MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

November 25, 2011

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

Attention: Mr. Jeffery A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-11410

Subject: MHI's Second Responses to US-APWR DCD RAI No. 804-5938 Revision 3 (SRP section 03.12)

Reference:
1) "Request for Additional Information No. 804-5938 Revision 3, SRP Section: 03.12 – ASME Code Class 1, 2, and 3 Piping Systems and Piping Components And Their Associated Supports," dated 8/11/2011.
2) "MHI's Responses to US-APWR DCD RAI No.804-5938," UAP-HF-11382, dated 11/10/2011.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Second Responses to Request for Additional Information No. 804-5938, Revision 3."

Enclosed are the responses to remaining 1 RAI contained within Reference 1. That is RAI 3.12-29. 3 responses to RAI of Question No.3.12-26, 27 and 28 contained with reference 1 are previously provided in Reference 2.

As indicated in the enclosed materials, this document contains information that MHI considers proprietary, and therefore should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential. A non-proprietary version of the document is also being submitted with the information identified as proprietary redacted and replaced by the designation "[ ]".

This letter includes a copy of the proprietary version (Enclosure 2), a copy of the non-proprietary version (Enclosure 3), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all materials designated as "Proprietary" in Enclosure 2 be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

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

Sincerely,

Y. Ogata

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

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Enclosures:

1. Affidavit of Yoshiki Ogata

2. Second Responses to Request for Additional Information No. 804-5938, Revision 3 (Proprietary)

3. Second Responses to Request for Additional Information No.804-5938, Revision 3 (Non-Proprietary)

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

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Enclosure 1

Docket No. 52-021
MHI Ref: UAP-HF-11410

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, state as follows:

1. I am General Manager, APWR Promoting Department, of Mitsubishi Heavy Industries, LTD ("MHI"), and have been delegated the function of reviewing MHI's US-APWR documentation to determine whether it contains information that should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential.

2. In accordance with my responsibilities, I have reviewed the enclosed document entitled "Second Responses to Request for Additional Information No. 804-5938, Revision 3" and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages contain proprietary information as identified with the label "Proprietary" on the top of the page, and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

3. The information identified as proprietary in the enclosed documents has in the past been, and will continue to be, held in confidence by MHI and its disclosure outside the company is limited to regulatory bodies, customers and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and is always subject to suitable measures to protect it from unauthorized use or disclosure.

4. The basis for holding the referenced information confidential is that it describes the information for performing the plant design of US-APWR.

5. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of information to the NRC staff.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information. Other than through the provisions in paragraph 3 above, MHI knows of no way the information could be lawfully acquired by organizations or individuals outside of MHI.

7. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without incurring the costs or risks associated with the design of the subject systems. Therefore, disclosure of the information contained in the referenced document would have the following negative impacts on the competitive position of MHI in the U.S. nuclear plant market:

A. Loss of competitive advantage due to the costs associated with the development of the methodology related to the analysis.

B. Loss of competitive advantage of the US-APWR created by the benefits of the approach to jet expansion modeling that maintains the desired level of conservatism.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information and belief.

Executed on this 25th day of November, 2011.

Y. Ogata

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

Enclosure 3

UAP-HF-11410
Docket No. 52-021

Second Responses to Request for Additional Information
No. 804-5938, Revision 3

November, 2011
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/25/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 804-5938 REVISION 3**
SRP SECTION:	**03.12 – ASME Code Class 1, 2, and 3 Piping Systems and Piping Components and Their Associated Supports**
APPLICATION SECTION:	**3.12**
DATE OF RAI ISSUE:	**08/11/2011**

QUESTION NO.: RAI 03.12-29

DCD Tier 2, Section 3.12.5.9 discussed thermal cycling in piping connected to the RCS. In their RAI 3.12-17 response (ML0933380324), MHI stated that the piping is routed such that a stratified surface boundary does not occur at horizontal pipe bends and elbows. However, MHI did not provide discussion for the piping routing methodology in the DCD mark-up. The staff is requesting MHI to provide piping stratification prediction method. In general, US utilities followed MRP-146 model for predicting and evaluating thermal cycling for PWR stagnant lines. MHI provided research results of ICONE 10- 23340 and ICONE 11-36214. However, ICONE 11-36214 identified that predictive accuracy of the evaluation method is about + 20 percent from the limited experiment data. If MHI model is not consistent with MRP-146 model which has shown benchmarking results to be effective in predicting the location of thermal cycling in branch line attached to RCL, MHI should identify the difference and provide justification.

In their RAI 3.12-17 response, MHI stated that as verification of actual equipment, confirmation is made such that there is no stratification surface boundary (top of cavity flow) at pipe bends and elbows migrating horizontally from vertical risers by performing temperature measurement for actual equipment of isolated branch pipes around RCS during initial startup test. However, the staff reviewed DCD Chapter 14 verification programs which did not mention any activity related to the stratification surface boundary verification. The staff is requesting MHI provide additional information for the test abstract including stating the standard operating conditions in DCD Chapter 14 that identifies the Objective, Prerequisites, Test Method, Data Required, and Acceptance Criteria for unisolable piping connected to the RCS to address NRC Bulletin 88-08. In general stratification verification monitoring activity shall be the COL's responsibility. However, this activity has not been listed as COL action item in the DCD. The staff asked MHI to clarify the responsibility. If this activity is to be completed by COL, the DCD should be modified to add this activity as a COL action item.

In the response, MHI provided the Table to address detection of leakage of the safety related valves, the staff is requesting MHI to put this information in DCD mark-up.

ANSWER:

If the top of the cavity flow (i.e., the swirling penetration flow) entering a branch pipe from the Reactor Coolant Loop (run pipe) is located at a bend and thermal stratification occurs, where the downward vertical pipe turns horizontal, adverse thermal cycling, induced by swirling penetration flow only, may occur without the process due to valve seat leakage. Therefore, since the package of "JSME S 017-2003 "Guideline for Evaluation of High-Cycle Thermal Fatigue of a Pipe" (hereinafter, "the JSME Guideline") does not include the knowledge for predicting and evaluating thermal cycling for the PWR stagnant lines, MHI will avoid this adverse thermal cycling through the piping routing design.

In order to prevent the top of the cavity flow from being located at a pipe bend or an elbow, MHI will predict the penetration depth of cavity flow in accordance with the JSME Guideline. ICONE-10-23340 and ICONE-11-36214 essentially explain the empirical knowledge contained in the JSME Guideline.

As for the piping routing methods, MHI will develop the ASME Class 1 pipe routing design based on the guidance in the JSME Guideline to meet either of the following criteria (See Attachment 1). If a piping route satisfies the criteria, no further evaluation is required in piping routing design. This design approach is the same as that of the MRP-146 guidance. MHI will add a description of this approach in the DCD Tier 2 Subsection 3.12.5.9 as follows:

a. The length of the vertical pipe between the Reactor Coolant Loop (run pipe) and the horizontal pipe should be longer than the predicted penetration depth (maximum depth) of the cavity flow. In this case, the stratification layer is stable in the vertical pipe and therefore significant thermal cycling does not occur.

b. The length of the vertical pipe between the Reactor Coolant Loop (run pipe) and the horizontal pipe should be shorter than the predicted penetration depth (minimum depth) of the cavity flow. In this case, the cavity flow at a temperature equal to the water temperature in the run pipe keeps the water temperate in the horizontal pipe high and therefore significant thermal cycling does not occur.

Furthermore, in the JSME Guideline, the design margins for the penetration depth of the cavity flow, as shown in Table 1, are set to assure the reliability of the predicted penetration depth of the cavity flow, taking into account quantitatively the following factors:

- Difference between the actual thermal-hydraulic test results and the results predicted by the JSME model (A)
- Reproducibility of the thermal-hydraulic tests (B).

Table 1 Prediction accuracy of the penetration depth of the cavity flow

In order to confirm that no thermal stratification layer is located at the pipe bend of stagnant lines connected to the RCL, MHI will provide an additional test abstract stating the standard operating conditions in DCD Chapter 14 that identifies the Objective, Prerequisites, Test Method, Data Required, and Acceptance Criteria. Since this temperature measurement is just for design verification, to confirm the top location of the cavity flow as an as-built reconciliation, this activity is not a COL action item.

In the response to RAI 465-3382 Question 3.12-17 submitted as UAP-HF-09547, MHI provided a Table to address detection of leakage of the safety-related valves. However, this response mentioned that the double isolation valves, assuming leakage from RCS with high temperature to downstream of the valves, are not necessary to be considered since the low probability of the thermal stratification, therefore, it is not necessary to expect the temperature detection. Further, the countermeasure or justification regarding the thermal stratification to the following portion is already applied to the US-APWR design.

The normally stagnant branch lines attached to the RCS piping (up-horizontal/horizontal) with a potential for in-leakage from a high-pressure source toward the RCS piping include an out-of-service charging line and an auxiliary pressurizer spray line.

- Out-of-service charging line
  -An out-of-service charging line does not exist in US-APWR, therefore, there is no concern for thermal stratification and thermal oscillation.
- Auxiliary pressurizer spray line
  -The velocities in the spray lines are not sufficient to produce thermal fatigue cycling due to insufficient swirl penetration into the auxiliary spray line. Also, the temperature difference between spray line flow and valve seat leakage flow is small. Therefore, the auxiliary spray line can be screened out for normal operating conditions.

Therefore, the Table will not be added to the DCD and the DCD will be revised to reflect the countermeasure or justification regarding thermal stratification as shown in Attachment 2.

Impact on DCD

See Attachment 2 for the mark-up of DCD Tier 2, Section 3.12 and Section 14, changes to be incorporated.

Impact on R-COLA

There is no impact on the COLA.

Impact on S-COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical/Topical Report

There is no impact on a Technical/Topical Report.

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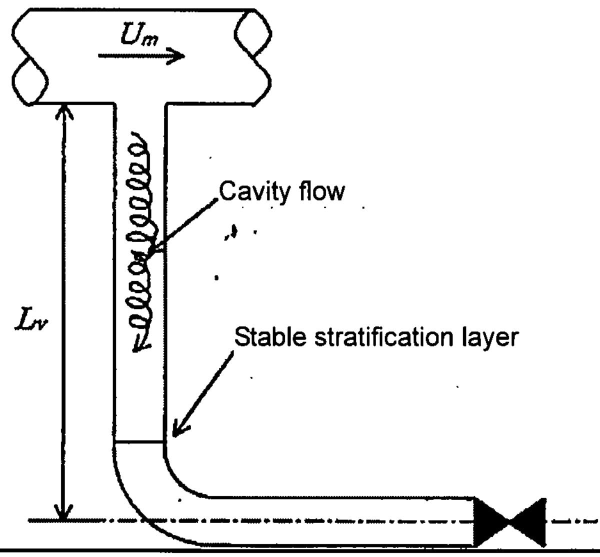
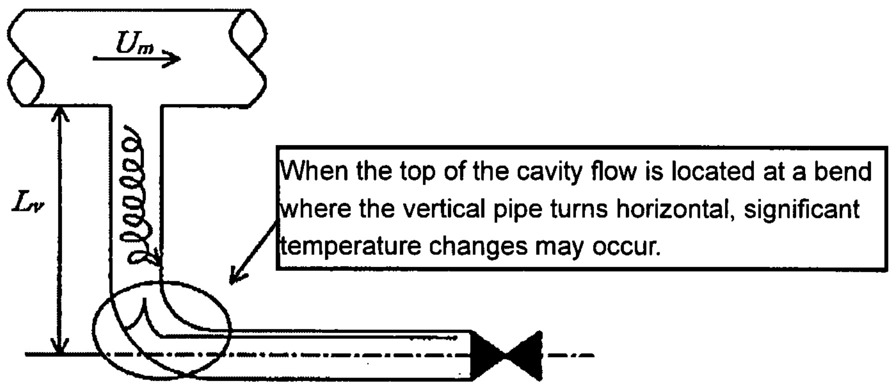
(1) When the top of the cavity flow is located in a vertical pipe



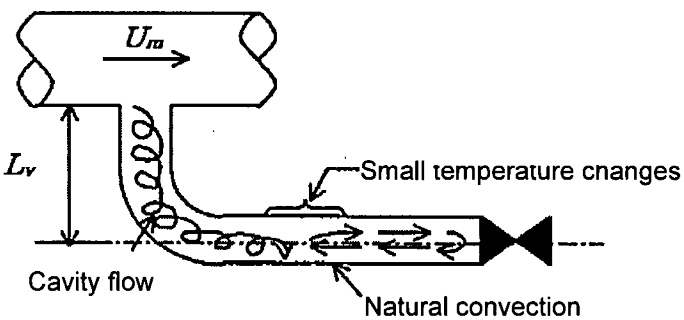
When the top of the cavity flow is located in a vertical pipe, the stratification layer is stable and no significant temperature changes occur. Therefore, the fluctuating stress does not reach the fatigue limit.*1

*1: This is due to that there is little change in the layer although the temperature difference across the surface is large.

(2) When the top of the cavity flow is located in a bend



(3) When the front of the cavity flow is located in a horizontal pipe



When the top of the cavity flow is located in a horizontal pipe, small temperature changes occur near the front but the fluctuating stress does not reach the fatigue limit. *2

*2: This is due to that the temperature difference at the top of the cavity flow is smaller than in (1) and the changes in cavity flow are smaller than in (2).

Fig. A1-1 Summary for evaluating the structural integrity at the top of the cavity flow

3.12.5.8 Fatigue Evaluation of ASME Code Class 2 and 3 Piping

ASME Code, Section III (Reference 3.12-2), Class 2 and 3 piping are not explicitly analyzed for calculation of cumulative usage factors in a manner similar to the ASME Code, Section III, Class 1 piping. ASME Code, Section III, Class 2 and 3 piping are evaluated following the requirements of NC/ND-3611.2 (Reference 3.12-2), which allows the reduction of allowable stress for thermal expansion stress ranges calculated using the requirements of NC/ND-3653.2(a) by stress range reduction factor (*f*) of Table NC/ND-3611.2(e)-1. The stress intensification factors used in the Class 2 and 3 Code Equations 10 and 11 for various piping products and components are based on fatigue testing. As such, they indirectly account for fatigue. The environmental impact on fatigue of Class 2 and 3 piping follows guidelines established by the NRC at the time of actual analysis.

3.12.5.9 Thermal Oscillations in Piping Connected to the Reactor Coolant System

As determined by NRC Bulletin 88-08 including Supplement 3 (Reference 3.12-27), the RCS is reviewed to identify places potentially affected by thermal stresses caused by thermal stratification or thermal oscillation.

Thermal stratification and oscillation has been evaluated relative to the design of the US-APWR. It has been found that no problem would occur due to the thermal stresses caused by thermal stratification or temperature changes in the closed branch piping, connected to RCS~~ (see Tables 3.12-7 and 3.12-8)~~. The following US-APWR design approach to address valve leakage is provided as assurance against thermal stratification and thermal oscillation. DCD_03.12-29

The normally stagnant branch lines attached to the RCS piping (up-horizontal/horizontal) with a potential for in-leakage from a high-pressure source toward the RCS piping such as out-of-service charging line and auxiliary pressurizer spray line. For the former, no out-of-service charging line exists in US-APWR, therefore, there are no concern for thermal stratification and thermal oscillation. For the latter, the velocities in the main spray lines are not sufficient to product thermal fatigue cycling due to insufficient swirl penetration into the auxiliary spray line. Also, the temperature difference between spray line flow and valve seat leakage flow is small. Therefore, the auxiliary spray line can be screened out for normal operating conditions. DCD_03.12-29

1. ~~A double isolation valve configuration successfully restricts leakage, preventing fatigue failure from thermal stratification or oscillation.~~

2. ~~For a single valve configuration, leakage can be detected by measuring the downstream temperature. Monitoring of downstream temperature is utilized to detect valve leakage. As a result of leakage detection, valve repair can be scheduled, thereby preventing fatigue failure from thermal stratification or oscillation.~~

3. In the case of a gate valve configuration, high-cycle fatigue could be caused by repeated leaks from the valve gland. Leaks would occur even when double isolation valves are installed in series. By permitting continuous leakage through the valve gland packing by valve disk position adjustment, valve disk expansion and contraction cycle is prevented and cyclic fatigue failure caused by thermal

stratification or thermal oscillation is eliminated (~~Reference 3.12-27~~see Table 3.12-8). DCD_03.12-29

Furthermore, high temperature water that flows from the RCS pipe run into ~~isolated~~the closed stagnant down-horizontal branch pipes occurs as swirling water, referred to as cavity flow induced by the high temperature pipe. In addition, there is a phenomenon in which stratification occurs at the surface boundary between the high temperature water entering the branch pipe and the low temperature water already in the branch. When the stratified surface boundary (top of cavity flow) occurs at ~~horizontal pipe bends and elbows~~pipe bend(s) or elbow(s) which directs flow horizontally from vertical and downward risers, there is a possibility of significant thermal fluctuation inducing high cycle thermal fatigue at this point. Therefore, the piping is routed such that a stratified surface boundary does not occur at ~~horizontal pipe bends and elbows~~pipe bend(s) or elbow(s) which directs flow horizontally from vertical and downward risers as predicted in the JSME guidelines (Reference 3.12-43). DCD_03.12-29 DCD_03.12-29

a. The length of the vertical line between the Reactor Coolant Loop (run pipe) and the horizontal pipe should be longer than the predicted penetration depth (maximum) of the cavity flow. In this case, the surface boundary becomes stagnant in the vertical pipe and therefore significant thermal cycling does not occur.

b. The length of the vertical pipe between the Reactor Coolant Loop (run pipe) and the horizontal pipe should be shorter than the predicted penetration depth (minimum) of the cavity flow. In this case, the cavity flow at a temperature equal to the water temperature in the run pipe keeps the water temperate in the horizontal pipe high and therefore significant thermal cycling does not occur.

3.12.5.10 Thermal Stratification

NRC Bulletin 79-13 (Reference 3.12-28) addresses the effect of thermal stratification that lead to the cracking of the feedwater line at D.C., Cook Nuclear Plant Unit 2.

Provisions of the thermal stratification of the feedwater nozzle are described in Subsection 5.4.2.1.2.12.

NRC Bulletin 88-11 (Reference 3.12-29) was issued after Portland General Electric Company experienced difficulties in setting whip restraint gap sizes on the pressurizer surge line at the Trojan plant.

At the horizontal portion of the pressurizer surge line, thermal stratification is expected to occur if the surge flow velocity is low, and to disappear if the velocity is high. At normal operation, a low flow-rate out-surge flow in the line connecting the pressurizer to the hot leg may occur due to a continuous spray, which could lead to a thermal stratification in the cross section of pressurizer surge line in accordance with the temperature difference between pressurizer and hot leg. When a high-flow rate out-surge flow or in-surge flow occurs during transient events, this thermal stratification disappears. The low flow-rate out-surge flow is recovered as soon as out-surge or in-surge ends, thus, reproducing the thermal stratification.

3.12-34 Structural Welding Code – Steel. AWS D1.1/D1.1M, American Welding Society, 2006.

3.12-35 Service Limits and Loading Combinations for Class 1 Linear-Type Supports. Regulatory Guide 1.124, Rev. 2, U.S. Nuclear Regulatory Commission, Washington, DC, February 2007.

3.12-36 Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports. Regulatory Guide 1.130, Rev. 2, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.12-37 ~~Code Requirements for Nuclear Safety Related Concrete Structures. ACI 349, American Concrete Institute, 2001.~~Code Requirements for Nuclear Safety-Related Concrete Structures, American Concrete Institute (ACI) 349-06 and Commentary, 2006. MIC-03-03-00066

3.12-38 Anchoring Components and Structural Supports in Concrete. Regulatory Guide 1.199, U.S. Nuclear Regulatory Commission, Washington, DC, November 2003.

3.12-39 IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations, IEEE Std 344-2004, Institute of Electrical and Electronic Engineers Power Engineering Society, New York, New York, June 2005.

3.12-40 Evaluation of Potential for Pipe Breaks, Report of U.S. NRC Piping Review Committee. NUREG-1061, Volume 4, U.S. Nuclear Regulatory Commission, Washington, DC, 1984.

3.12-41 Dynamic Analysis of Piping, Using the Structural Overlap Method. NUREG/CR-1980, U.S. Nuclear Regulatory Commission, Washington, DC, March 1981.

3.12-42 ASME Boiler and Pressure Vessel Code, Section II, 2001 Edition including 2003 Addenda, The American Society of Mechanical Engineers. DCD_03.12-27

3.12-43 Guideline for Evaluation of High-Cycle Thermal Fatigue of a pipe, JSME S 017-2003, November 2003. DCD_03.12-29

Table 14.2-1 Comprehensive Listing of Tests (Sheet 5 of 5)

Section	Test
14.2.12.2.4.10	Thermal Power Measurement and Statepoint Data Collection Test
14.2.12.2.4.11	Ventilation Capability Test
14.2.12.2.4.12	RCS Flow Measurement Test
14.2.12.2.4.13	Process and Effluent Radiation Monitoring System Test
14.2.12.2.4.14	Primary and Secondary Chemistry Test
14.2.12.2.4.15	Biological Shield Survey Test
14.2.12.2.4.16	Load Swing Test
14.2.12.2.4.17	Loss of Offsite Power (LOOP) at Greater Than 10 % Power Test
14.2.12.2.4.18	Plant Trip from 100 % Power Test
14.2.12.2.4.19	100% Load Rejection Test
14.2.12.2.4.20	Dynamic Response Test
14.2.12.2.4.21	Ultimate Heat Sink Heat Rejection Capability Test
14.2.12.2.4.22	Automatic High Power SG Water Level Control Test
14.2.12.2.4.23	Confirmation of the Top Location of the Cavity Flow

DCD_03.12-29

1. The high power SG water level controller responds to maintain a stable SG level at setpoint (see Subsection 7.7.1.1.9).

2. The feedwater pump speed controller responds to maintain a stable feedwater pump discharge pressure at setpoint.

3. The main feedwater control valves open or close and stabilize in response to level setpoint changes.

14.2.12.2.4.23 Confirmation of the Top Location of the Cavity Flow

DCD_03.12-29

A. Objective

1. To verify the un-isolable and closed branch piping connected to the Reactor Coolant System has no adverse thermal oscillations induced by thermal stratification induced by the cavity flow at 100% power operation.

B. Prerequisites

1. The penetration depths of the cavity flow are predicted based on the empirical data from thermal-hydraulic tests (see Subsection 3.12.5.9).

2. To avoid adverse thermal oscillations, the piping is routed such that no stratified surface boundary induced by cavity flow occurs at a pipe bend or elbow directing flow horizontally from vertical and downward risers.

3. This test is conducted at 100% power operation.

4. Supports, restraints, and hangers are installed.

5. All insulations are installed.

6. Reference points are established. Temporary instrumentations are installed as required to confirm the temperature distribution along each pipeline under test.

C. Test Method

1. The temperature distribution along each pipeline is measured from the outer surface of the pipe to confirm the top location of the cavity flow at 100 % power condition, and the temperature data is recorded.

D. Acceptance Criterion

1. The top of the cavity flow from the Reactor Coolant Loop does not exist at the area of a pipe bend or elbow directing flow horizontally from vertical and downward risers. Temperature difference between cavity flow and stagnant water in each branch pipeline is observed at the thermal stratification induced by the cavity flow (see Subsection 3.12.5.9).

Table 14.2-3 Power Ascension Test Metrix

Power Level					
Test Number	0%	30%	50%	75%	100%
14.2.12.2.4.1	x	X	X	X	X
14.2.12.2.4.2		x	x	x	x
14.2.12.2.4.3			x	x	X
14.2.12.2.4.4	X	X	X	X	X
14.2.12.2.4.5			X		
14.2.12.2.4.6		≥10%			
14.2.12.2.4.7	To be determined in accordance with the manufacture's instructions.				
14.2.12.2.4.8	Not specified.				
14.2.12.2.4.9				x	x
14.2.12.2.4.10		X	X	X	X
14.2.12.2.4.11	X		x		x
14.2.12.2.4.12	Prior to criticality		X	X	X
14.2.12.2.4.13	Low power				x
14.2.12.2.4.14	X	X	X	X	x
14.2.12.2.4.15	Low power		X		X
14.2.12.2.4.16		X	X	X	x
14.2.12.2.4.17		X			
14.2.12.2.4.18					x
14.2.12.2.4.19					X
14.2.12.2.4.20	Not specified.				
14.2.12.2.4.21	Not specified.				
14.2.12.2.4.22		X			
14.2.12.2.4.23					X

DCD_03.12-29

Table 14A-1 Conformance Matrix of RG 1.68 Appendix A Guidance Versus Typical Test Abstracts (Sheet 17 of 17)

RG 1.68 Appendix A	Section Number	Typical Test
5.ii	-	Exception The complete loss of flow at full-power test will not be performed. Results for reactor coolant system (RCS) flowrates obtained in the flow coastdown test will verify that the RCS flowrates assumed in safety analysis (related to chapter15) are conservative.
5.jj	14.2.12.2.4.17	Loss of Offsite Power at Greater Than 10 % Power Test
5.kk	-	Exception The loss of or bypass of feedwater heaters test will not be performed because US-APWR does not have the bypass line for feedwater heaters and feedwater temperature reduction by the loss of feedwater heater is not occurred from a credible single failure or operator error.
5.ll	14.2.12.2.4.18	Plant Trip from 100 % Power Test
5.mm	-	Exception Tests item u and mm will not be performed as the results obtained will be similar to the results obtained during a plant trip from 100 % power which will be performed. The closure times for the MSIVs will be verified during hot functional and preoperational testing.
5.nn	14.2.12.2.4.19	100 % Load Reduction Test
5.oo	14.2.12.1.52 14.2.12.2.4.20 14.2.12.4.23	Thermal Expansion Test Dynamic Response Test Confirmation of the Top Location of the Cavity Flow

DCD_03.12-29