**REQUEST FOR ADDITIONAL INFORMATION 374-2446 REVISION 0**

5/21/2009

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 Reactor Pressure Vessel Internals

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

03.09.05-1

In DCD Tier 2, Subsection 3.9.5.1 the applicant stated that on the periphery of the upper core plate there are several top slotted columns and mixing devices designed to provide a uniform exit flow and temperature distribution to the outlet loop pipes. There are also two reactor vessel (RV) level instrumentation support tubes that measure the water level in the reactor vessel.

The staff reviewed Subsection 3.9.5.1 and found that the applicant did not provide sufficient information to allow the review of the supporting structures design and their liability to potential adverse flow effects. The DCD should explicitly state whether these structures and their operating environment are similar to those of the existing 4-loop reactor design. If this is not the case for some supporting structures, explain the differences and provide appropriate flow-induced vibration analysis for those structures. The applicant is requested to provide more details of the instrumentation supporting structures [e.g. thermocouple, water level sensor, in-core nuclear instrumentation system (ICIS), control and drive rod assembly] as well as the relevant flow-induced vibration analysis for these structures. The staff needs this information to assure conformance with GDC-1 and 4. Revise Section 3.9.5 of the DCD to include sufficient information about the instrumentation supporting structures and their relevant flow-induced vibration analysis.

03.09.05-2

The applicant stated in Subsection 3.9.5.1.1 of the DCD that the upper core support assembly is restrained vertically in the upward direction by the RV head flange and in the downward direction by the reactor internals hold-down spring. The preload in the hold-down spring during installation, is controlled by a fixed distance between the bottom of the upper core support flange and the top of the core barrel flange. The horizontal loads on the upper core support assembly due to flow, vibration, and seismic and pipe rupture events 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.

The staff's review of the DCD indicated that the applicant did not discuss the potential loss of preload in the hold-down spring due to stress relaxation during service and its potential effect on the functional and structural integrity of upper core support assembly.

The applicant is requested 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. Alternately, provide a reference document where this information is available. The staff needs this evaluation for the above mentioned plant components to assure conformance with GDC-1 and 4. Revise the DCD to include the requested information.

03.09.05-3

In DCD Tier 2, Subsection 3.9.5.1.1 the applicant provided a description of the US-APWR upper reactor internals assembly design arrangement, including the manner of positioning and securing of these items and providing for axial and lateral retention and support.

The staff reviewed Subsection 3.9.5.1.1 of the DCD and found that the applicant did not provide sufficient information to allow the review of the upper core plate design and its interfaces with other reactor components. The applicant is requested 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. Review of any design differences from the 4-loop design and consequent effects on potential adverse flow effects is needed to assure conformance with GDC-1 and 4. 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 4-loop reactor.

03.09.05-4

The applicant stated in Subsection 3.9.5.1.1 of the DCD that the guide tubes consist of two main assemblies, an upper and a lower guide tube, that provide horizontal restraint and guidance to the control rods and drive rod assembly, and allow parking of the drive rod during removal and installation after refueling. The upper and lower guide tubes have plates that guide the control rod spider during insertion and retraction of the rod cluster control assembly (RCCA).

The staff's review indicated that the applicant did not provide sufficient information about the control rod guide inside the upper and lower guide tubes. The applicant is requested to provide design details together with relevant flow-induced vibration analysis (if they are needed) for the plates that guide the control rod spider inside the upper and lower guide tubes. In particular, the applicant is requested to explain, with the aid of technical drawing/sketches, the design of the control rod guide and to clarify any differences of this design from that of the existing 4-loop reactor. Also, explain the effects of any design differences on potential flow excitation mechanisms. This information is needed to assure conformance with GDC-1 and 4. Revise Section 3.9.5 of the DCD to provide the requested information.

03.09.05-5

The DCD Tier 2, Subsection 3.9.5.1.1 presents a description of the guide tube assemblies. The applicant stated that the upper and lower guide tube flanges are fastened together by hold-down bolts threaded to the top of the upper core support plate. The lower guide tube is inserted through holes in the upper core support and restrained in the horizontal direction by a small clearance between the lower guide tube flange and upper core support plate hole. Also, the bottom of the lower guide tube is fastened by two large support pins with flexible leaves that slide vertically with a small amount of friction force, but are horizontally preloaded against the upper core plate holes to prevent excessive vibration and wear.

The applicant, however, did not include sufficient geometry/design details to allow the staff to evaluate the flow-induced response of the guide tubes. The applicant is requested to provide details of the geometry/design of the lower and upper guide tubes indicating the differences from the guide tubes of the current 4-loop reactors. Explain the effect of these differences on the flow-induced structural response of the guide tubes. Substantiate the response to this RAI by referring to the flow-induced vibration analysis which will be included in the response to this RAI by means of appropriate flow-induced vibration analysis for the guide tubes. The requested information will facilitate the assessment of the dynamic response of the guide tubes, which is necessary to assure conformance with GDC-1 and 4. Revise the DCD to include additional details about the geometry/design of the lower and upper guide tubes in Subsection 3.9.5.1, about their flow-induced vibration analysis in Subsection 3.9.2.3, and also about their design bases in Subsection 3.9.5.3.

03.09.05-6

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 is presented in Subsection 3.9.5.1.1 of the DCD. The applicant stated 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. The applicant is requested 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. This information is needed to review the safety analysis design requirements and thereby assure conformance with GDC-1 and 4. Revise Subsection 3.9.5.1 of the DCD to include the requested information.

03.09.05-7

The DCD Tier 2, Subsection 3.9.5.1.2 describes the lower core support plate assembly. The applicant stated that 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 the flow into the core should be limited so as not to 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 structure and reactor internals. The applicant is therefore requested 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. Alternately, provide a reference document where this information is available. This information is needed to review the safety analysis design requirements and thereby assure conformance with GDC-1 and 4. Revise Section 3.9.5 of the DCD to provide the requested information.

03.09.05-8

The applicant stated in Subsection 3.9.5.1.2 of the DCD that the energy absorber system and base plate have traditionally been used in PWR internals. Its purpose is to preclude overstressing the RV in the unlikely event of a failed core barrel weld. The drop distance between the bottom of the base plate and the energy absorber system RV bottom is carefully controlled to minimize the impact load and stresses on the RV bottom head. In Subsection 3.9.5.3.1 the applicant further stated that the safety analysis of this issue is a design requirement for the reactor vessel.

The staff reviewed Section 3.9.5 of the DCD and found that the applicant did not refer to any analysis of the impact load, which would result from a postulated core drop event. The applicant is requested to: (a) characterize the postulated core drop event as either a design basis accident required by NRC regulation, or as a beyond-design-basis event not required by regulation; (b) if the core drop event is considered a design basis accident, discuss the analysis performed and the measures undertaken to make sure that the impact load on the RV bottom head from a postulated core drop event would not adversely affect the integrity of the RV bottom head. This information is needed to review the safety analysis design requirements and thereby assure conformance with GDC-1 and 4. Revise Section 3.9.5 of the DCD to provide the requested information.

03.09.05-9

In Subsection 3.9.5.3.4 of the DCD the applicant stated that corrosion, stress corrosion cracking (SCC), radiation embrittlement, and degradation of fatigue strength are considered to be not an issue for the operating conditions of the US-APWR and that the potential for irradiation assisted stress corrosion cracking (IASCC) of the US-APWR core internals is very low. The applicant further stated that void swelling from neutron irradiation was a concern for components with high dose of neutron fluence, e.g., neutron reflector ring blocks, but the ring blocks were cooled to keep metal temperature low and minimize void swelling.

However, the applicant did not provide estimates of temperature and end-of-life neutron fluence for the various reactor internal components, and has not identified the components where void swelling, radiation embrittlement, IASCC, or degradation in fatigue strength is likely to be significant. The primary water stress corrosion cracking (PWSCC) of Ni-alloys, such as X-750, is not addressed in the DCD. Also, the DCD does not provide an assessment of environmental effects on the structural and functional integrity of reactor internals. The applicant is requested to (a) describe the environmental conditions, including estimates of the temperature and end-of-life neutron fluence, for the various reactor internal components, and (b) either provide an evaluation to verify that, under the operating conditions of the US-APWR, the effects of corrosion, SCC, IASCC, PWSCC, degradation of fatigue strength, radiation embrittlement, and void swelling, on the structural and functional integrity of the reactor internal components are not a concern during the design life of 60 years, or define an acceptable program for investigating and managing these environmental effects on reactor internals. Alternately, provide a reference document where this information is available. This information is needed to assure conformance with GDC-1 and 4. Revise the DCD to include the requested information or provide a reference where this information is available.

03.09.05-10

The DCD Tier 2, Subsection 3.9.5.3.4 includes the potential effects of irradiation stress relaxation in the list of environmental effects on reactor core internal materials caused by long term exposure to fast neutron irradiation. The applicant stated that neutron fluence and temperature limits are imposed on the tie-rods to preclude excessive loss of pre-load from irradiation stress relaxation.

The staff reviewed the DCD but did not find where the applicant had provided an evaluation of the loss of preload in various threaded fasteners due to irradiation stress relaxation or a reference where this information is available. The applicant did not identify the fasteners where the effect of irradiation stress relaxation is expected to be significant. Also, it is not clear how the pre-stress will be maintained in the preloaded components such as the ring block tie-rods or guide tube hold-down bolts. The applicant is requested to provide an assessment of the potential loss of preload due to irradiation stress relaxation in various threaded fasteners, in particular the guide tube hold-down bolts, guide tube support pins and the flexible leaves, and the neutron reflector tie-rods, and examine its effect on the structural and functional integrity of the components. Alternately, provide a reference document where this information is available. The requested information will assure conformance with GDC-1 and 4. Revise the DCD to include the requested information or provide a reference where this information is available.

03.09.05-11

In Subsection 3.9.5.2 of the DCD the applicant has identified the loading conditions that have been considered in the design of US-APWR core support structures and internals components. The list includes pressure differences due to coolant flow.

However, the applicant did not provide any details regarding the method used to determine the pressure differences for reactor internal components during different operating conditions or to validate the calculated values. The applicant is requested to provide a description and validation of the method for determining the maximum pressure differences for reactor internals during ASME Code, Section III, Level A, B, C, and D service conditions. Alternately, provide a reference document where this information is available. The requested information is needed to assure conformance with GDC-1, 2, 4, and 10. Revise the DCD to include the requested information or provide a reference where this information is available.

03.09.05-12

In Subsection 3.9.5.2 of the DCD the applicant stated that pressure differences due to the coolant flow have been taken into account in designing the US-APWR core support and internal structures. The complete list of loading conditions that have been considered in the reactor internals design is given in Table 3.9-11 of the DCD. The applicant further stated in Subsection 3.9.5.3.2 of the DCD that the thermal-hydraulic performance criteria require the pressure drops across the reactor internals to meet system requirements for all Level A and B service conditions.

The staff reviewed Section 3.9.5 of the DCD but did not find where the applicant had provided estimates of the maximum pressure differentials for the reactor internals, and verified that they meet the thermal-hydraulics performance requirements of Subsection 3.9.5.3.2. The applicant is therefore requested to describe the system requirements for pressure differentials across reactor internals, and provide an assessment of the maximum pressure differentials for the reactor internals with respect to the design basis system requirements. The requested information will assure conformance with GDC-1, 2, 4, and 10. Revise the DCD to include the requested information.

03.09.05-13

Subsection 3.9.5.2 of the DCD identifies the loading conditions that have been considered in the design of US-APWR core support and reactor internals components. In Subsection 3.9.2.5 the applicant stated that asymmetric LOCA loads for the reactor internals have been considered for the LOCA dynamic analysis. However, in Subsection 3.9.5.2 of the DCD, the applicant did not confirm that such loads have been included in the reactor internals dynamic analysis and that they do not exceed the design limits.

As stated in Section 3.9.5 of the SRP the reactor internals should be designed to accommodate asymmetric blowdown loads from postulated pipe ruptures. Furthermore, the applicant’s evaluation of such loads should demonstrate that these loads do not exceed the limits imposed by the applicable codes and standards. The applicant is requested to verify whether the asymmetric blowdown loadings on reactor internals due to pipe ruptures at postulated locations not excluded in leak-before-break analyses, have been evaluated in the design in accordance with the acceptance criteria of SRP Section 3.9.5, SRP Acceptance Criteria Subsection II.5. Review of the requested information regarding the reactor internals design to withstand blowdown loads from postulated pipe rupture is necessary to assure conformance with GDC-1, 2, 4, and 10. Revise Subsection 3.9.5.2 of the DCD to include the requested information.

03.09.05-14

DCD Tier 2, Subsection 3.9.5.2 identifies the loading conditions that have been considered in the design of US-APWR core support structure and reactor internals components.

However, Subsection 3.9.5.2 does not include any error analysis. The applicant is therefore requested to provide detailed analysis of expected bias errors and random uncertainties included in predicting the vibration responses of reactor core support and internal structure, steam generator internal components, and of other plant systems and components. In response to this RAI, the applicant is expected to provide the total (or end-to-end) bias error and random uncertainties, and to substantiate the contributions of each of the following tasks to the total bias and uncertainties :

1. Modelling and validation of the forcing functions
2. Modelling and validation of the acoustic environment using SYSNOISE
3. FE modelling and validation of structural dynamic characteristics
4. Combining the forcing functions and system dynamic characteristics to estimate the dynamic response of structures and components
5. Experimental measurements which are used to validate models and analysis, whether these measurements are performed in-plant or in the laboratory by means of scale model testing.

The applicant is also expected to explain how the bias and uncertainties are implemented in the calculation of the minimum safety margin.

The Staff needs this information to evaluate the (minimum) margin of safety for the dynamic stress of various core support and reactor internals components and thereby assure conformance with GDC-1, 2, and 4. Revise Subsection 3.9.5.2 of the DCD to include analysis of bias errors and random uncertainties.

03.09.05-15

In Subsection 3.9.5.2.2 of the DCD the applicant stated that the service limits for reactor internals other than the core support structures (CSSs) are not addressed in the ASME Code, Section III. However, because the structural integrity of the reactor internals are 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, then the applicant proposes to utilize alternate acceptance criteria based on validation by testing, sound engineering judgment, and experience with similar design.

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. The applicant is requested 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."*
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. Information about these alternate acceptance criteria is needed to assure conformance with GDC-1, 2, 4, and 10. Revise Section 3.9.5 of the DCD to provide the requested information.

03.09.05-16

The applicant has stated in Subsection 3.9.5.3 of the DCD that the rules for design of the US-APWR core support structures (CSSs) and internal structures follow those in Section III, Subsection NG of the ASME Boiler and pressure Vessel Code (2001 Edition up to and including 2003 Addenda). Also, in DCD Section 3.9.3 the applicant stated that the environmental effects on fatigue of ASME Code, Section III, Class 1 components follow the guidance delineated in RG 1.207.

The staff's review of Section 3.9.5 of the DCD showed that the stress categories and stress intensity limits for CSS given in Table 3.9-12 do not include fatigue. The applicant should explain why fatigue evaluation was excluded from the design bases for CSSs and reactor internals, and provide a technical basis for the exclusion. The applicant is therefore requested to provide the reason and technical justification why fatigue evaluation, including the effects of PWR coolant environment, is not included in the list of CSS stress categories and stress intensity limits given in Table 3.9-12 of the DCD. Review of the requested information regarding the reactor internals design is necessary to assure conformance with GDC-1, 2, 4, and 10. Revise Subsection 3.9.5.3 of the DCD to include the requested information.

03.09.05-17

The load and displacement limits for the reactor internals that affect the safety and operability of the interface components are summarized in Table 3.9-2 of the DCD.

However, the DCD does not give any details how the deformation limits were determined, or provide the technical basis for these deformation limits. As stated in SRP Section 3.9.5, SRP Acceptance Criteria, deformation limits for reactor internals should be established by the applicant and presented in the safety analysis report, and the basis for these limits should be included. Also, the stresses for these displacements should not exceed the specified limits. The applicant is requested to provide the technical basis for defining the displacement limits listed in Table 3.9-2 of the DCD. Alternately, provide a reference document where this information is available. Review of the requested information regarding the reactor internals design is necessary to assure conformance with GDC-1, 2, 4, and 10. Revise Subsection 3.9.5.2.3 of the DCD to include the requested information or provide a reference where this information is available.

03.09.05-18

In accordance with the recommendations in Appendix A of SRP 3.9.5, the applicant is expected to evaluate the design, including dynamic response, stress, and design margin, of the steam generator internal components for potential adverse flow effects from flow-induced vibration and from acoustic resonance conditions in attached main steam piping and associated branch connections. This evaluation is expected to address potential adverse flow effects which can be caused by the flow in the steam generator as well as the main steam line flow past closed branch connection standpipes, such as those for the main steam safety relief valves. Past operating experience and analysis may be used to support the adequacy of design margins for steam generator internals (see Appendix A of SRP 3.9.5 for more details).

The staff's review indicated that Section 3.9.5 of the DCD neither included nor referred to a comprehensive dynamic analysis or past operating experience for the steam generator internals. The applicant is requested to discuss in detail the design methodology used to ensure that the structural integrity of the steam generator internals components will not be endangered due to adverse flow effects. The staff needs this information to assure conformance with GDC-1 and GDC-4. Revise Subsection 3.9.5.2 of the DCD to include a detailed evaluation of potential adverse flow effects on the structural integrity of the internal components of the US-APWR steam generators. Alternatively, the applicant may choose to provide this analysis in Section 3.9.2 of the DCD and refer to this in Subsection 3.9.5.2.

03.09.05-19

According to the recommendations in Appendix A of SRP 3.9.5, the applicant is expected to evaluate potential adverse flow effects on piping and components of plant systems.

The staff reviewed Section 3.9.5 of the DCD and found that the applicant did not include an evaluation of these effects. The applicant is requested to provide a detailed evaluation of potential adverse effects from flow-induced vibrations and acoustic resonances on piping and components of plant systems, including the reactor coolant, steam, feedwater, and condensate systems. Flow-induced vibrations of various sampling probes should also be evaluated. Also, substantiate any assumptions made in the analysis, particularly for damping coefficients of structural elements. The staff needs this evaluation for the above mentioned plant components to assure conformance with GDC-1 and GDC-4. Revise Subsection 3.9.5.2 of the DCD to include a detailed evaluation of potential adverse flow effects on piping and components of plant systems, including the sampling probes, or refer to this evaluation if it is included elsewhere in the DCD.

03.09.05-20

Appendix A of SRP 3.9.5 also recommends that the applicant should maintain monitoring of potential adverse flow effects on plant systems and components for a sufficient time period to verify that adverse flow effects are not occurring (See Appendix A, Item 7 of SRP 3.9.5 for more details).

The staff reviewed Section 3.9.5 of the DCD and found that the applicant did not discuss monitoring of potential adverse flow effects. The applicant is therefore requested to discuss the plans for monitoring potential adverse flow effects in the plant after the initial start-up period. Previous plant experience has shown that adverse flow effects might not appear for an extended period of time following initial start-up. The staff needs information about the monitoring program to complete the review and to evaluate conformance with GDC-1 and GDC-4. Revise Subsection 3.9.5.2 of the DCD to include adequate information about monitoring of potential adverse flow effects on plant systems and components.

03.09.05-21

Neither Section 3.9.2, nor Section 3.9.5, of the DCD provides any values of damping coefficient used in the assessment of the dynamic response of the reactor and steam generator internals. Instead, the document states that a "*damping coefficient smaller than the best estimate value*" is used.

The reliability and associated bias and uncertainty errors of the dynamic analysis of the reactor internals and steam generator internals depend on the damping coefficient assumed for various structural components. The use of appropriate damping values is therefore necessary to ensure that the reactor and steam generator internal structures are designed to quality standards commensurate with the importance of their safety functions. The applicant is requested to provide and substantiate the damping coefficient values used in the dynamic analysis of the reactor and steam generator internals. Support the response to this RAI by referring to available in-plant measurements of damping values for the current 4-loop reactors and steam generators. The applicant should discuss the damping values used in the following situations, together with the methods used to validate these values and the expected bias error and random uncertainties:

1. Calculations of the vibratory response of the scale model internals and comparison with the measured values of damping coefficient.
2. Calculations of the vibratory response of the US-APWR and comparison with the damping measured for the current 4-loop reactors.
3. Calculations of the vibratory response of the steam generator internals and comparison with the measured values from operational steam generators.

In order to facilitate assessment of the dynamic response of the reactor internals and steam generator internals, which is necessary to assure conformance with GDC-1 and GDC-4, the staff needs the requested information about the damping values and the method(s) used to validate these values. Revise the applicable Subsections of the DCD to include the damping values used in the analysis as outlined above.

03.09.05-22

The applicant states in DCD Tier 2, Subsection 3.9.5.3 that the rules for materials, design, fabrication, examination, and preparation of reports for the manufacture and installation of the US-APWR core support structures (CSSs) and internal structure follow those in Section III, Subsection NG of the ASME Boiler and Pressure and Vessel Code, 2001 Edition up to and including 2003 Addenda. Additional codes, standards, regulations, and guidelines from the NRC and the Utility Requirements Document are adhered to, and are listed in the Owner's design specification.

However, these additional design codes, code cases, and acceptance criteria are not identified in the DCD. Section 3.9.5 of the SRP states that if other guidelines (e.g., manufacturer standards or empirical methods based on field experience and testing) are the bases for the stress, deformation, and fatigue criteria, those guidelines should be identified and their use justified. The applicant is requested to provide a list and justification of the applicable codes, standards, regulations, and guidelines, for the design of US-APWR CSSs and reactor internals, if different from ASME III, Subsection NG requirements. The requested information is needed to assure conformance with GDC-1. Revise the DCD to include the requested information.

03.09.05-23

The reactor coolant flow path for the reactor internals is described in Subsection 3.9.5.3.2 of the DCD. The applicant states that the main coolant flow enters the bottom of the RV and turns upward, flowing past the diffuser plates and distributing into the lower core support plate orificed holes. The orifices are carefully designed to control the flow into the fuel assemblies and to minimize uneven flow distributions and hot spots.

The thermal-hydraulics performance criteria for the design of the US-APWR core support and internal structure, listed in Subsection 3.9.5.3.2, require that the distribution of main coolant inlet flow into the fuel assemblies during normal operation must meet fuel assembly core inlet requirements. However, the DCD does not provide any details about these requirements or how compliance with the requirements is verified. The applicant is requested to provide additional details regarding the fuel assembly core inlet requirements to explain how compliance with the requirements during service is verified. The requested information is needed to confirm compliance with the thermal-hydraulics design basis requirements for the design of the US-APWR core support and internal structure, and assure conformance with GDC-4, and -10. Revise the DCD to include the requested information.

03.09.05-24

In Subsection 3.9.5.3.2 of the DCD the applicant states that the main coolant flow then mixes in the upper plenum and exits from the core barrel outlet nozzles at an average fluid temperature of $T_{hot}$. The applicant further states that special flow columns are spaced on the periphery of the upper core plate near the core barrel outlet nozzles in order to improve mixing and minimize outlet fluid temperature mal-distribution.

The thermal-hydraulics performance criteria for the design of core support and internal structures, identified in Subsection 3.9.5.3.2, require that the main coolant flow into the outlet piping during normal operation meets the system requirements, specifically (a) exit fluid temperature striations are minimized, and (b) velocity criteria to prevent erosion are met. The DCD does not provide any details how compliance with these system requirements is verified. The applicant is requested to provide additional details regarding the system requirements for exit fluid velocity and temperature striations, and describe the procedure used to verify compliance with these requirements during service. The requested information is needed to confirm compliance with the thermal-hydraulics design basis requirement for the design of the US-APWR core support and internal structures and assure conformance with GDC-4 and GDC-10. Revise the DCD to include the requested information.

03.09.05-25

The applicant states in DCD Tier 2, Subsection 3.9.5.3.2 that some percentage of the main coolant flow is bypass flow which is either for cooling metal or leakage between gaps. The bypass flows from gap leakages are as follows: small gap between the core barrel outlet nozzle and RV outlet nozzle, neutron reflector ring block inside surface and the peripheral fuel assembly grids and nozzles, and neutron reflector small gaps between the ring blocks.

However, the applicant did not assess the liability of the core barrel flange to leakage flow-induced vibration. The applicant is requested to discuss the liability of the core barrel flange to flow-induced vibration caused by the leakage (or bypass) flow between the outlet nozzle of the core barrel flange and the RV exit nozzle. Since the diameter of the core barrel flange is larger than that of current 4-loop reactors, its shell modes may have lower frequencies. In addition, the leakage flow rate is higher in the US-APWR than in the 4-loop reactors. Provide evidence showing that the leakage flow between the outlet nozzle of the core barrel flange and the RV exit nozzle will not cause excessive vibration of the core barrel flange. This assessment is needed to assure conformance with GDC-4, and -10. Revise Section 3.9.5 of the DCD to include an assessment of the leakage flow effects on the core barrel flange.

03.09.05-26

The applicant states in Subsection 3.9.5.3.12 of the DCD that the pre-service inspection as well as the in-service inspection (ISI) plans follow the rules of ASME Code, Section XI.

However, the applicant does not examine or discuss the adequacy of the ASME Code, Section XI ISI plan to detect environmental effects on the structural and functional integrity of the core support and internal structures during their 60-year design life. For license renewal of operating reactors, the staff has reviewed the aging effects on components and structures, identified the relevant existing aging management programs

(AMPs), and evaluated the program attributes to determine where existing programs are adequate without modifications and where existing programs should be augmented for the extended period of operation. The evaluation results documented in the Generic Aging Lessons Learned (GALL) report, NUREG-1801, Rev. 1, indicate that the ASME Code, Section XI ISI program is inadequate to manage aging effects such as cracking due to SCC or IASCC, change in dimensions due to void swelling, loss of fracture toughness due to neutron irradiation embrittlement, or loss of pre-load due to irradiation stress relaxation. The applicant is requested to provide a commitment to:

(a) Review and evaluate the effect of environmental degradation processes such as SCC, IASCC, PWSCC, degradation of fatigue strength, radiation embrittlement, and void swelling on the structural and functional integrity of the reactor core support and internals components.
(b) Define the range of environmental and service conditions under which these environmental effects can be significant.
(c) Evaluate and implement the results of the industry programs for investigating and managing environmental effects as applicable to reactor core support and internal structures.
(d) Develop an inspection plan for reactor core support and internal structures that addresses these service conditions and environmental degradation issues.

This information is needed for timely detection the effects of environmental effects on the structural and functional integrity of the US-APWR core support and internal structures and components and assure conformance with GDC-1, -4, and -10. Revise Section 3.9.5 of the DCD to include the requested information.

03.09.05-27

A description of the US-APWR upper reactor internals assembly design arrangement, including the classification of the various upper reactor internals components is presented in Subsection 3.9.5.1 of the DCD. The applicant stated that both the upper core support assembly and lower core support assembly are classified as Core Support Structures (CSS). The design bases requirements provided in DCD Section 3.9.5.3 specifies that those reactor internals components classified as CSS conform to the materials, design, fabrication, examination, and documentation requirements of the ASME Boiler and Pressure Vessel Code, Section III (ASME III), Subsection NG, 2001 Edition through the 2003 Addenda.

The staff's review of DCD Subsection 3.9.5.1, together with DCD Subsection 3.9.5.1.3, revealed that the applicant did not clearly define the classification of the reactor internals hold-down spring, which is a load bearing component of the upper core support assembly. The jurisdictional boundaries of the reactor internals are defined in DCD Subsection 3.9.5.1.3. The fourth bullet item defines the boundary between the components classified as CSS and components classified as internal structures, following the guidance for boundaries of jurisdiction in the ASME Code Section III, Subsection NG-1000. The second line item under the fourth bullet states, *"Upper core support flange and core barrel flange with the reactor internals hold-down spring."* The staff review interprets this DCD statement to mean that the reactor internals hold-down spring is classified by the applicant as internal structure, as opposed to a core support structure, for purposes of specifying the applicable design code requirements for the hold-down spring. Based on the staff's interpretation of the DCD statement, the classification of the reactor internals hold-down spring appears to be inconsistent with the requirements of GDC-1 and 10 CFR 50.55a.

ASME III, Article NG-1121 defines core support structures as structures or parts of structures which provide direct support or restraint of the core (fuel and blanket assemblies) within the reactor pressure vessel. The staff considers the reactor internals hold-down spring, together with the upper core support assembly and lower core support assembly, to be complementary parts of the load bearing assembly providing support for the reactor core.

The applicant is requested 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 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. The staff requires this information to assure conformance with the regulatory requirements of GDC-1.