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

August 12, 2009
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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 143 and 145 related to Combined License Application (COLA) Part 2, Tier 2, Sections 12.1, 12.3, 12.4, and 12.5S. This submittal completes the response to RAI letter 143, and partially completes the response to RAI letter 145. Attachments 1 through 4 include the responses to the RAI questions listed below:

12.01-1	12.03-12.04-1	12.05-1
	12.03-12.04-2	

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.

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

Executed on 8/12/09

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

ccc
Attachments:
1. Question 12.01-1
2. Question 12.03-12.04-1
3. Question 12.03-12.04-2
4. Question 12.05-1

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STI 32519116

cc: w/o attachment except*
(paper copy)

Director, Office of New Reactors
U. S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

Regional Administrator, Region IV
U. S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, Texas 76011-8064

Kathy C. Perkins, RN, MBA
Assistant Commissioner
Division for Regulatory Services
Texas Department of State Health Services
P. O. Box 149347
Austin, Texas 78714-9347

Alice Hamilton Rogers, P.E.
Inspection Unit Manager
Texas Department of State Health Services
P. O. Box 149347
Austin, Texas 78714-9347

C. M. Canady
City of Austin
Electric Utility Department
721 Barton Springs Road
Austin, TX 78704

*Steven P. Frantz, Esquire
A. H. Gutterman, Esquire
Morgan, Lewis & Bockius LLP
1111 Pennsylvania Ave. NW
Washington D.C. 20004

*George F. Wunder
* Michael Eudy
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852

(electronic copy)

*George Wunder
*Michael Eudy
Loren R. Plisco
U. S. Nuclear Regulatory Commission

Steve Winn
Eddy Daniels
Joseph Kiwak
Nuclear Innovation North America

Jon C. Wood, Esquire
Cox Smith Matthews

J. J. Nesrsta
R. K. Temple
Kevin Pollo
L. D. Blaylock
CPS Energy

RAI 12.01-1:

QUESTION:

STP 3 & 4 COL FSAR Section 12.1.4.3, Occupational Radiation Exposures, states "occupational radiation exposures will be maintained ALARA by means of the Operational Radiation Protection Program described in Section 12.5S" and that "The operational plans, procedures and policies currently in use at STP 1 & 2 reflect industry experience and guidance. They will be used, in conjunction with the guidance contained in the documents identified above, to develop the policies, plans and procedures for STP 3 & 4. Many of these plans, procedures and policies will be common to all four units." In keeping with the above stated policy of utilizing "industry experience and guidance" in development of the Radiation Protection Program, please answer the following:

1) Verify that STP has reviewed Draft Nuclear Energy Institute (NEI) document NEI 07-08, "Generic FSAR Template Guidance for Ensuring That Occupational Radiation Exposures Are As Low As is Reasonably Achievable (ALARA)" for applicability and possible incorporation into the STP 3 & 4 COL.

2) NEI 07-03 alone is not enough to fully describe all elements of an ALARA program. NEI 07-08 supplements the ALARA program description in NEI 07-03 with information describing the roles and responsibilities of management and staff, training requirements, as well as keys elements of an effective ALARA program. If STP has reviewed NEI 07-08 and determined that it will not be incorporated into the FSAR, modify applicable FSAR Sections to fully describe all elements of the ALARA program, or justify an alternative. Otherwise, reference NEI 07-08 in the STP FSAR.

RESPONSE:

It is understood that NEI 07-08 is currently under review by the NRC and presently exists in draft form. STP 3 & 4 has reviewed NEI 07-08 and continues to monitor its development.

Section 12.5S of the STP 3 & 4 COLA will be revised as shown below:

12.5S Operational Radiation Protection Program

Nuclear Energy Institute Report No. NEI 07-03A, "Generic FSAR Template Guidance for Radiation Protection Program Description" provides the Operational Radiation Protection Program for STP 3 & 4. This NEI template is incorporated by reference with the following site-specific supplements. The NEI 07-03A template material is shown in italics.

NEI report no. NEI 07-08, "Generic FSAR Template Guidance for Ensuring that Occupational Radiation Exposures are as Low as is Reasonably Achievable

(ALARA)", provides additional operating policy and consideration guidance for developing and implementing an ALARA program. This NEI template is incorporated by reference.

Additional markups to the COLA will be made in a later revision as necessary following acceptance of NEI 07-08 by the NRC and its issuance as NEI 07-08A.

RAI 12.03-12.04-1:

QUESTION:

Refueling dose - STP 3 & 4 COL Section 12.5S.2.4, Methods to Maintain Exposures ALARA, states "Movement of irradiated fuel assemblies is accomplished with the assembly maintained underwater. By following these procedures, the normal radiation level on the refueling bridge is expected to be less than 5 mrem/hr." However, FSAR Section 12.4.2, Reactor Building Dose, and Table 12.4-1, Projected Annual Radiation Exposure, use a reference exposure rate of 2 μGy/h for evaluation of refueling occupational exposures. Please address the following in FSAR Section 12.4 and 12.5S:

a) Change the units used from μGy/h to μSv/h to correctly indicate an exposure rate.

b) If the expected exposure rate on the refueling bridge will be less than 5 mrem/hr, a more conservative estimate of occupational dose would use 5 mrem/hr in the calculation. Correct the refueling dose estimate in the applicable portions of FSAR Section 12.4 and 12.5S, or justify an alternative.

RESPONSE:

a) The certified ABWR DCD, Section 12.4, provides effective dose rates in units of μGy/h However, for consistency with departure STD DEP 11.5-1 and present nomenclature, the units will be changed to μSv/h throughout COLA Section 12.4 as an editorial change in a future COLA revision as shown in the response to RAI 12.03-12.04-2.

b) The effective dose rate of 2 μGy/h (2 μSv/h) for the refueling operations stated in FSAR subsection 12.4.2 and Table 12.4-1 is not changed from the DCD. The statement in subsection 12.5S.2.4 that the expected normal radiation level on the refueling bridge will be less than 5 mrem/h is a dose rate for zone designation. As explained in DCD subsection 12.3.1.3, the dose rates for zone designation should not be interpreted as the expected dose rates which would apply in all portions of that zone, or for all types of work within that zone.

Refer to the response to RAI 12.03-12.04-2 for the required COLA revisions as a result of this RAI response.

RAI 12.03-12.04-2:

QUESTION:

Radwaste Building dose – STD DEP 11.2-1 completely replaces the Liquid Radioactive Waste Processing System located in the Radwaste Building. FSAR Section 12.4.3 states that the section has been replaced in its entirety. The Section contains a limited discussion of Radwaste Building work activities and states that Table 12.4-1, Projected Annual Radiation Exposure, was updated using an average dose rate for workers. The revised Table 12.4-1 indicates a reduction of hours per year by over a factor of four, and no change in the average dose rate for workers. Provide additional information in FSAR Section 12.4 that fully discusses and documents the reduced work hours in the Radwaste Building including a breakdown of work activities using revised work times and area dose rates.

RESPONSE:

EPRI Technical Report 1003063 includes an assessment of radiation exposure associated with changes from the traditional radwaste treatment systems (that is, filters, evaporators, demineralizers, and solidification processes) to advanced radwaste treatment systems (mobile charcoal, membrane, and ion exchange processes) similar to the systems for STP Units 3&4. The results of the assessment indicate that there was a substantial reduction in radiation dose (one plant experienced a factor of eight reduction in radiation dose). Much of this reduction is attributed to the reduction in the time personnel are working in radiation environments because of simpler operation and reduced maintenance requirements. Based on this experience, it is estimated that the changes to the Liquid Waste Management System (LWMS) will result in a reduction of the Radwaste Building annual radiation dose by a factor of approximately four. To update Table 12.4-1 to reflect this decrease in the annual radiation exposure, the average radiation dose rate to workers was assumed to remain the same and the number of hours in the Radwaste Building radiation areas is changed from 4200 hours to 1000 hours per year. This results in a radiation dose associated with the Radwaste Building of 25 person-mSv/year (approximately a factor of four reduction), a total of 54,040 hours per year in radiation areas, and a total radiation exposure of 909 person-mSv/year. The reduction in radiation exposure associated with the Radwaste Building by approximately a factor of four is considered conservative, and bounded by the industry experience discussed above.

STP 3&4 COLA Part 2, Tier 2 Section 12.4 will be revised as a result of this response and the response to RAI 12.03-12.04-1, as shown on the following pages.

12.4 Dose Assessment

The information in this section of the reference ABWR DCD, including all subsections and tables, is incorporated by reference with the following departures and supplements.

STD DEP Admin (Table 12.4-1)

STD DEP 9.1-1

STD DEP 11.2-1 (Table 12.4-1)

12.4.1 Drywell Dose

STD DEP Admin

The following provides the basis by which the drywell dose estimates for occupational exposure were made.

(1)

Early studies on dose rates during MSIV maintenance showed increases in dose rate directly proportional to recirculation line activity. The ABWR has deleted the recirculation lines entirely, thereby removing the singly most significant source of radiation in the drywell. The second most significant dose for MSIV operations will be the deposited and suspended activity in the feedwater lines. The deposited activity in the feedwater lines is expected to be lower than typical BWRs owing to an enhanced condensate polishing *system ~~with full cleanup of all condensate water~~, a 2% CUW System, and titanium* or stainless steel *condenser tubes. Additionally, the ABWR is designed to limit the use of cobalt bearing materials on moving components which have historically been identified as major sources of in-water contamination. Overall, the feedwater line radiation is expected to be a factor of three lower than current BWRs. Because of these factors, it is expected that the effective dose rate in the drywell will be 18* ~~μGy/h~~ μSv/h *and 13* ~~μGy/h~~ μSv/h *in the steam tunnel outboard of the primary containment.*

12.4.2 Reactor Building Dose

STD DEP 9.1-1

STD DEP Admin

The following provides the basis by which the Reactor Building dose estimates for occupational exposure were made.

(2) *ABWR refueling is accomplished via an automated refueling ~~bridge~~* machine. *All operations for refueling are ~~accomplished from an enclosed automation center off the refueling floor~~* as described in Section 9.1.4.2.7.1. *Time for refueling is reduced from a typical 4,400 person-hours down to 2,000 person-*hours *and from an effective dose rate of 25 ~~μGy/h~~* μSv/h *to less than 2 ~~μGy/h~~* μSv/h.

12.4.3 Radwaste Building Dose

STD DEP 11.2-1

This subsection is replaced in its entirety with the following.

Radwaste Building work consists of water processing, pump and valve maintenance, shipment handling, radwaste management, and general cleanup activity. Radwaste building doses result from routine surveillance, testing, and maintenance of the solid and liquid waste treatment equipment. The liquid treatment system collects liquid wastes from equipment drains, floor drains, filter backwashes, and other sources within the facility. The solid treatment system processes resins, backwash slurries, and sludge from the phase separator. It also processes dry active waste from the plant. Some examples of radwaste activities include resin dewatering, movement of casks and liners, filter handling, resin movement, and installation and removal of mobile radwaste processing skids. Both waste treatment systems are based on current mobile radwaste processing technology and avoid complex permanently installed components. All radwaste tankage and support systems are permanently installed. More of the radwaste operations involve remote handling than in a typical BWR. This~~, as well as improved maintenance procedures and a~~ more flexible radwaste system and building design, ~~leads to the estimated value shown in Table 12.4-1 for maintenance tasks in the Radwaste Building. The average dose rate shown in Table 12.4-1 is estimated for all operations.~~ simpler operation, and improved maintenance procedures result in a reduction in the number of total hours in the Radwaste Building radiation areas. The results of an industry assessment indicate that there was a substantial reduction in radiation dose (one plant experienced a factor of eight reduction in radiation dose) relative to the doses specified in the reference DCD. Based on this experience, it is estimated that the departures involving the Liquid Waste Management System (LWMS) will result in a reduction of the Radwaste Building annual radiation dose by a factor of approximately four (Reference 12.4-5). The average radiation dose rate to workers is assumed to be the same as specified in the reference DCD and the number of hours in the Radwaste Building radiation areas is changed from 4200 hours to 1000 hours per year. This results in a radiation dose associated with the Radwaste Building of 25 person-mSv/year (approximately a factor of four reduction), a total of 54,040 hours per year in radiation areas, and a total radiation exposure of 909 person-mSv/year. This is presented in Table 12.4-1.

12.4.5 Work at Power

STP DEP Admin

Work at power typically requires 5,000 hours per year at an effective dose rate of 66 ~~*µGy/h*~~ µSv/h *for the BWR. This category covers literally all aspects of plant maintenance performed during normal operations from health physics coverage to surveillance, to minor equipment adjustment, and minor equipment repair. Overall, the ABWR has been designed to use more automatic and remote equipment. It is expected that items of routine monitoring will be performed by camera or additional instrumentation. Most equipment in the ABWR is palatalized, which permits quick and easy replacement and removal for decontamination and repair. Therefore, a reduction in actual hours needed at power is estimated at 1,000 hours less than the typical value. In the area of effective dose rate, the ABWR is expected to have significantly lower general radiation levels over current plants, owing to more stringent water chemistry controls, a full flow condensate flow system, a 2% cleanup water program, titanium* or stainless steel *condenser tubes, Fe feedwater control, and low cobalt usage. In addition, the ABWR has in the basic design, compartmentalized all major pieces of equipment so that any piece of equipment can be maintained or removed for maintenance without affecting normal plant operations. This design concept thereby reduces radiation exposure to personnel maintaining or testing one piece of equipment from both shine and airborne contamination from other equipment. Finally, the ABWR has incorporated in the basic design the use of hydrogen water chemistry (HWC) and the additional shielding necessary to protect from the factor of six increase in N-16 shine produced through the steamlines into the Turbine Building. For normally occupied areas, sufficient shielding is provided to protect from N-16 shine. In areas which may be occupied temporarily for specific maintenance or surveillance tasks and where additional shielding is not appropriate (for the surveillance function) or deemed reasonable, the HWC injection can be stopped causing the N-16 shine to decrease to within normal operating BWR limits within 90 seconds and thus permitting those actions needed. Overall, it is estimated that the effective dose rate for work at power will be slightly over two thirds the typical rate or 40* ~~*µGy/h*~~ µSv/h.

12.4.6 References

12.4-5 "Performance Evaluation of Advanced LLW Liquid Processing Technology, Boiling Water Reactor Liquid Processing," EPRI Technical Report 1003063, November 2001.

Table 12.4-1 Projected Annual Radiation Exposure

Operation Task	Tier 2 Section	hours per year	~~µGy/h~~ µSv/h	person-mSv/yr
Drywell				
MSIV	(1)	~4,200	15	63
SRV, RIP, etc	(2)	1,150	75	86
FMCRD	(3)	370	65	24
LPRM/TIP	(4)	200	500	100
ISI	(5)	1,200	55	66
Other	(6)	3,500	35	123
Total		10,620		462
Reactor Building				
Vessel	(1)	1,200	15	18
Refueling	(2)	2,000	2	4
RHR/CUW	(3)	400	54	22
FMCRD	(4)	120	45	5
Instrument	(5)	1,000	30	30
Other	(6)	4,400	15	66
Total		9,120		145
Radwaste Building		~~4200~~ **1,000**	25	~~105~~ **25**
Turbine Building				
Valve Maintenance	(1)	1,000	39	39
Turbine Overhaul	(2)	15,500	2	31
Condensate	(3)	1,000	35	35
Other	(4)	11,800	1	12
Total		29,300		117
Work at Power		4,000	40	160
Totals		~~57,240~~ **54,040**		~~989~~ **909**

RAI 12.05-1:

QUESTION:

FSAR Section 12.5S.4.4 identifies three Very High Radiation Areas (VHRA) requiring additional administrative controls for entry including access control, use of an RWP and additional monitoring. Section 12.5S also incorporates by reference Nuclear Energy Institute Report No. NEI 07-03, "Generic FSAR Template Guidance for Radiation Protection Program Description" (NEI 07-03). As specified in Section 12.5.4.4 of NEI 07-03, please include the following additional information in FSAR Section 12.3-4:

1) Anticipated frequency of accessing each of the Very High Radiation Areas.
2) Detailed drawings for each Very High Radiation Area that indicate physical barriers that completely enclose the respective area in a manner that is sufficient to thwart undetected entry into the area. Alternatively, if such detailed drawings are not available, describe how such barriers will be verified in the final design of the facility.

RESPONSE:

1) It is anticipated that the Very High Radiation Areas listed in FSAR Section 12.5S.4.4 are seldom if ever accessed due to the extremely high expected dose rates. It is further anticipated that entry to these locked areas, if required at all, would be done during shutdown/refueling, and then only after the tank contents were removed and the associated reduction of dose rates occurred.

2) As stated in Section 12.5S, the South Texas Project Units 3 & 4 COLA incorporates the guidance of NEI 07-03A, as indicated in the response to NRC question 12.05-4. NEI 07-03A guidelines are consistent with those in Regulatory Guide 8.38: "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants". Section 1.6 of Reg Guide 8.38 allows the use of shielding not readily removable to make a high or very high radiation area inaccessible, and further states that "shielding requiring a hoist or crane to move would not be considered readily removable".

Section 12.3.1.3 of the DCD states:

- Access to areas in the plant is controlled and regulated by the zoning of a given area. Areas with dose rates such that an individual would receive a dose in excess of 1 mGy in a period of one hour are locked and posted with "High Radiation Area" signs. Entry to these areas is on a controlled basis.

Section 12.3.1.4.1 of the DCD states the following:

- the CUW filter/demineralizers are located in separate concrete-shielded cubicles which are accessible through shielded hatches.

- the backwash receiving tank is shielded separately from the resin transfer pump, permitting maintenance of the pump without being exposed to the spent resins contained in the backwash receiving tank.

- the backwash receiving tank is fitted with a charcoal canister filter, and that the HVAC system is designed to limit the spread of contaminants from these shielded cubicles. Finally, that personnel access to the cubicles for maintenance of these components is on a controlled basis, whereby specific restrictions and controls are implemented to minimize personnel exposure.

Section 12.3.1.4.3 of the DCD states:

- The FPC System components are located in the Reactor Building, and that the filter demin units are the highest radiation level components in the system. Further, that each unit is located in a concrete-shielded cubicle which is accessible through a shielded hatch.

In NUREG 1503, Radiation Protection section, under Facility Design Features, Section 12.3.1, the following statement is made:

- The ABWR is designed so that operation will not require alternate high-radiation area controls (pursuant to 10 CFR 20.203(c)(5), as used in current operating BWRs.... All high radiation areas (with greater than 1.0mSV/hr (100mrem/hr) can be locked to control unauthorized access. This design position meets the requirements of 10 CFR Part 20 and is acceptable.

STP 3 & 4 took no departure from Section 12.3 of the DCD affecting these design features. STPNOC concludes that the issue of access to these shielded areas to preclude undetected entry is adequately addressed by the certified ABWR design and has finality.

No revision to the COLA is required as a result of this response.