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

April 2, 2009
U7-C-STP-NRC-090031

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 81 and 82 related to Combined License Application (COLA) Part 2, Tier 2, Section 5.3.

Attachments 1 through 3 address the responses to the RAI questions listed below.

RAI 05.03.01-1
RAI 05.03.01-3
RAI 05.03.02-1

RAI Response 05.03.01-1 modifies STP commitment COM 5.3-1 and the change is shown in Attachment 4.

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 response.

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

STI 32454165

D091
NRO

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

Executed on 4/2/09

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

jet

Attachments:

1. Question 05.03.01-1
2. Question 05.03.01-3
3. Question 05.03.02-1
4. Commitment Change

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
Texas Department of Health Services
Division for Regulatory Services
P. O. Box 149347
Austin, Texas 78714-9347

Alice Hamilton Rogers, P.E.
Inspections Unit Manager
Texas Department of 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
*Tekia Govan
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852

(electronic copy)

*George Wunder
*Tekia Govan
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 05.03.01-1:

QUESTION:

FSAR Section 5.3.4.1 states that fracture toughness data based on the limiting reactor vessel actual materials will be provided in an amendment to the FSAR in accordance with 10 CFR 50.71 (e) prior to receipt of fuels on site. The as-procured reactor vessel material properties will be available to the COL holder after the acceptance of the reactor vessel. In order to provide sufficient time for NRC review of the fracture toughness data, the staff requests that a more specific and timely milestone for submitting the fracture toughness data to the NRC be established. Accordingly, the staff proposes a Commitment be added to address the COL information item in Section 5.3.4.1 to state that, within a reasonable period of time following acceptance of the reactor vessel, the COL holder submit to the NRC staff the fracture toughness data, for example, one year after the acceptance of the reactor vessel.

RESPONSE:

STP agrees to provide the requested fracture toughness data in the first regular FSAR update that occurs one year after the on-site acceptance of the reactor vessel.

The COLA will be changed as follows:

5.3.4.1 Fracture Toughness Data

Fracture toughness data based on the limiting reactor vessel actual materials will be provided in the first regular amendment to the FSAR in accordance with 10 CFR 50.71(e) ~~prior to receipt of fuel on site~~ that occurs one year after the on-site acceptance of the reactor vessel. The data will be based on test results from the actual materials used in the RPV. (COM 5.3-1)

RAI 05.03.01-3:

QUESTION:

Table 13.4S-1 proposes that the reactor vessel material surveillance program (RVSP) be implemented at initial criticality. However, in order to verify that the requirements of the RVSP have been met and are in accordance with Appendix H of 10 CFR Part 50, the program must be implemented prior to fuel load by a license condition. Please revise the FSAR accordingly.

RESPONSE:

The FSAR will be revised to clarify that the requirements of the RVSP will be met prior to fuel load.

COLA Table 13.4S-1 will be revised as follows:

Table 13.4S-1 Operational Programs Required by NRC Regulation and Program Implementation

Item	Program Title	Program Source (Required By)	FSAR (SRP) Section	Implementation	
				Milestone	Requirement
1	Inservice Inspection Program	10 CFR 50.55a(g)	5.2.4 6.6	Commercial operation	10 CFR 50.55a(g) ASME Section XI IWA-2430(b)
2	Inservice Testing Program	10 CFR 50.55a(f) 10 CFR 50, App A	3.9.6 5.2.4	After generator on line on nuclear heat	10 CFR 50.55a(f) ASME OM Code
3	Environmental Qualification Program	10 CFR 50.49(a)	3.11	~~None specified~~ Fuel Load	License Condition
4	Preservice Inspection Program	10 CFR 50.55a(g)	5.2.4 6.6	Completion prior to initial plant startup	10 CFR 50.55a(g) ASME Section XI IWB-2200(a)
5	Reactor Vessel Material Surveillance Program	10 CFR 50.60 10 CFR 50, App H	5.3.1	~~None specified~~ ~~Initial criticality~~ Fuel Load	License Condition
6	Preservice Testing Program	10 CFR 50.55a(f)	3.9.6	~~None specified~~ Fuel Load	License Condition
7	Containment Leakage Rate Testing Program	10 CFR 50.54(o) 10 CFR 50, App A (GDC 32) 10 CFR 50, App J 10 CFR 52.47(a)(1)	6.2.6	Fuel load	10 CFR 50, App J Option A - Section III Option B - Section III.A

RAI 05.03.02-1:

QUESTION:

ABWR DCD, Revision 0, Section 5.3.4.3 states that the COL applicant will submit plant-specific calculations of RT_{NDT}, stress intensity factors, and pressure-temperature (P-T) curves similar to those in Regulatory Guide 1.99 and SRP Section 5.3.2. In response to COL License Information Item, South Texas Project Units 3 & 4 COL FSAR Section 5.3.4.3 states that plant-specific calculations of RT_{NDT}, stress intensity factors, and P-T curves similar to those in Regulatory Guide 1.99 and SRP Section 5.3.2 will be provided in an amendment to the FSAR in accordance with 10 CFR 50.71 (e) prior to receipt of fuel on site. The data will be based on test results from the actual materials used in the RPV.

The P-T limits or the pressure temperature limits report (PTLR) for South Texas Project Units 3 & 4 must be provided to the NRC for review and approval prior to the issuance of a COL license. Please provide all documentation (methodologies, calculations, reports, etc.) used in developing the P-T limits or PTLR to the NRC for review and approval.

RESPONSE:

A PTLR containing calculations of RT_{NDT}, stress intensity factors, and bounding pressure-temperature (P-T) curves similar to those in Regulatory Guide 1.99 and SRP Section 5.3.2 will be submitted by November 30, 2009. This PTLR will be based on the NRC approved methodology contained in the Structural Integrity Associates topical report prepared for the Boiling Water Reactor Owner's Group: SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," dated April 2007.

FSAR Figure 5.3-1 will be updated in a future COLA revision, following the November 30, 2009 PTLR submittal, to include the bounding pressure-temperature (P-T) curves developed as part of the PTLR.

Additionally, there are several COLA changes that are associated with the change in this methodology. These changes are provided below.

The bracketed item in Technical Specification 5.7.1.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)" requesting the identification of a "topical report, number, title, date and NRC staff approval document, or staff safety evaluation report for a plant specific methodology by NRC letter and date" will be completed by specifying SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," April 2007 in this space and removing the brackets as follows:

5.7.1.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The RCS pressure and temperature limits, including heatup and cooldown rates, criticality, and hydrostatic and leak test limits, shall be established and documented in the PTLR. LCO 3.4.9, RCS Pressure and Temperature (P/T) Limits addresses the reactor vessel pressure and temperature limits and the heatup and cooldown rates. The analytical methods used to determine the pressure and temperature limits including the heatup and cooldown rates shall be those previously reviewed and approved by the NRC in ~~[Regulatory Guide 1.99, Revision 2, and in accordance with 10 CFR 50, Appendix G]~~ SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," April 2007. *The reactor vessel pressure and temperature limits, including those for heatup and cooldown rates, shall be determined so that all applicable limits (e.g., heatup limits, cooldown limits, and inservice leak and hydrostatic testing limits) of the analysis are met. The PTLR, including revisions or supplements thereto, shall be provided upon issuance for each reactor vessel fluency period.*

The first paragraph of COLA Technical Specification Bases Applicable Safety Analyses in Section B 3.4.9 will be restored to the DCD wording as follows:

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, a condition that is unanalyzed. Reference 7 establishes the methodology for determining the P/T limits ~~Reference 7 establishes the methodology for determining the P/T limits~~.

The References to COLA Technical Specification Bases Section B 3.4.9 will be revised as follows:

1. *10 CFR 50, Appendix G.*
2. *ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.*
3. *ASTM E 185-82, July 1982.*
4. *10 CFR 50, Appendix H.*
5. *Regulatory Guide 1.99, Revision 2, May 1988.*
6. *ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.*
7. ~~*NEDO-21778-A, December 1978.*~~ SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," April 2007.

The third paragraph of COLA Part 2 Tier 2 Section 5.3.4.1 will be revised as follows:

> The evaluation methods will be in accordance with ~~Appendix G of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1989 Edition, excluding addenda,~~ 10 CFR 50 Appendices G and H, ~~and~~ Regulatory Guide 1.99 Rev. 2, and Reference 5.3-4.

The third paragraph of COLA Part 2 Tier 2 Section 5.3.4.3 will be revised as follows:

> The evaluation methods will be in accordance with ~~Appendix G of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1989 Edition, excluding addenda,~~ 10 CFR 50 Appendices G and H, ~~and~~ Regulatory Guide 1.99 Rev. 2, and Reference 5.3-4.

Reference 5.3-4 in COLA Part 2 Tier 2 Section 5.3.5 will be added as follows:

> 5.3-4 SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," April 2007~~"*Transient Pressure Rises Affecting Fracture Toughness Requirements for Boiling Water Reactors*", *January 1979 (NEDO-21778-A)*~~.

The Departures Report, Part 7, Section 2.2 will be revised as follows:

STD DEP 16.3-8, LCO 3.4.9 RCS Pressure and Temperature (P/T) Limits

> The LCO 3.4.9 Bases Applicable Safety Analyses ~~of~~ section states that, "Reference 7 establishes the methodology for determining the P/T limits." Reference 7 is NEDO 21778-A, December 1978, which is not the correct reference. The correct reference is SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," April 2007. ~~This document does not include the methodology for determining P/T Limits for the ABWR. Therefore it has been deleted as a reference.~~ Reference 7 has been changed from NEDO 21778-A, December 1978 to SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," April 2007.
>
> Additionally, the bracketed information in Section 5.7.1.6 has been replaced with ~~"Regulatory Guide 1.99, Revision 2 and in accordance with 10 CFR 50, Appendix G."~~ "SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," April 2007." ~~The methodology establishing the ABWR P/T Limits is described in Regulatory Guide 1.99 Revision 2 and is in accordance with 10 CFR 50, Appendix G, as described in the Bases Background section of LCO 3.4.9.~~

COMMITMENT CHANGE

Commitment Number	Commitment Summary	Milestone	COL Item
COM 5.3-1	Fracture toughness for actual materials in pressure vessel will be added to the FSAR.	~~Prior to Fuel Receipt~~ First regular FSAR update one year after on-site acceptance of the reactor vessel	5.4