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

May 19, 2010
U7-C-STP-NRC-100114

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 Request for Additional Information

Reference: Letter, Mark McBurnett to Document Control Desk, "Response to Request for Additional Information," dated July 29, 2009. U7-C-STP-NRC-090089 (ML092150965)

Attached are the responses to NRC staff questions included in Request for Additional Information (RAI) letter number 313 related to Combined License Application (COLA) Part 2, Tier 2, Section 3.9.2. This completes the response to the letter. Attachments 1 through 7 provide the responses to the RAI questions listed below:

RAI 03.09.02-9	RAI 03.09.02-12	RAI 03.09.02-14
RAI 03.09.02-10	RAI 03.09.02-13	RAI 03.09.02-15
RAI 03.02.02-11		

Additionally, Attachments 8 and 9 provide revised responses to NRC staff questions included in RAI letter numbers 144 and 147 related to Combined License Application (COLA) Part 2, Tier 2, Section 14.2. The Reference above provides the original responses which are revised by the attached responses to the following RAI questions:

RAI 14.02-6 RAI 14.02-8

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

There are no commitments in this letter.

D091
NRO

STI 32679745

If you have any questions regarding this response, 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 _____

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

jep

Attachments:

1. RAI 03.09.02-9
2. RAI 03.09.02-10
3. RAI 03.09.02-11
4. RAI 03.09.02-12
5. RAI 03.09.02-13
6. RAI 03.09.02-14
7. RAI 03.09.02-15
8. RAI 14.02-6, Revision 1
9. RAI 14.02-8, Revision 1

cc: w/o attachment except*
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RAI 03.09.02-9

QUESTION:

Supplement to Question 11648 (eRAI 03.09.02-2)

In the response to RAI 03.09.02-2, STP referenced two Toshiba reports:

1. RS-5126954, Revision 1, "Prototype ABWR Reactor Internals Flow Induced Vibration Test Report," that documents the ABWR prototype comprehensive vibration assessment program; and

2. RS-5126579, Revision 1, "STP 3 and 4 Reactor Internals Flow Induced Vibration Assessment Program," that documents STP 3 and 4's additional testing and analyses and their application of the prototype vibration assessment program.

The staff has reviewed these reports for their contents and the level of details described in Regulatory Guide 1.20. Please be advised that the valid prototype referenced in the STP COL FSAR is a foreign plant, as such, these reports do not meet the guidance described in Regulatory Guide 1.20. Please update and re-submit these reports.

RESPONSE:

The reports discussed in this RAI were prepared to support the use of a Japanese ABWR reactor as the prototype, and were prepared using the information that was available from the prototype testing that was discussed in the ABWR FSER (NUREG-1503). As noted in the FSER, the testing was not yet completed at the time of the ABWR certification. The test program that was performed at the Japanese ABWR was performed in the mid-1990's and included extensive testing for numerous test conditions at both pre-operational and power conditions. The testing development was performed on the basis of known information and issues at the time the test planning was done and the testing performed. Since that time, new issues have emerged (e.g., issues with BWR steam dryer acoustic resonance) that were not directly addressed in the test program.

STPNOC has re-evaluated the use of the Japanese ABWR plant as the ABWR prototype plant. This re-evaluation included review of the available information from the Japanese ABWR testing as compared to the guidance in Regulatory Guide (RG) 1.20 Rev. 3, the specific information that is requested in the DCD for COL Information Item 3.27 (DCD Tier 2 Subsection 3.9.7.1), and the clarification of NRC staff expectations as delineated in a meeting on December 17, 2009. As a result of this review, STPNOC has made a determination that STP 3 will be the prototype ABWR plant. STP 4 will be a non-prototype Category I plant, and will rely on STP 3 as the valid prototype. Therefore, instead of relying on the reports discussed in this RAI, new reports will be prepared and submitted. The contents and schedule for the new reports to be submitted in support of the STP 3&4 COLA review, and the resulting changes to the COLA, are discussed below.

This response completely supersedes the responses to RAIs 03.09.02-2 through 03.09.02-8, which were transmitted by STPNOC letter U7-C-STP-NRC-090088 (July 30, 2009).

STP 3: STP 3 will be the ABWR prototype reactor in support of the COL application. STPNOC will provide a final report (STP 3 ABWR Prototype Reactor Internals Flow-Induced Vibration Assessment Program, hereinafter referred to the STP 3 FIV Program Report) that documents the stress and vibration analysis program, the stress and vibration measurement program, and the inspection program. The stress and vibration analysis program will include the steam dryer, lower plenum components, and all other reactor internal components. In the development of the program, the test results and operating experience from the foreign reactor will be used to inform the STP 3 test program. The results of the foreign reactor FIV testing are directly applicable to the STP 3 prototype test planning because the reactor internals, power levels, flow rates, etc. are substantially the same. In addition, the foreign ABWR plant has over 10 years of successful operating experience.

The STP 3 FIV Program Report will be similar in scope and level of detail to the report developed for the AP-1000 reactor design (WCAP-15949-P, Rev. 2, "AP-1000 Reactor Internals Flow-Induced Vibration Assessment Program"). The STP 3 FIV Program report will include the predictive analysis results for the STP 3 ABWR Prototype Reactor internals, including specific information on the modeling for reactor internals analysis, and validation and benchmarking of the models. STP plans to use the results obtained from the extensive FIV testing performed on the reference Japanese ABWR to validate and benchmark the analytical models and forcing function predictions. The stress and vibration measurement plan will identify reactor internal components to be instrumented and the types and numbers of instruments to be used. Sensor types will be selected based upon prior application in a reactor environment and proven reliability. Test data will be collected and analyzed on-line and off-line during pre-operational and start-up test conditions as well as transient conditions. Inspections for such indications as wear, cracks, displaced/failed components, loosening of bolts, evidence of loose parts and foreign material will be performed prior to and after pre-operational test completion. This plan, which will be based on the predictive analysis and the reference Japanese ABWR tests, will be included in the STP 3 FIV Program Report. This report will also include the inspection plan, which will include identification of components and locations to be inspected, the inspection methods, and the method for documentation of results and comparison to the predictive analyses.

Calculations for development of the forcing functions, component modal analyses, and stress analyses will be prepared in support of the STP 3 FIV Program Report. These calculation packages will be made available for NRC review in accordance with the schedule provided in Table 03.09.02-9 below.

To address the potential of acoustic resonance for the steam dryer, which is discussed at length in RG 1.20 Rev. 3, the STP 3 FIV Program will also include explicit development of the predictive analysis for the ABWR steam dryer, including effects of acoustics of the main steam line. An initial analysis for acoustics (an acoustic screening analysis) will be completed, and a summary report will be prepared and submitted in accordance with the schedule provided in Table 03.09.02-9 below. A 1/8-scale, 4-line model test will be performed to provide input for the development of the steam dryer acoustic loads, based on the acoustic circuit model (ACM)

methodology. The ACM methodology is currently employed for power uprates of operating BWR reactors, and has been demonstrated as an acceptable method of developing predictive analysis loads for the steam dryers. A test plan will be developed for the 1/8-scale model test, which will be made available for NRC review. The subscale model testing schedule and location will be provided to NRC, and NRC is welcome to witness the subscale testing. The subscale test results will be documented in a test report that will be submitted in accordance with the schedule provided in Table 03.09.02-9 below. The resulting acoustic loads developed based on the subscale model tests will be used to develop the acoustic loads on the steam dryer, and to perform the analysis for the effects of high cycle fatigue. The Steam Dryer High Cycle Fatigue Analysis Report will document this analysis, and will be submitted in accordance with the schedule provided in Table 03.09.02-9 below.

The current schedule for submittal of the reports and the availability of the calculation packages are summarized in Table 03.09.02-9.

Table 03.09.02-9
STP 3 FIV Program Pre-COL Deliverable and Availability Schedule

Component / Document	Available for Review	Submittal to NRC
Steam Dryer		
Initial Acoustic Screening Report		1-June-2010
1/8-Scale 4-Line Model Test Plan	1-June-2010	
Performance of Subscale Test	June 2010 (TBD)	
Subscale Test Report		30-Sept-2010
Non-Steam Dryer Components		
Lower plenum CFD forcing function calc	18-Aug-2010	
Other components forcing function calc	21-Jul-2010	
Lower plenum modal analyses calc	18-Aug-2010	
Other component modal analyses calc	21-Jul-2010	
Predictive stress analysis calc	10-Nov-2010	
Measurement, Test and Inspection Plan	26-Nov-2010	
STP 3 ABWR FIV Program Report		15-Dec-2010
STP 3 Steam Dryer High Cycle Fatigue Rpt		15-Dec-2010

STP 4: STP 4 will be a non-prototype Category I plant, and will use STP 3 as the valid prototype. In support of the COLA, STPNOC will provide a final report (STP 4 Reactor Internals Flow-Induced Vibration Assessment Program, hereinafter referred to the STP 4 FIV Program Report). Because STP 4 is identical to STP 3, the test program for STP 3 will be completely

applicable to the STP 4 design. Per the guidance of RG 1.20 Rev. 3, the STP 4 FIV Program Report will include the stress and vibration analysis program and inspection program. This report is scheduled to be submitted to NRC by December 15, 2010.

Post-COL Deliverables: Preliminary and final reports documenting the results of the FIV testing will be prepared for the STP 3 prototype testing and the STP 4 non-prototype testing. The completion of the testing and the reports are dependent upon the schedule for start-up testing, which is dependent on the fuel load dates. The current schedule for STP 3 has an estimated fuel load date of early 2015. Thus the earliest estimated date for completion of the start-up testing related elements of the FIV program would be mid-2015. The schedule for STP 4 is currently one year later. Per the guidance of RG 1.20 Rev. 3 regulatory position C.2.5, the preliminary and final reports will be submitted 60 and 180 days, respectively, after the completion of the vibration testing. STPNOC provides regular updates of the schedule that can be used to track the dates for the testing completion and thus the estimated final report availability dates.

COLA Changes: In order to incorporate STP 3 as the prototype reactor and clarify that STP 4 is a non-prototype category I reactor, the following changes will be incorporated in a future revision of the COLA. The supplemental information provided in COLA Part 2, Tier 2, Subsections 3.9.2.3 and 3.9.2.4 will be revised to make the description consistent with the current approach as described in this RAI response. Subsection 3.9.2.6 will revert to the DCD section, which is incorporated by reference. The supplemental information provided in the COLA in response to COL Information Item 3.27, which requested that the results of the prototype testing be provided by the COL applicant, will be revised to provide the plan for the FIV testing of STP 3 as the prototype plant, and as such will state that STPNOC is addressing the required information to address the regulatory positions C.2.1 through C.2.4 of RG 1.20 Rev. 3 in the STP 3 FIV Assessment.

The changes to the STP 3&4 COLA are provided below. Changes to COLA Revision 3 are highlighted in gray shading.

3.9.2.3 Dynamic Response of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

The following standard supplement addresses Regulatory Guide (R.G.) 1.206, Rev. 0:

The plan to evaluate the response of the reactor internals due to operational flow transients and steady state conditions for the ABWR prototype, STP 3, is included in the STP 3 ABWR Prototype Reactor Internals Flow-Induced Vibration Assessment Program (Reference 3.9-13). For STP 4, STP 3 will be the valid prototype reactor, and STP 4 will be classified as "Non-Prototype, Category I." The plan for evaluation of STP 4 is included in the STP 4 Reactor Internals Flow-Induced Vibration Assessment Program (Reference 3.9-14).

~~The reactor internals design of the first ABWR plant, which has been in operation since 1995, is considered to be the 1350 MWe ABWR "Valid Prototype" defined in Regulatory Guide 1.20 because:~~

~~(1) The prototype ABWR plant reactor internals have successfully completed a comprehensive vibration assessment program during the pre-operational and initial startup testing. This vibration assessment program consisted of a vibration and fatigue analysis, a vibration measurement program, an inspection program, and a correlation of their results.~~

~~(2) The reactor internals of the prototype ABWR plant have experienced no adverse in-service vibration phenomena.~~

~~Also, Regulatory Guide 1.20 Section C-1.4 defines non-prototype, Category I as "a reactor internals configuration with substantially the same arrangement, design, size, and operating conditions as a specified Valid Prototype and for which nominal differences in arrangement, design, size, and operating conditions have been shown by test or analysis to have no significant effect on the vibratory response and excitation of those reactor internals important to safety." STP 3 and 4 reactor internals are substantially the same as those of the valid prototype. Also, the valid prototype has no significant effect on the vibratory response and excitation of those reactor internals important to safety. Therefore, STP 3 and 4 reactor internals are classified as "Non-Prototype, Category I" of the 1350 MWe ABWR.~~

~~From the guidance of Regulatory Guide 1.206 Section C.I.3.9.2.3 for non-prototype, a brief summary of the valid prototype test and analysis results are shown as follows.~~

~~Following the guidance of Regulatory Guide 1.20, Section C-2.1, the vibration analysis program was performed for those steady-state and anticipated transient conditions that correspond to preoperational and initial startup test and normal operating conditions. The dynamic analytical finite-element models were developed to predict the natural vibration frequency, modal displacement, and modal strain and stress for the following components.~~

~~(1) Control Rod Guide Tubes and Control Rod Drive Housings~~

~~(2) In-core Guide Tubes and Housings~~

~~(3) High Pressure Core Flooder Sparger and Coupling~~

~~(4) Core Shroud~~

~~(5) Steam Dryer Skirt, Drain Channel and Hood~~

~~From the analyses results, it was verified that the maximum vibration stress amplitudes were all below the allowable limit for all normal steady-state and transient operating conditions (including the combination of several pumps stopping).~~

~~Following the guidance of Regulatory Guide 1.20, Section C-2.2, a vibration measurement program was developed and implemented to verify the structural integrity of the reactor internals, to determine the margin of safety associated with steady-state and anticipated transient conditions for normal operation, and to confirm the results of the vibration analysis. Strain gages, accelerometers, and/or linear variable differential transformers were utilized to measure vibration related data on the following reactor internal components.~~

~~(1) Steam Dryer Skirt, Drain Channel, Support Ring, Hood and Vessel Dome Region Pressure~~

~~(2) High Pressure Core Flooder Sparger, Coupling and Thermal Ring~~

~~(3) Control Rod Guide Tube and Control Rod Drive Housing~~

~~(4) In-core Monitor Guide Tube and Housing~~

~~(5) Core Shroud~~

~~(6) Top Guide~~

~~For the selection of instrument components, the following criteria were considered:~~

~~(1) History of flow-induced vibration problems~~

~~(2) New design or new flow condition~~

~~(3) Difficulty to repair or replace~~

~~The measurements results showed that the maximum vibration stress amplitudes were all below the allowable limit for all normal steady-state and transient operating conditions (including the combination of several pumps stopping).~~

~~From the guidance of Regulatory Guide 1.20, Section C-2.3, the inspection program was implemented prior to and following operation at those steady-state and transient modes consistent with the test conditions for Regulatory Guide 1.20 Section C-2.2.2.~~

~~The reactor internals were removed from the reactor vessel for these inspections. For components in which removal was not feasible, the inspections were performed by means of examination equipment appropriate for in situ inspection.~~

~~The proposed design for STP 3 and 4 is substantially the same as the valid prototype reactor internal components. In addition, changes to the reactor internal components are not contemplated at this time. If any changes are determined necessary in the future, they will be addressed at the time the change is proposed with proper evaluation/ justification.~~

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

The following standard supplement addresses Regulatory Guide (R.G.) 1.206, Rev. 0:

As discussed in Subsection 3.9.2.3, STP 3 reactor internals are classified as Prototype, and the STP 4 reactor internals are classified as non-prototype, Category I. In accordance with the requirement of Regulatory Guide 1.206 Section C.I.3.9.2.4 for ~~non-~~prototype, ~~a brief summary of test and analysis results are shown in~~ Section 3.9.2.3 identifies the assessment program for STP 3 that addresses the flow modes, vibration monitoring and sensor types and locations, procedures and methods to be used to process and interpret the measured data, planned visual inspections, and planned comparisons of test results with analytical predictions. In addition, scale model tests will also be used for the development of the analyses of the steam dryers for acoustic loads.

For STP ~~3 and~~ 4 reactor internals components, an inspection program will be implemented in lieu of a vibration measurement program as discussed in paragraph C.3.1.3 of Regulatory Guide 1.20. Subsection 3.9.2.3 identifies the assessment program for the STP 4 non-prototype.

~~The inspection of the reactor internals shall be implemented prior to and following operation at those steady-state and transient modes including the unbalanced pump operating condition. The test operating duration shall be determined to have the reactor internal components accumulate at least 106 cycles of vibration prior to the final inspection. Also, the duration of testing shall be no less than that for the valid prototype reactor internals. This test operating duration is adequate, because the operating ABWRs have not experienced problems caused by flow-excited acoustic resonances and flow-induced vibrations, and the valid prototype measurement results show no significant responses. These flow test and inspections shall be implemented prior to fuel loading.~~

~~Testing shall be performed with the reactor internals important to safety and the fuel assemblies (or dummy assemblies that provide equivalent dynamic mass and flow characteristics) in position. The testing may be conducted without real or dummy fuel assemblies if it can be shown (by analytical or experimental means) that such conditions will yield conservative results. For the reactor internals for which removal is not feasible, the inspections shall be performed by means of examination equipment appropriate for in situ inspection.~~

~~These inspections, when completed prior to fuel load, will permit closure of that portion of Tier 1 Table 2.1.1d ITAAC #7 for as-built vessel internals. It shall be verified that the as-built vessel internals have no damage or loose parts affected by flow-induced vibration. Details of the inspection requirements shall be provided in a specification, which shall be submitted prior to pre-operational test.~~

~~Although the steam dryer design of STP 3 and 4 is identical to the valid prototype design, the configuration of the main steamline may have an influence on the steam dryer loads. Therefore, additional test and analysis requirements are voluntarily adopted in accordance with Regulatory Guide 1.20, Rev. 3. It is noted that Regulatory Guide 1.20, Rev. 2 is applicable to the ABWR, per Table 1.8-20.~~

~~Pursuant to the guidance of Regulatory Guide 1.20, Rev.3, analyses and scale model tests will be performed to address the effects of any differences in the main steam line configuration between STP 3 and 4 and the Prototype plant, specifically with respect to the steam dryer loads.~~

Also, as discussed in Regulatory Guide 1.20, Rev. 3, the main steam lines in STP 3 and 4 will be instrumented with strain gages to provide measurements of pressure fluctuations due to flow-induced vibrations. The measurements will be used by the Acoustic Circuit Methodology to analytically predict the steam dryer flow-induced vibration loads. The predicted loads will then be used with a finite-element model of the dryer to confirm the acceptability of the flow-induced vibration loads.

After the first operating cycle of STP 3 and 4, detailed inspections of the steam dryer will be performed to confirm the structural adequacy of the dryer for flow-induced vibration loads.

~~3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results~~

~~The following standard supplement addresses Regulatory Guide (R.G.) 1.206, Rev. 0:~~

~~As discussed in Section 3.9.2.3, correlation between reactor internals vibration tests and analytical results was performed for the valid prototype. The test results were compared to the analytical results, and both results showed that the maximum vibration stress amplitudes were all below the allowable limit for all normal steady-state and transient operating conditions (including the combination of several pumps stopping).~~

~~Analytical models used for these analyses were verified by comparing the calculated natural frequencies of each component with those values measured by the hammering test or alternate calculation.~~

3.9.7 COL License Information

3.9.7.1 Reactor Internals Vibration Analysis, Measurement and Inspection Program

The following standard supplement addresses COL License Information Item 3.27.

The results of the vibration assessment program for the first ABWR plant have been assessed and it was determined that the level of detail of the available information is inadequate to meet the level of information described in the guidance provided in RG 1.20 Rev. 3. Therefore, as described in Subsection 3.9.2.3, STP 3 is the prototype plant, and therefore a prototype reactor internals stress and vibration analysis, measurement and inspection program is provided. This program addresses the following regulatory positions of RG 1.20 Rev. 3:

C.2.1	Vibration and Stress Analysis Program
C.2.2	Vibration and Stress Measurement Program
C.2.3	Inspection Program
C.2.4	Documentation of Results

As described in Subsection 3.9.2.3, the STP 3 ABWR FIV Assessment Program (Ref. 3.9-13) provides the summary of the results of the vibration and stress analysis, and descriptions of the vibration and stress measurement program and inspection program. The preliminary and final reports, which together summarize the results of the vibration analysis, measurement, and inspection programs, will be submitted to the NRC within 60 and 180 days, respectively, following the completion of vibration testing in accordance with the guidance in RG 1.20 Rev. 3.

As described in Subsection 3.9.2.3, STP 4 is considered a non-prototype category 1 plant. Based on the guidance of RG 1.20 Rev. 3 regulatory position C.3, the STP 4 FIV Assessment Program (Ref. 3.9-14) provides the summary of the results of the vibration and stress analysis, and description of the inspection program. The preliminary and final reports, which together summarize the results of the vibration analysis and inspection programs, will be submitted to the NRC within 60 and 180 days, respectively, following the completion of inspection program in accordance with the guidance in RG 1.20 Rev. 3.

~~The results of the vibration assessment program for the valid prototype ABWR internals are shown in Section 3.9.2.3. In addition, a vibration assessment program summarized in Section 3.9.2.4 will be implemented for non-prototype, Category I ABWR.~~

3.9.8 References

3.9-13 "STP 3 ABWR Prototype Reactor Internals Flow-Induced Vibration Assessment Program," WCAP-17256.

3.9-14 "STP 4 Reactor Internals Flow-Induced Vibration Assessment Program," WCAP-17257.

RAI 03.09.02-10

QUESTION:

Supplement to Question 11649 (eRAI 03.09.02-3)

RAI 03.09.02-3 requested STP for the following:

> *"(1) A tabulation of all reactor internals components and local areas to be inspected. A description of the inspection procedure including the method of examination, method of documentation, provisions for access to the reactor internals, and the criteria which will be applied. The applicant should also discuss what actions will be taken as a result of these inspections.*
>
> *(2) In addition, the SRP recommends that walkdown inspections of the steam, feedwater, and condensate systems take place during hold points in the testing. The applicant should provide details of the planned walkdowns, what monitoring and testing equipment is required, and what actions will be taken as a result of these inspections."*

STP's response referenced the Toshiba report RS-5126579, Revision 1. Our review indicated that this report does not meet regulatory positions C.2.2(2)(a) and C.2.3 in Regulatory Guide 1.20. Please consider these regulatory positions and update the reports.

RESPONSE:

As stated in the response to RAI 03.09.02-9, STPNOC is revising its approach to the reactor internals flow-induced vibration (FIV) program and will make STP 3 the ABWR prototype reactor. As also noted in that response, the response to RAI 03.09.02-3 is completely superseded by the revised approach and the FIV test reports previously provided for review are being superseded by new reports for the STP 3 prototype and STP 4 non-prototype Category I reactors. These new reports will provide information regarding the vibration and stress measurement program, including the tabulation of the components and local areas to be inspected, as described in the response to RAI 03.09.02-9. These reports will also include a description of what is to be included in the inspection procedure.

The response to RAI 03.09.02-9 also provides the schedule for submittal of the FIV program reports.

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

RAI 03.09.02-11

QUESTION:

Supplement to Question 11650 (eRAI 03.09.02-4)

In the response to RAI 03.09.02-4, STP advised that the results of the initial acoustic screening analysis and confirmatory scale model testing are scheduled to be completed by December 2010. This schedule is not acceptable since it is beyond the Phase 4 schedule.

Please provide an updated schedule.

RESPONSE:

As stated in the response to RAI 03.09.02-9, STPNOC is revising its approach to the reactor internals flow-induced vibration (FIV) program and will make STP 3 the ABWR prototype reactor. As stated in the response to RAI 03.09.02-9, that response completely supersedes the response to RAI 03.09.02-4. That response includes the schedule for submittal of the associated reports in support of the STP 3 prototype analysis and tests. The initial acoustic screening analysis submittal schedule is also provided in that response.

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

RAI 03.09.02-12

QUESTION:

Supplement to Question 11651 (eRAI 03.09.02-5)

In RAI 03.09.02-5, the staff requested more detailed discussion on the acoustic circuit methodology. STP's response referenced Chapter III of the Toshiba report RS-5126579, Revision 1, and advised that the evaluation methodology is in compliance with EPRI report BWRVIP-194, which is currently under review by the NRC. Our review indicated that the Toshiba report does not have sufficient details. Please resubmit the Toshiba report including the detailed discussion of the acoustic circuit methodology.

RESPONSE:

As stated in the response to RAI 03.09.02-9, STPNOC is revising its approach to the reactor internals flow-induced vibration (FIV) program and will make STP 3 the ABWR prototype reactor. As also noted in that response, the steam dryer predictive analysis will be developed using the acoustic circuit model (ACM) methodology. As stated in the response to RAI 03.09.02-9, that response completely supersedes the response to RAI 03.09.02-5.

As noted in the response to RAI 03.09.02-9, the FIV program steam dryer high cycle fatigue analysis report is currently planned to be submitted to NRC on December 15, 2010. This report will include analysis of acoustic resonance of the steam dryer using the ACM methodology, and will include a discussion of the ACM methodology and its application for the STP analysis.

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

RAI 03.09.02-13

QUESTION:

Supplement to 11652 (eRAI 03.09.02-6)

In RAI 03.09.02-6, the staff requested the applicant to instrument the steam dryer to verify the analytically predicted loads. In the response, STP referenced the Toshiba report RS-5126579, Revision 1, and credited the successful operating experience of K-6, the valid prototype plant. Please note that our review of RS-5126579, revision 1 indicated that it does not have the level of details described in Regulatory Guide 1.20. In addition, STP Units 3 and 4's steam line configuration is not the same as K-6. Please address the differences in configuration and either re-consider providing instrumentation for the steam dryer or provide a better justification.

RESPONSE:

As stated in the response to RAI 03.09.02-9, STPNOC is revising its approach to the reactor internals flow-induced vibration (FIV) program and will make STP 3 the ABWR prototype reactor. As such, the difference between the K-6 main steam lines and the STP 3&4 main steam lines noted in the RAI is no longer an issue as the STP-3&4 Comprehensive Vibration Assessment Program Report will address the STP-3&4 main steam line layout. As stated in the response to RAI 03.09.02-9, the response to RAI 03.09.02-6 is completely superseded. The measurement plan for the prototype reactor internals, which includes the instrumentation information for the steam dryer, will be described in the STP 3 ABWR FIV Assessment Program, as discussed in the response to RAI 03.09.02-9.

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

RAI 03.09.02-14

QUESTION:

Supplement to Question 11653 (eRAI 03.09.02-7)

In ABWR FSAR, Tier 2, Section 3.9.7.1, Reactor Internals Vibration Analysis, Measurement and Inspection Program, it states that the first COL applicant will provide, at the time of the application, the results of the vibration assessment program for the ABWR prototype internals. NRC review and approval of the results, as specified in Regulatory Guide 1.20, will complete the vibration assessment program provision for prototype reactor internals. In addition to this information, the first COL applicant will provide the information on the schedule in accordance with position C.3 of Regulatory Guide 1.20. The staff's review of the FSAR section 3.9.2.3 and 3.9.2.4 did not include information on the schedule. In accordance with Regulatory Guide 1.20, the staff requests the applicant to provide a comprehensive schedule which includes the prototype test report and testing of the steam dryer.

The staff requested the above in RAI 03.09.02-7. STP's response referenced Toshiba report RS-5126954, Revision 1, and RS-5126579, Revision 1. The staff has reviewed these reports and did not\consider these reports meet the guidance in Regulatory Guide 1.20. Please update these reports and provide more details as described in the regulatory guide.

RESPONSE:

As stated in the response to RAI 03.09.02-9, STPNOC is revising its approach to the reactor internals flow-induced vibration (FIV) program and will make STP 3 the ABWR prototype reactor. That response also provides the schedule for the deliverables and completely supersedes the response to RAI 03.09.02-7.

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

RAI 03.09.02-15

QUESTION:

Supplement to Question 11654 (eRAI 03.09.02-8)

ITAAC #7 of DCD Tier 1, Table 2.1.1d states that a vibration type test will be conducted on the prototype RPV internals of an ABWR, and that a flow test and post-test inspection will be conducted on the as-built RPV internals. Since the prototype has been identified as the 1350 MWe ABWR, the staff requests that the prototype vibration assessment report be made available to be reviewed by the staff. The staff also requests the applicant to explain how the remaining portion of the ITAAC for the as-built RPV internals will be resolved. It should be noted that a COL applicant should submit the results from the vibration assessment program for the RPV internals in accordance with Regulatory Guide 1.20. STP's response to this RAI referenced the Toshiba report RS-5126954, Revision 1, which we did not consider adequate. Please update this report to meet the guidance in Regulatory Guide 1.20.

RESPONSE:

As stated in the response to RAI 03.09.02-9, STPNOC is revising its approach to the reactor internals FIV program and will make STP 3 the ABWR prototype reactor. As also noted in the reference response, the associated FIV test programs for the STP 3 prototype reactor internals and STP 4 reactor internals are being provided to NRC.

The ITAAC cited in the RAI states:

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. The RPV internals withstand the effects of FIV.	7. A vibration type test will be conducted on the prototype RPV internals of an ABWR.	7. A vibration type test report exists and concludes that the prototype RPV internals have no damage or loose parts as a result of the vibration type test.
	A flow test and post-test inspections will be conducted on the as-built RPV internals	The as-built RPV internals have no damage or loose parts.

The first ITA (Inspection, Test, Analyses) and acceptance criterion for this ITAAC apply for the preoperational prototype testing, as described in FSAR Subsection 14.2.12.1.52. The STP 3 FIV Assessment Program described in the response to RAI 03.09.02-9 includes a description of the vibration and stress measurement and inspection programs. The resulting report for the pre-operational testing portion of the STP 3 FIV program, which will document the test and

inspection results and the acceptability of the reactor internals, would be used for closure of this first ITA.

The second ITA and acceptance criterion for this ITAAC apply for the as-built RPV internals, and as such apply to both STP 3&4. The closure of this ITA for STP 3 is accomplished by completion of the first ITA. For STP 4, the STP 4 FIV Assessment Program described in the response to RAI 03.09.02-9 includes the inspection program. The resulting report for the pre-operational testing portion of the STP 4 FIV program, which will document the inspection results and the acceptability of the reactor internals, would be used for closure of this second ITA.

Note that the STP 3 and STP 4 FIV Assessment Programs also include testing that is to be performed during startup, consistent with the guidance of RG 1.20 Rev. 3, as described in FSAR Subsection 14.2.12.1.12. Such startup testing is not a subject of this ITAAC. The startup FIV testing requires fuel to be loaded and, as explained in DCD Subsection 14.3.2.2, the ITAAC do not include any inspections, tests, or analyses that are dependent upon conditions that only exist after fuel load. ITAAC by regulation are completed prior to issuance of the 10 CFR 52.103(g) finding. Since it is not possible to perform the startup testing prior to the 103(g) finding, the startup FIV testing is not part of the ITAAC requirement.

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

RAI 14.02-6, Revision 1

QUESTION:

The COL applicant supplemented FSAR Subsections 14.2.12.1.2 and 14.2.12.1.52 by deleting NEDO 33316, "Advanced Boiling Water Reactor (ABWR) Vibration Assessment Program in compliance with RG 1.20," and replacing it with a reference to Subsections 3.9.2.3 and 3.9.2.4 in the STP Units 3 and 4 FSAR. The information in STP FSAR Subsections 3.9.2.3 and 3.9.2.4 is not sufficient to provide reasonable confidence that these two preoperational tests for the Reactor Recirculation System and the Reactor Vessel Flow-Induced Vibration System will satisfy the NRC regulations.

For example, Criterion XI of Appendix B to 10 CFR Part 50 requires that a test program be established to ensure that all testing required to demonstrate that structures, systems, and components (SSCs) will perform satisfactorily in service is identified and performed in accordance with written test procedures that incorporate the requirements and acceptance limits in applicable design documents. The test program should include, as appropriate, proof tests before installation, preoperational tests, and operational tests during plant operation of SSCs. Test procedures should include provisions for ensuring that all prerequisites for the given test have been met, adequate test instrumentation is available and used, and the test is performed under suitable environmental conditions. Test results should be documented and evaluated to ensure that test requirements have been satisfied. The staff requests a comprehensive test program for these two test abstracts be submitted to the NRC for review.

REVISED RESPONSE (Revision 1):

The original response to this RAI was submitted with STPNOC letter U7-C-STP-NRC-090089, dated July 29, 2009. This revised response reflects the updated STP approach for the reactor internals flow induced vibration (FIV) program. This revised response replaces the original response in its entirety.

DCD Subsections 14.2.12.1.2 and 14.2.12.1.52 are incorporated by reference without any departures. DCD Subsection 14.2.12.1.52 explains that "This testing will fulfill the preoperational requirements of Regulatory Guide 1.20 for a vibration measurement and inspection program for prototype reactor internals, and applies only to the ABWR designated for testing of "prototype" reactor internals." The supplementary information referencing NEDO-33316 in the initial COLA (Revision 0), and referencing COLA Part 2, Tier 2, Subsections 3.9.2.3 and 3.9.2.4 in COLA Revision 3, were intended to support designation of STP 3 and STP 4 as Category I, non-prototype plants. As explained in the response to RAI 03.09.02-9, STP 3 is now designated as the prototype ABWR plant in accordance with the guidance in Regulatory Guide 1.20, Revision 3. STP 4 is a Category I, non-prototype plant. For each of these units, reports will be submitted to the NRC that summarize the analytical models and validation and predictive analysis results for the steam dryer, lower plenum components, and all other reactor internal components, including a summary of the measurement and inspection plans for STP 3 and the inspection plan for STP 4. Activities to be performed post-COL through

testing at power will be described. Additional details regarding the content of the FIV assessment program reports for STP 3&4, and the schedule for submittal of these reports are provided by STPNOC in the response to RAI 03.09.02-9.

The details of the FIV program for STP 3&4 as noted above also will be reflected in revised Tier 2 Subsections 14.2.12.1.2 and 14.2.12.1.52, as shown below. The supplement provided in Subsection 14.2.12.1.2 will be deleted, because DCD Subsection 14.2.12.1.2 already references Subsection 14.2.12.1.52, which will be revised to reflect this response. Changes from COLA Revision 3 are shown with gray shading.

14.2.12.1.2 Reactor Recirculation System Preoperational Test

~~The following supplement augments that provided by this subsection.~~

~~For STP 3 & 4 reactor internals testing requirements reference Tier 2 Subsections 3.9.2.3 and 3.9.2.4.~~

14.2.12.1.52 Reactor Vessel Flow-Induced Vibration Preoperational Test

The following supplement augments that provided by this subsection.

~~For STP 3 & 4 reactor internals testing requirements reference Tier 2 Subsections 3.9.2.3 and 3.9.2.4.~~

STP 3 is designated as the prototype ABWR plant in accordance with the guidance in Regulatory Guide 1.20, Revision 3. STP 4 is considered a Category I, non-prototype plant.

For STP 3, the report provided in Reference 3.9-13 summarizes the analytical portion of the program in terms of maximum vibrational response levels of overall structures and components and the measurement and inspection plans.

For STP 4, Reference 3.9-14 summarizes the analytical models and validation and predictive analysis results for the reactor internals, and includes the inspection plan.

RAI 14.02-8, Revision 1

QUESTION:

The COL applicant supplemented FSAR Subsection 14.2.12.2. 12 by deleting NEDO 33316, "Advanced Boiling Water Reactor (ABWR) Vibration Assessment Program in compliance with RG 1.20," and replacing it with a reference to Subsections 3.9.2.3, 3.9.2.4 and 3.9.2.6 in the STP Units 3&4 FSAR. The information in STP Unit 3&4 FSAR Subsections 3.9.2.3, 3.9.2.4 and 3.9.2.6 is not sufficient to provide reasonable assurance that startup testing for Reactor Internal Vibration will satisfy the NRC regulations. For example, Criterion XI of Appendix B to 10 CFR Part 50 requires, in part, that a test program be established to ensure that all testing required to demonstrate that structures, systems, and components (SSCs) will perform satisfactorily in service is identified and performed in accordance with written test procedures that incorporate the requirements and acceptance limits in applicable design documents. The test program should include, as appropriate, operational tests during plant operation of SSCs. Test procedures should include provisions for ensuring that all prerequisites for the given test have been met, adequate test instrumentation is available and used, and the test is performed under suitable environmental conditions. Test results should be documented and evaluated to ensure that test requirements have been satisfied. The staff requests a comprehensive startup test program be submitted to the NRC for review.

REVISED RESPONSE (Revision 1):

The original response to this RAI was submitted with STPNOC letter U7-C-STP-NRC-090089, dated July 29, 2009. This revised response reflects the updated STP approach for the reactor internals flow induced vibration (FIV) program. This revised response replaces the original response in its entirety.

DCD Subsection 14.2.12.2.12 is incorporated by reference without any departures. DCD Subsection 14.2.12.2.12 explains that "The extent to which reactor internals vibration testing is conducted during the power ascension phase is dependent on the classification of the reactor internals as prototype or not in accordance with Regulatory Guide 1.20 ..." The supplementary information referencing NEDO 33316 in the initial COLA (Revision 0), and referencing COLA Subsections 3.9.2.3, 3.9.2.4 and 3.9.2.6 in COLA Revision 3, were intended to support designation of both STP 3 and STP 4 as Category I, non-prototype plants. As explained in the response to RAI 03.09.02-9, STP 3 is now designated as the prototype ABWR plant in accordance with the guidance in Regulatory Guide 1.20, Revision 3. STP 4 is a Category I, non-prototype plant. For each of these units, reports will be submitted to the NRC that summarize the analytical models and validation and predictive analysis results for the steam dryer, lower plenum components, and all other reactor internal components, including a summary of the measurement and inspection plans for STP 3 and the inspection plan for STP 4. Activities to be performed post-COL through testing at power will be described. Additional details regarding the content of the FIV assessment program reports for STP 3&4, and the

schedule for submittal of these reports is provided by STPNOC in the response to RAI 03.09.02 9.

The details of the FIV program for STP 3&4 as noted above will be reflected in revised Part 2, Tier 2 Subsection 14.2.12.2.12, as shown below. Changes from COLA Revision 3 are shown with gray shading.

14.2.12.2.12 Reactor Internals Vibration

The following supplement augments that provided by this subsection.

~~For STP 3 & 4 reactor internals vibration assessment program reference Tier 2 Subsections 3.9.2.3, 3.9.2.4, and 3.9.2.6.~~

STP 3 is designated as the prototype ABWR plant in accordance with the guidance in Regulatory Guide 1.20, Revision 3. STP 4 is a Category I, non-prototype plant.

For STP 3, Reference 3.9-13 summarizes the analytical models, predictive analysis results, and the measurement and inspection plans.

For STP 4, Reference 3.9-14 summarizes the analytical models and predictive analysis results, and includes the inspection plan.