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

August 26, 2009
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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
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South Texas Project
Units 3 and 4
Docket Nos. 52-012 and 52-013
<u>Response to Requests for Additional Information</u>

Attached are responses to NRC staff questions included in Request for Additional Information (RAI) letter number 180 related to Combined License Application (COLA) Part 2, Tier 2, Sections 12.3 and 12.4. This submittal completes the response to RAI letter 180. Attachments 1, 2, and 3 contain responses to the RAI questions listed below:

12.03-12.04-6 12.03-12.04-7 12.03-12.04-8

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 8/26/09

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

scs
Attachments:

1. Question 12.03-12.04-6
2. Question 12.03-12.04-7
3. Question 12.03-12.04-8

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NRO

STI 32526862

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

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RAI 12.03-12.04-6:

QUESTION:

Section 12.3 of the COL FSAR references Turbine Building Radiation Area Zone maps, Figures 12.3-49 thru 12.3-53, 12.3-55, 12.3-70 thru 12.3.73, and 12.3-75 thru 12.3-77, and Table 12.3-7, Area Radiation Monitors Turbine Building, for the location and sensitivity ranges of the turbine building area radiation monitors. Additional information is needed by the staff concerning placement and monitor sensitivity ranges. In accordance with NUREG-0800 and RG-1.206 C.I.12.3.4, please provide the following additional information concerning the monitors:

1. How placement of the monitors was determined.
2. How the specified sensitivity ranges were determined.

RESPONSE:

The reference ABWR DCD Tier 2 Subsection 12.3.4.2, ARM Detector Location and Sensitivity, provides the area radiation monitor detector location and channel sensitivity range presented in units of Gy per hour. Because Gray is a unit of absorbed dose, which is by definition a biological effect that is calculated using effective dosages and biological/radiation factors, instead of a measurement, STD DEP 11.5-1 corrected the COLA units to those of an effective dose rate, Sv per hour, which is what is measured and displayed by the instruments. In summary, the placement of the monitors and specified sensitivity ranges are determined as provided by the reference ABWR DCD in Tier 2 Subsection 12.3.4.2.

Also, the reference ABWR DCD Tier 2, Subsection 12.2.1.3 Turbine Building Sources, was incorporated by reference into the COLA and discusses the turbine building sources.

NUREG-1503 Section 12.3.4 Area Radiation and Airborne Radioactive Monitoring Instrumentation states in part:

> "Open item 114 in the DSER (SECY-91-355), questioned the description of the ABWR area radiation monitoring system. GE revised the SSAR with the following information. The area radiation monitoring system consists of 25 gamma sensitive detectors and their associated digital monitors. The detectors are in key locations of the plant and will have operating ranges (sensitivity) commensurate with the expected radiation levels in the areas."

NUREG-1503 Section 12.3.5.1 Plant Shielding DAC states:

> "Chapter 12 of the SSAR contains layout drawings of the plant that indicate the designed maximum radiation level (or zone) for each room, equipment cubicle, and operating space during normal power operations, shutdown operations and accident conditions. As discussed in Section 12.2 above, the piping layout and component selection have not been set for the ABWR systems; therefore, parameters such as source strength and geometry needed to verify the adequacy of the radiation shields around these systems are not available. In addition, nitrogen-16 gammas from the plant can significantly contribute to offsite dose rates. The

adequacy of the plant shielding needed to comply with the radiation dose limits for individual members of the public in 10 CFR Part 20 cannot be verified since the turbine design and site-specific parameters such as distance to the site boundary are unknown.

"GE has submitted DAC for plant shielding in Table 3.2.a of the CDM. These DAC require the COL applicant to verify the adequacy of (1) the shielding around rooms and spaces during normal operations and shutdown conditions, (2) the shielding and temporary shield space provided between plant systems during maintenance activities, (3) the shielding provided around vital plant areas during accident conditions (Three Mile Island (TMI) Action Plan Item II.B.2 (10 CFR 50.34((f)(2)(vii)), and (4) the plant shielding needed to limit public dose. The staff's review indicates that the analysis, assumptions, methods, and acceptance criteria in these DAC are consistent with the criteria in the SRP. Therefore, the staff concludes that compliance with these DAC, as supplemented by the information in SSAR Sections 12.3.2, is acceptable to adequately address the relevant requirements in 10 CFR 50.34(b)(3) and 10 CFR Part 20 concerning the limitation of radiation exposures to personnel, including the requirement to maintain doses ALARA as supplemented by the guidance in RG 8.8 (Rev. 3); 10 CFR 50.34(f) and GDC 19 with respect to operator access to plant areas during and following a reactor accident as supplemented by guidance in Item II.B.2 of NUREG-0737; and GDC 61 regarding adequate shielding of fuel storage and handling, radioactive waste, and other systems which may contain radioactivity. GE provided a revised set of design descriptions and ITAAC (including DAC). The adequacy and acceptability of the ABWR Tier 1 material and ITAAC (including DAC) are evaluated in Section 14.3 of this report."

The DCD Tier 1 Table 3.2a, Plant Shielding Design, incorporates the Radiation Protection DAC. The STP 3&4 COLA, Rev. 2, has incorporated this table by reference, with no departures or supplements.

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

RAI 12.03-12.04-7

QUESTION:

ABWR DCD Section 12.3.1.4, Implementation of ALARA, states that the Reactor Water Cleanup (CUW) System backwash tank is vented through a charcoal filter canister to reduce the emission of radioiodines into the plant atmosphere. STD DEP 12.3-2 states that the text was revised to delete the ABWR DCD description of a charcoal filter canister on the backwash tank vent line. The applicant states that the current design intent is for the CUW backwash tank to be vented to the Reactor Building HVAC exhaust.

Operating experience has shown that venting tank and demineralizer vents directly into HVAC systems with no filtration, or moisture or resin traps, could result in intrusion of moisture or particulate matter into ventilation systems resulting in contamination of facilities or the environment. There is no discussion in the STP COL FSAR or in the Departures Report (Part 7 of the STP COL) to address this operating experience. In accordance with 10 CFR 20.1406(a), the COL applicant is responsible for documenting in the application how facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.

Please describe what design feature(s) will be included in the backwash tank vent line that will prevent or mitigate contamination of the ductwork and facility. If design features are not used, then provide the specific alternate approaches used and the associated justification.

RESPONSE:

The design of the RWCU System will include a charcoal filter canister on the backwash tank vent line. Accordingly, STD DEP 12.3-2 is deleted. The STP 3&4 COLA will be changed in a future revision as shown below, with changes in gray shading.COLA Part 2, Tier 2, Section 12.3

In the beginning of Section 12.3, the departure listing is being deleted from the COLA.

12.3 Radiation Protection Design Features

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

~~STD DEP 12.3-2~~

For Subsection 12.3.1.4.1, the departure listing is being deleted, and the text that was deleted by STD DEP 12.3-2 is being restored to the DCD text.

12.3.1.4.1 Reactor Water Cleanup (CUW) System

~~STD DEP 12.3-2~~

The CUW System is provided with chemical cleaning connections which can utilize the condensate system to flush piping and equipment prior to maintenance. The CUW filter/demineralizers can be remotely backflushed to remove spent resins and filter aid material. If additional decontamination is required, chemical addition connections are provided in the piping to clean piping as well as equipment prior to maintenance. The backwash tank employs an arrangement to agitate resins prior to discharge. The tank vent is fitted with a charcoal filter canister to reduce emission of radioiodines into the plant atmosphere. The HVAC System is designed to limit the spread of contaminants from these shielded cubicles by maintaining a negative pressure in the cubicles relative to the surrounding areas.

COLA Part 2, Tier 2, Section 19.2

In Table 19.2-2, the departure number STD DEP 12.3-2 and its describing columns are being deleted from the COLA.

Table 19.2-2 PRA Assessments of STP COLA Departures from ABWR DCD

Departure Number	Certified Design Basis (DCD)	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
~~STD DEP 12.3-2 Deletion of CUW Backwash Tank Vent Charcoal Filter~~	~~The CUW vent for CUW backwash is fitted with a charcoal filter canister to reduce the omission of radioiodines into the plant atmosphere.~~	~~The CUW system contains charcoal filter on its vent. The CUW backwash tank is vented into the reactor building HVAC System exhaust, exiting the plant via the plant stack as monitored release.~~	~~No effect on the PRA, not modeled.~~

For the COLA Part 7, STD DEP 12.3-2 writeup, the departure listing is being deleted from the COLA.COLA Part 7, Section 3.0

~~STD DEP 12.3-2, Deletion of CUW Backwash Tank Vent Charcoal Filter~~

~~Description~~

~~This departure deletes the statement in Subsection 12.3.1.4.1 which states that the vent off the CUW backwash tank is fitted with a charcoal filter canister to reduce the emission of radioiodines into the plant atmosphere. A review of the system diagrams for the CUW system show no such filter as part of the approved design. The current design intent is for the CUW backwash tank to be vented into the Reactor Building HVAC System exhaust, which eventually exits the plant via the plant stack as a monitored release.~~

~~Evaluation Summary~~

~~This departure has been evaluated and determined to comply withpursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5, as described previously. The change has no adverse impact.~~

~~This departure does not change any Tier 1, Tier 2* information, the Technical Specifications, any underlying design or other operational requirements.~~

~~This departure corrects the text description of the backwash tank vent system by deleting reference to a charcoal filter on that vent system which does not exist in the design. As noted earlier, the current design intent is for the CUW backwash tank to be vented into the Reactor Building HVAC System exhaust, which eventually exits the plant via the plant stack as a monitored release. Since this change does not affect any plant SSC, there is no affect on any accident previously evaluated in the plant specific DCD. Furthermore, it does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possiblility for an accident of a different type is not created. Also it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.~~

~~Therefore, this change has no adverse impact and does not require prior NRC approval.~~

RAI 12.03-12.04-8

QUESTION:

NUREG-0800 12.3-3 and Regulatory Guide 1.206 C.I.12.3.5 10, CFR 50.34(f)(2)(vii) and NUREG-0737 II.B.2, note that vital areas which require access by operators aiding, mitigating or recovering from the accident, need to be identified. NUREG-0737 II.B.2 notes that this is applicable to vital areas and equipment, other than the Control Room, such as the radwaste control stations, emergency power supplies, motor control centers, and instrument areas. These criteria are applicable even if the areas are accessed on an irregular basis and not continuous occupied for the duration of the event. Some of the areas that may reasonably be expected to require access include but are not limited to:

- Radwaste Control Panel
- Safety Injection Pumps
- SFP area
- Residual Heat Removal Pumps

Section 12.3 of the COL FSAR contains insufficient location, exposure rate, occupancy time information, and the associated mission doses, for those areas that plant operators or radiation protection personnel may need to access as noted above. In order to make a determination of reasonable assurance that the post-accident dose limit of 0.05 Sv (5 rem) for the duration of the accident can be met, additional information is needed by the staff.

In accordance with 10 CFR 50 GDC-19, NUREG-0800 and RG-8.8 C.I.12.3.5, please include figures, tables, and discussion in FSAR Section 12.3 that clearly identifies likely post-accident locations that will require access, travel pathways, occupancy times, and the exposure rate values associated with plant operation and monitoring, for the duration of the event, or provide the specific alternative approaches used and the associated justification.

RESPONSE:

NUREG 1503 Volume 1, Section 12.3.5.1, Plant Shielding DAC, states, in part:

> "GE has submitted DAC for plant shielding in Table 3.2.a of the CDM. These DAC require the COL applicant to verify the adequacy of . . . (3) the shielding provided around vital plant areas during accident conditions (Three Mile Island (TMI) Action Plan Item II.B.2 (10 CFR 50.34 ((f)(2)(vii)). . . The staff's review indicates that the analysis assumptions, methods, and acceptance criteria in these DAC are consistent with the criteria in the SRP. Therefore, the staff concludes that compliance with these DAC, as supplemented by the information in SSAR sections 12.3.2, is acceptable to adequately address the relevant requirements in 10 CFR 50.34 (b)(3) and 10 CFR Part 20 concerning the limitation of radiation exposures to

personnel, including the requirement to maintain doses ALARA as supplemented by the guidance in RG 8.8 (Rev.3); 10 CFR 50.34(f) and GDC 19 with respect to operator access to plant areas during and following a reactor accident as supplemented by the guidance in Item II.B.2 of NUREG-0737; and GDC 61 regarding adequate shielding of fuel storage and handling, radioactive waste, and other systems which may contain radioactivity, GE provided a revised set of design descriptions and ITAAC (including DAC). The adequacy and acceptability of the ABWR Tier 1 material and ITAAC (including DAC) are evaluated in Section 14.3 of this report."

NUREG 1503 Volume 1, Section 12.3.6, 10 CFR 50.34(f): TMI Related Items, states, in part:

"SSAR Section 12.3 addresses two items from the TMI Action Plan (NUREG-0660), II.F.1.3 (10 CFR 50.34(f)(2)(xvii)(D), and II.B.2 (10 CFR 50.34(f)(2)(vii)). . . Item II.B.2 specifies that radiation shielding be provided so that operators can access vital equipment in the plant during an accident without receiving excessive radiation dose. As discussed in Section 12.3.5.1 of this report, the DAC in Table 3.2.a of the CDM require the COL applicant to demonstrate compliance with Item II.B.2, as part of the analysis to verify the adequacy of the plant's radiation shielding.

"The COL applicant will be responsible for demonstrating compliance with II.B.2 because the final hardware and system design specifications need [sic] as inputs to shielding calculations are not available now."

NUREG 1503 Volume 1, Section 12.2.2, Certified Design Material, states, in part:

"As discussed in Section 12.2 of this report, the SSAR does not provide system layouts within rooms or cubicles or information about the type and size of components in these systems. Without this as-built or as-procured information, source term parameters needed to calculate radiation shielding for these systems cannot be provided as specified in the SRP. . . As an alternative, GE provided DAC that require the COL applicant to determine source term parameters that will be verified during plant construction. DAC are discussed in Section 12.3.5 of this report. DAC describing the bases for the source term are consistent with the SRP acceptance criteria. Compliance with these DAC, supplemented by the information in SSAR sections 12.2, and 12.3, is acceptable to adequately address the requirement to identify the kinds and quantities of radioactive materials expected to be produced by plant operation in 10 CFR 50.34(b)(3) and will ensure that the appropriate source terms (as supplemented by the guidance of RG 1.112 (Rev. 0)). NUREG-0737, and ANSI/ANS 18.1) are used to demonstrate that the ABWR design meets the relevant requirements in 10 CFR Part 20 concerning the limitation of radiation does [sic] to personnel; 10 CFR 50.34(f) and GDC 19 with respect to operator access to plant areas during and following a reactor accident; and GDC 61 regarding adequate shielding, containment and confinement of fuel storage and handling, radioactive waste, and other systems which may contain radioactivity. The adequacy and acceptability of the ABWR design descriptions and ITAAC (including DAC) are evaluated in Section 14.3 of this report."

NUREG 1503 Volume 1, Section 14.3.3.2, Radiation Protection DAC, states, in part:

"Item 4 in Table 3.2a ensures that adequate shielding is provided for those areas of the plant that may require occupancy to permit an operator to aid in the mitigation of or the recovery

> from an accident. . . Therefore, the staff concludes that the DAC in the CDM are necessary and sufficient to provide reasonable assurance that if the inspections, tests, and analyses are performed and the acceptance criteria met, the radiation protection aspects of SSCs important to safety in a facility that references the design have been designed, constructed and will operate in accordance with the design certification and applicable regulations."

The DCD Tier 1 Table 3.2a, Plant Shielding Design, incorporates the Radiation Protection DAC. The STP 3&4 COLA, Rev. 2, has incorporated this table by reference, with no departures or supplements.

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