Nuclear Operating Company

South Texas Project Electric Generating Station 4000 Avenue F – Suite A Bay City, Texas 77414

May 27, 2010
U7-C-STP-NRC-100119

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville MD 20852-2738

South Texas Project
Units 3 and 4
Docket Nos. 52-012 and 52-013
<u>Response to Request for Additional Information</u>

Reference:
1. Letter, Mark McBurnett to Document Control Desk, " Response to Request for Additional Information," dated July 13, 2009, U7-C-STP-NRC-090064 (ML092740559).
2. Letter, Scott Head to Document Control Desk, "Response to Request for Additional Information," dated September 15, 2009, U7-C-STP-NRC-090144 (ML092600154).
3. Letter, Scott Head to Document Control Desk, "Response to Request for Additional Information," dated January 4, 2010, U7-C-STP-NRC-100001 (ML100060691).
4. Letter, Mark McBurnett to Document Control Desk, " Response to Request for Additional Information," dated January 20, 2010, U7-C-STP-NRC-100023, (ML100250138)

Attachment 1 provides a response to NRC staff question 12.02-19 received in Request for Additional Information (RAI) letter number 441, related to Combined License Application (COLA) Part 2, Tier 2, Section 12.2. The attachment completes the response to letter 441.

This letter also supplements the responses to RAI 19-5 provided in References 1 and 2, and the response to RAI 19.01-31 provided in Reference 3. In addition, this letter revises the response to RAI 19-30 provided in Reference 4. Attachments 2, 3, and 4 address the following RAIs:

19-5, Supplemental Response 2
19.01-31, Supplemental Response
19-30, Revised Response

When a change to the COLA is indicated, it will be incorporated into the next routine revision of the COLA following NRC acceptance of the RAI response.

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STI 32682234

A summary of commitment, COM 19.9-30, which covers strategies for primary containment flooding in the emergency procedure guidelines specifically related to flooding in the lower drywell, is provided in Attachment 5.

If you have any questions regarding this submittal, please contact me at (361) 972-7136, or Bill Mookhoek at (361) 972-7274.

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

Executed on 5/27/10

Scott Head
Manager, Regulatory Affairs
South Texas Project Units 3 & 4

dws

Attachment:

1. RAI 12.02-19 Response
2. RAI 19-5, Supplemental Response 2
3. RAI 19.01-31, Supplemental Response
4. RAI 19-30, Revised Response
5. Summary of Commitment COM 19.9-30

cc: w/o attachment except*
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RAI 12.02-19

QUESTION:

The response to RAI 12.02-16 provided the basis for all values and assumptions used in the revised BWR-GALE code calculation of annual gaseous radioactive effluent releases. The response also provided the BWR-GALE code input parameters used to calculate the annual gaseous effluent releases.

The staff used the information provided by the applicant to verify the annual effluents released in the GALE86 Code Liquid Release Source Term Table, in curies per year. The independent verification indicated that all individual radionuclide liquid curie releases were in agreement except, the I-131 liquid release value of 6.50E+03 curies per year, compared to the NRC GALE code output of 6.50E-03 curies per year.

Also, the calculated Xe-133 and Xe-135 gaseous values in the GALE86 Code Gaseous Release Source Term Table, for STP 3&4 Annual Releases in MBq/yr/unit, do not agree with the indicated MBq/yr/unit values using the adjustment factors for noble gases.

The staff reviewed the four adjustments to the GALE gaseous source terms that included:

a) The conversion of curies per year to MBq per year for all source term radionuclides.
b) The increase of the gaseous effluent radionuclide source terms by a factor of 1.16 due to the assumed STP capacity factor of 0.93 versus the GALE Code assumption of 0.80.
c) An Iodine, I-131 adjustment factor based upon the I-131 concentration in the reactor water indicated in the ABWR DCD Subsection 12.2.2.1.
d) A noble gas adjustment factor based upon the noble gas release rate of 555 MBq/s indicated in ABWR DCD Subsection 12.2.2.1.

The staff requests for the applicant to provide their analysis of the following items:

1) The STP I-131 liquid effluent release value of 6.50E+03 curies per year, compared to the NRC GALE code I-131 output of 6.50E-03 curies per year.
2) The STP Xe-133 and Xe-135 gaseous effluent release quantities (MBq/yr/unit) do not agree with the values calculated by the staff using the noble gas adjustment factors provided by STP.
3) The technical justification for using the adjustment factors described in c) and d) above for the STP annual gaseous source term.

RESPONSE:

1. The liquid release source term table provided in the response to RAI 12.02-15 contains a typographical error. The I-131 entry for the GALE86 Annual Release should be 6.5E-03 Ci/yr/unit. All other entries in the table are correct. The corrected table is attached.

2. The information concerning gaseous releases in the response to RAI 12.02-16 consists of output from the GALE86 computer code that is adjusted for certain STP 3 & 4 site specific conditions and operating parameters that are different from the default values in the GALE86

computer code. A review of the information provided in the response has concluded that the some of the adjustments were calculated incorrectly. A corrected set of tables is attached. A description of the identified errors and a sample calculation are provided below.

The adjustment factors applied to the GALE86 output to determine the gaseous annual release quantities are listed in the NRC's question and summarized below.

a) Conversion of Ci/y to MBq/yr: The output of the GALE86 Code is in units of Ci/yr. These units are converted to MBq/yr to be consistent with the existing units in Table 12.2-20 of the STP 3 & 4 FSAR. The conversion factor is

$K_1 = (3.7 \times 10^{10}$ Bq/Ci x 1 MBq / 10^6 Bq) = 3.7×10^4 MBq/Ci

b) Increase due to assumed capacity factor: The assumed capacity factor for each STP ABWR unit is 0.93. Because the GALE86 computer code uses a capacity factor of 0.80 to calculate the gaseous release activities, the GALE86 output is increased by the factor

$K_2 = 0.93/0.80 = 1.1625$

c) I-131 adjustment factor: The I-131 concentration in reactor water is 0.085 MBq/kg (ABWR DCD Subsection 12.2.2.1). This parameter is not a direct input to the GALE86 Code, but is internally calculated as 1.92×10^{-3} μCi/g. To adjust the GALE86 Code iodine results to the I-131 concentration in reactor water of 0.085 MBq/kg, the gaseous release results for the iodines were multiplied by the following factor in addition to the adjustment factors calculated above.

K_3 = 0.085 MBq/kg /
(1.92×10^{-3} μCi/g x 1 Ci/10^6 μCi x 10^3 g/kg x 3.7×10^{10} Bq/Ci x 1 MBq / 10^6 Bq)
K_3 = 0.085 MBq/kg / 0.07104 MBq/kg = 1.196.

d) Noble gas adjustment factor: The ABWR DCD, Subsection 12.2.2.1, states a noble gas release rate of 555 MBq/s is used to calculate the expected releases whereas the GALE86 computer code utilizes a noble gas release rate of 1850 MBq/s. To adjust the GALE86 output to the noble gas release rate of 555 MBq/s, the noble gas release results were multiplied by the following factor

K_4 = 555 MBq/s / 1850 MBq/s = 0.3.

The results of gaseous releases in the output of GALE86 are broken down by release pathway and individual nuclides. To calculate the total adjusted release rate for a specific nuclide, the release rate for each pathway is multiplied by the appropriate correction factors and then summed to produce the total release rate. A detailed calculation for Xe-135 is presented below. The first column identifies the release pathway and the second column is the output from the GALE86 computer code. The adjustment factors for each pathway are identified in the third column. The first correction factor (K_1), which is simply a units conversion, applies to all nuclides. The correction factor for the capacity factor (K_2) applies to all pathways except for the mechanical vacuum pump. The mechanical vacuum pump is only used during startup and is therefore a function of the number of outages and not the capacity factor for the unit. Since this is a noble gas nuclide, the adjustment factor for the noble gas release rate (K_4) is also applied. For iodine nuclides, adjustment factor K_3 is used instead of K_4, and for nuclides that are not iodine or noble gases, neither K_3 nor K_4 is used. The final column is the adjusted results, which are also summarized in the attached table of detailed results. The total GALE86 results and adjusted results are included in the attached table of gaseous source terms.

Detailed Calculation for Xe-135

Pathway	GALE86 Results (μCi/yr)	Adjustment Factors	Adjusted Results (MBq/yr)
Containment Bldg	3.300E+01	K_1, K_2, K_4	4.258E+05
Turbine Bldg	3.300E+02	K_1, K_2, K_4	4.258E+06
Auxiliary Bldg	9.400E+01	K_1, K_2, K_4	1.213E+06
Radwaste Bldg	2.800E+02	K_1, K_2, K_4	3.613E+06
Gland Seal	0.000E+00	K_1, K_2, K_4	0.000E+00
Air Ejector	0.000E+00	K_1, K_2, K_4	0.000E+00
Mech Vac Pump	5.000E+02	K_1, K_4	5.550E+06
Total	1.237E+03		1.506E+07

These results differ from the results presented in the response to RAI 12.02-16 in two ways. In the previous response, the adjustment for the mechanical vacuum pump pathway was done incorrectly. The results for these nuclides have been corrected. The second change is that the total releases are now the sum of the releases from individual pathways. Since the GALE86 output contains both the individual pathways and the total, the previous RAI was based on the total releases. This introduced some inconsistencies due to round off. This change in approach results in minor changes to some of the total releases. Note that the net effect of these corrections is a slight decrease in the total ECL fraction (from 0.184 to 0.182), so the conclusions of RAI 12.02-16 are still valid.

3. The following is an expanded discussion of the technical basis for adjustments c) and d) that supplements the information provided in response to NRC question 2). For clarity, noble gas adjustment is discussed first followed by the I-131 adjustment.

   d) Noble gas adjustment factor: ABWR DCD Subsection 12.2.2.1 provides the noble gas release rate that is the basis for the annual average releases in DCD Table 12.2-20 (555 MBq/s at t=30 min). Note that this release rate is considerably less than the release rate used to determine the design basis steam concentrations in DCD Subsection 11.1.1. However, DCD Subsection 11.1.1 describes the release rate of 555 MBq/s as the expected release rate, so the average annual releases are based on the more realistic expected release rate. The value of the corresponding parameters in the GALE86 code are documented in NUREG-0016, "Calculation of Releases of Radioactive material in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-GALE Code)." Paragraph 2.2.3.2 of NUREG-0016, page 2-11, states that the noble gas source term is based on a release rate of 50,000 μCi/sec measured at 30 minutes decay. This corresponds to a release rate of 1,850 MBq/sec at 30 minutes. Since the amount of noble gas available for release from the plant is directly proportional to the release rate from the reactor, the GALE86 results are adjusted for this difference.

c) I-131 adjustment factor: ABWR DCD Subsection 12.2.2.1 also provides the I-131 parameters that are the basis for the annual average releases in DCD Table 12.2-20. These are an I-131 release rate of 3.7 MBq/s at t=0, which corresponds to a reactor coolant concentration of 0.085 MBq/kg. Note that this release rate is also considerably less than the release rate used to determine the reactor coolant concentrations in DCD Subsection 11.1.1. However, DCD Subsection 11.1.1 describes the release rate of 3.7 MBq/s as the expected release rate, so the average annual releases are based on the more realistic expected release rate. There is no discussion of the I-131 release rate in NUREG-0016 that is similar to the noble gas release rate. However, the GALE86 computer code calculates the I-131 reactor coolant concentration based on input parameters such as power level and cleanup rate. As indicated above, the concentration of I-131 calculated by GALE86 is slightly smaller than the bases described in the ABWR DCD. Since the amount of iodine available for release is directly proportional to the iodine in the reactor coolant, the GALE86 results for all iodines are adjusted by the ratio of the DCD technical basis to the GALE86 I-131 results.

GALE86 Code Liquid Release Source Term

Nuclide	GALE86 Annual Release (Ci/yr/unit)	STP 3 & 4 Annual Release (MBq/yr/unit)	STP 3 & 4 Concentration (MBq/ml)	Allowable Concentration (MBq/ml)	Fraction of Allowable Concentration
I-131	6.5E-03	3.35E+02	1.51E-13	3.70E-08	4.07E-06
I-132	2.1E-03	1.08E+02	4.87E-14	3.70E-06	1.32E-08
I-133	3.0E-02	1.54E+03	6.95E-13	2.59E-07	2.68E-06
I-134	2.8E-04	1.44E+01	6.49E-15	1.48E-05	4.39E-10
I-135	9.5E-03	4.89E+02	2.20E-13	1.11E-06	1.98E-07
H-3	7.0E+00	3.01E+05	1.36E-10	3.70E-05	3.66E-06
Na-24	4.1E-03	1.76E+02	7.94E-14	1.85E-06	4.29E-08
P-32	4.2E-04	1.81E+01	8.14E-15	3.33E-07	2.44E-08
Cr-51	1.2E-02	5.16E+02	2.32E-13	1.85E-05	1.26E-08
Mn-54	3.9E-03	1.68E+02	7.55E-14	1.11E-06	6.81E-08
Mn-56	2.1E-03	9.03E+01	4.07E-14	2.59E-06	1.57E-08
Co-58	8.2E-03	3.53E+02	1.59E-13	7.40E-07	2.15E-07
Co-60	1.5E-02	6.45E+02	2.91E-13	1.11E-07	2.62E-06
Fe-55	8.5E-03	3.66E+02	1.65E-13	3.70E-06	4.45E-08
Fe-59	2.2E-03	9.46E+01	4.26E-14	3.70E-07	1.15E-07
Ni-63	1.7E-03	7.31E+01	3.29E-14	3.70E-06	8.90E-09
Ni-65	1.0E-05	4.30E-01	1.94E-16	3.70E-06	5.24E-11
Cu-64	1.1E-02	4.73E+02	2.13E-13	7.40E-06	2.88E-08
Zn-65	2.6E-04	1.12E+01	5.04E-15	1.85E-07	2.72E-08
Zn-69m	7.6E-04	3.27E+01	1.47E-14	2.22E-06	6.63E-09
Br-83	2.3E-04	9.89E+00	4.46E-15	3.33E-05	1.34E-10
Sr-89	2.2E-04	9.46E+00	4.26E-15	2.96E-07	1.44E-08
Sr-90	2.0E-05	8.60E-01	3.87E-16	1.85E-08	2.09E-08
Sr-91	1.1E-03	4.73E+01	2.13E-14	7.40E-07	2.88E-08
Y-91	1.7E-04	7.31E+00	3.29E-15	2.96E-07	1.11E-08
Sr-92	4.5E-04	1.94E+01	8.72E-15	1.48E-06	5.89E-09
Y-92	1.3E-03	5.59E+01	2.52E-14	1.48E-06	1.70E-08
Y-93	1.1E-03	4.73E+01	2.13E-14	7.40E-07	2.88E-08
Zr-95	1.1E-03	4.73E+01	2.13E-14	7.40E-07	2.88E-08
Nb-95	1.9E-03	8.17E+01	3.68E-14	1.11E-06	3.32E-08
Mo-99	1.9E-03	8.17E+01	3.68E-14	7.40E-07	4.97E-08
Tc-99m	4.5E-03	1.94E+02	8.72E-14	3.70E-05	2.36E-09
Ru-103	3.2E-04	1.38E+01	6.20E-15	1.11E-06	5.58E-09
Ru-105	2.0E-04	8.60E+00	3.87E-15	2.59E-06	1.50E-09
Ru-106	8.9E-03	3.83E+02	1.72E-13	1.11E-07	1.55E-06
Ag-110m	1.2E-03	5.16E+01	2.32E-14	2.22E-07	1.05E-07
Te-129m	5.0E-05	2.15E+00	9.69E-16	2.59E-07	3.74E-09
Te-131m	7.0E-05	3.01E+00	1.36E-15	2.96E-07	4.58E-09
Cs-134	1.2E-02	5.16E+02	2.32E-13	3.33E-08	6.98E-06
Cs-136	7.4E-04	3.18E+01	1.43E-14	2.22E-07	6.46E-08

GALE86 Code Liquid Release Source Term

Nuclide	GALE86 Annual Release (Ci/yr/unit)	STP 3 & 4 Annual Release (MBq/yr/unit)	STP 3 & 4 Concentration (MBq/ml)	Allowable Concentration (MBq/ml)	Fraction of Allowable Concentration
Cs-137	1.8E-02	7.74E+02	3.49E-13	3.70E-08	9.42E-06
Cs-138	4.0E-05	1.72E+00	7.75E-16	1.48E-05	5.24E-11
Ba-139	1.1E-04	4.73E+00	2.13E-15	7.40E-06	2.88E-10
Ba-140	1.4E-03	6.02E+01	2.71E-14	2.96E-07	9.16E-08
Ce-141	2.7E-04	1.16E+01	5.23E-15	1.11E-06	4.71E-09
La-142	8.0E-05	3.44E+00	1.55E-15	3.70E-06	4.19E-10
Ce-143	2.0E-05	8.60E-01	3.87E-16	7.40E-07	5.24E-10
Ce-144	3.9E-03	1.68E+02	7.55E-14	1.11E-07	6.81E-07
Pr-143	5.0E-05	2.15E+00	9.69E-16	7.40E-07	1.31E-09
W-187	1.7E-04	7.31E+00	3.29E-15	1.11E-06	2.97E-09
Np-239	7.0E-03	3.01E+02	1.36E-13	7.40E-07	1.83E-07
				Total:	3.32E-05

GALE86 Code Gaseous Release Source Term

Nuclide	GALE86 Annual Release[1] (Ci/yr/unit)	STP 3 & 4 Annual Release[2] (MBq/yr/unit)	STP 3 & 4 Concentration (MBq/cm^3)	Site Wide 10CFR20 Limits (MBq/cm^3)	Fraction of Allowable Concentration
Kr-83m	0.000E+0	0.000E+0	0.00E+00	1.85E-06	0.00E+00
Kr-85m	8.900E+1	1.148E+6	2.95E-13	3.70E-09	7.97E-05
Kr-85	2.700E+2	3.484E+6	8.95E-13	2.59E-08	3.46E-05
Kr-87	6.300E+1	8.129E+5	2.09E-13	7.40E-10	2.82E-04
Kr-88	9.800E+1	1.265E+6	3.25E-13	3.33E-10	9.76E-04
Kr-89	6.110E+2	7.884E+6	2.03E-12	3.70E-11	5.47E-02
Xe-131m	1.800E+1	2.323E+5	5.97E-14	7.40E-08	8.06E-07
Xe-133m	0.000E+0	0.000E+0	0.00E+00	2.22E-08	0.00E+00
Xe-133	2.230E+3	2.643E+7	6.79E-12	1.85E-08	3.67E-04
Xe-135m	9.900E+2	1.277E+7	3.28E-12	1.48E-09	2.22E-03
Xe-135	1.237E+3	1.506E+7	3.87E-12	2.59E-09	1.49E-03
Xe-137	1.268E+3	1.636E+7	4.20E-12	3.70E-11	1.14E-01
Xe-138	1.010E+3	1.303E+7	3.35E-12	7.40E-10	4.52E-03
I-131	2.530E-1	1.239E+4	3.18E-15	7.40E-12	4.30E-04
I-133	3.290E+0	1.623E+5	4.17E-14	3.70E-11	1.13E-03
H-3	1.100E+2	4.731E+6	1.22E-12	3.70E-09	3.28E-04
C-14	9.500E+0	4.086E+5	1.05E-13	1.11E-10	9.45E-04
Ar-41	1.600E+1	6.882E+5	1.77E-13	3.70E-10	4.78E-04
Cr-51	2.701E-3	1.162E+2	2.98E-17	1.11E-09	2.69E-08
Mn-54	6.000E-3	2.581E+2	6.63E-17	3.70E-11	1.79E-06
Fe-59	7.900E-4	3.398E+1	8.73E-18	1.85E-11	4.72E-07
Co-58	1.500E-3	6.452E+1	1.66E-17	3.70E-11	4.48E-07
Co-60	1.300E-2	5.592E+2	1.44E-16	1.85E-12	7.76E-05
Zn-65	1.130E-2	4.861E+2	1.25E-16	1.48E-11	8.44E-06
Sr-89	6.050E-3	2.602E+2	6.68E-17	3.70E-11	1.81E-06
Sr-90	3.000E-5	1.290E+0	3.31E-19	2.22E-13	1.49E-06
Zr-95	1.840E-3	7.914E+1	2.03E-17	1.48E-11	1.37E-06
Nb-95	1.001E-2	4.306E+2	1.11E-16	7.40E-11	1.49E-06
Mo-99	6.800E-2	2.925E+3	7.51E-16	1.48E-10	5.08E-06
Ru-103	4.251E-3	1.828E+2	4.70E-17	3.33E-11	1.41E-06
Ag-110m	2.400E-6	1.032E-1	2.65E-20	3.70E-12	7.16E-09
Sb-124	2.200E-4	9.463E+0	2.43E-18	1.11E-11	2.19E-07
Cs-134	7.303E-3	3.141E+2	8.07E-17	7.40E-12	1.09E-05
Cs-136	6.019E-4	2.588E+1	6.65E-18	3.33E-11	2.00E-07
Cs-137	1.101E-2	4.735E+2	1.22E-16	7.40E-12	1.64E-05
Ba-140	3.202E-2	1.377E+3	3.54E-16	7.40E-11	4.78E-06
Ce-141	1.091E-2	4.691E+2	1.20E-16	3.70E-11	3.26E-06
				Total:	1.82E-01

Detailed Gaseous Release Source Term Rates (MBq/yr/unit)*

	Gaseous Release Rate (MBq/yr/unit)						
Nuclide	Containment Building	Turbine Building	Auxiliary Building	Radwaste Building	Gland Seal	Air Ejector	Mech. Vac. Pump
Kr-83m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	1.29E+04	3.23E+05	3.87E+04	0.00E+00	0.00E+00	7.74E+05	0.00E+00
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.48E+06	0.00E+00
Kr-87	0.00E+00	7.87E+05	2.58E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	1.29E+04	1.17E+06	3.87E+04	0.00E+00	0.00E+00	3.87E+04	0.00E+00
Kr-89	0.00E+00	7.48E+06	2.58E+04	3.74E+05	0.00E+00	0.00E+00	0.00E+00
Xe-131m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.32E+05	0.00E+00
Xe-133m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	3.48E+05	1.94E+06	1.07E+06	2.84E+06	0.00E+00	5.81E+06	1.44E+07
Xe-135m	1.94E+05	5.16E+06	5.81E+05	6.84E+06	0.00E+00	0.00E+00	0.00E+00
Xe-135	4.26E+05	4.26E+06	1.21E+06	3.61E+06	0.00E+00	0.00E+00	5.55E+06
Xe-137	5.81E+05	1.29E+07	1.81E+06	1.07E+06	0.00E+00	0.00E+00	0.00E+00
Xe-138	2.58E+04	1.29E+07	7.74E+04	2.58E+04	0.00E+00	0.00E+00	0.00E+00
I-131	5.66E+02	6.18E+03	1.13E+03	6.18E+02	0.00E+00	0.00E+00	3.90E+03
I-133	8.23E+03	8.75E+04	1.54E+04	8.23E+03	0.00E+00	0.00E+00	4.29E+04
H-3	2.37E+06	2.37E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.09E+05	0.00E+00
Ar-41	6.45E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.30E+04	0.00E+00
Cr-51	8.60E+00	3.87E+01	3.87E+01	3.01E+01	0.00E+00	0.00E+00	3.70E-02
Mn-54	1.72E+01	2.58E+01	4.30E+01	1.72E+02	0.00E+00	0.00E+00	0.00E+00
Co-58	4.30E+00	4.30E+01	8.60E+00	8.60E+00	0.00E+00	0.00E+00	0.00E+00
Fe-59	3.87E+00	4.30E+00	1.29E+01	1.29E+01	0.00E+00	0.00E+00	0.00E+00
Co-60	4.30E+01	4.30E+01	1.72E+02	3.01E+02	0.00E+00	0.00E+00	2.07E-02
Zn-65	4.30E+01	2.58E+02	1.72E+02	1.29E+01	0.00E+00	0.00E+00	1.26E-02
Sr-89	1.29E+00	2.58E+02	8.60E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	1.29E-01	8.60E-01	3.01E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-95	4.30E+01	2.58E-01	3.87E+02	1.72E-01	0.00E+00	0.00E+00	0.00E+00
Zr-95	1.29E+01	1.72E+00	3.01E+01	3.44E+01	0.00E+00	0.00E+00	0.00E+00
Mo-99	2.58E+02	8.60E+01	2.58E+03	1.29E-01	0.00E+00	0.00E+00	0.00E+00
Ru-103	8.60E+00	2.15E+00	1.72E+02	4.30E-02	0.00E+00	0.00E+00	0.00E+00
Ag-110M	1.72E-02	0.00E+00	8.60E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb-124	8.60E-01	4.30E+00	1.29E+00	3.01E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	3.01E+01	8.60E+00	1.72E+02	1.03E+02	0.00E+00	0.00E+00	1.18E-01
Cs-136	4.30E+00	4.30E+00	1.72E+01	0.00E+00	0.00E+00	0.00E+00	7.03E-02
Cs-137	4.30E+01	4.30E+01	2.15E+02	1.72E+02	0.00E+00	0.00E+00	3.29E-01
Ba-140	8.60E+01	4.30E+02	8.60E+02	1.72E-01	0.00E+00	0.00E+00	4.07E-01
Ce-141	8.60E+00	4.30E+02	3.01E+01	3.01E-01	0.00E+00	0.00E+00	0.00E+00

* Containment Building and Auxiliary Building terminology is used by the GALE86 Code and corresponds to the Reactor Building and Service and Control Buildings of the STP 3&4 power plants, respectively.

RAI 19-5

QUESTION:

In developing the technical basis for accident management procedures for STP 3 & 4, it will be necessary to identify departures from Revision 2 of the BWROG Accident Management Guidelines and capture the severe accident-related insights from the ABWR SSAR and the STP PRA. This is necessary to address potential changes in the emergency procedure guidelines (EPGs) and severe accident guidelines (SAGs), particularly with respect to strategies for flooding the containment by using the drywell flooder, or from the AC-independent water addition (ACIWA) system sprays. It will be necessary, for example, to avoid inadvertent operation of these features in order to assure that there is not a pool of water in the lower drywell into which molten core debris could pour during a severe accident and cause a large steam explosion. Please describe the necessary changes to the BWROG EPGs and SAGs, as applied to the STP 3 & 4 ABWRs, to ensure sound severe accident mitigation strategies and procedures.

RESPONSE SUPPLEMENT 2:

This response supplements the previous responses submitted in the letter from Mark McBurnett to Document Control Desk, "Response to Request for Additional Information," dated July 13, 2009, U7-C-STP-NRC-090064 (ML092740559), and in the letter from Scott Head to Document Control Desk, "Response to Request for Additional Information," dated September 15, 2009, U7-C-STP-NRC-090144 (ML092600154).

The lower drywell flooder (LDF) provides an alternate source of water to the lower drywell as a direct result of high temperatures in the lower drywell due to core melt and subsequent vessel failure. As stated in FSAR Subsection 9.5.12.5, operation of the LDF during severe accidents is confirmed by instrument readings in the containment including those which would record the reduction in drywell temperature and the lowering of suppression pool water level.

To address flooding in the lower drywell resulting from operation of the LDF, a supplement and new commitment will be added in FSAR Subsection 19.9.14, Accident Management, as the last two paragraphs. Changes shown in gray highlight will be included in the next COLA revision.

19.9.14 Accident Management

The following supplement addresses the lower drywell flooder (LDF) operation in the event of a severe accident scenario that involves a core melt and vessel failure.

Strategies for primary containment flooding in the emergency procedure guidelines will incorporate generic industry guidance as necessary and use existing site specific design features to the extent possible to provide indication of and address flooding in the lower drywell when the lower drywell flooder (1) does not operate, (2) does not operate as designed, (3) prematurely operates resulting in an inadvertent pool of water in the lower drywell, and (4) operates as designed during a severe accident scenario that involves a core melt and vessel failure. The procedures will be developed consistent with the plant operating procedure development plan in Section 13.5, and training on the procedures will be developed and implemented as described in Section 13.2. (COM 19.9-30)

RAI 19.01-31

QUESTION

The staff has reviewed the applicant's response to RAI 19-18 and 19-20 and has additional questions. The shared fire water system design departure impacts the shutdown and full power hurricane risk assessment for the site. In accordance with 10CFR Part 52.79(d)(1), the staff requests that the applicant provide:

(a) The shutdown and full power hurricane core damage frequency (CDF) and large early release frequency (LERF).

(b) (b) A description of the dominant sequences contributing to the shutdown and full power hurricane CDF and LERF estimates.

(c) The list of SSCs that were identified as risk significant for the Reliability Assurance Program with the supporting Fussell-Vesely (FV) and Risk Achievement Worth (RAW) for component basic events, human error probabilities, and common cause failures.

RESPONSE

The "Addenda to ASME/ANS RA-S–2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME/ANS RA-Sa–2009," approved February 2, 2009 (reference 1), contains screening criteria for external events other than fire and seismic events in Subsection 6-2.3. This Standard applies to an At-Power Level 1 PRA for operating nuclear power plants. An equivalent Low Power/Shutdown Standard is not yet approved; however, for the purposes of responding to this Request for Additional Information, the external event screening criteria in the published national standard are selected to provide a basis for the response provided below. In NUREG-1407 (reference 2), the NRC recommended a similar set of screening criteria for the Individual Plant Examination of External Events (IPEEE) required of all operating nuclear power plants.

In ASME/ANS RA-Sa-2009, Subsection 6-2.3, the fundamental criteria for screening external events other than fire and seismic events are as below:

"There are three fundamental screening criteria embedded in the requirements here, as follows. An event can be screened out either

(a) if it meets the criteria in the NRC's 1975 Standard Review Plan (SRP) or a later revision; or

(b) if it can be shown using a demonstrably conservative analysis that the mean value of the frequency of the design-basis hazard used in the plant design is less than $\sim 10^{-5}$/yr and that the conditional core damage probability is $<10^{-1}$, given the occurrence of the design-basis hazard event; or

(c) if it can be shown using a demonstrably conservative analysis that the CDF is $<10^{-6}$/yr."

The STP site is within the site parameters specified in the Design Control Document (DCD) for the ABWR for high winds and tornados. Therefore, the STP 3&4 design satisfies the requirements of the Standard Review Plan 3.3.1, Revision 3, which was in effect at the time of the Combined Operating License Application. Criterion (a) of ASME/ANS RA-Sa-2009 Subsection 6-2.3 is satisfied for high winds and tornados and these events are screened from the STP 3&4 PRA described in Chapter 19 of the Combined Operating License Application.

REFERENCES

1. Addenda to ASME/ANS RA-S–2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME/ANS RA-Sa–2009, February 2, 2009, American Society for Mechanical Engineers and American Nuclear Society.
2. "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," Report NUREG-1407, U.S. Nuclear Regulatory Commission (1991).

SUPPLEMENTAL RESPONSE

As a result of a public meeting with the NRC on Chapter 19 Open Items, the response provided to Request for Additional Information 19.01-31 in letter U7-C-STP-NRC-100001, dated January 4, 2010 (ML100060691) is supplemented as indicated below.

Background Information

Prior to hurricane arrival, the site implements hurricane mitigation strategies to put the units in a safe stable shutdown configuration in accordance with the Abnormal Procedure for Natural or Destructive Phenomena Guidelines, 0POP04-ZO-0002 . Starting approximately 36 hours prior to landfall (Hurricane Watch), the site starts making preparations for a controlled shutdown of all units. Preparations include topping off water supplies, fuel oil supplies, and other consumable inventories, site cleanup, staging of equipment such as portable fire equipment, fire brigade supplies, and ensuring the equipment necessary to establish and maintain safe shutdown is Operable. When a Hurricane Warning is received (landfall predicted within 24 hours) additional personnel are ensured to be available for the duration of the storm (the storm crew), monitoring of the grid status in coordination with the Transmission Distribution Service Provider (TDSP) is established, the equipment necessary to establish and maintain safe shutdown is again determined to be Operable, and predicted landfall between Corpus Christie and Galveston is confirmed. At twelve hours prior to landfall, the additional personnel move on-site for the duration of the storm. Two hurricane shutdown timelines are developed at least 8 hours prior to predicted landfall, one for wind speeds at the site greater than 73 mph and the other for greater than 96 mph. The determination of which timeline to follow is made by the Plant Manager after discussing with the TDSP the effect of taking the units offline prior to 73 mph on site. All exterior doors and hatches are verified closed or secured at least 4 hours prior to the projected arrival of winds in excess of 73 mph, and personnel move into Category I structures. At least 2 hours prior to wind speeds in excess of 73 mph (96 mph) the units are shutdown and cooled down to Mode 3. When expected time of winds in excess of 73 mph is less than 2 hours, one emergency diesel generator (EDG) in each unit is started and loaded onto its safety bus, and the bus disconnected from offsite power. If an unstable electrical grid develops or is predicted by the TDSP, the remaining diesel generators are started and loaded on their safety buses and the buses disconnected from offsite power. This is procedure is consistent with NUMARC 87-00, Rev. 1, Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors, Initiative 2, Procedures, and Section 2.11, Hurricane Preparations.

The specific shutdown requirements for STP Units 3&4 will be similar to the requirements established for STP Units 1&2 and are required as part of the abnormal procedure development described in Section 13.5.3.4.7, Abnormal Operating Procedures, and will satisfy the NUMARC 87-00 Guidelines.

The basic wind speed for Extreme Wind for STP Units 3&4 is 134 mph (3 second gust) (Response to RAI 03.03.01-1, U7-C-STP-NRC-090111, Attachment 10, Table 2.0-2, ML092430131). This design wind speed is applied to the combustion turbine generator structure (Table 1C-3; U7-CTG-M-SPEC-CTG-5002) and the 345kV switchyard (Section 8.2.1.2.1). The return period of the 3 second gust wind is one in a hundred years (Ref 2.3S.10, ASCE Standard ASCE/SEI-7-02, Minimum Design Loads for Buildings and Other Structures, Revision of ASCE 7-98, American Society of Civil Engineers (ASCE) and Structural Engineering Institute, January 2002; Ref 3.3-4, International Code Council, 2006 International Building Code). This is assumed to be the loss of offsite power initiating event frequency for this simplified screening assessment. In NUREG/CR-6890, Reevaluation of Station Blackout Risk at Nuclear Power Plants, Volume 1, December 2005, the weather related loss of offsite power (LOOP) initiating event frequency for the South Texas site is 3.84E-03/yr from Table D-1, Plant-specific LOOP frequencies for critical operation, 1997-2004.

Quantitative Screening

A quantitative screening assessment was performed as a further check on the effect of a hurricane on STP Units 3&4. Failure data for the screening assessment was taken from the emergency diesel generator analysis contained in the STP 1&2 PRA, Revision 6. The STP 1&2 EDG system failure results are slightly higher than the STP3&4 EDG only failure results, primarily due to the inclusion of other EDG support equipment (ventilation fans) and the generator output breaker in the system model. The electric power arrangement for STP 1 and 2 is similar to the arrangement for STP 3 and 4 and is summarized below.

	STP 1 and 2	STP 3 and 4
Number of emergency diesel generators/Required	3/1	3/1
Independent trains	Yes	Yes
Room Ventilation	Included in EDG model	Included as a basic event
Output Breaker	Included in EDG model	Included as a basic event
Common Cause – start, run 1st hour, run 24 hours, output breaker, ventilation	Included – MGL method for all active components	Included – Beta Factor for diesel
Modeling	RISKMAN (large fault tree, large event tree)	CAFTA (linked fault tree)

Two simple event trees were constructed and evaluated in EXCEL and are presented below. In the first case, the Combustion Turbine Generators (CTGs) for Units 3 and 4 are included in the model with a conditional failure likelihood of 0.5 given extreme high winds on the site, the second model does not credit operation of the CTGs, i.e., guaranteed to be failed. If the CTG for a Unit is successful, no further questions are asked in the event trees. If the CTG for a Unit fails, the EDG for the Unit are challenged. The STP 1 and 2 PRA EDG model used for this assessment includes the EDG failure modes start, run for the first hour, and run for 23 hours, the EDG ventilation system start and run for 24 hours, and the EDG output breaker to its Class 1E bus. One EDG is assumed to be running and loaded on its Class 1E bus in accordance with the Abnormal Procedure described above, which removes the start and run for the first hour failure modes for that EDG, and the ventilation fan start and breaker close failure modes for the same EDG. The appropriate MGL parameters were adjusted to remove one train from the start or close MGL set. No planned maintenance or testing would be in progress prior to or during a hurricane that affects the site.

In the screening assessment, other systems, such as Reactor Core Isolation Cooling (RCIC), which are available to provide core cooling in the ABWR, are not included. RCIC is designed to operate for a least eight hours after a Station Blackout, which would provide additional time to provide inventory makeup from the AC Independent Water Addition (ACIWA) function of the fire protection system or recover offsite power following a loss of the offsite grid. No recovery of the offsite grid is included in this simplified screening assessment.

The ACIWA function is included in the final calculation of core damage frequency for each Unit, as the basis for this Request for Additional Information is the departure for the shared fire water system, STP DEP 1.1-2. The dual unit core damage sequence does not credit the operation of the ACIWA.

Note, the containments for STP Units 3&4 are expected to remain in the state they were in prior to the arrival of the hurricane. If the plants are operating, the containments will remain inerted during a forced shutdown due to a hurricane in anticipation of restoring the units to operation after the hurricane has passed. If one of the Units were shutdown for refueling prior to the arrival of a hurricane, the containment would be deinerted to support refueling operations and would remain deinerted for the duration of the hurricane event.

Results of the Quantitative Screening

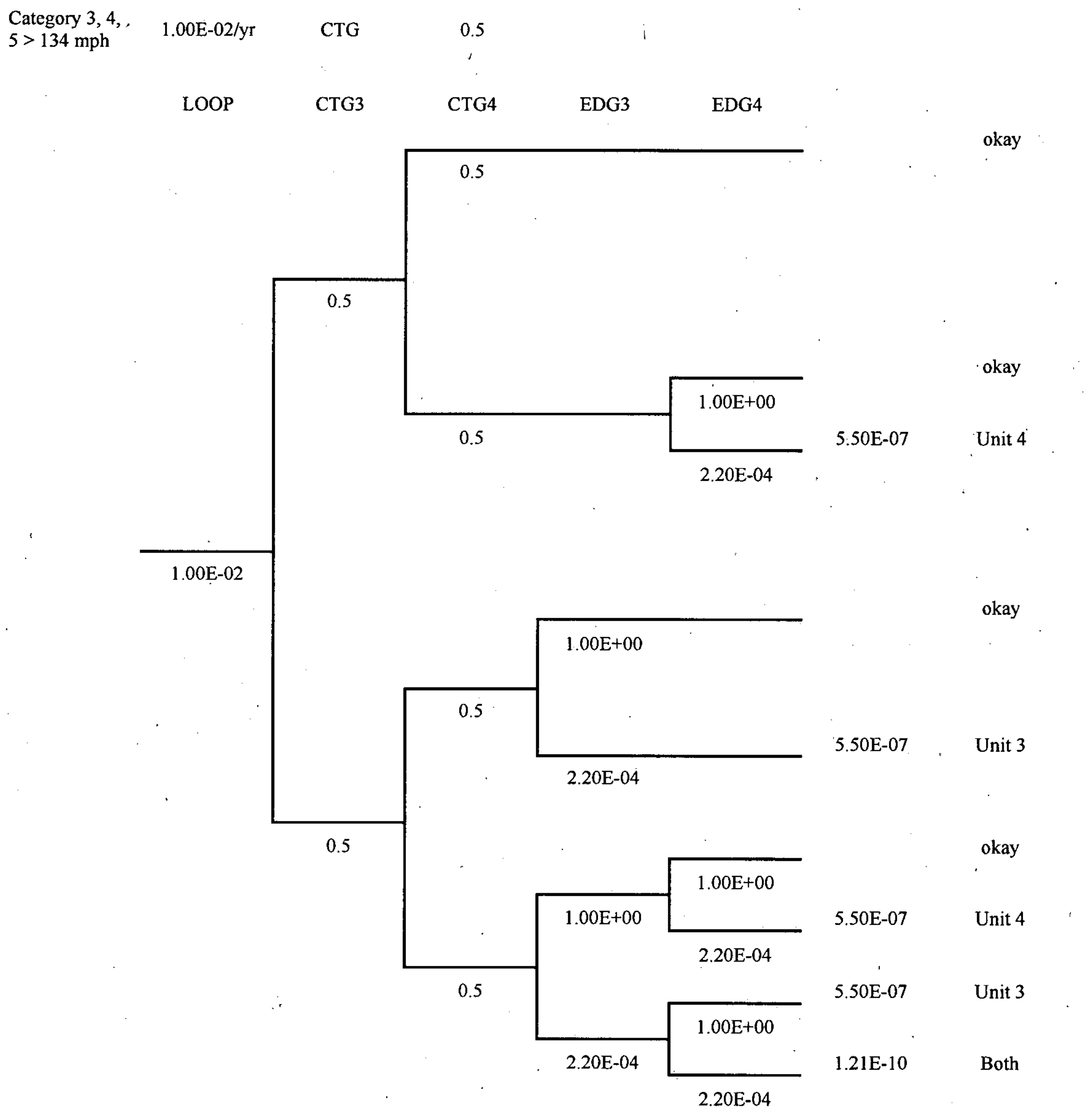
Using the simplified events trees below, the core damage frequency with credit for the ACIWA function with and without credit for the CTGs are:

	CTG = 0.5	CTG = 1.0
Unit 3	1.1E-08	2.2E-08
Unit 4	1.1E-08	2.2E-08

These results are well below the quantitative screening criteria established in ASME/ANS RA-Sa-2009, Subsection 6-2.3. No evaluation of Large Release Frequency is necessary with screening results this low.

No COLA changes are required as a result of this RAI response.

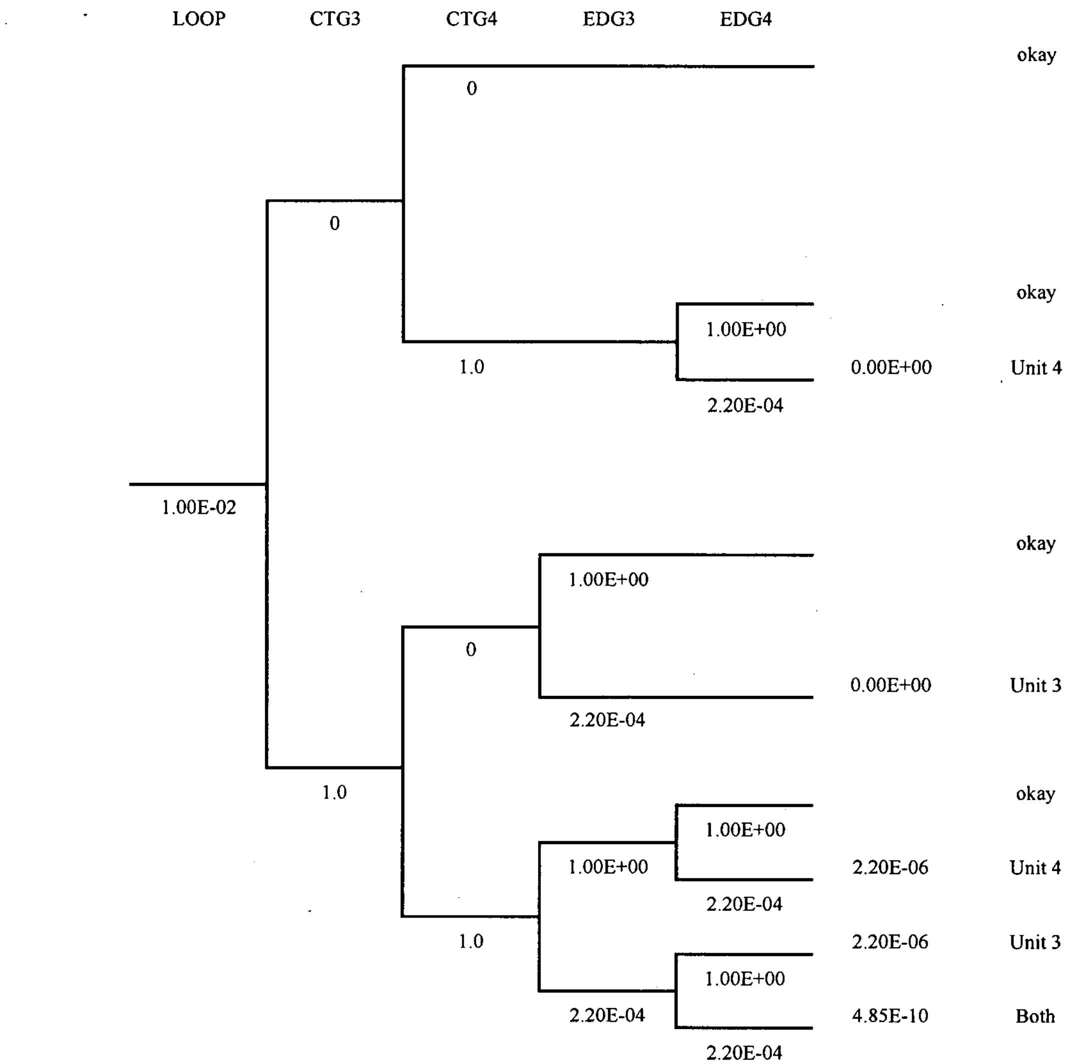
Hurricane Screening – CTG at 0.5



		*ACIWA (.01)
Unit 3	1.10E-06	1.10E-08
Unit 4	1.10E-06	1.10E-08
Both	1.21E-10	

Hurricane Screening – CTG at 1.0

Category 3, 4, 5 > 134 mph 1.00E-02/yr CTG 1.0



		*ACIWA (.01)
Unit 3	2.20E-06	2.20E-08
Unit 4	2.20E-06	2.20E-08
Both	4.85E-10	

RAI 19-30

QUESTION

At the staff audit of the South Texas Projects Unit 3 and Unit 4 PRA on September, 23, 2009, the staff reviewed the calculation, "External Flooding Event Breach of the Main Cooling Reservoir (MCR)". The calculation was dated April 20, 2009 and was referenced in the applicant's RAI response to 19.01-10 which discussed the PRA for external flooding due to MCR breach. The staff then reviewed Section 2.4S.4.1.2 of the FSAR which evaluates postulated failure of the MCR. Based on staff review of these two documents, the staff requests that the applicant address the following questions:

1. Section 2.4S.10 of the FSAR states: "All safety-related facilities in the power block are designed to be water tight at or below elevation 40.0 ft MSL. All water tight doors and hatches are normally closed under administrative controls and open outward. ... <u>An MCR embankment breach near the STP 3 & 4 power block area would not provide sufficient time for implementation of emergency operating procedures or flood warning systems.</u> As all water-tight doors and hatches are to remain in a closed position, no emergency operating procedures or plant Technical Specifications (plant shutdown), which are discussed in Subsection 2.4S.14, are required for implementation of flood protection measures." The MCR external flooding PRA analysis described in Section 19R of the FSAR is not consistent with the above statement in that under Section 19R the water tight door between the service building and the control building is normally open and takes credit for emergency operating procedures and operator action to close this water tight door during MCR breach. Please clarify this inconsistency and revise the FSAR as appropriate.

2. In STP's response to RAI 19.01-10, STP stated that the overtopping, slope protection erosion, and sliding failure modes are not applicable to the MCR design. Please justify why these failure modes are not applicable to the MCR design, and provide the basis for the reductions in dam failure frequency as a result of excluding these failure modes. In your discussion on why the MCR cannot overtop, please include the following information:

    a. The maximum pumping capacity to the MCR from the Colorado River and the maximum discharge capacity to the Colorado River.
    b. The frequency at which the MCR levels are monitored and how this information is alarmed/displayed in the control room.
    c. The procedures used to control MCR level, and the response procedures if MCR level becomes too high.

3. Section 19R.7.4.1 of the FSAR states: "A breach of the main cooling reservoir could occur suddenly or progress over many minutes." This section of the FSAR also discusses other dam breaches noting that the failure time of most breaches is 15 minutes to 1 hour, and some breaches become fully developed in as little as 6 minutes. A sudden breach of the MCR (e.g., seismic liquidfication) may not provide sufficient time for the

operator to close the water tight door between the service building and the control building (i.e., basic event OCD = 1.0). Please address the external flooding analysis due to sudden MCR breaches.

4. Please assess the impact of Category 4 and 5 hurricanes on the frequency of MCR breach. Address how a storm surge from such a hurricane would affect the MCR levee system and the exterior side of the reservoir that has no liner.

5. Please provide your data sources for dam failures that include infantile dam's failures that were used to support your reduction factor for satisfactory operation of the MCR for five years. Based on staff review of dam failures from the National Performance of Dams Program (NPDP), developed by the Department of Civil and Environmental Engineering at Stanford University, including the Taum Sauk dam failure in 2005, the inclusion of infantile dam failures would result in generic dams break frequencies greater than 1E-4 per year. In addition, it appears that the reduction you credited for satisfactory operation of the MCR seems to be double-counting. Please address these issues in your response.

6. Please justify the factor of three reduction you used, based on the assumption that the location of a breach is limited to a thousand foot section. Please explain why any thousand foot section in the 16,250 foot perimeter facing the safety related buildings can not cause a flood.

7. Please assess the impact of a MCR breach during cold shutdown and refueling if secondary and primary containment has open penetrations to facilitate maintenance. Please consider the elevations of these penetrations in your assessment.

8. Please document if the assumptions, insights, or conclusions in the referenced calculation change given the revised MCR breach evaluation in Section 2.4.4.1.2 of the FSAR.

9. The staff needs more information on the probability (basic event- OCD) of the operator failing to close the single normally open flood door between the service building and the control building. To justify the human error probability 0.1, please provide the following information:

   a. The criterion that you will supply to the guard at security house to determine if the MCR has breached.

   b. The process by which these procedures will be controlled.

   c. The potential for ambiguous visual indication on the occurrence of a MCR breach including: the occurrence of local ponding due to heavy rains and the ability of the guard to identify increased flood levels due to reduced visibility during heavy rain storms, fog, etc., particularly at night time.

d. Section 19R.7.5.1 of the FSAR states: "...a minimum available warning time from water at the South Security Gate House, approximately El. 32.0' MSL, to water at the entrances to safety-related buildings, El. 35.0' MSL. At least 30 minutes is available for operator action to close the normally open access door between the Service Building and the Control Building once water reaches the South Security Gate House." Please sufficiently justify the operator action time of at least 30 minutes.

REVISED RESPONSE

Based upon discussions and feedback provided by the NRC during a Chapter 19 Open Items meeting, the response provided to Request for Additional Information (RAI) 19-30 in U7-C-STP-NRC-100023, dated January 20, 2010, (ML100250138), is revised as described below. This revision changes the normal status from Open to Closed for the three watertight doors that provide access to the Control Building from the Service Building (2 doors) and to the Radwaste Building (1). In addition, additional information supporting the development of the Main Cooling Reservoir (MCR) breach failure frequency described in the original RAI response to item (5) is provided.

Based on the change in watertight door status, the following COLA Sections will be revised.

FSAR Section 2.4S.10 will be revised as shown below.

2.4S.10 Flooding Protection Requirements

~~An MCR embankment breach near the STP 3 & 4 power block area would not provide sufficient time for implementation of emergency operating procedures or flood warning systems. As all water-tight doors and hatches are to remain in a closed position, no emergency operating procedures or plant Technical Specifications (plant shutdown), which are discussed in Subsection 2.4S.14, are required for implementation of flood protection measures.~~

FSAR Section 2.4S.14 will be revised as shown below.

2.4S.14 Technical Specifications and Emergency Operation Requirements

Specific flood protection measures are described in Subsection 2.4S.10. To withstand the static and dynamic forces as a result of the MCR embankment breach, watertight flood protection measures and structural measures are applied to any STP 3 & 4 facilities that have an open passageway to any safety-related facility. ~~Since all~~ All watertight doors and hatches for these facilities, at or below 40.0 ft. MSL are to remain in a closed position under administrative control. ~~, no emergency operating procedures or plant technical specifications (plant shutdown) are required for implementation of flood protection measures.~~ The emergency procedures for MCR breach described in Section 19.9.3 will be developed consistent with the plant operating procedure development plan in Section 13.5.

FSAR Appendix 19R will be revised as shown below.

19R.6.4 Operator Actions

(4) ~~Ensure~~ Verify that the ~~*Close*~~ *watertight door at* the *entrance to the control room area* and the two watertight doors at the entrances to the Reactor Building Access Corridor ~~is~~ are closed *if floods in the turbine building result in service building flooding.*

19R.7 External Flooding Evaluation

Summarized in the sections below is the external flooding PRA analyses for the STP 3 & 4 plants. External flooding is defined as intrusion of water from sources outside of plant buildings such that the ability of the plant to achieve safe shutdown is affected. The analysis determined the potential core damage frequency (CDF) that could result from external flooding events for each of the new units and was developed assuming that the watertight doors providing normal access to the main control room and the two watertight doors in the Reactor Building Access Corridor are ~~is~~ open. This assumption provides a conservative and bounding assessment of risk from external flooding because the administrative controls will require the doors to be closed except when in actual use.

19R.7.4.1 Main Cooling Reservoir Breach

Note that this analysis is developed assuming that the watertight doors providing normal access to the main control room and the two watertight doors in the Reactor Building Access Corridor are ~~is~~ open. This assumption provides a conservative and bounding assessment of risk from external flooding.

~~With the exception of the normally open access door to the control building from the service building,~~ All external access points to the control and reactor buildings are provided with normally-closed, watertight barriers or doors designed to withstand the maximum loadings of any potential main cooling reservoir breach.

The normal access to the main control ~~building~~ room is via the service building through a watertight door on the 2950 mm elevation (elevation 35.0). In addition, there are two access doors to the Reactor Building Access Corridor in the Control Building, one from the Service Building and one from the Radwaste Building (elevation 18' 6 ½"). As discussed above, this analysis assumes that ~~this~~ these doors are ~~door is~~ open even though administrative controls will require that the doors be closed except when in actual use. The doors are ~~door is~~ oriented such that water external to the control building will seal the door. In addition, there are other normally-closed watertight doors that provide access to the control building from the service building and that are located either at or below grade. Since the service building is not designed to withstand flooding, it is assumed that a main cooling reservoir breach would result in water entering the service building. If any one of the doors from the service building to the control building is not closed or fails, then water could enter the control building and cause failure of all three divisions of reactor cooling water (RCW) or DC power since these are located below grade. Since there are no internal watertight barriers to protect the rooms below grade in the control building, it is conservatively assumed that failure of one of the watertight doors on the control building would result in core damage.

When notified of a main cooling reservoir breach by security personnel, the operators in the main control room staff would ~~ensure~~ verify that the normally-~~open~~closed watertight control room access door and the two watertight doors in the Reactor Building Access Corridor are ~~is~~ closed. Closing these doors ~~this door~~ prevents water from entering the control building. As discussed above, failure ~~to close~~ of these doors ~~this door~~ would result in submerging the control building and is conservatively assumed to result in core damage.

If the door to the main control room and the doors to the Reactor Building Access Corridor are ~~is~~ closed, then the event progresses as a loss of offsite power since it is assumed that the MCR breach causes a loss of offsite power.

19R.7.5.1 Main Cooling Reservoir Breach Accident

OCD - Operator Action To Close Control Room Watertight Access Door or RB/CB External Doors Fail

This top event represents failure of the watertight doors to prevent flood waters from entering either the control building or the reactor building. Failure of this top event can occur from two causes. First, even though the watertight doors will normally be closed, it is assumed for the purposes of this analysis that the doors are open. The ~~the~~ operators ~~can~~ could fail to close the normally ~~open~~ closed, watertight door that provides main control room access from the service building and the two watertight doors that provide access to the Reactor Building Access Corridor. As described in section above, security personnel are stationed such that they will have a clear view of the area between the main cooling reservoir and plant buildings. This analysis assumes that the security staff is trained and that procedures are in place for them to alert the control room if there are indications of a breach of the main cooling reservoir. Procedures are also assumed to be in place to direct that the main control room access door and the Reactor Building Access Corridor doors be verified to be closed immediately on notification of a potential external flooding event (Refer to Section 19.9.3). Furthermore, the analysis assumes that the area between the main cooling reservoir and plant buildings is lighted to an extent that any flow of water from a breach of the main cooling reservoir would be clearly visible to the security personnel at night.

The main cooling reservoir breach analysis described in Section 2.4S.4 was used to develop a minimum available warning time from water at the South Security Gate House, approximately El. 32.0' MSL, to water at the entrances to safety-related buildings, El. 35.0' MSL. At least 30 minutes is available for operator action to verify closed the normally closed ~~open~~ access doors between the Service Building and the Control Building and the Reactor Building Access Corridor doors once water reaches the South Security Gate House. Once the security staff notifies the control room of the breach, to verify the closing and securing of the watertight doors takes less than five minutes ~~one minute~~. Therefore, it is assumed that a moderate and adequate amount of time is available to effect the actions to verify closed the ~~control room~~ access doors. Then the failure probability for this event was assigned using the values in the Standard Safety Analysis Report (SSAR) Table 19R-4.

Even if operator action to verify the closure of ~~close~~ the normally-~~open~~ closed doors is successful, failure of any one of the watertight doors that allow access to the reactor building or control building could randomly fail. Using the values in the SSAR Table 19R-4, the probability of random door failures that allow water to enter either the control building or the reactor building was calculated.

19R.7.7 Operator Actions Related to External Flooding

One operator action is important to external flooding risk. This action, timely verification of closure of the watertight doors at the entrance to the main control room and the two doors in the Reactor Building Access Corridor is similar to the event included in section 19R.6.4. However, the cues to initiate the action for the external flooding event ~~is~~ are different than for internal flooding.

19R.7.9 Conclusions

The conclusions from the ABWR probabilistic external flooding analysis are that the risk from external flooding is acceptably low, even with the assumption that the normally-closed watertight ~~normal~~ access door to the control room and the two watertight access doors to the Reactor Building Access Corridor are ~~is~~ open and that operator action is required to close the doors. It is also concluded that the incremental risk from external flooding events is within the goals for an increase in CDF or LERF.

FSAR Appendix 19K will be revised as shown below.

19K.10 Identification of Important Capabilities Outside the Control Room

The identified activities outside the control room are:

(8) Verifying the closure of ~~Closing~~ the normally-~~open~~ closed watertight door to the control room and the two normally closed watertight doors on the Reactor Building Access Corridor on notification of a main cooling reservoir breach.

FSAR Section 19.4 will be revised as shown below.

19.4.5 ABWR Probabilistic Flooding Analysis

Failure of any watertight door to prevent water from entering the control building was assumed to result in core damage because all three essential DC divisions and the main control room are located below grade and there are no internal watertight barriers that would prevent water that enters the control building from failing all three DC divisions or the main control room. For a breach of the main cooling reservoir, timely operator action is required to verify the closure of ~~close~~ the normally-~~open~~ closed main control room access door and the two access doors to the Reactor Building Access Corridor.

FSAR Section 19.8 will be revised as shown below.

19.8.5.3 Features Selected

Operator Check Watertight Doors are Dogged

The flooding analysis assumes that all watertight doors except the normally-~~open~~ closed main control room access door and the two normally closed access doors to the Reactor Building Access Corridor, *are closed and dogged to prevent floods from propagating from one area to another* or from outside to the inside.

View of the Main Cooling Reservoir

Plant buildings are located such that security personnel will have a clear and unobstructed view of the main cooling reservoir. Having such a view allows for prompt notification of the main control room so that the normally-~~open~~ closed watertight door to the main control room and the two normally closed access doors to the Reactor Building Access Corridor can be verified closed before failure of the main cooling reservoir could be expected to threaten the plant. The area between the plant and the main cooling reservoir is lighted so that clear views are provided at night.

Operator Actions to Ensure Integrity Against External Floods

In addition to having unobstructed views of the main cooling reservoir, security personnel will be trained to alert the main control room immediately to any indication of main cooling reservoir failure. On such notification, ~~personnel in the main control room~~ operators will ~~ensure~~ immediately verify that the access door to the main control room and the two access doors to the Reactor Building Access Corridor are ~~is~~ closed ~~immediately~~. Also, all external doors located below the maximum flood level will be closed and verified on notification of any upstream dam failures. The emergency procedures for Severe External Flooding ensure that watertight barriers are in place and external opening sandbagged prior to the arrival on site of high water levels from external flooding (COM 19.9-3).

FSAR Section 19.9 will be revised as shown below.

19.9.3 Event Specific Procedures for Severe External Flooding

(1) Procedures and training will be developed to ensure that observation of the main cooling reservoir is conducted such that main control room personnel will be alerted on indications of a main cooling reservoir breach. These procedures will also direct that the main control room access door and the two access doors on the Reactor Building Access Corridor will be verified closed immediately on receipt of such notification.

FSAR Section 19.11 will be revised as shown below.

19.11 Human Action Overview

A new human action is modeled by the STP 3 & 4 external flooding analysis (Appendix 19R) to verify the closure of ~~close~~ the control room watertight access door and the two watertight access doors to the Reactor Building Access Corridor in the event of an external flood. These doors are normally closed but are assumed open for the MCR breach flooding assessment. This action has been found to be important and meets the provisions identified in Subsection 19D.7 for important human actions and critical tasks. In addition, Subsection 19.9.3 documents the actions to be completed to ensure the human action's reliability.

The response to Item 5 of RAI 19-30 is supplemented with the following information.

5. The dam failure information was developed to support the Individual Plant Examination for External Events (IPEEE) performed for STP Units 1 and 2 and transmitted to the NRC under STP letter ST-AE-HL-93526, August 31, 1993. As described in Section

3.4.6.5 of the IPEEE, the primary data sources are "Baecher, G. B., M. E. Pate, and R. de Neufuille, "Risk of Data Failure in Benefit-Cost Analysis, Water Resources Research," Vol. 16, No. 3, Pg. 449-456, June 1980," and "Von Thun, J. L., Bureau of Reclamation, Engineering and Research Center, "Application of Statistical Data from Dam Failures and Accidents to Risk-Based Decision Analysis on Existing Dams," October 1985." The base failure rate developed for the IPEEE included all dam failures and noted that approximately one-half of dam failures occur during the first five years after initial fill. A 50% reduction in failure rate, is appropriate based upon this information and the successful operation of the MCR for 25+ years.

As a check on the initial failure rate development, a search was performed to find new information on failure rates for embankment dams. The U. S. Army Corps of Engineers (USACE) developed the Engineering and Design Reliability Analysis and Risk Assessment for Seepage and Slope Stability Failure Modes for Embankment Dams, ETL 1110-2-561, published 31 January 2006, that provides a screening methodology for seepage and slope failure for embankment dams using the latest dam failure data from the "Analysis of Embankment Dam Incidents," UNIGIV Report No. R-374; School of Civil and Environmental Engineering. The University of New South Wales, Sydney 2052, Australia, September 1998, ISBN: 85841 3493. Foster, M. A., Fell, R., and Spannagle, M. Appendix H, Historical Frequency of Occurrence Assessment of Embankment Failure due to Piping, presents the data and a screening method for determining dam failure frequency due to piping failure.

The screening method requires the determination of a base failure frequency for a particular dam type given the operating age of the dam and for three different piping failure modes, piping through the embankment, piping through the foundation, and piping from the embankment into the foundation. The three failure modes are qualitatively evaluated based on specific engineering details associated with dam engineering and construction. The final failure rate for each piping failure mode is determined by multiplying the engineering and construction adjustment factors together with the base failure rate for the failure mode to determine the final failure rate. The failure rate for the MCR breach is the sum of the rates for the individual failure modes. The four tables from ETL 1110-2-561 Appendix H are recreated below.

Table H-1 **ETL 1110-2-561**

(Table 11.1 UNSW): Average Probability of Failure of Embankment Dams by Mode of Piping and Dam Zoning.

ZONING CATEGORY	EMBANKMENT			FOUNDATION			EMBANKMENT INTO FOUNDATION		
		Average Annual P_e (x10-6)			Average Annual P_f (x10-6)			Average Annual P_{ef} (x10-6)	
	Average P_{Te} (x10-3)	First 5 Years Operation	After 5 Years Operation	Average P_{Tf} (x10-3)	First 5 Years Operation	After 5 Years Operation	Average P_{Tef} (x10-3)	First 5 Years Operation	After 5 Years Operation
Homogenous earthfill	16	2080	190						
Earthfill with filter	1.5	190	37						
Earthfill with rock toe	8.9	1160	160						
Zoned earthfill	1.2	160	25						
Zoned earth and rockfill	1.2	150	24						
Central core earth and rockfill	(<1.1)	(<140)	(<34)	1.7	255	19	0.18	19	4
Concrete face earthfill	5.3	690	75						
Concrete face rockfill	(<1)	(<130)	(<17)						
Puddle core earthfill	9.3	1200	38						
Earthfill with corewall	(<1)	(<130)	(<8)						
Rockfill with corewall	(<1)	(<130)	(<13)						
Hydraulic fill	(<1)	(<130)	(<5)						
ALL DAMS	3.5	450	56	1.7	255	19	0.18	19	4

Notes: (1) P_{Te} , P_{Tf} , and P_{Tef} are the average probabilities of failure over the life of the dam.
(2) P_e , P_f and P_{ef} are the average annual probabilities of failure.

Ref: Foster, Fell, & Spanagle 1998

STP Main Cooling Reservoir – Earthfill with filter

First Five Years = (190 + 255 + 19) x 1E-06 = 4.6E-04

Late (>5 yrs) = (37 + 19 + 4) x 1E-06 = 6.0E-05

Table H-2 **ETL 1110-2-561**

(Table 11.2 UNSW): Summary of the Weighting Factors for Piping Through the Embankment Mode of Failure.

FACTOR	GENERAL FACTORS INFLUENCING LIKELIHOOD OF FAILURE				
	MUCH MORE LIKELY	MORE LIKELY	NEUTRAL	LESS LIKELY	MUCH LESS LIKELY
ZONING	Refer to Table 11.1 for the average annual probabilities of failure by piping through the embankment depending on zoning type				
EMBANKMENT FILTERS $W_{E(filt)}$		No embankment filter (for dams which usually have filters (refer to text) (2)	Other dam types (1)	Embankment filter present - poor quality (0.2)	Embankment filter present - well designed and constructed (0.02)
CORE GEOLOGICAL ORIGIN $W_{E(cgo)}$	Alluvial (1.5)	Aeolian, Colluvial (1.25)	Residual, Lacustrine, Marine, Volcanic (1.0)		Glacial (0.5)
CORE SOIL TYPE $W_{E(cst)}$	Dispersive clays (5) Low plasticity silts (ML) (2.5) Poorly and well graded sands (SP, SW) (2)	Clayey and silty sands (SC, SM) (1.2)	Well graded and poorly graded gravels (GW, GP) (1.0) High plasticity silts (MH) (1.0)	Clayey and silty gravels (GC< GM) (0.8) Low plasticity clays (CL) (0.8)	High plasticity clays (CH) (0.3)
COMPACTION $W_{E(cc)}$	No formal compaction (5)	Rolled, modest control (1.2)	Puddle, Hydraulic fill (1.0)		Rolled, good control (0.5)
CONDUITS $W_{E (con)}$	Conduit through the embankment – many poor details (5)	Conduit through the embankment - some poor details (2)	Conduit through embankment - typical USBR practice (1.0)	Conduit through embankment - including downstream filters (0.8)	No conduit through the embankment (0.5)
FOUNDATION TREATMENT $W_{E(FT)}$	Untreated vertical faces or overhangs in core foundation (2)	Irregularities in foundation or abutment, Steep abutments (1.2)		Careful slope modification by cutting, filling with concrete (0.9)	
OBSERVATIONS OF SEEPAGE $W_{E(obs)}$	Muddy leakage Sudden increases in leakage (Up to 10)	Leakage gradually increasing, clear, Sinkholes, Seepage emerging on downstream slope (2)	Leakage steady, clear or not observed (1.0)	Minor leakage (0.7)	Leakage measures none or very small (0.5)
MONITORING AND SURVEILLANCE $W_{E(mon)}$	Inspections annually (2)	Inspections monthly (1.2)	Irregular seepage observations, inspections weekly (1.0)	Weekly - monthly seepage monitoring, weekly inspections (0.8)	Daily monitoring of seepage, daily inspections (0.5)

Ref : Foster, Fell, & Spanagle 1998.

Weighting Total – Upper: 0.02 * 1.5 * 0.3 * 0.5 * 0.5 * 1 * 0.7 * 0.8 = 0.00126
Weighting Total – Expected: 0.02 * 1.5 * 0.3 * 0.5 * 0.5 * 1 * 0.5 * 0.5 = 0.0005625

Table H-3 **ETL 1110-2-561**

(Table 11.3 UNSW): Summary of Weighting Factors for Piping Through the Foundation Mode of Failure.

FACTOR	GENERAL FACTORS INFLUENCING LIKELIHOOD OF FAILURE				
	MUCH MORE LIKELY	MORE LIKELY	NEUTRAL	LESS LIKELY	MUCH LESS LIKELY
ZONING	Refer to Table 11.1 for the average annual probabilities of failure by piping through the embankment depending on zoning type				
FILTERS $W_{F(filt)}$		No foundation filter present when required (1.2)	No foundation filter (1.0)	Foundation filter(s) present (0.8)	
FOUNDATION TYPE (below cutoff) $W_{F(fnd)}$	Soil foundation (5)		Rock – clay infilled or open fractures and/or erodible rock substance (1.0)	Better rock quality	Rock - closed fractures and non-erodible substance (0.05)
CUTOFF TYPE (Soil foundation) $W_{F(cts)}$ OR CUTOFF TYPE (Rockfill foundation) $W_{F(ctr)}$	Sheetpile wall Poorly constructed diaphragm wall (3)	Shallow or no cutoff trench (1.2) Well constructed diaphragm wall (1.5)	Partially penetrating sheetpile wall or poorly constructed slurry trench wall (1.0) Average cutoff trench (1.0)	Upstream blanket, Partially penetrating well constructed slurry trench wall (0.8) Well constructed cutoff trench (0.9)	Partially penetrating deep cutoff trench (0.7)
SOIL GEOLOGY TYPES (below cutoff) $W_{F(sg)}$ OR ROCK GEOLOGY TYPES (below cutoff) $W_{F(rg)}$	Dispersive soils (5) Volcanic ash (5) Limestone (5) Dolomite (3) Saline (gypsum) (5) Basalt (3)	Residual (1.2) Tuff (1.5) Rhyolite (2) Marble (2) Quartzite (2)	Aeolian, Colluvial, Lacustrine, Marine (1.0)	Alluvial (0.9) Sandstone, Shale, Siltstone, Claystone, Mudstone, Hornfels (0.7) Agglomerate, Volc. Breccia (0.8)	Glacial (0.5) Conglomerate (0.5) Andesite, Gabbro (0.5) Granite, Gneiss (0.2) Schist, Phyllite, Slate (0.5)
OBSERVATIONS OF SEEPAGE $W_{F(obs)}$ OR OBSERVATIONS OF PORE PRESSURES $W_{F(obp)}$	Muddy leakage, Sudden increases in leakage (up to 10) Sudden increases in pressures (up to 10)	Leakage gradually increasing, clear, Sinkholes, Sandboils (2) Gradually increasing pressures in foundation (2)	Leakage steady, clear or not observed (1.0) High pressures measured in foundation (1.0)	Minor leakage (0.7)	Leakage measures none or very small (0.5) Low pore pressures in foundation (0.8)
MONITORING AND SURVEILLANCE $W_{F(mon)}$	Inspections annually (2)	Inspections monthly (1.2)	Irregular seepage observations, inspections weekly (1.0)	Weekly - monthly seepage monitoring, weekly inspections (0.8)	Daily monitoring of seepage, daily inspections (0.5)

Ref : Foster, Fell, & Spanagle 1998.

Weighting Total – Upper: 0.8 * 5 * 1.0 * 0.9 * 0.7 * 0.8 = 2.016
Weighting Total – Expected: 0.8 * 5 * 1.5 * 0.9 * 0.5 * 0.5 = 1.35

Table H-4 **ETL 1110-2-561**

(Table 11.4 UNSW): Summary of Weighting Factors for Piping From the Embankment into the Foundation – Accidents and Failures

FACTOR	GENERAL FACTORS INFLUENCING LIKELIHOOD OF FAILURE				
	MUCH MORE LIKELY	MORE LIKELY	NEUTRAL	LESS LIKELY	MUCH LESS LIKELY
ZONING	Refer to Table 11.1 for the average annual probabilities of failure by piping through the embankment depending on zoning type				
FILTERS $W_{EF(filt)}$	Appears to be independent of presence/absence of embankment or foundation filters (1.0)				
FOUNDATION CUTOFF TRENCH $W_{EF(cot)}$	Deep and narrow cutoff trench (1.5)		Average cutoff trench width and depth (1.0)	Shallow or no cutoff trench (0.8)	
FOUNDATION TYPE $W_{EF(fnd)}$		Founding on or partly on rock foundations (1.5)			Founding on or partly on soil foundations (0.5)
EROSION CONTROL MEASURES OF CORE FOUNDATION $W_{EF(ecm)}$	No erosion control measures, open jointed bedrock or open work gravels (up to 5)	No erosion control measures, average foundation conditions (1.2)	No erosion control measures, good foundation conditions (1.0)	Erosion control measures present, poor foundations (0.5)	Good to very good erosion control measures present and good foundation (0.3 - 0.1)
GROUTING OF FOUNDATIONS $W_{EF(gr)}$		No grouting on rock foundations (1.3)	Soil foundation only - not applicable (1.0)	Rock foundations grouted (0.8)	
SOIL GEOLOGY TYPES $W_{EF(sg)}$, OR ROCK GEOLOGY TYPES $W_{EF(rg)}$	Colluvial (5) Sandstone interbedded with shale or limestone (3) Limestone, gypsum (2.5)	Glacial (2) Dolomite, Tuff, Quartzite (1.5) Rhyolite, Basalt, Marble (1.2)	Agglomerate, Volcanic breccia Granite, Andesite, Gabbro, Gneiss (1.0)	Residual (0.8) Sandstone, Conglomerate (0.8) Schist, Phyllite, Slate, Hornfels (0.6)	Alluvial, Aeolian, Lacustrine, Marine, Volcanic (0.5) Shale, Siltstone, Mudstone, Claystone (0.2)
CORE GEOLOGICAL ORIGIN $W_{EF(cgo)}$	Alluvial (1.5)	Acolian Colluvial (1.25)	Residual, Laucustrine, Marine, Volcanic (1.0)		Glacial (0.5)
CORE SOIL TYPE $W_{EF(cst)}$	Dispersive Clays (5) Low plasticity silts (ML) (2.5) Porrly and well graded sands (SP, SW) (2)	Clayey and silty sands (SC, SM) (1.2)	Well graded and poorly graded gravels (CW, CP) (1.0) High plasticity silts (MH) (1.0)	Clayey and silty gravels (GC, GM) (0.8) Low plasticity clays (CL (0.8)	High plasticity clays (CH) (0.3)
CORE COMPACTION $W_{EF(CC)}$	Appears to be independent of compaction - all compaction types (1.0)				
FOUNDATION TREATMENT $W_{EF(ft)}$	Untreated vertical faces ior overhangs in core foundation (1.5)	Irregularities in foundation or abutment, Steep abutments (1.1)		Careful slope modification by cutting, filling with concrete (0.9)	
OBSERVATIONS OF SEEPAGE $W_{EF(obs)}$	Muddy leakage, Sudden increases in leakage (up to 10)	Leakage gradually increasing, clear, Sinkholes (2)	Leakage steady, clear or not observed (1.0)	Minor leakage (0.7)	Leakage measured none or very small (0.5)
MONITORING AND SURVEILLANCE $W_{EF(mon)}$	Inspections annually (2)	Inspections monthly (1.2)	Irregular seepage observations, inspections weekly (1.0)	Weekly - monthly seepage monitoring, weekly inspections (0.8)	Daily monitoring of seepage, daily inspections (0.5)

Ref : Foster, Fell, & Spanagle 1998.

Weighting Total – Upper: 1.0 * 1.0 * 0.5 * 1.0 * 1.0 * 0.5 * 1.5 * 0.3 * 1.0 * 0.7 * 0.8 = 0.063

Weighting Total – Expected: 1.0 * 0.8 * 0.5 * 0.3 * 1.0 * 0.5 * 1.5 * 0.3 * 1.0 * 0.5 * 0.5 = 0.00675

Adjusted MCR Failure Rate

STP Main Cooling Reservoir – Earthfill with filter, Upper bound

Late (>5 yrs) = [(37 x 1.26E-03) + (19 x 2.016) + (4 x 6.3E-02)] x 1E-06 = 3.9E-05/yr

Weighting Total – Upper: 0.02 * 1.5 * 0.3 * 0.5 * 0.5 * 1 * 0.7 * 0.8 = 0.00126
Weighting Total – Upper: 0.8 * 5 * 1.0 * 0.9 * 0.7 * 0.8 = 2.016
Weighting Total – Upper: 1.0 * 1.0 * 0.5 * 1.0 * 1.0 * 0.5 * 1.5 * 0.3 * 1.0 * 0.7 * 0.8 = 0.063

STP Main Cooling Reservoir – Earthfill with filter, Expected

Late (>5 yrs) = [(37 x 5.625E-04) + (19 x 1.35) + (4 x 6.75E-03)] x 1E-06 = 2.6E-05/yr

Weighting Total – Expected: 0.02 * 1.5 * 0.3 * 0.5 * 0.5 * 1 * 0.5 * 0.5 = 0.0005625
Weighting Total – Expected: 0.8 * 5 * 1.5 * 0.9 * 0.5 * 0.5 = 1.35
Weighting Total – Expected: 1.0 * 0.8 * 0.5 * 0.3 * 1.0 * 0.5 * 1.5 * 0.3 * 1.0 * 0.5 * 0.5 = 0.00675

Using the geometric reduction factor presented in the original response, 1000ft/16250 ft, the expected MCR breach frequency for the section of the MCR that faces Units 3 & 4 is approximately 1.6E-06 per year. This is consistent with the value used in the PRA screening assessment originally performed for the MCR breach at Units 3 & 4.

Conservative Assumptions in the Main Cooling Reservoir Breach Analysis

The PRA screening assessment includes several conservative assumptions from the design basis MCR breach calculation that overestimate the effects of a MCR breach.

1. The starting water level in the MCR considered for the breach analysis was 50.9 feet. This level corresponds to the response of the MCR to one-half probable maximum precipitation (PMP) on the normal maximum operating level plus the effect of wind set-up produced by the 2-year wind speed (50 mph) from the south (Reference 2.4S.4-7).The maximum operating water level of the MCR is 49 feet with all four units operating. Reservoir overflow from the discharge structure starts at 49.5 feet.

2. Discharge flow. The design basis MCR breach flow is approximately 130,000 cfs. The maximum outflow from historical failures for a hydraulic depth of 25 feet is approximately 20,000 cfs. The maximum outflow is related to the MCR design basis breach width. The design breach width is the highest determined from several breach development models.

3. The design basis breach time to complete failure is approximately 1.7 hours with a breach erosion rate of 112 ft/hr on both sides of the breach. Active monitoring systems would detect breach flow prior to water reaching the south access point. No credit is assumed in the PRA model for detection of a breach prior to water level reaching the south access point at el. 32 ft.

4. The calculated maximum flood height for the design basis MCR breach, 38.8 ft., is on the south face of the Ultimate Heat Sink. Water level at the entrance to the control building from either breach scenario is approximately 37.5 ft. while the flood level at the north end of the turbine building is approximately 36 ft.

STP 3&4 is actively pursuing a contract with dam safety engineers to develop a PRA screening assessment based on current methodologies in dam safety.

SUMMARY OF COMMITMENT COM 19.9-30

Commitment	Description	Completion Date
COM 19.9-30 CR 10-11778 Action 1	Strategies for primary containment flooding in the emergency procedure guidelines will incorporate generic industry guidance as necessary and use existing site specific design features to the extent possible to provide indication of and address flooding in the lower drywell when the lower drywell flooder (1) does not operate, (2) does not operate as designed, (3) prematurely operates resulting in an inadvertent pool of water in the lower drywell, and (4) operates as designed during a severe accident scenario that involves a core melt and vessel failure. The procedures will be developed consistent with the plant operating procedure development plan in section 13.5, and training on the procedures will be developed and implemented as described in section 13.2.	Prior to fuel load