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Subject: **Submittal of ABWR Licensing Topical Report (LTR) NEDO-33316 "Advanced Boiling Water Reactor (ABWR) Vibration Assessment Program in compliance with The United States Nuclear Regulatory Commission Regulatory Guide 1.20"**

Reference: Letter MFN 017-97, J. Quirk to NRC, *ABWR Design Control Document, Revision 4*, dated March 28, 1997, Docket No. 52-001

The enclosed Licensing Topical Report (LTR) is submitted for NRC generic review and approval as a Combined License (COL) license information item as required by the current ABWR certified design (referenced), Docket No. 52-001. The regulatory basis for this submittal is discussed below.

This is the seventh of a number of ABWR-related LTRs GE plans to submit and which have been discussed in South Texas Project 3&4 project meetings with the NRC. In support of the ABWR Design Centered Working Group (DCWG) plans, GE requests a generic review and approval of the subject LTR in advance of any future combined license applications (COLA) submittals. Note that the submittal is the result of design detailing performed for ABWRs in the US and in Asia and provides for the generic resolution of a COL license information item, thereby contributing to standardization.

This LTR is submitted in response to DCD Tier 2, Section 3.9.7.1, COL license information item – 3.27. The information contained in this LTR is typical for an ABWR Vibration Assessment Program. It should be noted that plant specific information as

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required by COL license information item - 3.27 will be provided following preoperational testing. The plant specific information will provide assessment results in accordance with the applicable portions of position C.3 of Regulatory Guide 1.20 for non-prototype internals.

The enclosure contains no information that GE considers proprietary although full copyright protection applies.

If you have any questions about the information provided here, please contact me at 910-602-1885.

Sincerely,

Joseph A Savage
Project Manager, ABWR Licensing

Enclosure: NEDO-33316 "*Advanced Boiling Water Reactor (ABWR) Vibration Assessment Program*" April 2007 – Non-Proprietary

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Revision 0

LICENSING TOPICAL REPORT

Advanced Boiling Water Reactor (ABWR) Vibration Assessment Program in compliance with The United States Nuclear Regulatory Commission Regulatory Guide 1.20

INFORMATION NOTICE

This document, NEDO-33316, Revision 0, contains no proprietary information.

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

PLEASE READ CAREFULLY

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Summary

The Advanced Boiling Water Reactor (ABWR) Design Control Document (DCD) specifies that the ABWR reactor internals vibration assessment program will comply with the U.S. Nuclear Regulatory Commission Regulatory Guide 1.20, "Comprehensive Vibration Assessment program for Reactor Internals During Preoperational and Initial Startup Testing."

Specifically, the DCD requires the first Combined Operating License (COL) applicant for the ABWR to provide the test results required by RG 1.20 for the ABWR Valid Prototype reactor. This Licensing Topical Report (LTR) references the comprehensive vibration assessment program results, including test results, for the ABWR Valid Prototype reactor and is intended to be referenced by the first ABWR COL applicant.

This LTR provides the basis for closure of that portion of Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) #7 of DCD Tier 1 Table 2.1.1d that pertains to prototype testing.

Based on Regulatory Guide 1.20 classification, non-prototype subsequent ABWRs are classified as having "Non-Prototype, Category I" reactor internals of the Valid Prototype Standard 1350 MWe Plant. For Non-Prototype, Category I ABWR units, the vibration assessment program consists of (1) a vibration analysis program and (2) an inspection program prior to and following the pre-operational tests. This program complies with Regulatory Guide 1.20.

These inspections, when completed prior to fuel load, will permit closure of that portion of ITAAC #7 for as-built vessel internals. Details of the inspection requirements are provided in LTR-NEDO-33309, Licensing Topical Report ABWR Pre-operational Test Specification.

1.0 Introduction

The non-prototype subsequent ABWR units will have the same reactor internals design as the two ABWR plants, which have been in operation since 1995 and 1996, respectively. The internals design of the first operating ABWR plant qualifies as the valid prototype design as described in the DCD. The second operating plant design was successful in flow testing and inspection as a non-prototype design.

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 analysis, a vibration measurement, an inspection, and a correlation of their results. Also, since its initial operation in 1995, the prototype ABWR plant reactor internals have experienced no adverse inservice vibration phenomena. Having satisfied these two requirements, the prototype ABWR plant reactor internals configuration is considered to be the 1350 MWe ABWR "Valid Prototype" as classified in Regulatory Guide 1.20. A GE report, NEDC-32780P, "Flow-Induced Vibration of

ABWR Reactor Internals," (Reference 2.2.6.1) documents the evaluation of the vibration test data collected in testing of the prototype internals.

The non-prototype subsequent ABWRs and the non-prototype operating ABWR plant reactor internal configurations are identical to that of the prototype ABWR plant in arrangement, design, size and operating conditions. With the existence of the prototype ABWR plant as a "Valid Prototype", the non-prototype ABWR plant reactor internal configurations qualify as "Non-Prototype, Category I" as classified in Regulatory Guide 1.20. For "Non-Prototype, Category I" reactor internals, Regulatory Guide 1.20 requires a comprehensive vibration assessment program consisting of (1) a vibration analysis and (2) either a vibration measurement or a full inspection prior to and following pre-operational and initial startup testing. For non-prototype subsequent ABWRs, the analyses performed for the prototype ABWR plant are applicable. An inspection program will be conducted for non-prototype subsequent ABWRs as specified by Regulatory Guide 1.20 during the pre-operational testing.

Thus, with the implementation of the inspection program, together with the extensive vibration assessment programs conducted for the prototype ABWR plant, and the successful operating experience with the operating ABWR plants, non-prototype subsequent ABWRs reactor internals meet requirements of Regulatory Guide 1.20 as specified in the DCD.

Regulatory Guide 1.20 (Revision 2, May 1976) is divided into 4 main sections labeled A ("Introduction"), B ("Discussion"), C ("Regulatory Position") and D ("Implementation"). The guidelines against which compliance can be measured are concentrated in Section C. Furthermore, non-prototype subsequent ABWRs are classified as "Non-Prototype, Category I" plants of the 1350 MWe ABWR. Therefore, only the portions of Section C of the Regulatory Guide that apply to "Non-Prototype, Category I" will be addressed herein.

2.0 Vibration Assessment Program

In this section, the features of the non-prototype subsequent ABWRs vibration assessment program will be discussed in detail and compared against Regulatory Guide 1.20 guidelines on a section-by-section basis.

2.1 *Section C.1, "Classification of Reactor Internals Relative to the Comprehensive Vibration Assessment Program"*

Non-prototype subsequent ABWRs are classified as "Non-Prototype, Category I" plants of the 1350 MWe ABWR by virtue of the fact that they meet this definition of Section C.1.4: "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."

Sections C.1.1 and C.2 apply to the operating prototype ABWR plant. Section C.3 (Non-Prototype), as it applies to non-prototype subsequent ABWRs, is discussed below.

2.2 Section C.3, "Comprehensive Vibration Assessment Program for Non-Prototype Reactor Internals"

Section C.3 states that Non-Prototype reactor internals important to safety should be subjected during the pre-operational and initial startup test program to all significant flow modes associated with normal steady state and anticipated transient operation under the same test conditions imposed on the applicable Prototype. Evaluation of the effects of such operation on the structural integrity of the Non-Prototype reactor internals should be based on the results of a comprehensive vibration assessment program for "Non-Prototype Category I".

This vibration assessment program should consist of (1) a vibration analysis, and (2) a vibration measurement program or an inspection program. The specific guidelines for each of these elements are addressed further in the following subsections.

2.2.1 Section C.3.1.1, "Vibration Analysis Program"

This section requires that the applicable "Valid Prototype" should be specified and sufficient evidence should be provided to support the classification as "Non-Prototype, Category I". Also, the vibration analysis for the Valid Prototype, which includes a summary of the anticipated structural and hydraulic responses and test acceptance criteria, should be modified to account for the nominal differences that may exist between the Non-Prototype, Category I and the Valid Prototype reactor internals.

A report, NEDC-32780P, "Flow Induced Vibration of ABWR Reactor Internals", is prepared that satisfies this analysis provision for the Valid Prototype. As mentioned above, the non-prototype subsequent ABWRs internals are identical to those in the operating prototype and non-prototype plants.

2.2.2 Section C.3.1.2, "Vibration Measurement Program"

This section specifies that for "Non-Prototype, Category I" plant the vibration measurement program may be omitted if the inspection program is implemented.

For non-prototype subsequent ABWRs, an inspection program will be implemented and is discussed in the next subsection.

2.2.3 Section C.3.1.3, "Inspection Program"

This subsection specifies that for a "Non-Prototype, Category I" plant, if an inspection program is implemented in lieu of a vibration measurement program, the guidelines for the inspection program delineated in regulatory position for "Prototype" reactor internals should be followed. An inspection of internals should be performed before and after operation at steady state and transient modes, per subsections C.2.3 and C.2.2.2. These test operating conditions should include:

a. "All steady-state and transient modes of operation."

b. "The planned duration of all testing in normal operating modes to ensure that each critical component will have been subjected to at least 10^6 cycles of vibration (i.e., computed at the lowest frequency for which the component has a significant structural response) prior to the final inspection of the reactor internals. The duration of testing for non-prototype reactor internals should be no less than that for the applicable reference reactor internals (i.e., Valid, Conditional, or Limited valid Prototype)."

c. "Disposition of fuel assemblies. (Testing should be performed with the reactor internals important to safety and the fuel assemblies (or dummy assemblies which provide equivalent dynamic mass and flow characteristics) in position. The test 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 non-prototype subsequent ABWRs, a wide variety of steady state and transient operating conditions that might affect flow-induced vibration will be simulated during the tests. This will include numerous pump operating configurations and pump trip combinations. Non-prototype subsequent ABWRs test operations at or above 100% rated core flow for over 51 hours (the required duration of testing for the prototype reactor internals) shall be performed to assure that at least 10^6 cycles of vibration are accumulated. For those portions of the reactor internals, where appropriate and conservative, testing and inspection will be performed without fuel in the reactor, prior to fuel load. Otherwise, testing will be performed with real or dummy fuel assemblies in the reactor, as required. If testing and inspection is performed without fuel in the reactor, the Final Safety Analysis Report will be updated in accordance with 10 CFR 50.71(e) to describe an evaluation that demonstrates that such conditions yield conservative results. Such evaluation will be based on test and/or finite element analysis, as appropriate. The acceptance criteria will be to ensure that the expected reactor internals equipment flow induced vibration stress amplitudes are at or below the acceptable fatigue limit stress amplitude for the material.

Section C.2.3 further states that the reactor internals should be removed from the reactor vessel for inspection. If removal is not feasible, the inspections should be performed by means of examination equipment appropriate for in-situ inspection. It lists specific provisions as:

1. "A tabulation of all reactor internal components and local areas to be inspected, including:"

   a. "All major load-bearing elements of the reactor internals relied upon to retain the core support structure in position."

   b. "The lateral, vertical, and torsional restraints provided within the vessel."

   c. "Those locking and bolting components whose failure could adversely affect the structural integrity of the reactor internals."

d. "Those surfaces that are known to be or may become contact surfaces during operation."

e. "Those critical locations on the reactor internal components as identified by the vibration analysis."

f. "The interior of the reactor vessel for evidence of loose parts or foreign material."

2. "A tabulation of specific inspection areas that can be used to verify segments of the vibration analysis and measurement program."
3. "A description of the inspection procedure, including the method of examination (e.g. visual and nondestructive surface examination), method of documentation, access provisions on the reactor internals, and specialized equipment to be employed during the inspections to detect and quantify evidence of the effects of vibration."

   An inspection plan and procedure will be developed to include all of these details for non-prototype subsequent ABWRs reactor internals inspection program.

2.2.4 Section C.2.4, "Documentation of Results"

This section specifies that a summary of results should be submitted in the form of preliminary and final reports as follows:

The preliminary report should summarize an evaluation of the raw and, as necessary, limited processed data and the results of the inspection program with respect to the test acceptance criteria. Anomalous data that could bear on the structural integrity of the internals should be identified in the report, as should the method to be used for evaluation of such data.

The non-prototype subsequent ABWRs Inspection Reports will follow the stated guidelines for the content of the preliminary report and will be prepared after completion of the pre-operational testing.

The final report will include the following items:

a. "A description of any deviations from the specified measurement program and/or inspection programs, including instrumentation reading and inspection anomalies, instrumentation malfunctions, and deviations from the specified operating conditions."

b. "A comparison between the measured and analytically determined modes of structural and hydraulic response (including those parameters from which the input forcing function is determined) for the purpose of establishing the validity of the analytical technique."

c. "A determination of the margin of safety associated with normal steady-state and anticipated transient operation."

d. "An evaluation of measurements that exceeded acceptable limits not specified as test acceptance criteria or of observations that were unanticipated and the disposition of such deviations."

The "Flow Induced Vibration of ABWR Reactor Internals" report, NEDC-32780P, prepared for the operating prototype ABWR plant, follows the above stated final report guidelines to document the results of the vibration measurement program. As mentioned above, the non-prototype subsequent ABWRs program will be the inspection program in lieu of the vibration test program. Therefore, NEDC-32780P report and the STP 3 and 4 Inspection Report mentioned above will constitute the non-prototype subsequent ABWRs final report.

Section C.2.4.3 specifies that, if (a) inspection of the reactor internals reveals defects, evidence of unacceptable motion, excessive or undue wear, (b) the results from the measurement program fail to satisfy test acceptance criteria, or (c) the results from the analysis, measurement, and/or inspection programs are inconsistent, the final report should include an evaluation and description of planned modifications or actions in order to justify the structural adequacy of the reactor internals. This provision will be followed in the non-prototype subsequent ABWRs inspection programs in the event that unacceptable results are found.

Section C.2.4.4 further specifies that the collection, storage, and maintenance of all records relevant to the vibration assessment program should be in accordance with the acceptable Quality Assurance Program. All records that are part of the non-prototype subsequent ABWR vibration assessment program discussed in this report will be stored in accordance with the non-prototype subsequent ABWR Quality Assurance Program.

2.2.5 Section C.2.5, "Schedule"

This section specifies that a schedule should be established and submitted during the construction permit phase. It is to provide for classifying the reactor internals status as "Prototype", "Valid Prototype" or "Non Prototype" and agreeing on scope of the vibration assessment program at the beginning of the project.

Also, this schedule should provide a description of the vibration measurement and/or inspection programs, and should be submitted sufficiently before the program begins in order to allow a 90-day period for review and comment.

This section also specifies that the preliminary report be submitted within 60 days of the completion of testing, and that the final report be submitted within 180 days of the completion of testing.

The USNRC has designated this activity in the ABWR DCD as a combined operating license (COL) information item as shown in section 3.9.7.1 to be followed during the construction phase and after the pre-operational tests. This assessment and test plan report and the "Flow Induced Vibration of ABWR Reactor Internals" report, NEDC-32780P, are submitted to satisfy the COL information requirement. The final inspection reports for each unit will be made available for NRC inspection after the pre-operational tests of non-prototype subsequent ABWR are completed in accordance with ITAAC #7.

2.2.6 References

2.2.6.1 Flow-Induced Vibration of ABWR Reactor Internals, NEDC-32780P, December 2003 (GE Proprietary).

3.0 Conclusion

From the above, it can be concluded that the Vibration Assessment Program for non-prototype subsequent ABWRs has incorporated all of the necessary steps to comply with the provisions of Regulatory Guide 1.20.

With the non-prototype subsequent ABWR inspection program, the prototype ABWR plant extensive vibration testing and assessment program, and the successful prototype ABWR plant and the non-prototype ABWR plant operating experience, it can be concluded that the non-prototype subsequent ABWR Vibration Assessment Program is in compliance with Regulatory Guide 1.20.