate that the initial and boundary conditions for each accident scenario are appropriate for the given simulation. This demonstration should focus on the simulations performed under the 10 CFR Part 50 Appendix B Quality Assurance Program. Provide a discussion of the confirmation of the initial and boundary conditions, and describe how these conditions reflect the conditions in the plant. Provide a discussion on the treatment of uncertainties. Provide appropriate references. *If this demonstration relies on comparisons with results from other computer codes, please provide (1) a description of the code, (2) confirmation that the code has been approved by the NRC, (3) a summary of the simulations the code has been approved to analyze, and (4) an analysis addressing each initial and boundary condition, and how a deviation in that condition would be reflected in the code comparison.* *Please confirm that the steady state simulation is consistent with plant operation (e.g., pressure drop around the loop). Confirm that important system parameters are being applied with their TS values or values assumed in the UFSAR as appropriate (e.g., flow rates, temperatures).*					
Criterion	1.4	**Reference**	SRP, III.3c		

SNPB-3-7

Initial and Boundary Conditions for the Long-Term Phase					
Please demonstrate that the initial and boundary conditions for each accident scenario at the beginning of the long-term phase are consistent with those conditions which are expected. This demonstration should analyze the RELAP5-3D calculations for the conditions at the beginning of the reflood stage, and show that those calculations are reasonable compared with known behavior. This analysis should include a comparison between the conditions calculated by RELAP5-3D and the current large and small break LOCA safety analyses.					
Criterion	1.4	**Reference**	SRP, III.3c		

SNPB-3-8

How are the Phenomena Modeled					
Please summarize how the important phenomena are being modeled in the LTCC EM. This discussion should provide the phenomena and a summary of how it is being modeled (e.g., through the field equations, by an identified closure relationship).					
Criterion	2.2	**Reference**	SRP, III.3a		

SNPB-3-9

Reference and Limits of Closure Relationships					
Please demonstrate that each closure relationship is associated with an appropriate reference providing its limits of applicability.					
Criterion	2.3	**Reference**	SRP, III.3a		

SNPB-3-10

User Manual					
Please provide the user manual and/or similar guidance for analysts performing simulations using the LTCC EM. The user manual should provide (a) detailed instructions about how the computer code is used, (b) a description of how to choose model input parameters and appropriate code options, (c) guidance about code limitations and options that should be avoided for particular accidents, components, or reactor types, and (d) if multiple computer codes are used, documented procedures for ensuring complete and accurate transfer of information between different elements of the EM.					
Criterion	2.4	**Reference**	SRP, III.3a		

SNPB-3-11

Modeling of Important Phenomena					
Please provide a summary of the important phenomena and discuss how the LTCC EM models these phenomena.					
Criterion	3.2	**Reference**	SRP, III.3b		

SNPB-3-12

Field Equations					
Please define and provide a summary of the field equations for the LTCC EM. This should include identification of the of the conservation equation (e.g., mass, momentum) and the number of dimensions of the equation. For portions of the RCS model that change in nodalization (e.g., 1-D to 3-D), a separate description may be necessary. Additionally, demonstrate that these equations are able to model the necessary phenomena.					
Criterion	3.3	**Reference**	SRP, III.3b		

SNPB-3-13

Validation of Closure Relationships			
For the closure relationships identified, please provide appropriate validation for the use of this relationship over its expected application domain. This validation should include comparisons to separate effects tests and/or integral test data and appropriately address the model's uncertainty. Where appropriate, discuss any similarity criteria, scaling rationale, assumptions, simplifications, and/or compensating errors.			
Criterion	3.4, 3.8, 3.9, 4.3,4.6, 5.2, 5.4, 5.5, 5.6	**Reference**	SRP, III.3b, d, e

SNPB-3-14

Simplifying and Averaging					
Please provide a summary of the key simplifying and averaging assumptions used in the generation of the mathematical models used in the LTCC EM and demonstrate that they are appropriate for the accident scenarios being modeled.					
Criterion	3.5	**Reference**	SRP, III.3b		

SNPB-3-15

Level of Detail					
Please confirm that the level of detail (e.g., phenomena modeled, initial and boundary conditions, overall assumptions) is consistent between STP's LOCA licensing basis analysis and the simulations performed in the LTCC EM.					
Criterion	3.6	**Reference**	SRP, III.3b		

SNPB-3-16

Single Version of the Evaluation Model					
Please confirm that a single version of the EM was used during the simulations of the given accident scenarios. This includes confirming that the code version was frozen and the manner for calculating or obtaining inputs did not change.					
Criterion	4.1	**Reference**	SRP, III.3d		

SNPB-3-17

Validation of the Evaluation Model			
Please provide appropriate validation demonstrating that the LTCC EM will result in a reasonable prediction of the important figures of merit for the accident scenarios considered. Demonstrate that the validation covers the range of the accident scenarios used in the LTCC EM. This validation should include comparisons to integral test data and appropriately address the model's uncertainty. Where appropriate, discuss any similarity criteria, scaling rationale, assumptions, simplifications, and/or compensating errors.			
Criterion	4.2, 3.8, 3.9, 4.3,4.6, 5.2, 5.4, 5.5, 5.6	**Reference**	SRP, III.3b, d, e

SNPB-3-18

Mesh Size Sensitivity					
Please demonstrate that the LTCC results are independent of mesh size for the accident scenarios under consideration.					
Criterion	4.7	**Reference**	SRP, III.3d		

SNPB-3-19

Initial Test Cases					
Please provide a summary of the assessment cases performed in order to demonstrate that RELAP5-3D has been installed and is being used appropriately.					
Criterion	4.7	**Reference**	SRP, III.3d		

SNPB-3-20

Specific Sensitivity Studies					
During the audit, the NRC staff identified a number of sensitivity studies that would be important for the NRC staff review of the proposed LTCC evaluation methodology. STP is requested to perform the following sensitivity studies and submit plots of the relevant figures of merit and important timings for LTCC analysis: *a) Appendix K decay heat load with single worst failure and steam generator tube plugging* *b) Axial power shape* *c) Break sensitivity study with appropriate break size resolution* d) *No bypass blockage*					
Criterion	4.7	**Reference**	SRP, III.3d		

SNPB-3-21

<table>
<tr><td colspan="6" align="center">Important Sources of Uncertainty

Please demonstrate that the important sources of uncertainty are appropriately accounted for in the LTCC EM.</td></tr>
<tr><td>Criterion</td><td>5.3, 5.2</td><td>Reference</td><td>SRP, III.3e</td><td></td><td></td></tr>
</table>

SNPB-3-22

<table>
<tr><td colspan="6" align="center">Uncertainness and Design Margin

Please provide a discussion on the impact of the uncertainties considered on the important figures of merit (e.g., peak centerline temperature) for each of the accident scenarios and the margin to the design limit.</td></tr>
<tr><td>Criterion</td><td>5.7</td><td>Reference</td><td>SRP, III.3e</td><td></td><td></td></tr>
</table>

SNPB-3-23

Evaluation Model in an Appendix B Quality Assurance (QA) Program

To address Generic Letter (GL) 2004-02, STP demonstrates its compliance with 10 CFR 50.46(b)(5) Long term core cooling, including the impact of debris, using the following two step approach:

(1) The hot-leg large break, hot-leg medium break, hot-leg small break, and cold-leg small break will be demonstrated to be in compliance with 10 CFR 50.46(b)(5) by ensuring that the long-term core temperature does not exceed 800 degrees Fahrenheit (°F) assuming a fully blocked core. This is demonstrated by using deterministic analysis performed with RELAP5-3D.

(2) The cold-leg large break and cold-leg medium break will rely on a risk-informed approach.

The hot-leg large break, hot-leg medium break, hot-leg small break, and cold-leg small break analyses are used to demonstrate compliance with 10 CFR 50.46(b)(5). Therefore certain design control measures are required, as specified in 10 CFR 50, Appendix B (III):

> *Design control measures shall be applied to items such as the following: reactor physics, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for inservice inspection, maintenance, and repair; and delineation of acceptance criteria for inspections and tests.*

However, it is not apparent that the RELAP5-3D analysis was performed under a QA program satisfying the requirements of Appendix B.

Please demonstrate that the RELAP5-3D analysis was performed under a QA program which satisfies the requirements of 10 CFR 50, Appendix B, or provide a similar analysis that was performed under such a program.

Criterion	6.1	**Reference**	SRP, III.3f		

SNPB-3-24

Input Verification

Please provide details of how STP's QA program controls over the input deck for the LTCC EM. How are the input values verified? What inputs are users given permission to change and how are such changes controlled?

Criterion	6.1	**Reference**	SRP, III.3f		

SNPB-3-25

Proper Convergence					
Please explain how the QA program ensures the code converged properly. Such indicators commonly include nonphysical state properties and excessive mass error. Demonstrate that if the code did not converge numerically, the analysts would be alerted to the error messages and act appropriately.					
Criterion	6.1	**Reference**	SRP, III.3f		

SNPB-3-26

Non-physical Results					
Please explain how the QA program ensures identification of non-realistic results such as liquid over vapor, unphysical oscillations that could be numerically induced, or any other nonphysical results that may lead to erroneous conclusions concerning the code's calculated thermal-hydraulic behavior.					
Criterion	6.1	**Reference**	SRP, III.3f		

SNPB-3-27

Realistic Results					
Please explain how the QA program ensures the physical results are realistic. Where the calculated flow regimes and heat transfer modes should be studied to ensure that the code is not assuming unrealistic conditions?					
Criterion	6.1	**Reference**	SRP, III.3f		

SNPB-3-28

Boundary Conditions as Prescribed					
Please explain how the QA program ensures that the boundary conditions are occurring as prescribed. Boundary conditions and others that control the direction of the transient (e.g., valves opening, pumps beginning to coast down, or heater rod power turning off) should be checked by the user to ensure expected performance.					
Criterion	6.1	**Reference**	SRP, III.3f		

SNPB-3-29

Thoroughly Understood Results					
Please explain how the QA program ensures that every aspect of the calculation is thoroughly understood. The depressurization rate, various indications of core heatup, drain rate of the system at various locations, liquid holdup, indications of condensation or evaporation, transition from subcooled to two-phase break flow, and other conditions should all be explainable. Also, the results of the user's calculation should be understood from the perspective of previous calculations done on the same or similar facilities.					
Criterion	6.1	**Reference**	SRP, III.3f		

SNPB-3-30

Quality Assurance Program Documentation					
Please demonstrate that the documentation for the QA program includes procedures to address all relevant areas including, but not limited to, design control, document control, software configuration control and testing, and corrective actions.					
Criterion	6.2	**Reference**	SRP, III.3f		

SNPB-3-31

Independent Peer Review					
Please demonstrate that the QA program used independent peer review in the key steps appropriately. This should include a description of the steps where independent peer review was applied and how independence was defined and obtained.					
Criterion	6.2	**Reference**	SRP, III.3f		

SNPB-3-32

Important Sources of Uncertainty					
Please identify the important sources of uncertainty in the LTCC EM.					
Criterion	5.1	**Reference**	SRP, III.3e		

SNPB-3-33

In response to SNPB RAI 10 and SSIB RAI 66 dated March 3, 2015 (ADAMS Accession No. ML14357A171), STPNOC stated in its August 20, 2015, submittal that the current risk-over-deterministic (RoverD) analysis relies on the hot leg switchover (HLSO) timing of 5.5 hours as stated in STP UFSAR Section 15.6.5.2. As such, the NRC staff believes that changes to the HLSO timing would impact the RoverD analysis and thus the risk-informed resolution to GSI-191. In order to better understand the basis for the current HLSO timing, the NRC staff has audited STPNOC's current boric acid precipitation (BAP) control analysis on two occasions, in March 2015 and February 2016.

During the February 2016 thermal-hydraulic audit at Westinghouse offices in Rockville, Maryland, the NRC staff were presented with testing data supporting STPNOC's contention that

the barrel-baffle region mixes with the core under certain conditions. This region has not been previously credited in a BAP analysis. Given this, the NRC staff is not satisfied that a sufficient level of quantitative support has been provided for inclusion of any portion of the barrel-baffle region in the mixing volume. The NRC staff requests that STPNOC either:

a. Provide additional quantitative justification for the use of the barrel-baffle region in the mixing volume, including discussion of the applicability of test data to the STP plants (e.g., scaling of the tests used and the design of the test facilities relative to the design of STP), or

b. Perform a sensitivity analysis of the impact of omitting the barrel-baffle region from the mixing volume, to demonstrate that the current STP HLSO timing of 5.5 hours would be supported.

the NRC staff's final decision on your requested actions. All responses are requested by July 31, 2016, to allow the staff to complete its review and make a decision by December 31, 2016.

If you have any questions, please contact me at 301-415-1906 or via e-mail at Lisa.Regner@nrc.gov.

Sincerely,

/RA/

Lisa M. Regner, Senior Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosures:
1. RAI Phased Response Tables
2. RAI

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ASmith, NRR/DSS/SSIB
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ADAMS Accession No. ML16082A507 * by email

OFFICE	NRR/DORL/LPL4-1/PM	NRR/DORL/LPL4-1/LA	NRR/DSS/SCVB/BC*	NRR/DSS/SSIB/BC*	NRR/DSS/SNPB/BC*
NAME	LRegner	JBurkhardt	RDennig	VCusumano	JDean
DATE	3/26/16	3/23/16	3/31/16	4/5/16	4/6/16
OFFICE	NRR/DE/ESGB/BC*	NRR/DSS/STSB/BC*	NRR/DE/EPNB/BC*	NRR/DORL/LPL4-1/BC	NRR/DORL/LPL4-1/PM
NAME	GKulesa	RElliott	DAlley	RPascarelli	LRegner
DATE	3/31/16	3/17/16	3/30/16	4/11/16	4/11/16

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