Nuclear Operating Company

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

July 28, 2016
NOC-AE-16003397
10 CFR 50.12
10 CFR 50.90

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

South Texas Project
Units 1 & 2
Docket Nos. STN 50-498, STN 50-499
Response to RAI 4-2 of APLA Round 4 Requests for Additional Information
Regarding STP Risk-Informed GSI-191 Licensing Application
(TAC NOs MF2400 and MF2401)

References:

1. Letter, G. T. Powell, STPNOC, to NRC Document Control Desk, "Supplement 2 to STP Pilot Submittal and Requests for Exemptions and License Amendment for a Risk-Informed Approach to Address Generic Safety Issue (GSI)-191 and Respond to Generic Letter (GL) 2004-02", August 20, 2015, NOC-AE-15003241, ML15246A126

2. Letter, Lisa Regner, NRC, to Dennis Koehl, STPNOC, "South Texas Project, Units 1 and 2- Request for Additional Information Related to the Risk Review of the Request for Exemptions and License Amendments to Resolve the Issue of Potential Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors", May 26, 2016, ML16125A290

3. Letter, G. T. Powell, STPNOC, to NRC Document Control Desk, "Responses to APLA Round 4 Requests for Additional Information Regarding STP Risk-Informed GSI-191 Licensing Application", July 20, 2016, NOC-AE-16003390

Reference 2 transmitted Probabilistic Risk Assessment Licensing Branch (APLA) Branch RAIs regarding the South Texas Project Nuclear Operating Company (STPNOC) application in Reference 1. Reference 3 responded to all the APLA RAI in Reference 2 except RAI 4-2. This submittal responds to RAI 4-2.

The attached evaluation shows that the indirect seismic contribution to LOCA frequencies and the potential impact on the change in CDF and change in LERF due to sump debris effects is insignificant.

There are no commitments in this submittal.

A147
NRR

STI 34348952

If there are any questions, please contact Mr. Wayne Harrison at 361-972-8774.

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

Executed on: July 28, 2016

James Connolly

James Connolly
Site Vice President

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Attachments:

1. Response to APLA RAI 4-2
2. Definitions and Acronyms

cc:

(paper copy)

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

Lisa M. Regner
Senior Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North (O8H04)
11555 Rockville Pike
Rockville, MD 20852

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

(electronic copy)

Morgan, Lewis & Bockius LLP
Steven P. Frantz, Esquire

U. S. Nuclear Regulatory Commission
Lisa M. Regner

NRG South Texas LP
Chris O'Hara
Jim von Suskil
Skip Zahn

CPS Energy
Kevin Pollo
Cris Eugster
L. D. Blaylock

Crain Caton & James, P.C.
Peter Nemeth

City of Austin
Elaina Ball
John Wester

Texas Dept of State Health Services
Helen Watkins
Robert Free

Attachment 1

Response to APLA RAI 4-2

ABS Consulting Letter Report
Response to RAI APLA-4-2 – Indirect LOCAs
(STP-3698088-O-02)

Response to
RAI APLA-4-2 – Indirect LOCAs

APLA-4-2

NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," April 2008 (ADAMS Accession No. ML082250436), includes only breaks caused by long-term material degradation. Other potential contributors to LOCA frequency, such as seismically induced LOCA (both direct and indirect), should be evaluated separately. A "direct" seismically induced LOCA involves rupture of a piping or non-piping component caused by the seismic event itself. For example, an "indirect" seismically induced LOCA is caused by failure of piping or component supports that leads to the consequential failure of the piping or non-piping component.

In its May 22, 2014, response to a U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) dated April 14, 2014 (ADAMS Accession No. ML 14087A075), STP provided an estimate of the frequency of seismically induced LOCA (Attachment 1, p. 24/86, Reference 1). However, the response did not appear to consider indirect seismically induced LOCAs. One acceptable approach for evaluating indirect seismically induced LOCAs is for the analyst to use the method described by NUREG-1903, "Seismic Considerations for the Transition Break Size," Section 4.6, "Indirectly Induced Piping Failures," February 2008 (ADAMS Accession No. ML080880140). "Representative" values in the NUREG could be replaced with site-specific fragility and hazard information that, as appropriate, accounts for any effects of material degradation or aging. Alternatively, it may be demonstrated that the representative values are bounding for the site with consideration of effects due to material degradation or aging.

Please clarify whether the analysis documented in the RAI response considered indirect seismic LOCAs. If not, provide an analysis accounting for indirect damage mechanics eventually leading to rupture of piping and non-piping systems and LOCA events. For both direct and indirect seismically induced LOCAs, estimate, bound, or screen any increase in seismic risk due to debris.

Response

The earlier response in Attachment 1, p. 24/86, referenced in the RAI above developed the seismic caused occurrence frequency of three different LOCA sizes: small, medium, and large. The fragility curves used were based on a much older study that did not explicitly state whether indirect failures caused by support failures were also included. However, it is expected that it was the intent to include indirect failures because anchorages were included in the assessment. For completeness, the approach described in NUREG-1903 is also considered below.

NUREG-1903 (Reference 2) refers to an earlier Lawrence Livermore National Laboratory (LLNL) report prepared for the NRC on pipe failure probabilities, NUREG/CR-3660 (Reference 3). NUREG/CR-3660 concluded that the direct contributors to larger LOCAs from seismic events were negligible. This is noted on page 3-7 of NUREG-1903 and repeated below:

> *"Overall, the LLNL study concluded that the probability of a direct DEGB in RCS piping is very low for both PWR and BWR plants. LLNL further concluded that the probability of a leak or direct DEGB in the RCS piping is negligibly affected by earthquakes, to the extent that direct DEGB and earthquake can be considered independent random events."*

For indirect effects on Westinghouse plants (similar to what they did for CE plants), NUREG-1903 also goes on to say the following on page 4-50:

> *"As for the CE plant, the staff's approach to the scoping study was to integrate the fragility function over the seismic hazard curve. Integrating over the seismic hazard curve from 0g to 1.5g, as specified for IPEEE evaluations, the staff determined that the mean probability of failure of the lowest-capacity component support is 7.6x10-7 per year. The staff also defined the fragility function by the median of 1.88g and a composite uncertainty of 0.42 (i.e., the value derived in NUREG/CR-3663 [NRC, 1985b])."*

NUREG/CR-3663 is listed as Reference 4 below. Assuming that the Beta coefficients for randomness and uncertainty are roughly equal, setting them both to 0.3 yields a beta composite value of 0.42. The high confidence of low probability of failure (HCLPF) value, or HCLPF, is then 0.7g for a median acceleration value of 1.88g and with these 0.3 beta values.

To examine the risk from sump debris at South Texas Project during a seismically caused LOCA, the frequency of a seismic caused indirect large LOCA can be obtained by integrating the seismic hazard curve applicable to South Texas Project weighted by the fragility failure probability over each acceleration interval covering the full range of accelerations. The seismic hazard curve assumed for this calculation is the same as in the response to the previous RAI (Reference 1). For this integration, the hazard curves were extrapolated to 1.6g.

The integration was performed discretely over 12 acceleration ranges using the RISKMAN™ software's "Fragility" module (Reference 5). The 12 intervals of acceleration were each further divided into 100 sub-intervals resulting in 1,200 points total used for the integration. No truncation of the fragility's lowest probability tails was applied. The integrated results for each mean seismic hazard curve are shown in Table 1.

Table 1. Seismic Generated Indirect LOCA Frequencies; HCLPF = 0.7g

Frequency of Seismically Generated Indirect LOCAS Summed Over all Accelerations (1/yr.)			
Fragility Group	**Conditional Probability of LOCA Given Support Failure**		
	1.0 (upper bound)	**0.2 (small)**	**0.05 (large)**
Unconditional Indirect LOCA Frequency; HCLPF = 0.7g	8.27E-9	1.65E-9	4.14E-10
Unconditional Seismic SBO Frequency	1.98E-7	1.98E-7	1.98E-7
Unconditional Indirect LOCA Frequencies with SBO Removed	3.65E-09	7.29E-10	1.82E-10

The unconditional frequency of an indirect LOCA occurrence by seismic events is presented in the first row. Three columns of results are provided. The first column, labeled upper bound, presents the frequency of an indirect support failure with a probability of 1.0 that the support failure leads to a break in the reactor coolant system (RCS) boundary and therefore a LOCA. The second column gives the LOCA frequency assuming a more realistic estimate of a small break occurring given support failure; i.e., 0.2. The third column presents the frequency assuming a more realistic estimate of a large break occurring given support failure; i.e., 0.05. These estimates for a small break or a large break given pipe support failure were first estimated in Reference 6 for balance of plant piping. They are assumed here to reflect the chance of a RCS pipe break given failure of its weakest support.

These three frequencies developed above are overly conservative to use for GSI-191 applications because there is still some probability less than 1.0 that sump debris fails sump recirculation, and because much of the seismic induced indirect LOCA frequency comes from accelerations greater than 0.7g, where other plant components would also fail. If other plant equipment failures occur and they alone result in core damage, then the impact of an additional, indirect LOCA caused core damage due to debris in the sump adds no additional frequency to the core damage frequency.

For this assessment, an estimate of the frequency for such other failures was evaluated, again using the RISKMAN software fragility module (Reference 5). Seismic component failures, along with a seismic induced loss of the offsite grid, which result in a station blackout, were evaluated. Such failures are judged to lead directly to core damage even without the occurrence of an indirect large LOCA. Further, with a station blackout, the sump debris issue is moot since none of the recirculation pumps have AC power. The component seismic fragilities updated in a severe accident management alternatives (SAMA) analysis for South Texas Project (Reference 7) were examined. Combined with a seismic failure of the offsite grid, failure of any one of the six components listed below were determined to result in an extended station blackout and hence core damage. RISKMAN subtracts out the overlapping probability of failure occurrence of multiple seismic failures at the same time when calculating the conditional probability of a station blackout in a given seismic interval. The components whose failure would result in a station blackout when combined with a loss of offsite power are listed below with the weakest at the top and the highest capacity component at the bottom.

- FOTANK – Diesel Generator Fuel Oil Tanks
- INVERTER – Inverters and Battery Chargers
- ECWPUMP – Essential Cooling Water Pump Breaker Cabinets
- 125VDC - 125V DC Batteries
- 4KVSWGR – 4160v Switchgear
- EDGS – Emergency Diesel Generators

The second row in Table 1 shows the frequency of a seismic station blackout for the most recent seismic hazard curve for South Texas Project of Reference 1 summed over all 12 acceleration intervals. These frequencies are much greater than those for a seismically caused indirect LOCA. By removing the frequency of seismic station blackout events when evaluating the indirect LOCA frequencies, the frequency of indirect LOCAs that could add to the seismic core damage frequency is much reduced.

The third row of numbers in Table 1 shows the resulting indirect LOCA frequencies after removing the contribution of station blackouts. These added frequencies are in the absence of station blackouts and therefore are directly applicable to the GSI-191 sump plugging issue. The added total frequency of a seismic caused indirect LOCA is very small.

Electric Power Research Institute (EPRI) NP-6041-SL (Reference 8) provides a table (Table 2-4) of screening requirements for various equipment and subsystems. For nuclear steam supply system (NSSS) primary coolant system piping and vessels and their supports, the equipment is expected to have a five-percent-damped peak spectral acceleration HCLPF of greater than 1.2g. This translates to approximately a HCLPF in terms of peak ground acceleration of 0.5g. Notes to Table 2-4 state that evaluation is not required if the supports are designed from combined loading determined by dynamic safe shutdown earthquake and pipe break analysis. Another note states that for a pressurized water reactor (PWR), the pressurizer supports should still be evaluated.

In October of 1990, a seismic walkdown effort performed in February 1987 for the South Texas Project was documented (Reference 9) with the specific objective of examining key equipment and supports, to screen high capacity equipment, and to select the equipment and structures for which seismic failures would most likely govern the seismic risk. In the process, the walkdown was to verify component support and anchorage and identify any unique design features which would limit the components seismic capacity to a median failure capacity of less than 2.0g.

No such weaknesses were found among NSSS components. The SIS accumulators were identified for possible analysis, but later, during the fragility analysis, were found to have median acceleration capacities in excess of 6.0g.

Regarding the pressurizer at South Texas Project, it was found to be a standard Westinghouse design and support and so was screened from further analysis as having a median capacity greater than 2.0g. A median capacity of 2.0g implies a much greater HCLPF than 0.5g.

Although the seismic capacity of indirect LOCAs is expected to be more robust than a fragility curve represented by a 0.5g HCLPF, a sensitivity case assuming a fragility curve with 0.5g HCLPF, and the same beta values as in the base case above, was performed as a further bound on the risks of sump debris. The results of this sensitivity are shown in Table 2. The same methods used to create the results in Table 1 were also used to create Table 2.

The frequency of indirect LOCAs is still small. If the seismic station blackout frequency is removed and the conditional probability of a large pipe break occurs given its support fails is applied (0.05) then the frequency of an indirect Large LOCA is only 9.8E-10 per year.

Table 2. Seismic Generated Indirect LOCA Frequencies; HCLPF = 0.5g

Frequency of Seismically Generated Indirect LOCA Summed Over all Accelerations			
Fragility Group	**Conditional Probability of LOCA Given Support Failure**		
	1.0 (Upper Bound)	**0.2 (Small)**	**0.05 (Large)**
Unconditional Indirect LOCA Frequency; HCLPF = 0.5g	3.56E-08	7.12E-09	1.78E-09
Unconditional Seismic SBO Frequency	1.98E-07	1.98E-07	1.98E-07
Unconditional Indirect LOCA Frequencies with SBO Removed	1.96E-08	3.92E-09	9.81E-10

Conclusions

For both the 0.7g and 0.5g HCLPF fragility curves, the results reported in Tables 1 and 2, respectively, bound the risk from indirect LOCA caused sump debris plugging even without considering the conditional probability of sump debris plugging given that an indirect LOCA occurs. There are still other contributors to seismic core damage besides station blackout events that were not removed. This additional probability varies with the actual break size. For break sizes less than 10 inches in diameter, this additional conditional probability that sump debris leads to sump recirculation failure could be noticeably less than 1.0.

Also, this bounding analysis does not consider any correlation between the failure of RCS component boundary and that of another. If all four RCS piping loops were assumed failed in response to a seismic event, then again the effects of sump debris would be moot since failure of all four RCS loops cannot be mitigated by the emergency core cooling system systems anyway.

A summary of the seismic indirect LOCA frequencies and their potential impact on the change in core damage frequency and change in large early release frequency due to sump debris effects is provided in Table 3. The indirect small and large LOCA frequencies consider the factors of 0.2 and 0.05 presently earlier, conditional on seismic caused support failure. The frequencies for core damage have removed the contribution from station blackouts since sump debris effects have no role to play when there is no AC power for the recirculation pumps. Finally, the large early release frequencies are obtained from the bounds on added core damage frequency by multiplying by the conditional probability of a large early release given core damage at South Texas Project; i.e., 2.5E-3.

Table 3. Summary of Results

Summary of Frequencies per year	Base Case 0.7g HCLPF	Sensitivity Case 0.5g HCLPF
Seismic Caused Indirect LOCA Frequencies		
Small LOCA	1.65E-09	7.12E-09
Large LOCA	4.14E-10	1.78E-09
Bound on Added Core Damage Frequency with SBO Removed	1.82E-10	9.81E-10
Bound on Added Large Early Release Frequency with SBO Removed i.e., CDF*P(P = Large Early release given Core Damage)	4.55E-13	2.45E-12

References

1. Letter from G.T. Powell to NRC, "Seismic Hazard and Screening Report (CEUS Sites), Response NRC Request for Information Pursuant to 10CFR50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force, Review of Insights from the Fukushima Dai-ichi Accident," NOC-AE-14003114, dated March 31, 2014. (ML14099A235)

2. NUREG-1903, "Seismic Considerations for the Transition Break Size," Section 4.6, "Indirectly Induced Piping Failures," February 2008 (ADAMS Accession No. ML080880140).

3. NRC, "Probability of Pipe Failure in the Reactor Coolant Loops of Westinghouse PWR Plants," NUREG/CR-3660, UCID-19988, Vols. 1–4, U.S. Nuclear Regulatory Commission, Washington, DC, 1985a

4. NRC, "Probability of Pipe Failure in the Reactor Coolant Loops of Combustion Engineering, PWR Plants," NUREG/CR-3663, UCRL-53500, Vols. 1–3, U.S. Nuclear Regulatory Commission, Washington, D.C, 1985b.

5. ABSG Consulting Inc., "RISKMAN for Windows User Manual, V Fragility Module", Version 14.4; December 2015.

6. D. A. Brand, "Long Term Seismic Program Completion," Final Report, PG&E letter DCL-88-192 to the NRC, July 31, 1988. Docket No. 50-275, OL-DPR-80 and 50-323, OL-DPR-82.

7. Letter from D.W. Recurrel to NRC, "Supplemental Responses to Requests for Additional Information for the South Texas License Renewal Application – SAMA (TAC Nos. ME4938 and ME5122)," NOC-AE-11002772, dated January 19, 2012; ML12030A081.

8. EPRI NP-6041-SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1)," Final Report, August 1991.

9. PLG, Inc., "Seismic Analysis Walkdown Report," prepared for Houston Lighting & Power Company, transmittal letter dated October 16, 1990; ST-RL-HL-0354, for work performed in February 1987.

Attachment 2

Definitions and Acronyms

Definitions and Acronyms

ANS	American Nuclear Society
ARL	Alden Research Laboratory
ASME	American Society of Mechanical Engineers
BA	Boric Acid
BAP	Boric Acid Precipitation
BC	Branch Connection
BEP	Best Efficiency Point
B-F	Bimetallic Welds
B-J	Single Metal Welds
BWR	Boiling Water Reactor
CAD	Computer Aided Design
CASA	Containment Accident Stochastic Analysis, also a short name for the CASA Grande computer program that uses the analysis methodology
CCDF	Complementary Cumulative Distribution Function or Conditional Core Damage Frequency
CCW	Component Cooling Water
CDF	Core Damage Frequency
CET	Core Exit Thermocouple(s)
CHLE	Corrosion/Head Loss Experiments
CHRS	Containment Heat Removal System
CLB	Cold Leg Break or Current Licensing Basis
CRMP	Configuration Risk Management Program
CS	Containment Spray
CSHL	Clean Strainer Head Loss
CSS	Containment Spray System (same as CS)
CVCS	Chemical Volume Control System
DBA	Design Basis Accident
DBD	Design Basis Document
D&C	Design and Construction Defects
DEGB	Double Ended Guillotine Break
DID	Defense in Depth
DM	Degradation Mechanism
ECCS	Emergency Core Cooling System (also ECC)
ECWS	Essential Cooling Water System (also ECW)
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedure(s)
EPRI	Electric Power Research Institute
EQ	Equipment Qualification
ESF	Engineered Safety Feature
FA	Fuel Assembly(s)
FHB	Fuel Handling Building
GDC	General Design Criterion(ia)
GL	Generic Letter
GSI	Generic Safety Issue
HHSI	High Head Safety Injection (ECCS Subsystem)
HLB	Hot Leg Break
HTVL	High Temperature Vertical Loop
HLSO	Hot Leg Switchover
HVAC	Heating, Ventilation & Air Conditioning
ID	Inside Diameter
IGSCC	Intergranular Stress Corrosion Cracking
ISI	In-Service Inspection
IOZ	Inorganic Zinc
LAR	License Amendment Request
LBB	Leak Before Break
LBLOCA	Large Break Loss of Coolant Accident (also LLOCA)
LCO	Limiting Condition for Operation
LDFG	Low Density Fiberglass
LERF	Large Early Release Frequency
LHS	Latin Hypercube Sampling
LHSI	Low Head Safety Injection (ECCS Subsystem)
LOCA	Loss of Coolant Accident
LOOP/LOSP	Loss of Off Site Power
MAAP	Modular Accident Analysis Program
MAB/MEAB	Mechanical Auxiliary Building or Mechanical Electrical Auxiliary Building
MBLOCA	Medium Break Loss of Coolant Accident (also MLOCA)
NIST	National Institute of Standards and Technology

Definitions and Acronyms

NLHS	Non-uniform Latin Hypercube Sampling
NPSH	Net Positive Suction Head, (NPSHA – available, NPSHR – required)
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
OD	Outer Diameter
OQAP	Operations Quality Assurance Plan
PCI	Performance Contracting, Inc.
PCT	Peak Clad Temperature
PDF	Probability Density Function
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owner's Group
PWSCC	Primary Water Stress Corrosion Cracking
QA	Quality Assurance
QDPS	Qualified Display Processing System
RAI	Request for Additional Information
RCB	Reactor Containment Building
RCFC	Reactor Containment Fan Cooler
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RI-ISI	Risk-Informed In-Service Inspection
RMI	Reflective Metal Insulation
RMS	Records Management System
RMTS	Risk Managed Technical Specifications
RVWL(S)	Reactor Vessel Water Level (System)
RWST	Refueling Water Storage Tank
SBLOCA	Small Break Loss of Coolant Accident (also SLOCA)
SC	Stress Corrosion
SI/SIS	Safety Injection, Safety Injection System (same as ECCS)
SIR	Safety Injection and Recirculation
SR	Surveillance Requirement
SRM	Staff Requirements Memorandum
SSE	Safe Shutdown Earthquake
STP	South Texas Project
STPEGS	South Texas Project Electric Generating Station
STPNOC	STP Nuclear Operating Company
TAMU	Texas A&M University
TF	Thermal Fatigue
TGSCC	Transgranular Stress Corrosion Cracking
TS	Technical Specification(s)
TSB	Technical Specification Bases
TSC	Technical Support Center or Technical Specification Change
TSP	Trisodium Phosphate
UFSAR	Updated Final Safety Analysis Report
UNM	University of New Mexico
USI	Unresolved Safety Issue
UT	University of Texas (Austin)
V&V	Verification and Validation
VF	Vibration Fatigue
WCAP	Westinghouse Commercial Atomic Power
ZOI	Zone of Influence