Nuclear Operating Company

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

September 3, 2009
U7-C-STP-NRC-090125

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
Response to Requests for Additional Information

Attached are responses to NRC staff questions included in Request for Additional Information (RAI) letter numbers 188, 194, 195, 197, 198, 199, 393, and 394, related to Combined License Application (COLA) Part 2, Tier 2, Sections 11.2, 11.3, 11.4, 11.5, 12.03, and 12.04. This letter completes responses to the letters listed. Attachments 1 through 9 contain responses to the RAI questions listed below:

11.02-4	11.03-4	11-05
11.03-2	11.03-5	11.05-1
11.03-3	11.04-4	12.03-12.04-9

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

There are no commitments in this letter.

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

STI 32527444 D091 NRO

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

Executed on 9/3/09

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

scs
Attachments:

1. Question 11.02-4
2. Question 11.03-2
3. Question 11.03-3
4. Question 11.03-4
5. Question 11.03-5
6. Question 11.04-4
7. Question 11-05
8. Question 11.05-1
9. Question 12.03-12.04-9

cc: w/o attachment except*
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RAI 11.02-4

QUESTION:

Please explain how the application demonstrates that the site can meet the general environmental radiation standard in 40 CFR Part 190 (per 10 CFR 20.1301(e)), and provide sufficient information for the staff to evaluate the bases and assumptions used in the applicant's analysis. Please incorporate this analysis into the FSAR or justify its exclusion.

RESPONSE:

A description of the bases and assumptions used in the calculation of the radiological effects of routine releases from STP 3 & 4 is provided in COLA Part 3 (ER), Section 3.5. Demonstration of compliance with 40 CFR Part 190 requires the consideration of the effects of the entire fuel cycle on the annual doses. The only additional fuel cycle facilities that affect the general public in the vicinity of STP 3 & 4 are the existing nuclear units, STP 1 & 2. To document compliance with 40 CFR 190, the COLA will be changed as follows.

12.2.3 COL License Information

12.2.3.1 Compliance with 10 CFR 20 and 10 CFR 50 Appendix I

The following supplement addresses COL License Information Item 12.5.

Using site-specific parameters, the average annual liquid releases and the average annual airborne releases to the environment have been computed and are shown in Tables 12.2-20 through 12.2-23. The average annual liquid and airborne releases are in compliance with 10 CFR 20 and 10 CFR 50 Appendix I.

The following is a summary of the annual dose contributions to the maximally exposed individual (MEI) from STP 1 & 2 and STP 3 & 4. The MEI is a child at residences in the vicinity of STP. The contributions from STP 1 & 2 are from the "2005 Radiological Effluent Release Report" (April 27, 2006). The total doses are in compliance with 40 CFR 190.

		Dose (mrem/year)		
		Total Body	Thyroid	Other Organ - Bone
STP 3 & 4	Direct Radiation	5.0	NA	NA
	Liquid	0.00025	0.00011	0.0023
	Gaseous	0.70	4.54	1.94
	Total	5.70	4.54	1.94
STP 1 & 2	Liquid	0.0042	0.0041	0.00077
	Gaseous	0.0080	0.0097	0.0011
	Total	0.012	0.014	0.0019
	Site Total	5.71	4.55	1.94
	40 CFR 190 Limit	25	75	25
	Fraction of Limit	0.23	0.06	0.08

RAI 11.03-2

QUESTION:

FSAR Section 15.7 Radioactive Release from Subsystems and Components has a sub section 15.7.1.1 Basis and Assumptions that states; "Therefore, inadvertant operator action with bypass of the delay charcoal beds is analyzed for compliance to ESTB 11-5."

Effluent Systems Treatment Branch (ESTB) no longer exists and ESTB-11 has been revised and now exists as SRP BTP 11-5, version 3.

a) What version of SRP BTP 11-5 was used to used to analyze for compliance with this section?

b) What source term was utilized for the determination of the "radioactive flow" through the offgas system.

How were the radioactivity levels in excess of environmental limits determined, "which are defined by 10CFR20as not greater than 2 x 10 -2 m Sv/h at the site boundary."? Pease provide the details, such as source term, flow estimate and X/Q dispersion as to how he radioactivity levels are determined.

RESPONSE:

STP Units 3&4 did not take a departure from Subsection 15.7.1.1 of the certified design. The DCD Subsection 15.7.1.1 analyses are based on inadvertent bypass of the downstream charcoal delay beds. Therefore the DCD Subsection 15.7.1.1 event is identical to the COLA Subsection 15.7.1.1 event. NUREG-1503 Subsection 11.3.2 provides the NRC's determination that the Offgas System is acceptable regarding the Subsection 15.7.1.1 accident.

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

RAI 11.03-3

QUESTION:

In Technical Rationale Section 5 of the Standard Review Plan (SRP) Section 11.3, "Gaseous Waste Management System," it states, "Compliance with GDC 60 requires that design provisions be included in the nuclear power unit to control releases of radioactive materials in gaseous effluents to the environment during normal reactor operation, including anticipated operational occurrences."

STD DEP 10.4-5, Section 11.3.3.3, "Process Facility," proposes to remove the phrase, "If any of these conditions cannot be met with reactor condensate, the coolant should be supplied by a closed cooling water system of reliability and quality equal to that of reactor condensate." The staff notes that Section 11.3.3.3, "Process Facility," changes the coolant for the offgas condensers from reactor condensate to turbine building cooling water (TCW). In the FSAR, provide additional discussion on how the TCW provides equivalent reliability and quality as reactor condensate as a suitable coolant for the offgas condensers, and provide discussion on how the offgas condensers are cooled in the event TCW is not available.

RESPONSE:

The current COLA provides the necessary information to support the change as provided in STD DEP 10.4-5. The detailed explanation related to reliability, quality and availability follows.

1. <u>Comparison of Turbine Cooling Water (TCW) System and Condensate System reliability:</u>

   The COLA provides the following design features:

   - COLA Tier 2 Subsection 10.4.5 Circulating Water System (CWS) shows that the Condensate System is cooled by the circulating water system.

   - COLA Tier 2 Subsection 10.4.5.1.1 Safety Design Basis states that the CWS does not serve or support any safety function and has no safety design basis.

   - COLA Tier 2 Subsection 10.4.7.1.1.Safety Design Basis states that the condensate-feedwater system does not serve or support any safety function and has no safety-design basis.

   - COLA Tier 2 Subsection 10.4.7.3 Evaluation states that the Condensate and Feedwater System does not serve or support any safety function. Systems analyses show that failure of this system cannot compromise any safety-related system or prevent safe shutdown.

   - COLA Tier 2 Subsection 9.2.14.1.1 Safety Design Basis states that the Turbine Building Cooling Water (TCW) System serves no safety function and has no safety design basis.

- COLA Tier 2 Table 9.2-11 Turbine Island Auxiliary Equipment shows that the TCW System removes heat from components including the:
  - Condensate pump motor coolers
  - Reactor feed pump and auxiliary coolers
  - Standby reactor feed pump motor coolers

Conclusion:

The above information shows that the Condensate System, like the TCW System, has no safety function or safety design basis. Therefore, they are similarly reliable from the standpoint that both systems have no safety design basis. Additional indication of the reliability of the TCW System is provided in that the TCW System provides condensate pump motor, reactor feed pump and standby reactor feed pump motor cooler heat removal. Therefore, logically, the reliability of the condensate pump (which is the motive force for offgas cooling in the DCD), reactor feed pump and standby reactor feed pump is directly related to the reliability of the TCW System, which is the pumps' collective heat removal mechanism. Therefore, the TCW System is at least as reliable as the Condensate System in its operation to cool the offgas system. The Condensate System is used in the DCD to cool the offgas system. Therefore, the TCW System that is used to cool the offgas system in the COLA can be expected to be at least as reliable as the Condensate System that is used to cool the offgas system in the DCD.

2. Comparison of TCW System and Condensate System quality:

   DCD Tier 2 Subsection 11.3.3.3 Process Facility (1) states that the temperature of the condensate supplied to the offgas condenser should not exceed 56.6°C during periods of normal operation.

   COLA Tier 2 Subsection 9.2.14.1.2 Power Generation Design Basis states:

   - The TCW System provides corrosion-inhibited, demineralized cooling water.
   - During power operation, the TCW System operates to provide a continuous supply of cooling water, at a maximum temperature of 41°C to the Turbine Island auxiliary equipment with a service water inlet temperature not exceeding 37.8°C. The main condenser, at its design temperature condition, provides ~49°C condensate cooled by 37.8°C circulating water. The main condenser and TCW share the same heat sink. Therefore, TSW is preferable as coolant of the offgas condenser.
   - COLA Tier 2 Subsection 9.2.14.2.1 and COLA Tier 2 Subsection 9.2.16.2.3.1 describe three 50% capacity TCW pumps, heat exchangers and TSW pumps. Therefore, a single failure of this equipment does not reduce the capacity of the system.

Conclusion:

Condensate is demineralized prior to cooling the offgas system as described in the DCD. TCW is demineralized and corrosion-inhibited prior to cooling the offgas system as proposed in the STP COLA. The TCW heat exchangers and main condenser share the same heat sink. When the heat sink water temperature is 37.8°C, the TCW heat exchangers supply 41°C cooling water to equipment while the main condenser supplies ~49°C condensate. In addition, the TCW and TSW systems have a standby pump and heat exchanger. Therefore, it is evident that the TCW is a higher quality and cooler water than the condensate water for cooling the offgas system.

3. Discussion on how the offgas condensers are cooled in the event TCW is not available:

   The COLA provides the following design features:

   - COLA Tier 2 Table 9.2-11 Turbine Island Auxiliary Equipment shows that the TCW System removes heat from components including the:

     - Generator stator coolers, hydrogen coolers, and generator gas dryer coolers
     - Turbine lube coolers
     - Reactor feed pump and auxiliary coolers
     - Standby reactor feed pump motor coolers
     - Condensate pump motor coolers

   COLA Tier 2 Subsection 10.2.2.6 Turbine-Generator Supervisory Instruments shows that turbine supervisory instrumentation monitors:

   - Oil system pressures, levels and temperatures
   - Bearing metal and oil drain temperatures
   - Hydrogen temperature, pressure, and purity
   - Stator coolant temperature and conductivity
   - Stator-winding temperature

   COLA Tier 2 Subsection 10.2.2.5 Turbine Protection System shows that the main stop and control valves and the intermediate stop and intercept valves close to shut down the turbine on:

   - Loss of stator coolant

COLA Tier 2 Subsection 7.2.1.1.4.2 Initiating Circuits shows that the RPS will initiate a reactor scram when any one or more of the following conditions occur or exist within the plant:

- Turbine Stop Valve Closed

COLA Tier 2 Subsection 10.2.4 Evaluation shows that the turbine-generator is not nuclear safety-related and is not needed to effect or support a safe shutdown of the reactor.

Conclusion:

The above features show that TCW cools the generator stator coolers, among other components. The turbine supervisory instrumentation monitors stator coolant temperature and conductivity. The turbine stop valves close on loss of stator coolant (TCW) which trips the turbine. RPS initiates a reactor scram on turbine stop valve closure which stops the production of hydrogen and oxygen in the reactor and eliminates the production of main steam for the motive force of the steam jet air ejectors. Therefore, the hydrogen, oxygen and steam motive force feeding the offgas system is eliminated on loss of TCW that effectively shuts down the offgas system. This effect is similar to the loss of condensate eventually causing a reactor scram and the subsequent removal of hydrogen, oxygen and steam motive force feeding the offgas system which effectively shuts it down when using the DCD described Condensate System for the offgas coolant. Therefore, regarding the effect on the offgas system, the loss of TCW is similar to the loss of condensate water.

As detailed above, the current COLA provides sufficient information to support the change made by STD DEP 10.4-5.

No COLA change is required as a result of this RAI response.

RAI 11.03-4

QUESTION:

In Standard Review Plan (SRP) Section 11.3, "Gaseous Waste Management System," Section 4 of SRP Acceptance Criteria, it states, "System designs should describe features that will minimize, to the extent practicable, contamination of the facility and environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste in accordance with Regulatory Guide 1.143, for gaseous wastes produced during normal operation and anticipated operational occurrences, and the requirements of 10 CFR 20.1406 or the DC application, update in the SAR, or the COL application to the extent not addressed in a referenced certified design."

• STD DEP 10.4-5, Table 11.3-3, "Equipment Malfunction Analysis," removes equipment from consideration and only includes the preheater, recombiner, and condenser. In the FSAR, provide justification or clarification on why the other equipment analyzed in Table 11.3-3 of the DCD are no longer included in Table 11.3-3, "Equipment Malfunction Analysis," of the FSAR.

• STD DEP 10.4-5 of the FSAR, Figures 11.3-1, "Offgas System PFD," and 11.3-2 "Offgas System P&ID," make substantial changes from the same figures given in the DCD. For example, Figure 11.3-2, "Offgas System P&ID," of the DCD, note 6 states, "After any valve closes due to high radiation signal, it shall remain closed until reset by manual switch." Note 6 is given with reference to the last valve before the final vent of the Gaseous Waste Management System (GWMS).

• In the FSAR, provide justification and clarification of the omission of these notes, including but not limited to note 6, annunciator alarms, piping specifications, and other design information that was provided in the DCD.

RESPONSE:

COLA Table 11.3-3, Equipment Malfunction Analysis, depicts changes to the names of some major offgas equipment. Specific name changes are from Recombiner preheater to Preheater; from Recombiner catalyst to Recombiner; and from Recombiner condenser to Condenser. The naming and malfunction analysis of other equipment in Table 11.3-3 were not changed.

Figures 11.3-1, Offgas System PFD, and 11.3-2, Offgas System P&ID, have been completely replaced in the COLA to correctly depict the redesigned STP 3 & 4 Offgas System. The DCD Offgas System PFD, Offgas System P&ID, and their respective notations, therefore, are not applicable to the new Offgas System design and have been removed in their entirety. New Offgas System PFD and Offgas System P&ID have been added. Specifically, Note 6 in DCD Figure 11.3-2 is not applicable in the COLA design because other design improvements automatically perform a similar function.

Part 2 Tier 2 of the COLA will add a new subsection 11.3.2.1 Offgas System Compliance With Part 20. that will address that the Offgas System complies with 10 CFR 20.1406. Changes to COLA R2 are shown in gray highlighting.

11.3.2.1 Offgas System Compliance With Part 20.1406

The Offgas System meets the requirements of 10 CFR 20.1406. The Offgas System design will minimize, to the extent practicable, contamination of the facility and the environment by removing the radioactive isotopes of noble gases and gaseous iodines drawn from the condenser and allowing them to decay to acceptable levels before the offgas stream discharges the decayed radionuclides to the environment via the plant stack. The Offgas System includes discharge monitoring that will automatically isolate the Offgas System from the environment in the unlikely event an unacceptable level of activity is detected in the Offgas System effluent. The Offgas System will be operated using approved plant procedures.

The Offgas System is designed to minimize the generation of radioactive waste in that the charcoal in the carbon beds is not intended to be replaced during the entire plant life. To minimize carbon usage and to greatly reduce the potential for the Offgas System to spread contamination, the radiolysis-generated hydrogen and oxygen removed from the condenser by the Offgas System is recombined in a controlled manner in piping designed to withstand hydrogen and oxygen detonation. The resulting offgas is cooled and partially condensed to reduce the total influent mass and increase the relative radionuclide concentration that the carbon delay beds process. This increases the efficiency of the carbon delay beds and, therefore, reduces the mass of charcoal adsorbent. In addition, the cooler gas and resultant overall lower gas flow rate produces conditions for radionuclides to be more efficiently retained on the carbon in the delay beds. Additionally, a guard bed is positioned at the front end of the series of charcoal delay beds and is designed to bear the brunt of potential offgas operating events so that the follow on beds are minimally affected during the event

The initial loading of carbon is designed to adsorb the radionuclide gases present in the offgas stream without requiring periodic replacement. This design provides an objective that, during the operating life of the plant, carbon in the Offgas System delay beds should not contribute to or be processed into solid radwaste. The design objective can provide, at the end of the plant life, Offgas System decommissioning that primarily entails the removal of carbon adsorbent from the delay beds and processing it as dry solid radwaste.

RAI 11.03-5

QUESTION:

ESTB 11-5 is being utilized as a reference for Section 15.7.1.1. ESTB 11-5 is not a current version document. SRP BTP 11-5 Revision 3 is the most current.

Based upon the changes included in the current version of BTP11-5, are there any calculations in this Section 15.7.1.1 that may need changed?

RESPONSE:

Refer to the response to RAI 11.03-2.

No COLA changes will result from this RAI.

RAI 11.04-4

QUESTION:

STP 3 and 4 COL FSAR Section 11.4.3, Plant Specific Solid Radwaste Information, Item (2), states "The wet waste solidification process and the spent resin and sludge dewatering process will result in products that comply with 10 CFR 61.56 for STP 3 and 4 as provided in Radioactive Waste Process Control Program (PCP). The PCP utilized by Units 1 and 2 is provided with the COL application and the latest revision will be provided as per the schedule in Table 13.4S-1."

FSAR Section 11.4.3, Item (3) states, "Establishment and implementation of a process control program (PCP) for the dewatering processing of the spent resins and filter sludges for STP 3 and 4 is provided in Radioactive Waste Process Control Program (PCP). The PCP utilized by Units 1 and 2 is provided with the COL application and the latest revision will be provided as per the schedule in Table 13.4S-1."

Therefore, the PCP will be common to all four units on site. In keeping with the policy of utilizing a site PCP, please answer the following:

1) Verify that STP has reviewed Nuclear Energy Institute (NEI) document NEI 07-10A, "Generic FSAR Template Guidance for Process Control Program (PCP)" for applicability and possible incorporation into the STP 3 & 4 COL.

2) NEI 07-10A "identifies the administrative and operational controls for waste processing, process parameters, and surveillance requirements which assure that the final waste product meets the requirements of applicable Federal, State and Disposal Site waste form requirements for burial at a 10 CFR 61 licensed Low Level Waste (LLW) disposal site." If STP has reviewed NEI 07-10A and determined that it will not be incorporated into the FSAR, modify all applicable FSAR Sections to fully describe all elements of the PCP program, or justify an alternative. Otherwise, reference NEI 07-10A in the STP FSAR.

RESPONSE:

NEI 07-10A will be incorporated in the PCP that is applicable to STP 3 & 4. The COLA will be revised as follows.

11.4.3 Plant-Specific Solid Radwaste Information

(1) STP 3 & 4 do not utilize an incinerator system.

(2) The wet waste solidification process and the spent resin and sludge dewatering process will result in products that comply with 10 CFR 61.56 for

STP 3 & 4 as provided in Radioactive Waste Process Control Program (PCP). The PCP utilized by Units 1 & 2 is provided with the COL application and the latest revision will be provided as per the schedule in Table 13.4S-1. The PCP will incorporate the guidance from NEI 07-10A, "Generic FSAR Template Guidance for Process Control Programs (PCP)."

(3) Establishment and implementation of a process control program (PCP) for the dewatering processing of the spent resins and filter sludges for STP 3 & 4 is provided in Radioactive Waste Process Control Program (PCP). The PCP utilized by Units 1 & 2 is provided with the COL application and the latest revision will be provided as per the schedule in Table 13.4S-1. The PCP will incorporate the guidance from NEI 07-10A, "Generic FSAR Template Guidance for Process Control Programs (PCP)."

RAI 11-05

QUESTION:

FSAR section 11.5.5.2, Calibration, states that "Calibration can also be performed on the applicable instrument by using liquid or gaseous radionuclide standards or by analyzing particulate iodine or gaseous grab samples with laboratory instruments."

RG 1.21 states, "Calibrations of measuring equipment should be performed using reference standards certified by the National Bureau of Standards or standards that have been calibrated against standards certified by the National Bureau of Standards." "Periodic inservice calibrations should also be performed to relate monitor "readings" to the concentrations and/or release rates of radioactive material in the monitored release path.

Distinction should be made in section 11.5.5.2 to delineate a calibration of a radioactive effluent process monitor versus a periodic inservice calibration.

Also, Regulatory Guide 4.15 describes the calibration process for radioactive effluent process monitors in Section 7.1 which could be utilized as guidance for calibrations of these monitors.

Please expand or revise this section 11.5.5.2 to describe the calibration process for radioactive effluent process monitors.

RESPONSE:

To provide distinction between the calibration process for radioactive effluent process monitors versus periodic inservice calibration, COLA Part 2 Tier 2, Subsection 11.5.5.2 will be revised to change the last sentence of the first paragraph to specifically discuss inservice calibrations. Changes to COLA Rev. 2 are shown below in gray highlighting.

11.5.5.2 Calibration

Calibration of radiation monitors is performed using certified commercial radionuclide sources traceable to the National Institute of Standards and Technology. ~~The overall reproducibility of calibration is limited to ±15%. The source detector geometry during primary calibration will be mechanically precise enough to ensure that positioning errors of either instruments or radiation source do not affect the calibration accuracy by more than ±3%~~ Calibrations are performed in accordance with manufacturers' requirements and controlled by approved plant procedures. *Each continuous monitor is calibrated during plant ~~operation~~* shutdown *or during the refueling outage if the detector is not ~~readily~~ accessible* during power operation. *Calibration can also be performed on the applicable instrument by using liquid or gaseous radionuclide standards ~~or by~~.* Periodic inservice calibrations are performed through appropriate methods including *analyzing particulate iodine or gaseous grab samples with laboratory instruments.*

RAI 11.05-1

QUESTION:

STP 3 and 4 COL FSAR Section 11.5.7S, Additional Information, states "An offsite dose calculation manual (ODCM) for STP 1 and 2 has been reviewed and approved by the NRC. It contains descriptions of the methodology and parameters used for calculation of offsite doses resulting from gaseous and liquid effluents. It also describes how liquid and gaseous effluent release rates are derived and parameters used in setting instrumentation alarm setpoints to control or terminate effluent releases. The ODCM also contains the radiological environmental monitoring program which samples and analyzes radiation and radionuclides in the environs of the existing plant, using local land use census data in identifying all potential radiation exposure pathways associated with radioactive materials present in liquid and gaseous effluents and direct external radiation from the plant. The ODCM for STP 3 and 4 will be integrated into the 1 and 2 ODCM, taking into account the appropriate differences between the existing and new units."

Therefore, the ODCM will be common to all four units. In keeping with that policy of utilizing a site ODCM, please answer the following:

1. Verify that STP has reviewed Nuclear Energy Institute (NEI) document NEI 07-09A, "Generic FSAR Template Guidance for the Offsite Dose Calculation Manual (ODCM) Program Description" for applicability and possible incorporation into the STP 3 and 4 COL.

2. NEI 07-09A "provides a complete generic program description for use in developing construction and operating license (COL) applications. The document reflects contemporary Nuclear Regulatory Commission (NRC) guidance, including Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants," and industry-NRC discussions regarding the applicable standard review plan section. A main objective of this program description is to assist in expediting NRC review and issuance of the combined license." If STP has reviewed NEI 07-09A and determined that it will not be incorporated into the FSAR, then the applicant should modify all applicable FSAR Sections to fully describe all elements of the ODCM program, or justify an alternative. Otherwise, the applicant should reference NEI 07-09A in the STP FSAR and provide any additional supplemental or site-specific information as needed.

RESPONSE:

As discussed in COLA Part 2 Tier 2, Subsection 11.5.7S, and acknowledged in this RAI, the intent is to maintain a site ODCM that integrates the STP 3 & 4 units into the existing approved ODCM for the licensed and operating STP 1 & 2 units. The STP 1 & 2 ODCM was provided for NRC review in COLA Rev 0. STPNOC has reviewed NEI 07-09A for applicability. Where possible, the ODCM template will be in alignment with NEI 07.09A. That alignment will be accomplished by the due date listed in COLA Part 2 Tier 2, Table 13.4S-1.

COLA Part 2 Tier 2, Section 11.5.7S will be revised to add a sentence to the end of the first paragraph as shown below. Changes to COLA R2 are shown in gray highlighting.

11.5.7S Additional Information

An offsite dose calculation manual (ODCM) for STP 1 & 2 has been reviewed and approved by the NRC. It contains descriptions of the methodology and parameters used for calculation of offsite doses resulting from gaseous and liquid effluents. It also describes how liquid and gaseous effluents release rates are derived and parameters used in setting instrumentation alarm setpoints to control or terminate effluent releases. The ODCM also contains the radiological environmental monitoring program which samples and analyzes radiation and radionuclides in the environs of the existing plant, using local land use census data in identifying all potential radiation exposure pathways associated with radioactive materials present in liquid and gaseous effluents and direct external radiation from the plant. The ODCM for STP 3 & 4 will be integrated into the 1 & 2 ODCM, taking into account the appropriate differences between the existing and new units. Where possible, the ODCM will align with NEI 07.09A (Revision 0), "Generic FSAR Template Guidance for Offsite Dose Calculation Manual (ODCM) Program Description," March 2009.

RAI 12.03-12.04-9

QUESTION:

RG 1.206, Section C.I.12.3.5 requests that an applicant should provide estimated annual dose to construction workers in a new unit construction area, as a result of radiation from onsite sources from the existing operating plant(s). The applicant should provide bases, models, assumptions, and input data.

STP 3 and 4 COL FSAR Section 12.3.8 provides a description of the potential sources of exposure to construction workers and comparison to the limits in 10 CFR 20.1301 and 40 CFR 190.10 for members of the public. However, the COL FSAR does not contain the bases, models and assumptions used to calculate construction worker dose. The staff needs additional information to make a determination of reasonable assurance. Please provide the following bases information:

1. Specific construction activities and the number of workers used in construction worker dose calculations.
2. Bases for the values used to calculate dose from direct radiation, gaseous effluents, and liquid effluents.
3. Calculated annual person-Sievert dose for each unit.

RESPONSE:

The following information is provided in response to this RAI request. Additional details are available in the STP 3&4 COLA Part 3, Sections 3.9S, 3.10S, and 4.5, and in the response to ER RAI 5.4.2-1.

1. STPNOC anticipates the construction activities will include the following, listed in expected sequence:

   - Planning and exploration activities, including such site activities as soil boring/sampling and monitoring wells or additional geophysical borings as allowed by 10 CFR 50.10(a)(2) and the removal and/or relocation of existing facilities in the new plant footprint.
   - Site preparation activities, including installation of temporary facilities, construction support facilities, service facilities, utilities, docking and unloading facilities, excavations for facility structures and foundations, the installation of a slurry wall around the entire excavation, and construction of structures, systems and components (SSCs) that do not constitute construction activities as defined by 10 CFR 50.10(a)(1).
   - Subsurface preparation, placement of backfill and concrete within an excavation, and installation of foundations prior to the issuance of the COL.
   - Major power plant construction activities under the COL.

The construction workforce would consist of two components: (1) Field Craft Labor; and (2) Field Nonmanual Labor. Field craft labor is the largest component of the construction workforce, with approximately 79% of the field workforce makeup in conventional Advanced Boiling Water Reactor (ABWR) nuclear plant construction. The field craft labor force comprises civil, electrical, mechanical, piping, and instrumentation personnel employed during the installation and startup of STP 3 & 4. The field non-manual labor makes up the balance of the construction workforce, consisting of approximately 21%. The non-manual labor force comprises field management, field supervision, field engineers, quality assurance/quality control (QA/QC), environmental, safety & health, and administrative/clerical staff.

The total onsite construction workforce for sequential construction of two units at the STP site is estimated to be approximately 20 jobhours per kilowatt of generating capacity. The schedule assumes 12 months for site preparation, 12 months for additional pre-COL site activities, and 45 months from COL issuance to Unit 3 fuel load, and 9 months for startup. Unit 4 fuel load is scheduled 12 months after Unit 3 for a total schedule duration of 90 months. Based on this schedule, the peak onsite construction workforce for Units 3 & 4 is estimated to be 5,950 people, and will be at this peak for 10 months during construction activities for both units. The average maximum workforce for one year is estimated to be 3,174 people during Unit 3 only construction, 5,929 people during Unit 3 and 4 construction, and 1,885 people during Unit 4 construction when Unit 3 is in operation.

2. Dose rates at the construction site are estimated based on dose rate measurements and calculations. Although the construction workers will occupy a large area over the course of the construction period, dose rates are estimated based on average distances from radiation sources.

   - Direct radiation: The direct radiation dose rates from STP 1 & 2 sources are based on TLD measurements taken at various onsite locations from 2002 through 2006. This 5-year period provides sufficient data to be representative of plant conditions. Since the construction location for STP 3 & 4 is farther away from STP 1 & 2 than are the respective TLD stations where dose rates are measured from each source, the STP 1 & 2 Offsite Dose Calculation Manual (ODCM) is used to extrapolate the dose rates from the TLD locations to the STP 3&4 location. In determining direct radiation dose rates, it is assumed that the worker is located in the center of the construction area of the unit (either STP 3 or 4) nearest to the source. Given that workers will move about the construction area over the course of a year, it is reasonable to select the center of the area as a representative location for occupancy. No credit is taken for any shielding provided by structures under construction. The estimated dose rate to Units 3 and 4 construction workers due to operation of Units 1 and 2 is 2.4 mrem/yr. The estimated dose rate to Unit 4 construction workers due to operation of Units 1, 2 and 3 is 7.9 mrem/yr.

- Gaseous effluents: The annual dose rates to the maximally exposed member of the public at the site boundary or the nearest residence due to the release of gaseous effluents is based on the STP 1 & 2 REMPs for 2002 to 2006. The composite maximum annual dose rate for each organ over these 5 years was calculated using the methodology found in the STP 1 & 2 ODCM. These offsite dose rates are used to estimate construction doses. Using the atmospheric dispersion factors in FSAR Section 2.3, the estimated total effective dose (TEDE) rate to construction workers from operation of Units 1&2 is 1.9 mrem/yr and operation of Units 1,2 and 3 is 8.9 mrem/yr.

- Liquid effluents: The annual dose rates to the maximally exposed member of the public at the site boundary or the nearest residence due to the release of liquid effluents is based on the STP 1 & 2 REMPs for 2002 to 2006. The composite maximum annual dose rate for each organ over these 5 years was calculated using the methodology found in the STP 1 & 2 ODCM. The offsite dose rates from STP 1, 2, and 3 are calculated at the Little Robbins Slough area due to sport fish ingestion and shoreline exposure. These dose rates are used to estimate construction location doses. The estimated total effective dose (TEDE) rate to construction workers from operation of Units 1,2 and 3 is 0.042 mrem/yr.

3. The calculated annual person-Sievert doses are provided in the table below. Note that the manpower estimates for the timeframe when construction on both Units 3 and 4 is in progress are provided for both units, as it is not feasible to break the workforce estimates by unit. The estimated doses for each of the three construction phases shown in the table are based on the maximum average annual workforce during that phase.

Maximum Annual TEDE (Person-Sieverts)			
	Unit 3 Construction Only(1)	**Unit 3 & 4 Construction(2)**	**Unit 4 Construction Only(1)**
Direct radiation	0.076	0.142	0.149
Gaseous Effluents	0.060	0.113	0.168
Liquid Effluents	0.001	0.002	0.001
Total(3)	0.138	0.257	0.318

Notes:
(1) Dose for construction of one unit.
(2) Dose for construction of two units.
(3) Values may not add exactly due to round-off.

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