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January 21, 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-11012

Subject: MHI's Response to US-APWR DCD RAI No. 663-4996 (SRP 03.09.05)

References: 1) "Request for Additional Information No. 663-4996 Revision 0, SRP Section: 03.09.05 – Reactor Pressure Vessel Internals, Application Section: DCD, Tier 2 – Section 3.9.5," dated 11/15/2010.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Response to Request for Additional Information No. 663-4996 Revision 0."

Enclosed are the responses to questions 03.09.05-28 through 34 of the RAI (Reference 1).

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.

Enclosures:

1. Response to Request for Additional Information No. 663-4996, Revision 0

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

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Enclosure 1

UAP-HF-11012
Docket No. 52-021

Response to Request for Additional Information No. 663-4996,
Revision 0

January, 2011

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

1//21/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 663-4996 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**3.9.5**
DATE OF RAI ISSUE:	**11/15/2010**

QUESTION NO.: 03.09.05-28

The staff requested the applicant in RAI 374-2446, Question 03.09.05-27 (#10958) to:
(a) provide clarification for the classification of the reactor internals hold-down spring;
(b) provide technical justification for any classification which would not require use of the design, fabrication, examination, and documentation requirements of the ASME Code Section III, Subsection NG for design and construction of the hold-down spring; and
(c) revise DCD Sections 3.9.5.1 and 3.9.5.1.3, and DCD Table 3.2-2, including the requested information.

In MHI's response, dated July 17, 2009, the applicant stated that the hold-down spring is classified as an internal structure. The applicant provided the following information as technical justification for classifying the hold-down spring as an internal structure:

> "The primary functions of the hold-down spring are to allow compliance for thermal expansion between the reactor vessel and the reactor internals (upper support flange and core barrel flange), and provide sufficient preload to the flanges to prevent excessive vibration or sliding during operation. These are considered to be not core support requirements, but functional requirements.
> The reason the hold-down spring has always been classified as an internal structure is because it is not required to directly support the core. The computational modeling of the reactor internals, for example, has the load path of the hold-down spring not connected in series but in parallel with those of the fuel assemblies.
> One other point should be made. Even if the hold-down spring loses all its preload from stress relaxation, the shape of the hold-down spring will remain unchanged - and the vertical loads from the core can still be transferred through the hold-down spring to the upper support and core barrel flanges and then to the vessel head and vessel flange. This extreme example of complete loss of preload is undesirable from a functional standpoint because of the potential adverse effects on vibration and sliding of the reactor internals, but does not warrant re-classifying the hold-down spring as a core support structure."

The applicant further stated that response to questions on the hold-down spring classification and function are also explained in the response to RAI 374-2446, Question 03.09.05-2 (#10080). The applicant also stated that Table 3.2-2 of the DCD would be changed to include internal structures, based on the response to RAI 03.02.01-14.

In its review of the applicant's response, including the response to Question 03.09.05-2, the staff noted that the applicant had not provided adequate technical justification for classifying the hold-down spring as an internal structure, and not as a core support structure. ASME Section III, Article NG-1121 defines core support structures as structures or parts of structures, which provide direct support or restraint of the core within the reactor pressure vessel. Subsection 3.9.5.1.1 of the DCD states that the horizontal loads on the upper core support assembly are transmitted from the upper core support flange to the RV head and hold-down spring by friction or direct contact with the RV flange. Furthermore, in its response the applicant stated: "This extreme example of complete loss of preload is undesirable from a functional standpoint because of the potential adverse effects on vibration and sliding of the reactor internals, ." Therefore, the staff's concerns summarized in the original RAI question are not resolved. Consequently RAI 374-2446, Question 03.09.05-27 remains open.

In this supplementary question (03.09.05-27.1), the applicant is requested to further justify why the hold-down spring is not considered to be a component contributing to support of the reactor core.

References: (1) MHI's Response to US-APWR DCD RAI No. 374-2446; MHI Ref: UAP-HF09387; July 17, 2009; ML092040046.

ANSWER:

The functions of the hold-down spring are (i) to allow compliance for thermal expansion between the reactor vessel and (ii) the reactor internals (upper core support flange and core barrel flange) and (provide sufficient preload on the core barrel flange to prevent lift-off from the support ledge of the reactor vessel due to the hydraulic force from the view point of long term vibration and wear of reactor internals.

Even if the preload of the hold-down spring is lost, the core barrel flange would not lift during normal operating conditions because the hydraulic lift force is smaller than downward loads such as the dead weights and the hold-down spring force of the fuel assemblies. In abnormal conditions or seismic/accident conditions, the core barrel may lift off from the vessel support ledge, but the lift displacement is limited by the small stroke of the hold-down spring. Vertical restraint of the core barrel is established on the top surface of the core barrel flange when the core barrel flange is lifted. The restraint provided by the upper core support flange is not affected.

Thus, the core support and restraint functions are accomplished by the lower and upper core support assemblies even if the preload of the hold-down spring is completely lost as discussed further in the response Question 03.09.03-29. The preload of the hold-down spring is required not for the core support function but for the long-term reliability of the reactor internals. Therefore, the hold-down spring is not considered to be a core support structure.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

1/21/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 663-4996 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESSEL INTERNALS**
APPLICATION SECTION:	**3.9.5**
DATE OF RAI ISSUE:	**11/15/2010**

QUESTION NO.: 03.09.05-29

The staff requested the applicant in RAI 374-2446, Question 03.09.05-2 (#10080) to provide an assessment of the potential loss of preload of the hold-down spring due to stress relaxation during the design lifetime, and discuss its effect on the horizontal and vertical restraints of the upper core support and core barrel assemblies. Revise the DCD to include the requested information or, alternately, provide a reference document where this information is available.

In MHI's response, dated July 17, 2009, the applicant stated the following:

> "The hold-down spring in PWR operating plants have been observed in the industry to have some loss of preload from inelastic deformation of the contact surfaces during initial bolt-up and subsequent stress relaxation from plant operation. Since the material and design of the USAPWR hold-down spring is similar to that which has been successfully used in many operating PWR plants, it is not expected that the loss of preload will affect the hold-down spring functionality."

The applicant further stated that the third paragraph in the DCD Section 3.9.5.1.1, Upper Reactor Internals Assembly Design Arrangement, will be changed as shown below to read:

> "For loads in the upward vertical direction, the upper core support assembly is vertically restrained by the RV head flange and in the downward direction by the RV flange. A toroidal-shaped hold-down spring is sandwiched between the upper core support flange and the core barrel flange. The primary function of the hold-down spring is to accommodate the thermal expansion differences between the RV and the reactor internals upper core support flange and core barrel flange. A vertical preload in the hold-down spring is developed during installation of the upper internals and is controlled by a fixed distance between the bottom of the upper core support flange and the top of the core barrel flange. Vertical loads on the upper core support assembly come from dead weights less buoyant forces, the upper core support and the upper core plate differential pressure loads, vibration loads on the components, fuel assembly spring and lift loads, and seismic and postulated LOCA loads. There is a designed radial gap between the upper core support flange and the RV inside diameter. The gap is large enough to prevent contact from thermal expansion of the upper core support flange relative to the RV flange during operation. Horizontal

loadings from flow loads, vibration loads, and seismic and pipe-rupture loads are transmitted from the upper core support flange to the RV head and hold-down spring by friction or direct contact with the RV flange. Head and vessel alignment pins also transmit some of the horizontal loads to the RV head and RV flanges."

In response to the effects of loss of preload in the hold-down spring on the horizontal and vertical restraints of the upper core support and core barrel assemblies, the applicant stated that,

"... it is not expected that the loss of preload will affect the hold-down spring functionality.",

but did not provide any technical basis for this statement. Furthermore, the applicant's response to this RAI seems to contradict the response to RAI 374-2446, Question 03.09.05-27 (#10958), where the applicant stated,

"This extreme example of complete loss of preload is undesirable from a functional standpoint because of the potential adverse effects on vibration and sliding of the reactor internals. "

The staff also noted that the proposed changes in DCD Subsection 3.9.5.1.1 basically describe the function of the hold-down spring and do not discuss the potential loss of preload of the hold-down spring and its effect on the horizontal and vertical restraints of the upper core support and core barrel assemblies. Therefore, the staff finds the applicant's response inadequate and Question 03.09.05-2 remains open.

The applicant is requested to respond to the original question, and discuss the effect of potential hold-down spring load relaxation on the horizontal and vertical restraint of the upper core support and core barrel assemblies, and ultimately the support of the reactor core.

References: (1) MHI's Response to US-APWR DCD RAI No. 374-2446; MHI Ref: UAP-HF09387; July 17, 2009; ML092040046.

ANSWER:

If the preload on the hold-down spring is completely lost, the following are phenomena and potential impacts:

1. Normal Operating Conditions

a. Possibility of core barrel lift

The core barrel flange does not lift off under the normal operation because the hydraulic forces on the core barrel, the neutron reflector, and the fuel assemblies do not exceed the downward loads from the dead weights of these components and the hold-down spring force of the fuel assemblies.

b. Effects on lower core support structures

The core barrel vertical alignment remains by the contact between the bottom surface of the core barrel flange and the reactor vessel support ledge. The core barrel flange primary lateral restraint under horizontal loadings is the reactor vessel flange. Prior to contact with the reactor vessel flange, any lateral sliding may be absorbed by friction between the core barrel flange and reactor vessel support ledge. However, if the friction is insufficient to restrain the lateral motion of the core barrel flange, the vessel alignment pins would further restrict the motion of the core barrel flange when the gap between the vessel pins and the reactor vessel slot is closed.

c. Effects on upper core support structures

The upper core support is aligned and restricted in the vertical direction at the upper surface of the upper core support flange in contact with the reactor vessel head. Similar to the core barrel flange, the upper core support flange primary lateral restraint is the reactor vessel. Before making contact with the reactor vessel, friction between the vessel head and the upper core support flange must be overcome as well as the vessel head-pin deformation. It should be noted that a vertical contact load from the fuel assembly hold down springs should maintain a frictional restraint between the vessel head and upper core support flange.

2. In abnormal or seismic/LOCA conditions

a. Possibility of core barrel lift

With loss of hold-down spring preload, the core barrel flange may momentarily lift-off in the abnormal condition with higher flow conditions. Also in seismic and LOCA conditions, some momentary lift-off may occur by from large vertical loadings. In no way is it expected that this momentary lift-off will damage the reactor internals or result in preventing shut-down of the reactor

b. Effects on lower core support structures

The core barrel flange vertical displacement may by dynamically affected but the lift-off displacement is limited within the structural (stiffness) vertical displacement of the hold down spring. This motion is easily absorbed by the deflection of the fuel assembly hold down spring which is typically ten times larger than the potential core barrel lift, therefore, the restraint of the fuel assemblies are accomplished. Sliding lateral motion of the core barrel flange is also possible, in seismic or LOCA conditions, these dynamic displacements of the core barrel flange could occur even if the preload of the hold down spring is maintained and therefore, are accounted for in the present dynamic analysis.

c. Effects on upper core support structures

As for the LCSP, the dynamic downward vertical motion of the upper core support may be possible with loss of hold-down spring preload, but the maximum lift-off displacement is limited by the stiffness of the hold down spring. This vertical displacement is easily absorbed by the stiffness of the fuel assembly hold down springs and the restraint of the fuel assemblies is therefore accomplished. Some sliding lateral motion of the upper core support flange is also possible. In seismic or LOCA conditions, these dynamic displacements of the upper core support flange could occur even if the preload of the hold down spring is maintained and therefore, are accounted for in the present dynamic analysis.

As discussed above, the functions to support and restrain the core are accomplished by the upper core support structures (upper core support column and upper core plate) and the lower core support structures (the core barrel and lower core support plate) even if the preload of the hold-down spring is completely lost.

As stated in 1. b., the lateral vibration of the core barrel may be affected by the loss of preload of the hold-down spring. Even though the lateral displacement is limited by the vessel alignment pins, wear of the pins may still be of concern. However, this concern is not related to the core support function but rather for the long-term reliability of the reactor internals.

Therefore, it is designed that the sufficient pre-load of the hold-down is provided on the core barrel flange to prevent lift-off from the support ledge of the reactor vessel due to the hydraulic force from the view point of long-term vibration and wear on the vessel alignment pins.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

1/21/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 663-4996 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESSEL INTERNALS**
APPLICATION SECTION:	**3.9.5**
DATE OF RAI ISSUE:	**11/15/2010**

QUESTION NO.: 03.09.05-30

The staff requested the applicant in RAI 374-2446, Question 03.09.05-3 (#10081) to provide sufficient details about the design of the upper core plate and its interface with the fuel assemblies, core barrel, upper support columns, and lower guide tubes. Also, explain any differences from the existing 4-loop design, and how these differences are evaluated against possible excitation mechanisms of flow-induced vibration. Revise Section 3.9.5 of the DCD to include sufficient information about the design arrangement of the upper core plate and a discussion of the differences, if there are any, in its loading conditions from the reference 4-loop reactor.

In MHI's response, dated July 17, 2009, the applicant provided the following information:

> "The US-APWR upper core plate and its interface with the fuel assemblies, core barrel, upper support columns, and lower guide tubes are similar to those of the existing 4-loop design. So, there is expected to be little impact on the flow-induced vibration due to the structural design changes around the upper core plate.
>
> More detail of discussions about the design differences of the US-APWR reactor internals from current 4-loop and effects on the flow-induced vibration are described in Chapter 2.1 of MUAP-07027 (R1): Comprehensive Vibration Assessment Program for US-APWR Reactor Internals."

Since the design of the upper core plate and its interface with other components is similar to that used in existing 4-loop reactors, the staff's concerns regarding the upper core plate design are resolved. However, the applicant did not commit to revising the DCD to include this information. The applicant is requested to revise DCD Section 3.9.5 to include this additional information. This supplementary question (03.09.05-3.1) will track this action as a confirmatory action until completed.

Reference: MHI's Response to US-APWR DCD RAI No. 374-2446; MHI Ref: UAP-HF09387; July 17, 2009; ML092040046.

ANSWER:

As requested in this RAI, DCD will be revised to include the information in the previous response to RAI Question 03.09.05-3.1.

Impact on DCD

The following paragraph will be added to the end of DCD Subsection 3.9.5.1.1, "Upper Reactor Internals Assembly Design Arrangement"

"The US-APWR upper core plate and its interface with the fuel assemblies, core barrel, upper support columns, and lower guide tubes are not different from those of the existing 4 loop design. So, there is little impact on the flow-induced vibration due to the structural design changes around the upper core plate. More detail about the design differences between the US-APWR reactor internals and current 4 loop and effects on flow-induced vibration are described in Chapter 2.1 of Reference 3.9-22."

Reference 3.9-22 Comprehensive Vibration Assessment Program for US-APWR Reactor Internals. MUAP-07027-P(R1), Mitsubishi Heavy Industries, May 2009.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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US-APWR Design Certification

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Docket No. 52-021

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APPLICATION SECTION:	**3.9.5**
DATE OF RAI ISSUE:	**11/15/2010**

QUESTION NO.: 03.09.05-31

DCD Subsection 3.9.5.1.1 presents a description of the US-APWR upper reactor internals assembly design arrangement, including the manner of positioning and securing of these items and coolant flow through the reactor internal assemblies. It states that the "Exit flow core pressure difference between the fuel assemblies is limited by the design to an acceptable cross-flow velocity to prevent vibratory damage to the fuel rods, thimbles, or RCCAs." The staff's review of Subsection 3.9.5.1.1 showed that the applicant did not explain how the thermal-hydraulic design requirement regarding the fuel assembly exit core flow would be verified. As stated in Subsection 3.9.5.3.2 of the DCD, the thermal-hydraulic performance criteria require that the "Core outlet flows from the fuel assemblies are to be designed to minimize horizontal velocities that may contribute to vibration of the RCCA rodlets." In RAI 374-2446, Question 03.09.05-6 (#10084), the staff requested the applicant to describe the procedure that is to be used to verify that the exit flow from the fuel assemblies does not lead to unacceptable cross-flow velocities that may cause vibration of the fuel rods, thimbles, or RCCAs. The applicant was also requested to revise Subsection 3.9.5.1 of the DCD to include the requested information.

In MHI's response, dated July 17, 2009, the applicant stated the following:

> "It is verified that the fuel assembly exit cross-flow velocity is acceptable for the US-APWR by operating plants with similar design features to the US-APWR fuel assemblies and upper internals.
>
> The design of the upper core plate flow holes, fuel assembly loss coefficients, and the fuel assembly design of the US-APWR is not significantly different from those of existing 4 loop plants. So the cross flow velocities at the core outlet are expected to be similar to the existing 4 loop plants. From the experience of existing 4 loop plants, the adverse flow effects on the vibration of the fuel rods or RCCA is acceptably limited."

The staff finds the response to this RAI unacceptable because it is vague and does not provide quantitative information about the design differences that may cause adverse flow effects of the fuel rods and thimbles. It is not clear what is meant by the phrase "the design is not significantly different" from existing 4-loop plants. Therefore, the applicant is requested in this supplementary question (03.09.05-6.1), to provide a quantitative assessment of the effect of differences from existing 4-loop plants on the cross-flow excitation of the fuel rods, thimbles, and RCCAs of the US APWR.

Reference: MHI's Response to US-APWR DCD RAI No. 374-2446; MHI Ref: UAP-HF09387; July 17, 2009; ML092040046.

ANSWER:

An assessment of the potential adverse cross-flow effect at the core outlet of US-APWR was performed in comparison with the existing plants as follows.

1. Cross-flow condition

A comparison of the flow parameters with the typical current 4-loop design is shown in Table 1.

Although the total flow rate of US-APWR is larger than that of the existing plants, the flow rate per fuel assembly is approximately 10% lower for its larger number of fuel assemblies.

The expected cross-flow velocity due to mal-distribution of the core outlet flow of US-APWR is also 10% lower than the current 4-loop designs the same as the fuel assembly flow rate.

2. Vibration characteristics of the fuel assemblies and RCCA

The mechanical properties of the fuel assembly for US-APWR are compared with the typical current 4-loop designs in Table 1. The fuel rod diameter and the pitch are equivalent. "Typical support span" is defined as "Effective fuel length" divided by "Number of grids per assembly". In general, the shorter support span leads to the higher natural frequency and larger margin for flow-induced vibrations. Because the typical support span of the US-APWR fuel assembly is shorter than the current 4-loop designs as shown in Table 1, the margin for cross-flow vibration is expected to be improved from the current 4-lloop designs for both the fuel rods and thimbles.

As for the RCCA rods, the vibration characteristics do not different from the current design because the mechanical properties such as rod diameter and cladding material thickness are equivalent as shown in Table 1.

3. Conclusion

In the currently operating 4-loop plants, there has been no observed component degradation, such as a high cycle fatigue by flow-induced vibration due to cross-flow at the core outlet.

From the discussions above, there is no concern for the cross-flow due to the core outlet flow mal-distribution in the US-APWR design.

Table 1 Comparison of Reactor Design Parameters

Parameter	US-APWR	Typical 12-ft 4-loop PWR	Typical 14-ft 4-loop PWR
Flow Parameters			
Vessel thermal design flow (10^6 lbm/hr)	168.2	139.4	145.0
Core bypass flow (%)	9.0	6.4	8.5
Number of fuel assemblies	257	193	193
Flow rate per assembly (10^6 lbm/hr)	0.596	0.676	0.687
Fuel Assemblies			
Assembly overall dimensions (in) Fuel rod pitch (in)	8.426 x 8.426 0.496	8.426 x 8.426 0.496	8.426 x 8.426 0.496
Fuel assembly array Number of fuel rods	17x17 264	17x17 264	17x17 264
Fuel rod OD (in)	0.374	0.360	0.374
Effective fuel length (in)	165.4	143.7	168
Number of grids per assembly	11	8	10
Typical support span (in) = Fuel length / grid numbers	15.0	18.0	16.8
Rod Cluster Control Assemblies			
Number of absorber rods per cluster	24	24	24
Absorber diameter (in)	0.341	0.341	0.366
Cladding material thickness, for Ag-In-Cd (in)	Type 304 SS 0.0185	Type 304 SS 0.0185	Type 304 SS 0.0185

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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US-APWR Design Certification

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DATE OF RAI ISSUE:	**11/15/2010**

QUESTION NO.: 03.09.05-32

DCD Subsection 3.9.5.1.2 states that

> "The lower core support assembly consists of a lower core support plate, six radial support keys, and fuel alignment pins. The lower core support plate has orificed flow holes to reduce mal-distribution of the flow into the core."

The safety analysis design requirements for US-APWR internals listed in Subsection 3.9.5.3.1 of the DCD state that

> "Mal-distribution of flow to the core should be limited so as not impact core safety limits"

in Chapter 15 of the DCD. However, the applicant did not refer to any safety analysis that would ensure compliance with this safety requirement for the design of US-APWR core support structures and core internals. Therefore, the staff requested, in RAI 374-2446, Question 03.09.05-7 (#10085), the applicant to discuss the analysis performed and the measures undertaken to make sure that the mal-distribution of the flow into the core shall be limited so as not to impact the US-APWR core safety limits. The applicant was also requested to revise DCD Section 3.9.5 to provide the requested information or, alternately, provide a reference document where this information is available. In MHI's response, dated July 17, 2009, the applicant stated the following (public version):

> "Mal-distribution of flow into the core is limited by meeting several reactor internals design requirements. These design requirements include the allowable minimum and maximum fuel assembly inlet flow rate, and the allowable difference in inlet flow rates between adjacent fuel assemblies. The design target values are . [ ]% of nominal flow rate for the minimum flow rate; . [ ]% of nominal for the maximum flow rate; and . [ ]% for difference between adjacent fuel assemblies. These design requirements are similar to those in operating 4-loop US plants. Confirmatory testing was performed for the US-APWR, Reference (1), and the results show that the minimum assembly flow was [ ]%; the maximum assembly flow was [ ]%; and the difference between adjacent fuel

assemblies was [ ]%. From the test results, it is concluded that the inlet core flow distribution is such as to preclude adverse effects such as core tilt, flow starvation, or undesirable inlet cross-flow distribution."

The staff finds MHI's response to this RAI acceptable because it indicates that well defined design targets are established for the flow distribution into the core. These design targets are similar to those used for the operating 4-loop US plants. In addition, confirmatory tests have been performed to validate that the design parameters of the US APWR are within the acceptable range to avoid mal-distribution of flow into the core. However, in its response, the applicant did not identify Reference 1. Therefore, as a confirmatory action, the applicant is requested in this supplementary question (03.09.05-7) to revise the DCD by identifying Reference 1, which includes confirmatory test results for the US-APWR core internals, and include it in the appropriate list of DCD references.

Reference: MHI's Response to US-APWR DCD RAI No. 374-2446; MHI Ref: UAP-HF09387; July 17, 2009; ML092040046.

ANSWER:

As requested in this RAI, DCD will be revised to include the information in the previous response to RAI Question 03.09.05-7.

Impact on DCD

The second, third, and fourth paragraphs of DCD Subsection 3.9.5.3.2 "Thermal-Hydraulic Design Basis" will be modified as shown below to include the requested information.

"A discussion of the reactor coolant flow path is described below.

The reactor coolant flow path for the reactor internals is depicted in Figure 3.9-8. Primary coolant flow at T_{cold} enters into the downcomer, the annular space between the reactor vessel inside wall, and the core barrel outside surface. The main coolant flow then enters the bottom of the reactor vessel and turns upward, flowing past the diffuser plates and distributing into the lower core support plate orifice holes. The orifices are carefully designed to control the flow into the fuel assemblies and to minimize uneven flow distributions and hot spots. Mal-distribution of flow into the core is limited by three design criteria. The first criterion is the minimum flow rate determined from the view point of thermal hydraulic design. The second criterion is the maximum fuel assembly inlet flow rate for the limit of the fuel assembly lift. And the third criterion is that the difference in inlet flow rates between adjacent fuel assemblies is determined to limit the cross-flow-induced vibration of the fuel rods. These design requirements are similar to those in the operating 4-loop plants. A confirmatory testing was performed for US-APWR (Reference1.5-3). From the test results, it is concluded that the core inlet flow distribution is sufficiently uniform to satisfy the design requirements mentioned above.

The coolant is heated in each fuel assembly to a fluid temperature depending on its core location in the core loading pattern. The hot assembly flow exits the fuel assemblies and enters the holes in the upper core plate. No fuel assembly exit temperature exceeds the water saturation temperature. This is to preclude bulk-boiling in the main coolant flow. The upper core plate has two types of flow holes. One type is circular in shape and the other in rectangular shape. The circular shape is for open exit flow or exit flow below the upper support columns. The rectangular shape is for exit flow below the guide tubes. Most of the main flow that enters the guide tube exits through the "windows" into the upper plenum cavity. Some of the flow exits through the controlled gap between the bottom of the guide tube flange and the top of the upper core plate. The guide tubes and the support columns are carefully configured to minimize the pressure drop and the cross-flow from the core exit fluid."

Reference 1.5-3 US-APWR Reactor Vessel Lower Plenum 1/7 Scale Mode Flow Test Report, MUAP-07022-P(R0), June, 2008, MHI

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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DATE OF RAI ISSUE:	**11/15/2010**

QUESTION NO.: 03.09.05-33

The reactor internal loads are categorized according to the design and service loading conditions for the plant. The ASME Code, Section III load combinations for core support structures (CSSs) and threaded structural fasteners are given in Table 3.9-11 of the DCD, and the stress categories and service limits are given in Table 3.9-12. In DCD Tier 2, Subsection 3.9.5.2.2, the applicant stated that the service limits for reactor internals other than the CSSs are not addressed in the ASME Code, Section III. However, because the structural integrity of the reactor internals is important-to-safety, the stress limits for CSSs are also applied to the reactor internals. If the stress limits for the internal structure do not meet the ASME Code, Section III limits for the CSSs, the applicant proposes to utilize alternate acceptance criteria "...based on validation by testing, sound engineering judgment, and experience with similar designs." The staff's review of the DCD showed that the applicant neither provided sufficient information about the proposed alternate acceptance criteria nor on the resulting safety margin. In RAI 374-2446, Question 03.09.05-15 (#10094) the staff requested the applicant to explain in more detail the meaning of the following statement, which is given in Subsection 3.9.5.2.2 of the DCD:

> "However, if the stress limits for the internal structure do not meet the ASME Code, Section III (Reference 3.9-1) limits for the core support structures, then alternate acceptance criteria are employed based on validation by testing, sound engineering judgment, and experience with similar designs."

The applicant was also requested to provide a list of all components, which did not meet the ASME Code for stress limits and explain the alternate design criteria used for these components. Also, the applicant was requested to revise Section 3.9.5 of the DCD to provide the information.

In MHI's response, dated July 17, 2009, the applicant stated that the loading conditions and stress limit for the Class CS were applied for the reactor internals, except the secondary core support structures. The applicant further stated that the function of the secondary core support assemblies was to limit the stroke of the drop and the impact force on the lower vessel head in the postulated core drop event. Therefore, the design of the secondary core support structures including the lower diffuser plate are determined with the impact force in the core drop event as a beyond-design basis accidents. The applicant also provided a table showing the load combination and acceptance criteria for the secondary core support structures, including the stress limit for design and beyond design basis accidents.

The staff finds the response acceptable because the applicant provided the details regarding the stress limits and design criteria for the reactor internals. However, the applicant did not include this information in the revised DCD as requested in the original RAI question. Therefore, as a confirmatory action, the applicant is requested in this supplementary question (03.09.05-15.1) to include this information in the next revision of the US-APWR DCD.

Reference: MHI's Response to US-APWR DCD RAI No. 374-2446; MHI Ref: UAP-HF-09387; July 17, 2009; ML092040046.

ANSWER:

As requested in this RAI, DCD will be revised to include the information in the previous response to RAI Question 03.99.05-15.1.

The DCD will revise to include the original RAI response.

Impact on DCD

DCD Subsection 3.9.5.2.2 Design and Service Limits is modified as follows:

3.9.5.2.2 Design and Service Limits

The reactor internals loads are categorized according to the design and service loading conditions for the plant. Table 3.9-11 and Table 3.9-12 list the ASME Code, Section III (Reference 3.9-1) load combinations and service limits for core support structures and threaded structural fasteners, respectively.

Internal structure service limits are not addressed in the ASME Code, Section III (Reference 3.9-1). However, because of their importance to the safe operation of the reactor internals, the stress limits for core support structures are applied except for the secondary core support structures. The function of the secondary core support assemblies is to limit the stroke of the drop and the impact force on the lower vessel head in the postulated core drop event. Therefore, the design of the secondary core support structures including the lower diffuser plate is determined by the impact force associated with the core drop event as a beyond-design basis accidents. The load combinations and stress limit for the secondary core support structures are shown in Table 3.9-15.
~~However, if the stress limits for the internal structure do not meet the ASME Code, Section III (Reference 3.9-1) limits for the core support structures, then alternate acceptance criteria are employed based on validation by testing, sound engineering judgment, and experience with similar designs.~~

Table 3.9-15 Load combinations and stress limits for Secondary Core Support Structures

Operating conditions	Load conditions	Occurrence	Stress limit	Remarks
Design	Design load combination for CS(1)	-	Design for CS(2)	(1) Table 3.9-11 (2) Table 3.9-12
Beyond-design basis accidents	Core drop load	1	Level D for CS(3) + No buckling	(3) Table 3.9-12

Note: 'CS' means Core Support Structures.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

1/21/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 663-4996 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESSEL INTERNALS**
APPLICATION SECTION:	**3.9.5**
DATE OF RAI ISSUE:	**11/15/2010**

QUESTION NO.: 03.09.05-34

The staff requested the applicant in RAI 374-2446, Question 03.09.05-17 (#10096) to provide the technical basis for defining the displacement limits listed in DCD Table 3.9-2 and to revise Subsection 3.9.5.2.3 of the DCD to include the requested information or provide a reference document where the

requested information is available. In MHI's response, dated July 17, 2009, the applicant stated that the technical basis of the loads and deformation limits in Table 3.9-2 of DCD are explained as follows.

> "(a) Allowable horizontal load of the RCCA guide tube should not impede insertion of the RCCA after the LOCA event.
> Technical Basis: The horizontal load limit provides assurance that after a SSE + LOCA combined event, the inelastic deformation of the guide tube is such that the control rods will be unimpeded during rod drop insertion. The horizontal load or displacement limit is determined from testing.
> (b) Upper core barrel radial displacement to prevent impeding emergency core cooling flow in RV downcomer.
> Technical Basis: The limit of the radial outward deformation of the upper core barrel, 60 mm, is determined such that the flow area of the connection part of the inlet nozzle to the downcomer is not smaller than the inlet pipe section area.
> (c) RV and upper head flange loads; Lower radial key loads; and Postulated core drop bottom of RV impact load and bearing area.
> Technical Basis: Lower radial key loads are limited by the reactor vessel radial restraints. Postulated core drop bottom of RV impact load and bearing area are also limited by the reactor vessel bottom head stresses.
> (d) The maximum vertical displacement of the upper core plate relative to the upper support plate should preclude buckling of the guide tube
> Technical Basis: The maximum relative displacement between the upper core plate and the upper core support plate 3 mm is based on the axial clearance of the shoulder of GT support pin and the upper core plate to avoid the axial loading on the guide tube.
> (e) Upper core barrel permanent displacement should not prevent loss of function of the RCCA by radial inwardly deforming the upper guide tube.

Technical Basis: The maximum inward radial deformation of the upper core barrel of 270 mm is determined based on the horizontal distance between the lower guide tube and the core barrel inside wall to prevent the interaction with the guide tube."

The staff finds that the applicant has provided the technical basis for defining the displacement limits listed in DCD Table 3.9-2, and the applicant's responses are acceptable with the exception of items (a) and (c). In item (a) of the response the applicant stated that the horizontal load or displacement limit is determined from testing but did not commit to providing this test report as a reference. Also, in item (c) of the response the applicant discussed only the lower radial key loads and postulated core drop bottom of RV impact loads and bearing area but not the RV and upper head flange loads.

Therefore, in this supplementary question (03.09.05-17.1) the applicant is requested to provide:

(a) the test report, used for determining the horizontal load and displacement limits, for staff review, and include it in the appropriate list of DCD references, and
(b) the technical basis for defining the loads and displacement limits for the RV and upper head flange.

Reference: MHI's Response to US-APWR DCD RAI No. 374-2446; MHI Ref: UAP-HF-09387; July 17, 2009; ML092040046.

The reactor internal loads are categorized according to the design and service loading conditions for the plant. The ASME Code, Section III load combinations for core support structures (CSSs) and threaded structural fasteners are given in Table 3.9-11 of the DCD, and the stress categories and service limits are given in Table 3.9-12. In DCD Tier 2, Subsection 3.9.5.2.2, the applicant stated that the service limits for reactor internals other than the CSSs are not addressed in the ASME Code, Section III. However, because the structural integrity of the reactor internals is important-to-safety, the stress limits for CSSs are also applied to the reactor internals. If the stress limits for the internal structure do not meet the ASME Code, Section III limits for the CSSs, the applicant proposes to utilize alternate acceptance criteria "...based on validation by testing, sound engineering judgment, and experience with similar designs." The staff's review of the DCD showed that the applicant neither provided sufficient information about the proposed alternate acceptance criteria nor on the resulting safety margin. In RAI 374-2446, Question 03.09.05-15 (#10094) the staff requested the applicant to explain in more detail the meaning of the following statement, which is given in Subsection 3.9.5.2.2 of the DCD:

"However, if the stress limits for the internal structure do not meet the ASME Code, Section III (Reference 3.9-1) limits for the core support structures, then alternate acceptance criteria are employed based on validation by testing, sound engineering judgment, and experience with similar designs."

The applicant was also requested to provide a list of all components, which did not meet the ASME Code for stress limits and explain the alternate design criteria used for these components. Also, the applicant was requested to revise Section 3.9.5 of the DCD to provide the information.

In MHI's response, dated July 17, 2009, the applicant stated that the loading conditions and stress limit for the Class CS were applied for the reactor internals, except the secondary core support structures. The applicant further stated that the function of the secondary core support assemblies was to limit the stroke of the drop and the impact force on the lower vessel head in the postulated core drop event. Therefore, the design of the secondary core support structures including the lower diffuser plate are determined with the impact force in the core drop event as a beyond-design basis accidents. The applicant also provided a table showing the load combination and acceptance criteria for the secondary core support structures, including the stress limit for design and beyond design basis accidents.

The staff finds the response acceptable because the applicant provided the details regarding the stress limits and design criteria for the reactor internals. However, the applicant did not include this information in the revised DCD as requested in the original RAI question. Therefore, as a confirmatory action, the applicant is requested in this supplementary question (03.09.05-15.1) to include this information in the next revision of the US-APWR DCD.

Reference: MHI's Response to US-APWR DCD RAI No. 374-2446; MHI Ref: UAP-HF-09387; July 17, 2009; ML092040046.

ANSWER:

(a) The report for RCC Guide Tube Mechanical Test is on going. The test report will be submitted by the end of March 2011.

(b) The loads and its bearing area at the reactor vessel and the upper head flange are limited by the bearing stress (contact force per bearing area) on the core barrel flange and the upper core support flange. The bearing stress limit for each operating condition is specified in Table 3.9-12 in DCD Subsection 3.9.5.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.
