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March 30, 2009

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-09092

Subject: MHI's Amended Response to US-APWR DCD RAI No.70 Revision 0

References: 1) "MHI's Responses to US-APWR DCD RAI No.70 Revision 0 (MHI Ref: UAP-HF-08207)" dated September 25, 2008.
2) "Request for Additional Information No. 70 Revision 0, SRP Section: 14.02– Initial Plant Test Program – Design Certification and New License Applicants, Application Section: 14.2," dated September 8, 2008.
3) "Request for Additional Information No. 271 Revision 0, SRP Section: 14.02 – Initial Plant Test Program – Design Certification and New License Applicants, Application Section: 14.2 Initial Test Program" dated March 10, 2009.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Amended Response to Request for Additional Information No.70 Revision 0."
Enclosed is the amended response to Question 14.02-87 that is contained within Reference 2. The previous response (Reference 1) is replaced with this amended response that is developed to reflect the recommendation in RAI No.271 (Reference 3).

Please contact Dr. C. Keith Paulson, Senior Technical Manager, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of the submittals. His contact information is below.

Sincerely,

Y. Ogata

Yoshiki Ogata,
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

Enclosure:
1. Amended Response to Request for Additional Information No.70 Revision 0

CC: J. A. Ciocco
C. K. Paulson

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NRO

Contact Information

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Enclosure 1

UAP-HF-09092
Docket No. 52-021

Amended Response to Request for Additional Information No. 70
Revision 0

March 2009

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

3/30/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 70 REVISION 0**
SRP SECTION:	**14.02 – Initial Plant Test Program – Design Certification and New License Applicants**
APPLICATION SECTION:	**14.2**
DATE OF RAI ISSUE:	**9/8/2008**

QUESTION NO.: 14.02-87

RAI 1152, Question 3969 addresses MHI's original response to RAI 550, Question 1895. RAI 550, Question 1895 was originally forwarded to MHI as RAI No. 27, Question No. 14.02-9. The subject of RAI No. 27, Question No. 14.02-9 was MHI's designation of the Natural Circulation Test documented in DCD Subsection 14.2.12.2.3.9 as a "First-Plant-Only Test."

The NRC staff does not agree with MHI's response to RAI No. 27, Question No. 14.02- 9. MHI's response reiterates that natural circulation testing for the US-APWR is a first plant- only test. On the basis stated below, the NRC staff requests, once more, that MHI revise the relevant portions of the US-APWR DCD to require natural circulation testing to be conducted on every plant.

The primary objectives of a suitable initial test program are to (1) provide additional assurance that the facility has been adequately designed and, to the extent practical, to validate the analytical models and verify the correctness or conservatism of assumptions used to predict plant response to anticipated transients and postulated accidents, and (2) to provide assurance that construction and installation of equipment in the facility have been accomplished in accordance with design. Other key objectives are to familiarize the plant's operating and technical staff with the operation of the facility and to verify by trial use, to the extent practical, that the facility operating procedures and emergency procedures are adequate.

Appendix A, 4.t, of RG 1.68 recommends the performance of natural circulation tests of the reactor coolant system to confirm that the design heat removal capability exists, or to verify that flow (without pumps) or temperature data are comparable to prototype designs for which equivalent tests have been successfully completed.

In Westinghouse's AP1000 design, natural circulation heat removal to cold conditions using the steam generators is not relied upon as a safety-related design feature unlike current PWR plants. Because of its "passive" nature, natural circulation in the AP1000 is achieved through the passive residual heat removal (PRHR) system. Because the PRHR heat exchanger serves as the safety-related heat sink for the AP1000 design, the staff determined that natural circulation testing through the PRHR, rather than the reactor coolant system and steam generators, met the intent of Appendix A, 4.t, of RG 1.68.

Regarding the US-APWR design, the staff notes that natural circulation is not a design feature in the US-APWR design, nor does it constitute a new, or unique design feature as confirmed by MHI's original response to RAI 550, Question 1895. In traditional PWR designs, natural circulation test demonstration not only provides benchmark data (temperature, flow) for the operator simulator, but also allows for operator training and familiarity of plant behavior under these unique circumstances. This is consistent with TMI Action Item I.G.1, that requires the development and implementation of procedures and training programs during the low-power testing phase of the initial test program. The goal of these programs is to increase the capability of shift crews to operate facilities in a safe and competent manner by assuring that the training for plant changes and offnormal events was conducted. These programs provide "hands on" training for plant evaluation and off-normal events for each operating shift.

Therefore, the staff disagrees with MHI's determination that natural circulation testing for the USAPWR is a first-of-a-kind test as defined in RG 1.68, Appendix A.6. Please revise the relevant portions of Subsections 14.2.8.1, 14.2.8.1.2 and 14.2.12.2.3.9 of the US-APWR DCD to reflect that this test will be conducted on every plant.

ANSWER:

MHI will revise the DCD to reflect that this test will be conducted on every plant.

However, MHI will include discussion of the option to propose omitting the test for subsequent plants based on verification that that flow and temperature data are comparable to prototype designs for which equivalent tests have been successfully completed, consistent with RG 1.68, Appendix A section 4, Low Power Testing, item t. MHI notes that this option has been utilized in the past. For example, natural circulation testing was not performed at Vogtle Electric Generating Plant Unit 2 based on the results from Unit 1. Similarly, natural circulation testing was not performed on Comanche Peak Steam Electric Station Units 1 or 2 based upon the similarity of the plants to Diablo Canyon Unit 1.

Impact on DCD

This revision impacts Revision 1 of the DCD in Subsections 14.2.8, 14.2.8.1 and 14.2.8.1.2, 14.2.12.2.3.9 and 14.2.12.2.4.12. (See Attachment 1, FSAR Mark-up.)

(1) Revise the second paragraph of section 14.2.8.1 as follows (only the affected portion is shown):

First-Plant-Only Test	Subsection
Reactor Internals Vibration Test	14.2.12.1.7
~~Natural Circulation Test~~	~~14.2.12.2.3.9~~
Rod Cluster Control Assembly (RCCA) Misalignment Measurement and Radial Power Distribution Oscillation Test	14.2.12.2.4.5

(2) Revise Subsection 14.2.8.1.2 as follows:

14.2.8.1.2 **DELETED** ~~Natural Circulation Test (14.2.12.2.3.9)~~

~~The natural circulation test using steam generators (SGs) is performed in the startup test phase of the first US-APWR as a prototype test. It is unnecessary for following plants to compare flow and temperature data (without reactor coolant pumps [RCPs]) and temperature data to that of this plant because no design differences exist that would significantly affect natural circulation capabilities.~~

(3) Add Subsection 14.2.8.2:

14.2.8.2 Prototype Test Results
14.2.8.2.1 Natural Circulation Testing

For subsequent plants, the COL applicant shall either perform the test or shall provide a justification for not performing the test based on an evaluation of the results of previous natural circulation tests and comparison of RCS hydraulic resistance coefficients applicable to normal flow conditions provided that

- **Test results from the US-APWR reference prototype plant indicate that natural circulation flow rates are adequate to ensure that core decay heat removal, boron mixing, plant cooldown/ depressurization, and stable natural circulation conditions are maintained throughout the test.**
- **The as-built plant and US-APWR reference prototype plant configurations are the same relative to the general configuration of the piping and components in each reactor coolant loop, the general arrangement of the reactor core and internals, and similar elevation head represented by these components and the system piping.**
- **The hydraulic resistance coefficients applicable to normal flow conditions and temperature data, and loss of coolant flow delay-time data (as measured during the RCS Flow Measurement Test in Subsection14.2.12.2.4.12 and during the RCS Flow Coastdown Test in Subsection 14.2.12.2.1.13) are comparable with the US-APWR reference prototype plant.**
- **The results of the natural circulation test from the US-APWR reference prototype plant are incorporated into a plant-referenced simulator that meets the requirements of 10CFR55.46(c) and used in the operator training program to provide training on plant evaluation and off-normal events for each operating shift.**

(4) Revise Subsection 14.2.12.2.4.12 as follows (only the added portion is shown):

C. Test Method

3. **RCS hydraulic resistance coefficients are calculated and recorded based on the RCS flow measurement data and temperature data of RCS is recorded.**

(5) Revise Subsection 14.2.12.2.3.9 as follows (only the affected portion is shown):

14.2.12.2.3.9 Natural Circulation Test

~~(first plant only test)~~ **(Perform on first plant. For subsequent plants, see discussion in Subsection 14.2.8.2.)**

Impact on COLA

This revision impacts revision 0 of the COLA FSAR (Docket #s 52-034 and 52-035) in Subsection 14.2.8.2.1.

Impact on PRA

There is no impact on the PRA.

Attachment 1

US-APWR DCD Chapter 14 Mark-up

AMENDED RESPONSE TO RAI No. 70 Revision 0

14.2.7 Conformance of Test Program with RGs

The development of the preoperational and startup test program adheres to the guidance of the NRC RGs associated with the ITP. These RGs are listed in Table 14.2-2. Conformance with the guidance of RGs are defined in Table 1.9.1-1, US-APWR Conformance with Division 1 RGs.

14.2.8 Utilization of Reactor Operating and Testing Experience in the Development of Test Program

Because the US-APWR plant is based on the development of previous pressurized-water reactor (PWR) plants, the US-APWR plants have the benefit of experience acquired with the successful and safe startup of many previous PWR plants in Japan. The operational experience and knowledge gained from these plants and other reactor types are factored into the design and test system information of US-APWR equipment and systems that are demonstrated during the preoperational and startup test programs. Additionally, reactor operating and testing experience of similar nuclear power plants obtained from NRC Licensee Event Reports, Institute of Nuclear Operations correspondence, Significant Operating Event Reports, and through other industry sources is utilized to the extent practicable in developing and carrying out the ITP. Special importance is attached to repeat reportable occurrences experienced involving safety concerns and other operating experiences that could potentially impact the performance of the test program.

14.2.8.1 Preoperational and/or Startup Testing for Unique or First-of-a-Kind Principal Design Features

First-of-a-kind tests are special tests performed to verify unique performance parameters for new design features. Since these design features are new, the tests have not been performed for the design certification of previous plants. Because of the standardization of the US-APWR design, the parameters will not change from plant to plant and thus the tests are performed only on the first plant containing the unique design.

These first-plant-only tests are identified in the individual test descriptions in Subsection 14.2.12. The following is a listing of the first plant only tests, and the corresponding subsection in which they appear. <u>The COL holder for the first plant is to perform these tests. For subsequent plants, either these tests are performed, or the COL applicant provides a justification that the results of the first-plant only tests are applicable to the subsequent plant and are not required to be repeated.</u>

<u>First-Plant-Only Test</u>	<u>Subsection</u>
Reactor Internals Vibration Test	14.2.12.1.7
~~Natural Circulation Test~~	~~14.2.12.2.3.9~~
Rod Cluster Control Assembly (RCCA) Misalignment Measurement and Radial Power Distribution Oscillation Test	14.2.12.2.4.5

There could be other special tests that further establish a unique performance parameter of the US-APWR design features beyond the testing performed for the design certification for previous plants and, which will not change from plant to plant, that are performed for the first three plants. There are no first-three-plants tests required.

The justifications for the first-plant-only tests and the justifications for there being no first-three-plant tests required are provided below:

14.2.8.1.1 Reactor Internals Vibration Test (14.2.12.1.7)

Preoperational vibration test of reactor internals is performed in accordance with RG 1.20 (Reference 14.2-13). This program is discussed in Subsection 3.9.2.

This test is conducted only during the hot functional test prior to fuel loading because the vibration responses under normal operating conditions with core are predicted to be almost the same or slightly lower than those under hot functional tests without the core loaded.

Justification for performing this on the first plant only is provided in Subsections 3.9.2.3 and 3.9.2.4.

~~14.2.8.1.2 Natural Circulation Test (14.2.12.2.3.9)~~Deleted

14.2.8.1.2 ~~The natural circulation test using steam generators (SGs) is performed in the startup test phase of the first US-APWR as a prototype test. It is unnecessary for following plants to compare flow and temperature data (without reactor coolant pumps [RCPs]) and temperature data to that of this plant because no design differences exist that would significantly affect natural circulation capabilities.~~

14.2.8.1.3 RCCA Misalignment Measurement and Radial Power Distribution Oscillation Test (14.2.12.2.4.5)

RCCA misalignment measurements and radial power distribution oscillation tests are performed in the power ascension test phase for the first US-APWR. The test is required only for the first plant because the stability of the radial power distribution is dependent upon the core diameter only. This test validates the calculation tools and instrument responses.

14.2.8.2 Prototype Test Results

14.2.8.2.1 Natural Circulation Testing

For subsequent plants, the COL applicant shall either perform the test or shall provide a justification for not performing the test based on an evaluation of the results of previous natural circulation tests and comparison of RCS hydraulic resistance coefficients applicable to normal flow conditions provided that

- Test results from the US-APWR reference prototype plant indicate that natural circulation flow rates are adequate to ensure that core decay heat removal, boron mixing, plant cooldown/depressurization, and stable natural circulation conditions are maintained throughout the test.

- The as-built plant and US-APWR reference prototype plant configurations are the same relative to the general configuration of the piping and components in each reactor coolant loop, the general arrangement of the reactor core and internals, and similar elevation head represented by these components and the system piping.

- The hydraulic resistance coefficients applicable to normal flow conditions and temperature data, and loss of coolant flow delay-time data (as measured during the RCS Flow Measurement Test in Subsection 14.2.12.2.4.12 and during the RCS Flow Coastdown Test in Subsection 14.2.12.2.1.13) are comparable with the US-APWR reference prototype plant.

- The results of the natural circulation test from the US-APWR reference prototype plant are incorporated into a plant-referenced simulator that meets the requirements of 10 CFR § 55.46 (c) and used in the operator training program to provide training on plant evaluation and off-normal events for each operating shift.

14.2.9 Trial Testing of Plant Operating and Emergency Procedures

Plant operating and emergency procedures are, to the extent practical, developed, trial-tested, and corrected during the ITP prior to fuel loading to establish their adequacy. Preoperational and startup test procedures utilize plant operating, surveillance, emergency, and abnormal procedures either by reference or verbatim incorporation in the performance of tests. This verifies the plant procedures by actual use and provides experience to the plant personnel.

The COL applicant provides a schedule for the development of plant procedures that assures that required procedures are available for use during the preparation, review and performance of preoperational and startup testing.

14.2.9.1 Operator Training during Special Low-Power Testing

At approval to load fuel, by virtue of being licensed by the NRC to operate the plant, the ROs/SROs have a responsibility for the operation of the plant. Therefore, at this point, the plant operations organization assumes responsibility for the plant. This period is used to further the training of licensed operators and provide training for operator trainees. This includes identifying the specific operator training to be conducted as a part of the use-testing during the special low power testing program required by the resolution of NUREG-0737 (Reference 14.2-7) TMI action plan item I.G.1. Meeting this requirement includes identifying proposed tests to be conducted, submitting analysis to support the test, submitting the test procedure, training to the test procedure and evaluating and documenting the results of the training.

14.2.10 Initial Fuel Loading and Initial Criticality

Fuel loading and initial criticality is conducted in accordance with the guidance of RG 1.68 (Reference 14.2-10). This phase of the ITP is performed in a controlled manner as prescribed in approved detailed written procedures. These procedures specify the following:

14.2.12.2.3.9 Natural Circulation Test

(~~first-plant-only test~~Perform on first plant. For subsequent plants, see discussion in Subsection 14.2.8.2.)

A. Objectives

1. To demonstrate the capability to remove decay heat by natural circulation.

B. Prerequisites

1. RCPs are operating.

2. Primary system is at normal operating temperature and pressure.

C. Test Method

1. The test is initiated by tripping all RCPs.

2. Natural circulation is verified by observing the response of the hot leg and cold leg temperature instrumentation in each loop for natural circulation stabilization period and the ability to maintain the cooling mode.

D. Acceptance Criteria

1. Decay heat removal capability is demonstrated by maintaining natural circulation conditions.

14.2.12.2.3.10 Automatic Low Power SG Water Level Control Test

A. Objective

1. To verify the stability of the automatic low power SG water level control system following simulated transients at low power conditions.

B. Prerequisites

1. The reactor is critical, and in the low power level.

2. The SG water level control system is checked and calibrated.

3. SG alarm setpoints are set for each SG.

C. Test Method

1. Induce simulated SG water level transients to verify SG water level control response.

D. Acceptance Criteria

2. Concrete surface temperatures are maintained in accordance with Subsection 9.4.6.1.2.3.

14.2.12.2.4.12 RCS Flow Measurement Test

A. Objectives

1. As for RCS flow rate measurement, RCS flow rate is determined based on the correlation between data obtained by measuring RCP motor input power and the differential pressure across the reactor coolant line elbow tap, for the purpose of confirming reactor coolant flow is equal to or greater than the design flow specified in Section 5.1.

2. To perform calorimetric flow measurements at 50%, 75%, and 100% power, for the purpose of confirming RCS flow is equal to or greater than the design flow in Section 5.1.

B. Prerequisites

1. Required instrument calibration is completed.

2. Required support systems are operational.

3. The reactor core is installed, and the plant is at normal operating temperature and pressure prior to initial criticality.

C. Test Method

1. Prior to criticality, operating all RCPs and any combination of them including a single operation, input power of each operating RCP motor and relating RCS line elbow tap differential pressure is measured. RCS flow rate is calculated using the correlation between obtained RCP motor input power and RCS line elbow tap differential pressure.

2. Calorimetric flow measurements are performed at the 50%, 75%, and 100% power plateau and RCS flow is calculated.

3. RCS hydraulic resistance coefficients are calculated and recorded based on the RCS flow measurement data and temperature data of RCS is recorded.

D. Acceptance Criterion

1. Reactor coolant flowrate in each loop is equal to or greater than the design flow specified in Section 5.1.

14.2.12.2.4.13 Process and Effluent Radiation Monitoring System Test

A. Objectives

1. To verify the operation of the process and effluent radiation monitor against an acceptable standard.