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March 28, 2014

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

Attention: Mr. Perry Buckberg

Docket No. 52-021
MHI Ref: UAP-HF-14035

Subject: Transmittal of the Response to March 4, 2014 ACRS Comments on DCD Chapter 14

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") an official document entitled "Response to March 4, 2014 ACRS Comments on DCD Chapter 14."

Please contact Mr. Joseph Tapia, General Manager of Licensing Department, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of this submittal. His contact information is provided below.

Sincerely,

K. Takeuchi for

Yoshiki Ogata
Executive Vice President
Mitsubishi Nuclear Energy Systems, Inc.
On behalf of Mitsubishi Heavy Industries, Ltd.

Enclosure:

1. Response to March 4, 2014 ACRS Comments on DCD Chapter 14

CC: P. Buckberg
J. Tapia

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Enclosure 1

UAP-HF-14035
Docket No. 52-021

Response to March 4, 2014 ACRS Comments on DCD Chapter 14

March 2014

RESPONSE FOR ACRS SUBCOMMITTEE MEETING

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: **14**

CHAPTER TITLE: **VERIFICATION PROGRAMS**

DATE OF MEETING: **3/4/2014**

QUESTION: Item 1

Why are the advanced accumulator and GTG not considered as first-of-a-kind tests?

ANSWER:
The advanced accumulator (ACC) and gas turbine generator (GTG) tests will be performed for every plant, as a result they are not considered as first-of a-kind tests.

Impact on DCD
There is no impact on the DCD.

Impact on R-COLA
There is no impact on the R-COLA.

Impact on PRA
There is no impact on the PRA.

Impact on Topical/Technical Reports
There is no impact on the topical and technical reports.

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QUESTION: Item 2

How does a COL applicant demonstrate that one of the first-of-a-kind tests (namely items 3 and 4 on MHI Presentation Side 11) do not require performance for subsequent COL?

ANSWER:
For item 3 (natural circulation test), the method for justifying not repeating the test by subsequent COL applicants is described in DCD Subsection 14.2.8.2.1. The justification for not performing the test will be 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 certain similarities, as outlined in Section 14.2.8.2.1, between the prototype test and the as-built plant can be justified.

For item 4 (pressurizer surge line test), the Chapter 14 description does not describe the justification requirement in detail; however, DCD Subsection 3.12.5.10 describes in detail how the structural integrity of the pressurizer surge line of the US-APWR plant is to be assured for the first US-APWR plant constructed.

Per the description in DCD Subsection 3.12.5.10, if the fatigue evaluation results comply with the ASME Code Section III; the physical configuration of surge line piping, pressurizer, and hot leg are the identical to the first plant; and the subsequent US-APWR plants the same plant heatup and cooldown procedures as the first US-APWR plant, the subsequent plants will not need to repeat the first-of-a-kind tests.

Impact on DCD
There is no impact on the DCD.

Impact on R-COLA
There is no impact on the R-COLA.

Impact on PRA
There is no impact on the PRA.

Impact on Topical/Technical Reports
There is no impact on the topical and technical reports.

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QUESTION: Item 3

The Chapter 15 safety analyses use the flow coast down from a single RCP trip as an input. Why is there no hot functional test (HFT) to verify the coast-down flow from a single RCP pump trip?

ANSWER:
The RCS low flow rate signal is confirmed during the test of RCS flow coast down described in DCD Subsection 14.2.12.2.1.13. The test abstract description in Subsection 14.2.12.2.1.13 describes that all RCPs trip. For the US-APWR 4 loop plant, the actual RCS flow coast down test will also be accomplished by tripping two RCPs, consistent with the assumption in DCD Section 15.3.1.1, partial loss of forced reactor coolant flow. The actual test procedure for the US-APWR will include the all RCPs tripped condition and the two RCPs tripped condition.

Impact on DCD
There is no impact on the DCD.

Impact on R-COLA
There is no impact on the R-COLA.

Impact on PRA
There is no impact on the PRA.

Impact on Topical/Technical Reports
There is no impact on the topical and technical reports.

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CHAPTER: **14**

CHAPTER TITLE: **VERIFICATION PROGRAMS**

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QUESTION: Item 4

Test abstract 14.2.12.1.60 "Essential Chilled Water System Preoperational Test" does not verify the ability to restart the chiller after a loss of offsite power (LOOP). These types of large chillers often have time delay recycler timers; DCD Chapter 9 does not mention this feature for the US-APWR design. Do the chillers in the US-APWR design have this type of delay embedded in the controller?

ANSWER:

When LOOP occurs and the Essential Chillers are actuated by the Blackout signal, the restart prevention interlock (30 minutes of recycle time) will be bypassed allowing the chillers to be actuated immediately upon availability of electrical power.

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Topical/Technical Reports

There is no impact on the topical and technical reports.

RESPONSE FOR ACRS SUBCOMMITTEE MEETING

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CHAPTER: 14

CHAPTER TITLE: VERIFICATION PROGRAMS

DATE OF MEETING: 3/4/2014

QUESTION: Item 5

Ventilation system tests 14.2.12.1.96 through 14.2.12.1.103, 14.2.12.1.106, 14.2.12.110, and 14.2.12.111: Currently the Acceptance Criteria for each of the individual HVAC tests states something like "design airflow is achieved". How does this test verify functional performance? Why isn't the Acceptance Criteria related to the ability of the system to maintain temperatures within a certain range?

ANSWER:

The Initial Test Program (ITP) is designed to demonstrate that the plant construction is in compliance with the design and that the plant systems perform in accordance with the design. DCD Chapter 9 Table 9.4-1 tabulates the area temperature and relative humidity during the plant normal and abnormal conditions, including design basis accident and LOOP conditions. The corresponding equipment design data for each HVAC system includes airflow capacity as one the parameters credited with maintaining local conditions within the design temperature range specified in Table 9.4-1. The capacity of the HVAC fans are calculated using maximum outside temperature and maximum heat input from equipment, including some margin.

The objective of each of the referenced ventilation system test abstracts says "To demonstrate operation of the XXX HVAC system", where XXX is a specific HVAC system name. In each case, the test specific acceptance criteria are written to verify that the system subcomponents operate as designed, which includes demonstration that the design airflow is achieved by the system. Since the design airflow capacity is determined to ensure the required temperature control, pre-operational verification of the flow rate of the fan is sufficient to verify the capability of the fan to ensure the required temperature control.

Additionally, temperature control is not only the required function for these fans; ventilation, personnel comfort, controlling the concentration of airborne radioactive material, etc., are required functions too. Flow rate is the principal parameter used to verify these additional important functions. Therefore, flow rate is used instead of temperature control in the ITP acceptance criteria.

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA
There is no impact on the R-COLA.

Impact on PRA
There is no impact on the PRA.

Impact on Topical/Technical Reports
There is no impact on the topical and technical reports.

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QUESTION: Item 6

Prerequisite #9 for ITP test 14.2.12.2.1.3 for initial fuel loading states "Water level in the reactor vessel is maintained to a level approximately equal to the center of the vessel outlet nozzles." What is the basis for this requirement?

ANSWER:
Unused fuel can be moved in air because there is no decay heat. Therefore the initial fuel load will be carried out without filling the refueling cavity. The intention of the description in DCD Section 14.2.12.2.1.3, which is "approximately equal to center of the vessel outlet nozzles", is to indicate that the RCS water level is intentionally low considering the expected water level increase due to the fuel loading. The description is not meant to imply the, so called, mid-loop level.

Impact on DCD
There is no impact on the DCD.

Impact on R-COLA
There is no impact on the R-COLA.

Impact on PRA
There is no impact on the PRA.

Impact on Topical/Technical Reports
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DATE OF MEETING: **3/4/2014**

QUESTION: Item 7

A generic comment was made that MHI made no attempt to verify success criteria defined in the PRA as part of the initial plant test program or the ITAAC. Four specific examples of this concern were identified by the ACRS.

a. PZR safety/depressurization valves – no test of just one valve to verify ability to remove decay heat via the feed and bleed operation described in the PRA.
b. RHR test (#22) – The design requirement is 2 out of 4. There is no test to verify single RHR subsystem ability to remove decay heat as described in the PRA.
c. Containment fan cooler HFT (#69) does not verify passive heat remove capability with no fans running.
d. UHS rejection test (#21) – PRA credits the heat removal capability of single train, but there is no test of this condition. All of the tests look at more than 1 train in operation.

ANSWER:
Thermal-hydraulic (T-H) analyses for the PRA success criteria are performed using design information described in DCD Tier 2 and the analysis condition do not deviate from the design information. By confirming that components are installed in the nuclear power plant in accordance with design information, it is verified that the components have the ability to perform as they are credited in the Thermo-Hydraulic (T-H) analysis. It can be confirmed that the components are installed in accordance with design information in an initial plant test program or ITAAC, and the success criteria defined in the PRA are also verified in this initial plant test program or ITAAC.

The following provide responses to the four specific items identified in the Question above.

a. Operation of the PZR safety/depressurization valves in accordance with the design is verified in Tier 1 Table 2.4.2-5 ITAAC 12.a.i. The test is implemented in accordance with DCD Subsection 14.2.12.1.4, which requires proper demonstration of the RCS depressurization function for both cold and hot conditions. The analysis condition used in the PRA success criteria (one valve) is covered by this test; therefore, the capability of the valves for feed and bleed operation can be verified by this test.

b. The heat removal capability of the CS/RHR heat exchangers is verified to be in accordance with the design by Tier 1 Table 2.4.5-5 ITAAC 8.a.i. The test is implemented in accordance with DCD Subsection 14.2.12.1.22, which requires verification of RHRS performance characteristics during RCS circulation. The PRA success criteria analyses use the heat removal capability of the RHR system, and validity of the PRA success criteria can be verified by this test.

Each of safety-related system of the US-APWR consists of four independent divisions. Although the initial test is implemented with multiple trains in operation, measured parameters such as the product of the overall heat transfer coefficient and the effective heat transfer area, UA, are verified for each division in the ITAAC. Therefore, the capability of single train can be verified by the test.

c. For alternate containment cooling, the ability of the containment fan cooler system (CFCS) depends on the fan cooler unit design specification (e.g., size, number of tubes and fins). The heat removal capacity of the alternate containment cooling was discussed in MHI's response to RAI 480-3711 Question 19-*** (2) (Ref. UAP-HF-09537, dated November 26, 2009). The heat removal capacity calculated by ACOOHT, a calculation tool validated by an experimental database, is applied to the T-H analysis for the PRA success criteria. It is verified that the CFCS is manufactured in accordance with the design specification during an inspection. Additionally, Tier 1 Table 2.7.5.3-1 ITAAC 1 ensures that the system is properly installed in the plant. By confirming the design specification, it is verified that the system has ability required in the PRA success criteria analysis.

d. The cooling chain for the PRA success criteria analyses is modeled such that the cooling water design flow rate is supplied to the components such as the SI pump and CS/RHR heat exchanger. Water supply and heat removal capability for each train of the CCWS is verified in Tier 1 Table 2.7.3.3-5 ITAAC 7.ii and 7.i, respectively. Similarly, the water supply capability of each train of the ESWS is verified in Tier 1 Table 2.7.3.1-5 ITAAC 7.ii. As documented in Tier 1 DCD Subsection 3.2.1, the heat removal capability of the UHS is also verified. The heat removal capacities of these systems are also verified in the preoperational test described in Tier 2 DCD Subsections 14.2.12.1.87 for CCWS, 14.2.12.1.34 for ESWS and 14.2.12.2.4.21 for the UHS. The parameters verified in these tests and the corresponding ITAAC are consistent with the analysis condition for the PRA success criteria.

Impact on DCD
There is no impact on the DCD.

Impact on R-COLA
There is no impact on the R-COLA.

Impact on PRA
There is no impact on the PRA.

Impact on Topical/Technical Reports
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QUESTION: Item 8

Test abstract 14.2.12.1.19 RTD/Thermocouple Cross-Calibration Preoperational Test – How do you determine the valid test instrument used to ensure RTD resolution and accuracy?

ANSWER:
The RTD test method is described in DCD Subsection 14.2.12.1.19, but the TC/RTD calibration test is also described in DCD Subsection 7.1.3.14. MHI follows the requirements in BTP 7-13, which provides guidance for safety-related RTD calibration methods. The preoperational test is the responsibility of the vendor, so the calibration test will be performed by a contractor or vendor engineer.

Impact on DCD
There is no impact on the DCD.

Impact on R-COLA
There is no impact on the R-COLA.

Impact on PRA
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QUESTION: Item 9

MHI provided a fairly comprehensive response to the previous Chapter 7 ACRS SC meeting discussion related to one way communication between the station bus and the unit bus, however, it was noted that the response did not include a test to verify proper functioning of the one way communication. Did MHI consider the need for verification via test? If so, what was the basis for not adding a test?

ANSWER:
"MHI's Response to ACRS Subcommittee Questions on April 25-26, 2013 Regarding DCD Chapter 7," UAP-HF-13232, dated September 20, 2013 addresses the ACRS Subcommittee's concern regarding the data flow between the station bus and data bus in Item13. The provided response commits to change DCD Tier 1, Table 2.5.6-1 (Sheet 1 of 2) as indicated in Attachment 13-1 of the response and replicated below for convenience.

The committed change adjusts the acceptance criteria of Tier 1 Table 2.5.6-1 ITAAC 3 to state that a report will exist and conclude the as-built isolation device is placed between the UMC and the station bus that only allows outbound communication from the as-built unit bus to external networks. MHI believes the committed change to the DCD is sufficient to verify proper functioning of the one way communication device.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. The DCS provides external networks with a communications link via the unit management computer (UMC) which is connected to the unit bus. The isolation device, which is located between the UMC and the station bus, provides a hardware-based unidirectional interface, which allows only outbound communication from the unit bus to external networks.	3. Inspection and analyses of the as-built DCS will be performed.	3. A report exists and concludes that: (1) the as-built DCS provides external networks with a communications link via the as-built unit management computer (UMC), which is connected to the as-built unit bus; (2) the as-built isolation device, which is located between the UMC and the station bus, provides a hardware-based unidirectional interface,

There are no other connections from external sources to the DCS.		which allows only outbound communication from the as-built unit bus to external networks; and (3) there are no other connections from external sources to the as-built DCS.

Impact on DCD
There is no impact on the DCD.

Impact on R-COLA
There is no impact on the R-COLA.

Impact on PRA
There is no impact on the PRA.

Impact on Topical/Technical Reports
There is no impact on the topical and technical reports.

RESPONSE FOR ACRS SUBCOMMITTEE MEETING

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CHAPTER: **14**

CHAPTER TITLE: **VERIFICATION PROGRAMS**

DATE OF MEETING: **3/4/2014**

QUESTION: Item 10

The ACRS requested additional information regarding the PRA assumptions associated with the flooding of the reactor cavity. Please provide additional information regarding the following:

a. What is the trigger to begin flooding the cavity? It is manual or automatic?
b. What MAAP scenarios are evaluated to determine the water level?
c. What determines whether it is adequate / how do you know that you added sufficient water in time to prevent core melt?

ANSWER:

Question a:

The trigger for reactor cavity flooding is identification of significant core damage. The reactor cavity is flooded by gravity when the containment spray system (CSS) is operated. The CSS is automatically initiated when the containment pressure reaches its setpoint. Therefore, if the CSS is already initiated when significant core damage is identified, then there is no need for operating additional systems in order to flood the reactor cavity. However, it is also possible that significant core damage occurs before the containment pressure reaches the CSS setpoint. In that case, the firewater system (FSS) is manually initiated to flood the reactor cavity. In the Level 2 PRA, failure of reactor cavity flooding is defined as when both CSS and FSS fail.

Question b:

Severe accident progression analysis results are documented in Chapter 14 of the PRA Technical Report MUAP-07030. Please refer to this document for reviewing the SA scenarios evaluated to study core debris cooling.

Question c:

The MAAP code is capable of evaluating how much concrete erosion occurs due to the MCCI process. Failure or success of core debris cooling is determined by reviewing the MAAP calculation results for the reactor cavity floor concrete erosion depth. It typically includes three scenarios to study MCCI for US-APWR, (1) reactor cavity is not flooded (i.e. no systems are available to flood reactor cavity, although some water exists in the cavity, which is spilled from the RCS); (2) reactor cavity is flooded before RV failure; and (3) reactor cavity is flooded after RV failure. Significant concrete erosion occurs for the scenario (1) after water boils off so that no water exists in the reactor cavity. Core debris cooling is not achievable without water in the reactor cavity. To the contrary, it is demonstrated that significant concrete erosion does not occur for scenarios (2) and (3) according to the MAAP calculation results provided in Chapter 14 of the PRA Technical Report. The amount of water available in the reactor cavity is not considered to judge debris coolability because MCCI does not occur even for the cases when reactor cavity flooding operation is initiated after RV failure. As discussed in the answer to Question a above, the trigger for reactor cavity flooding for the US-APWR is determined as identification of significant core damage. In accordance with the MAAP calculation results for the MCCI, it may be acceptable to set the triggering timing as identification of RV failure. This may allow operators more time for preparing the reactor cavity flooding operation. However, in reality, it is practically impossible to detect RV failure using current technology. It is therefore reasonable to set the trigger for cavity flooding as the identification of core damage since core exit temperature higher than 1200°F is a clearly established and detectable criterion for significant core damage.

Severe accident progression analysis in Chapter 14 of the PRA Technical Report is based on best-estimate assumption for core debris cooling. Sensitivity studies are therefore performed for the heat transfer between core debris and coolant water, as provided in Section 15.4.3 of the PRA Technical Report. It is demonstrated that the NRC-suggested success criterion for core debris cooling is satisfied even for very conservatively assumed cases. The severe accident progression analysis results provided in Chapter 14 of the PRA Technical Report are therefore considered acceptable for evaluating the core debris cooling process.

Impact on DCD
There is no impact on the DCD.

Impact on R-COLA
There is no impact on the R-COLA.

Impact on PRA
There is no impact on the PRA.

Impact on Topical/Technical Reports
There is no impact on the topical and technical reports.