

REQUEST FOR ADDITIONAL INFORMATION 663-4996 REVISION 0

11/15/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

SRP Section: 03.09.05 - Reactor Pressure Vessel Internals

Application Section: 3.9.5

QUESTIONS for Engineering Mechanics Branch 1 (AP1000/EPR Projects) (EMB1)

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

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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.

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

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 US-APWR 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

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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.

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

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

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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-HF-09387; July 17, 2009; ML092040046.

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

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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-HF-09387; July 17, 2009; ML092040046.

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

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allowable difference in inlet flow rates between adjacent fuel assemblies.

The design target values are $\geq []\%$ of nominal flow rate for the minimum flow rate; $\geq []\%$ of nominal for the maximum flow rate; and $\leq []\%$ 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 (03.09.05-7.1) 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-HF-09387; July 17, 2009; ML092040046.

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

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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.

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.

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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.