MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

June 19, 2009

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-09335

Subject: MHI's Response to US-APWR DCD RAI No. 374-2446

References: 1) "Request for Additional Information No. 374-2446 Revision 0, SRP Section: 03.09.05 – Reactor Pressure Vessel Internals, Application Section: DCD, Tier 2 – Section 3.9.5," dated 5/21/2009.

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. 374-2446 Revision 0."

Enclosed are the responses to questions 1, 5, 13, 21 and 25 of the RAI (Reference 1). The responses to the remaining 22 questions of this RAI has a 60-day response time, as agreed to between the NRC and MHI, and will be issued at a later date by a separate transmittal.

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

This letter includes a copy of the proprietary version (Enclosure 2), a copy of the non-proprietary version (Enclosure 3), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all materials designated as "Proprietary" in Enclosure 2 be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) and 10 C.F.R.§ 9.17 (a)(4).

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

Sincerely,

Y. Ogata

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

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NRO

Enclosures:

1. Affidavit of Yoshiki Ogata
2. Response to Request for Additional Information No. 374-2446, Revision 0 (Proprietary)
3. Response to Request for Additional Information No. 374-2446, Revision 0 (Non-Proprietary)

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

Contact Information

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Enclosure 1

Docket No. 52-021
MHI Ref: UAP-HF-09335

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, state as follows:

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

2. In accordance with my responsibilities, I have reviewed the enclosed document entitled "Response to Request for Additional Information No. 374-2446, Revision 0", dated June 2009, and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages contain proprietary information are identified with the label "Proprietary" on the top of the page, and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

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

4. The basis for holding the referenced information confidential are as follows:

   A. They include the know-how and outputs obtained from analyses or testing which required significant cost to MHI. It required the performance of detailed design calculations, supporting analyses and testing extending over several years. The referenced information is not available in public sources and could not be gathered readily from other publicly available information. MHI knows of no way the information could be lawfully acquired by organizations or individuals outside of MHI.

   B. They include the information directly referred from documents or books the copyrights of which are reserved.

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

6. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without the costs or risks associated with the design

of new systems and components. Disclosure of the information identified as proprietary would therefore have negative impacts on the competitive position of MHI and the Licensors in the U.S. nuclear plant market.

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

Executed on this 19th day of June 2009.

Y. Ogata

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

Enclosure 3

UAP-HF-09335
Docket No. 52-021

Response to Request for Additional Information No. 374-2446,
Revision 0

June, 2009
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

6/19/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESSEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 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.

ANSWER:

For the thermo-couple and ICIS, the upper core support columns are used as the guide structures in the upper plenum. These structures are similar to the upper support columns in the existing 4- loop reactor design.

Control and drive rod assembly informational responses were provided in RAI-2296-4.6-1 through RAI-2296-4.6.11.

The two reactor vessel (RV) level instrumentation support tubes that measure the water level in the reactor vessel are similar to the RCCA guide tubes in the support conditions and vibration characteristics. For the both structures, the top end is fixed on the upper core support plate with bolts and the bottom end is pin supported on the upper core plate. The fundamental mode frequency is [ ] Hz for the RV level instrumentation support tube and [ ] Hz for the RCCA guide tube, based on the scale model test as shown in Table 6-1 of Reference (1). Both of the two RV level instrumentation support tubes are not located near the outlet nozzles, adverse flow effects from cross flow velocities are bounded by the RCCA guide tubes nearest the outlet nozzles. Also, the operating environment experience with similar RV level instrumentation support tubes in existing 4 -loop plants gives confidence in the structural and functional design.

Assessments for the adverse flow effects such as the lock-in with vortex shedding or the fluid elastic instability of the RCCA guide tubes and the upper support columns were performed based on the ASME design guide lines. As the results, both structures of the US-APWR has sufficient margins for the adverse flow effects as described in Subsection 3.9.2.3 of DCD and 3.4 of Reference (2). In addition, the margin of safety for high cycle fatigue was confirmed based on the response analysis. Detail of the high cycle fatigue evaluations are described in Subsection 3.3.3.2 of Reference (2).

Reference (1): MUAP-070023 R1 "APWR Reactor Internals 1/5 Scale Model Flow Test Report"

Reference (2): MUAP-070027 R1 "The comprehensive vibration assessment program for the US-APWR Reactor Internals"

Impact on DCD

Information about the guide structures for the instrumentations will be added in the last part of Subsection 3.9.5.1.2 Upper Reactor Internals Assembly Design Arrangement. See the attachment for the changes to be incorporated to DCD Subsection 3.9.5.

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

6/19/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: NO. 374-2446 REVISION 0

SRP SECTION: 3.9.5 – REACTOR PRESSURE VESSEL INTERNALS

APPLICATION SECTION: 03.09.05

DATE OF RAI ISSUE: 5/21/2009

QUESTION NO.: 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.

ANSWER:

The lower RCCA guide tube of the US-APWR has same design with that of the current 4-loop reactor in the layouts and dimensions of the enclosure and guide structures. The upper RCCA guide tube is extended about 1 ft from that for the 12-ft core design, as like the 14-ft core 4 -loop reactors existing in US. So the vibration characteristics of the RCCA guide tubes for the US-APWR are equivalent with the existing guide tubes. The fundamental mode frequencies for the upper and lower guide tube are described in Table 3.3.1-1 of Reference (1).

Because the lower RCCA guide tube is located in the cross flow field of the upper plenum, assessments for the adverse flow effects such as the lock-in with vortex shedding or the fluid elastic instability were performed based on the ASME design guidelines. From the results, the RCCA guide tube of the US-APWR has sufficient margins for the adverse flow effects as described in Subsection 3.9.2.3 of DCD and 3.4 of Reference (1). In addition, the margin of safety for high cycle fatigue was confirmed based on the vibration response analysis. Detail of the high cycle fatigue evaluation of the RCCA guide tube in the US-APWR are described in Subsection 3.3.3.2 of Reference(1).

Reference (1): MUAP-070027 R1 "The comprehensive vibration assessment program for the US-APWR Reactor Internals"

Impact on DCD

Information about the RCCA guide tubes design change from the existing 4-loop plants and its impact on the vibration characteristics will be included in the 7th paragraph of Subsection 3.9.5.1.2 Upper Reactor Internals Assembly Design Arrangement. See the attachment for the changes to be incorporated to DCD Subsection 3.9.5.

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

6/19/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESSEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 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.

ANSWER:

The asymmetric blowdown loadings on reactor internals due to pipe ruptures at postulated locations not excluded in leak-before-break analyses were evaluated as described in MUAP-09002-P "Summary of Seismic and accident Load Conditions for Primary Components and Piping" ,Revision 1.

From the dynamic response analysis and the stress evaluation, it was confirmed that the reactor internals design withstood the blowdown loadings due to pipe rupture events. The summary of the stress analysis was reported in MUAP-09004-P "Summary of Stress Analysis Results for Core Support Structures", Revision 0.

Impact on DCD

In Subsection 3.9.5.2, the sentence of "Transient pressure difference loads, such as those which result from rupture of a branch pipe" will be replaced with "Transient pressure difference loads, such as asymmetric blowdown loadings due to pipe ruptures at postulated locations ". See the attachment for the changes to be incorporated to DCD Subsection 3.9.5.

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

6/19/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: NO. 374-2446 REVISION 0

SRP SECTION: 3.9.5 – REACTOR PRESSURE VESSEL INTERNALS

APPLICATION SECTION: 03.09.05

DATE OF RAI ISSUE: 5/21/2009

QUESTION NO.: 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.

ANSWER:

1. Reactor internals scale model test and analysis
a. Measured values of the damping ratio were [ ] % to the critical damping for the core barrel and the neutron reflectors as described in Table 6-1 of Reference (1).
b. In the calculation of the vibration response of the scale model internals, the ratio of [ ] % to the critical damping was applied as described in Table 3.2.3-1 of Reference (2). This was based on the measured value as mentioned above.

2. Calculations of the US-APWR reactor internals
a. In the calculations of the vibratory response of the US-APWR, [ ]% to the critical damping were applied as described in Table 3.3.3-1 of Reference (2).
b. No measured damping ratio is available in the current 4-loop reactors.

3. Steam generator internals
The damping ratio used in the steam generator tube vibration analysis is [ ]%, which is suggested in ASME Sec.III Appendix-N 1331-3 as conservative value based on the laboratory test data base for avoiding fluid-elastic instabilities of tube arrays.

Reference (1): MUAP-070023 R1 "APWR Reactor Internals 1/5 Scale Model Flow Test Report"

Reference (2): MUAP-070027 R1 "The comprehensive vibration assessment program for the US-APWR Reactor Internals"

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

6/19/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESSEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 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.

ANSWER:

There has been no reported evidence of nozzle gap by-pass flow being a major contributor to the core barrel vibration response through the experience of previous plants operation or testing.

The bypass flow from the outlet nozzle gap between the Core Barrel / RV has little effect on the core barrel vibration because the flow rate and the flow contact area of the gap are much smaller than those of the downcomer as discussed in the response to RAI 206-1576 (QUSTION NO.:RAI3.9.2-43) and Appendix-A of MUAP-07027-R1 "Comprehensive Vibration Assessment Program for US-APWR Reactor Internals".

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.

Attachment of the response to RAI No.374-2446 revision 0

3.9.5 Reactor Pressure Vessel Internals

This section discusses the US-APWR reactor internals design arrangements; design basis loading for all service conditions; the acceptance criteria of stresses and functional requirements; the computational methods used in the static mechanical and thermal analyses, dynamic analyses, vibration analyses, and computational flow analyses; the testing and alternate methods to confirm computational inputs to the design, the interface load and displacement criteria of the reactor internals and interfacing components; and preservice and inservice inspection plans.

The term "reactor internals" used in this subsection refers to the core support and internal structures and to all structural and mechanical elements inside the RV with the following exceptions:

- Reactor fuel elements and the reactivity control elements out to the coupling interface with the drive unit, except for the structural and interfacing aspects of the reactor fuel assemblies with the reactor internals, which are within the scope of this subsection.
- Control rod drive elements except for the guide tubes, which are within the scope of this subsection.
- In-core and thermocouple instrumentation except for the in-core instrumentation supports and thermocouple support structures which are within the scope of this subsection.

3.9.5.1 Design Arrangements

The reactor internals for the US-APWR can be divided into two major assemblies, the upper reactor internals assembly, and the lower reactor internals assembly. A separate part that is captured between the upper and lower reactor assembly flanges and provides vertical pre-load and frictional restraint to the flanges is the reactor internals hold-down spring.

Figure 3.9-4 illustrates the reactor internals general assembly configuration. The figure illustrates the design arrangement of the reactor internals and the interfacing components such as the RV, fuel assembly, and incore nuclear instrumentation system (ICIS) thimble assembly.

The US-APWR is a four loop PWR plant with 257 fuel assemblies having 17 by 17 fuel rod arrays, with 14 foot active fuel length. The basic design of the US-APWR reactor internals evolved from the existing four loop plant technologies. Figure 3.9-4 shows that the vertical fuel assembly cavity is established by an upper core plate that compresses the fuel assembly hold-down springs and a lower core support plate that supports the fuel assemblies. Also shown is the ICIS thimble assembly and the detector guide thimbles that house the in-core movable detectors.

Figure 3.9-4 shows that the horizontal fuel assembly cavity is formed by the neutron reflector inside surface. The neutron reflector also shields the RV from excessive neutron fluence.

The materials to be used in the construction of the US-APWR reactor internals are discussed in DCD Subsection 4.5.2. The reactor internals are classified as core support structures, threaded structural fasteners, and internal structures. Although there are many reactor internals parts, there are only a few classified components designed as core support structures and threaded structural fasteners. The core support structures and threaded structural fasteners classified components are summarized below:

Core support structures and threaded structural fasteners:

- Upper Core Support Assembly
  - Upper core support plate, flange, and skirt cylinder
  - Upper core plate

  - Upper core plate fuel alignment pins
  - Upper support columns, and threaded structural fasteners
  - Upper core plate clevis and threaded structural fasteners
- Lower Core Support Assembly
  - Core barrel flange
  - Upper and lower core barrel
  - Lower core support plate
  - Radial support key and clevis
  - Lower core support plate fuel alignment pins

The material of construction is mainly austenitic 304 stainless steel, which is selected for its resistance to corrosion from PWR water chemistry and its manufacturability (i.e., ease of welding and machining). The weld metal is 308 stainless steel. Strain hardened 316 stainless steel is the material of choice for the threaded structural fasteners because of its additional increased strength necessary for maintaining preload in a vibratory environment. Other materials such as Alloy X-750 used as an alternative for the guide tube support pins and nuts, and 403 stainless steel for the hold-down spring are selected for higher strength applications. Special materials such as a cobalt alloy hard-facing are applied to reduce material wear from vibration effects. All materials other than the cobalt alloy are controlled to a cobalt content not to exceed 0.2%.

3.9.5.1.1 Upper Reactor Internals Assembly Design Arrangement

Figure 3.9-5 shows the upper reactor internals assembly design arrangement. The major sub-assemblies of the upper reactor internals are the upper core support assembly, upper core plate assembly, upper support column assemblies, top slotted columns and mixing devices, guide tube assemblies, RV level instrumentation system assemblies, ICIS detector guide thimbles and thimble assemblies, and thermocouple conduit support column assemblies.

The upper core support assembly has an upper core support flange welded to the top of a cylindrical skirt. The upper core support flange has flow holes to allow cooling flow to enter into the RV head plenum. There are also slots in the flange that allow for guidance of the upper core support assembly during installation. These slots are engaged and guided by head and vessel alignment pins that are attached to the core barrel flange. Also, there are threaded roto-lock inserts in the upper core support flange for lifting of the upper support assembly. The cylindrical skirt is also welded at its bottom to the upper core support plate. There are holes in the upper support plate to allow installation of the lower guide tubes, and upper support columns. Each guide tube assembly is secured to the upper core support plate by hold-down bolts and each upper support column with a large nut.

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 a 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. Vertical loads on the upper core support assembly come from dead weights less buoyant forces, upper core support and upper core plate differential pressure loads, vibration loads on the components, fuel assembly spring and lift loads, and seismic and postulated LOCA loads. The upward vertical loads are transmitted from the upper core support flange to the RV head and the downward vertical loads to the RV flange. 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.

The vertical cavity between the upper core support and the upper core plate is dimensionally controlled by upper support columns that are fastened to the upper core plate at the bottom of the column and to the top of the upper core support by a single extended tube with a threaded nut that bears on the upper core support. Some upper support columns have either the detector guide thimbles or thermal couple conduits. In addition, there are several top slotted columns, and mixing devices on the periphery of the upper core plate. These columns are designed to provide a more uniform exit flow and temperature distribution to the outlet loop pipes. There are also two RV level instrumentation support tubes to measure the water level in the reactor vessel.

The upper core plate has circular flow holes for fuel assembly exit core flow. There are also circular flow holes for the exit core flow below the upper support columns and there are rectangular shaped holes for core exit flow into the guide tubes. 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 upper core plate has two fuel assembly pins for each fuel assembly that are shrink-fitted and provide guidance for the fuel assembly nozzles during refueling and installation. The fuel assembly pins also provide some horizontal restraint to the fuel assembly top nozzle. The upper core plate horizontal position, relative to the fuel assemblies, is controlled by a tight fit between the upper core plate clevis and the upper core plate guide pins. The upper core plate clevis and the upper core plate guide pins transmit horizontal loads during normal operation and during seismic and pipe rupture events. However, the upper core plate guide pins are limited in their load capacity, so that loads exceeding the capacity of the pins are transmitted by contact of the upper core plate rim with the core barrel inside diameter.

The guide tubes provide horizontal restraint and guidance to the control rods and drive rod assembly, as well as allow the parking of the drive rod during removal and installation after refueling. All guide tubes are designed for removal and replacement in the event they sustain damage during operation or refueling. The guide tubes have two main assemblies; an upper guide tube and a lower guide tube. The upper and lower guide tube flanges are fastened together by 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. 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 upper and lower guide tubes have plates that guide the control rod spider during insertion and retraction of the RCCA. The lower guide tube has a continuous section of C-tubes and sheaths just above the upper core plate hole that avoids excessive vibrations of the RCCA due to the flow from fuel assemblies. The lower guide tube has "windows" to allow the flow to egress to the plenum between the upper core support and upper core plate. These layouts and dimensions of the lower guide tube are not changed from those of the existing 4-loop reactors. The upper guide tube is extended about 1 ft from that for the 12-ft core design, as like the 14-ft core reactors existing in US. So the vibration characteristics of the RCCA guide tubes for the US-APWR are equivalent with those used in the existing plants.

The two reactor vessel (RV) level instrumentation support tubes that measure the water level in the reactor vessel are designed similar to the RCCA guide tubes in the vibration characteristics. Because both of the two RV level instrumentation support tubes are not located near the outlet nozzles, adverse flow from cross flow velocities are bounded by the RCCA guide tubes nearest the outlet nozzles. Also, the operating environment experience with similar RV level instrumentation support tubes in 4-Loop plants gives confidence in the structural and functional design.

For the thermo-couple and ICIS, the upper core support columns are used as the guide structures in the upper plenum. These structures are similar to the upper support columns in the existing 4- loop reactor design.

3.9.5.1.2 Lower Reactor Internals Assembly Design Arrangement

Figure 3.9-6 shows the lower reactor internals assembly design arrangement. The major sub-assemblies of the lower reactor internals assembly are the core barrel assembly; the lower core support assembly; the neutron reflector assembly; irradiation specimen guide assembly; and the secondary core support assembly.

The core barrel assembly consists of a forged flange that is welded to the upper core barrel. The upper core barrel is welded to the lower core barrel. The core barrel flange has flow nozzles that are welded to the flange and provides a cooling flow from the RV annulus to the RV head plenum. Lifting of the core barrel assembly is accomplished by threading the lifting fixtures into the roto-lock inserts in the flange. The head and vessel alignment pins are bolted to the flange to provide guidance for the core barrel assembly during installation and removal. The head and vessel alignment pins are guided and aligned by slots in the RV and RV head. The flange has holes for access to the irradiation specimens. The upper core barrel has four welded core barrel outlet nozzles to provide an exit flow path to the RV outlet nozzles. In addition, four safety injection pads are attached to the core barrel to divert the safety injection flow from directly impinging on the barrel during a safety injection event. The lower core barrel receives the most neutron fluence from the core during normal operation. The lower core barrel has irradiation specimen guides that are fastened to the outside of the core barrel at specific locations.

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 is welded to the lower core barrel. The lower core support plate has orificed flow holes to reduce mal-distribution of the flow into the core. Six radial keys are attached to the outside rim of the lower core support plate. These keys engage the RV clevis inserts. The keys and clevis inserts provide alignment during installation, resistance to vibration from flow, and transmit asymmetric flow loads and dynamic loads from seismic and postulated LOCA forces to the reactor vessel. The lower core support plate supports the fuel assemblies and has two fuel alignment pins per fuel assembly for alignment and horizontal restraint of the bottom fuel nozzle. The fuel alignment pins are attached to the top of the lower core support plate and restrained by a locking device.

The neutron reflector assembly offers a significant reduction in the number of threaded fasteners, and an improvement in neutron reflectivity, from the design of currently operating PWR plants. The neutron reflector consists of multiple stacked ring blocks that are supported in the vertical downward direction by the lower core support plate and by tie rods and neutron reflector mounting bolts in the vertically upward direction. The inside surface of the ring blocks establishes the core cavity profile for the fuel assemblies. The small gaps between ring blocks are designed to be aligned with the fuel assembly grids. The stacked ring blocks are connected to each other by ring block alignment pins mounted on their top and bottom surfaces for alignment and shear restraint. The neutron reflector upper alignment pin and lower alignment pins are guided into position by clevises attached to the core barrel. This arrangement provides horizontal restraint for mechanical loads, similar to the upper core plate arrangement. The ring blocks are carefully designed with cooling holes to assure that void swelling and distortion are minimized. Bypass cooling flow is directed into the bottom ring block from holes machined in the lower core support plate. The holes in the lower core support plate are also orificed to provide a pressure drop that minimizes the pressure difference between the core and the neutron reflector flow paths. The holes are also sized to prevent debris from entering or blocking the cooling holes in the ring blocks. Tie-rods provide vertical restraint for mechanical loads while the neutron reflector mounting bolts secure the bottom ring block to the lower core support plate. The tie-rods are captured by a nut bearing on the top ring block. The tie-rods pass through holes in the blocks and are threaded into the lower core support plate. Fluence and temperature limits are also imposed on the tie-rods to preclude excessive loss of pre-load from irradiation relaxation.

The irradiation specimen guides are fastened to the core barrel by long socket head cap screws (to accommodate bending). The specimen capsules inside the specimen guides are held in place by springs and a threaded cap. RV surveillance test specimens are periodically removed during outage for examination of RV neutron fluence embrittlement.

The secondary core support assembly consists of secondary core support columns, diffuser plate support columns, a base plate, and energy absorber system. The diffuser plates are bolted to the diffuser plate support columns and those columns are fastened to the bottom of the lower core support plate. The energy absorber and base plate are supported by columns that are bolted to the bottom of the lower core support plate. The energy absorber system and base plate have traditionally been used in PWR internals. Their 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.

3.9.5.1.3 Jurisdictional Boundaries of the Reactor Internals

The jurisdictional boundaries between the core support structures, the internal structure, and the interfacing components such as the RV, CRDMs, fuel assemblies and thermocouple, and ICIS follow the guidance for boundaries of jurisdiction in the ASME Code, Section III (Reference 3.9-1), Subsection NG-1000. Figure 3.9-7 illustrates the boundaries of jurisdiction by the heavy black line between reactor internals and interfacing components.

The jurisdictional boundaries between the core support structures, threaded structural fasteners with the internal structures, and interfacing structures are summarized below:

- Core support structures and threaded structural fasteners boundary with the RV
  - Core barrel and upper core support flanges with the RV and RV head flanges.
  - The radial support key and clevis with the RV
- Core support structures and threaded structural fasteners boundary with the fuel assemblies
  - Bottom of the upper core plate with the fuel assembly

  - Top of the lower core support plate with the fuel assembly
  - Upper core plate fuel assembly alignment pins with the fuel assembly top nozzle holes
  - Lower core support plate fuel assembly alignment pins with the fuel assembly bottom nozzle holes

- Core support structures and threaded structural fasteners boundary with the ICIS
  - Upper core support column inside hole with the ICIS thimble
  - Upper core support with the in-core and thermocouple thimbles and conduit supports

- Core support structures and threaded structural fasteners boundary with the internal structures
  - Upper core support plate and threads with the lower guide tube flange and bolts
  - Upper core support flange and core barrel flange with the reactor internals hold-down spring
  - Core barrel flange with head and vessel alignment pins
  - Core barrel flange with flow nozzles
  - Lower core support plate with secondary core support columns and bolts
  - Core barrel threads and irradiation specimen guide holder bolts
  - Lower core support plate threads with the neutron reflector tie-rods
  - Core barrel alignment pins with the neutron reflector clevis and bolts

- Internal structure boundary with the interface components
  - Core barrel outlet nozzle with RV outlet nozzle
  - Detector guide thimble with the in-core support housing plates
  - Thermocouple conduits with the thermocouple supports
  - Upper guide tube plates, lower guide tube guide plates, continuous section sheaths and C-tubes with the RCCA rodlets

- Upper guide tube top plate hole with the drive rod
- Parking button on the lower guide tube sheaths with the drive rod button for refueling
- Secondary core support base plate with the RV bottom head

3.9.5.2 Loading Conditions

The US-APWR reactor internals loading conditions, load combinations, and acceptance criteria, namely, the reactor internals design and service limits, and displacement limits are discussed below.

The loading conditions that are taken into account in designing the US-APWR reactor internals structures listed in Table 3.9-11 are summarized as follows:

- Pressure differences due to coolant flow
- Weight of the reactor internals
- Superimposed loads such as those due to other structures such as the reactor core (fuel assemblies), control rod assemblies; and ICIS and thermocouple instrumentation supports
- Earthquake loads or other loads which result from motion of the RV
- Reactions from restraints, supports, or both
- Thermal loads from reactor coolant flow, thermal transients, irradiation gamma heating, and differential thermal expansion
- Loads resulting from the impingement of flow or reactor coolant, or other contained or surrounding fluids
- Transient pressure difference loads, such as asymmetric blowdown loadings due to pipe ruptures at postulated locations ~~those which result from rupture of a branch pipe~~
- Vibratory loads from flow induced vibration, and pump induced vibration
- Handling loads experienced in preparation for or during refueling or inservice inspection

3.9.5.2.1 Loading Combinations

All combinations of design and service loadings (e.g., operating differential pressure and thermal effects, potential adverse flow effects (flow-induced vibrations and acoustic resonances), seismic loads, and transient pressure loads of postulated LOCA) are accounted for in the design of the reactor internals. The distribution of the design and service loadings acting on the reactor internals components and structures are described below. Table 3.9-11 summarizes the loading combinations for the reactor internals. As an example of a loading combination for a loading condition, for the Level A loading condition, the loads to be combined are those loads marked with an X in the Level A column, unless otherwise indicated.

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. 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.

3.9.5.2.3 Interface Load and Displacement Limits

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

3.9.5.3 Design Bases

The rules for materials, design, fabrication, examination, and preparation of reports for the manufacture and installation of the US-APWR core support structures 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 (Reference 3.9-1).

Additional codes, standards, regulations, and guidelines from the NRC and the Utility Requirements Document are adhered to and are listed in the Owners design specification.

The design basis for the operability of the US-APWR internals are listed below and discussed in detail under the following sections:

- Safety analysis
- Thermal-hydraulic performance
- Core loading pattern
- Environmental conditions including irradiation
- RCS transients
- Interface design requirements

3.9.5.3.1 Safety Analysis Design Basis

The safety analysis design requirements and limits for the US-APWR internals are as follows:

- Mal-distribution of flow to the core should be limited so as not impact core safety limits in Chapter 15.
- RCCA drop times or insertion during normal service conditions should not be adversely affected.
- RCCA are to be inserted without impediment after an unanticipated accident, or a seismic and postulated LOCA event.
- There should be no impediment of the reactor internals to the emergency core cooling flow after a seismic and postulated LOCA event.
- The impact load on the RV bottom head from a postulated core drop event should not adversely affect the integrity of the RV bottom head.
- The reactor internals are to provide fast neutron fluence protection to the RV to preclude excessive embrittlement.
- The water volume is to be monitored at all times.

3.9.5.3.2 Thermal-Hydraulics Design Basis

The reactor internals are to be designed for the following thermal-hydraulic performance parameters:

- The flow conditions are as follows:
  - Thermal design flow
  - Best estimate flow
  - Mechanical design flow
  - Hot pump overspeed
  - Hot functional testing
- Pressure drops across the reactor internals are to meet system requirements for all Level A and B service flow conditions.
- Bypass flow is to be minimized and must not exceed system requirements for normal operation.
- Distribution of main coolant inlet flow into the fuel assemblies during normal operation must meet fuel assembly core inlet requirements.
- Core outlet flows from the fuel assemblies are to be designed to minimize horizontal velocities that may contribute to vibration of the RCCA rodlets.
- Main coolant flow into the outlet piping during normal operation is to meet system requirements, specifically (1) to minimize exit fluid temperature striations, and (2) meet the velocity criteria to prevent erosion.
- Bulk temperature of the main coolant flow is not to exceed pressurized water saturation temperature.
- Fluid temperature increase in the bypass flow may be credited to the reactor power output.

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 RV inside wall, and the core barrel outside surface. The main coolant flow then 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 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 type is rectangular in 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 the main flow that enters the guide tube exits through "windows" into the upper plenum cavity. Some of the flow exits through a controlled gap between the bottom of the guide tube flange and the top of the upper core plate. The guide tubes and support columns are carefully configured to minimize the pressure drop and cross-flow from the core exit fluid.

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}. 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.

Some percentage of the main coolant flow is bypass flow. The bypass flow is either for cooling metal or leakage between gaps. The bypass flows for cooling metal are as follows:

- Neutron reflector blocks and tie rods
- RV head

The bypass flows from gap leakages are as follows:

- Small gap between the core barrel outlet nozzle and RV outlet nozzle
- Gap between the neutron reflector ring block inside surface and the peripheral fuel assembly grids and nozzles
- Neutron reflector small gaps between the ring blocks

3.9.5.3.3 Core Loading Pattern and Axial Power Distribution Design Basis

The core loading pattern and the axial power distribution have a design impact on the reactor internals gamma heating, fluid temperatures, metal temperatures, and neutron fluence. The reactor internals that are affected by the core loading pattern and fluence are as follows:

- Core barrel
- Upper core plate
- Lower core support plate
- Neutron reflector ring blocks
- Neutron reflector ring block alignment pins
- Neutron reflector tie rods and mounting bolts
- Irradiation specimen guide and bolts

3.9.5.3.4 Environmental Conditions Design Basis

The environmental conditions that are addressed in the design of the reactor internals are from the following two sources:

- Water chemistry
- Fast neutron irradiation

The environmental effects of water chemistry on reactor internals materials are as follows:

- Corrosion
- Stress corrosion cracking
- Fatigue strength reduction

The fast neutron irradiation environmental effects on reactor internal materials are:

- Irradiated assisted stress corrosion cracking
- Irradiation stress relaxation
- Irradiation embrittlement

- Gamma heating
- Radiation exposure of the RV
- Void swelling

The environmental effects of corrosion and stress corrosion cracking of austenitic stainless steel materials in the reactor internals are generally not a concern based on experimental and operational experiences. Also fatigue strength degradation due to water chemistry conditions has likewise not been observed.

The fast neutron effects resulting from irradiated assisted stress corrosion cracking has been shown to occur on former bolts of the baffle which attach in core region of reactor internals. However, the US-APWR uses a neutron reflector instead of the former baffle structure thus eliminating high stress bolts. Therefore, the potential of irradiated assisted stress corrosion cracking is very low.

Another environmental effect from fast neutron exposure on reactor internals is irradiation stress relaxation. The US-APWR neutron reflector is fastened axially by tie rods of strain hardened 316 stainless steel.

Irradiation embrittlement can cause a decrease in ductility and an increase in yield and ultimate strength over the design life of the plant. The amount of irradiation embrittlement is highly dependent on the fluence, metal temperature, and stress condition. For the materials selected for reactor internals, the reduction in ductility and the increase in yield and ultimate strength as well as the fatigue strength has not been shown to be an issue for the operating conditions of the US-APWR plant.

Gamma heating from fast neutron irradiation is accounted for in the design of the reactor internals.

The RV is subjected to fast neutron exposure. Protection from excessive fluence comes from (1) the water in the annulus between the core barrel outside diameter and the inside of the RV, and (2) the neutron reflector.

Void swelling from irradiation is a concern for materials with high dose rates. The neutron reflector ring blocks are subjected to high fluence dose rates and the ring blocks are cooled by flow inside cooling holes to minimize void swelling. This environmental issue is addressed for the US-APWR.

3.9.5.3.5 RCS Transient Design Basis

The RCS transient design basis is discussed in Subsection 3.9.1.1.

3.9.5.3.6 Reactor Internals Vibration Design Basis

The reactor internals vibration loads come from a dynamic computer model that inputs the pressure difference across components and the pump rotating speed and pump-induced vibration effects. The mechanical loads and displacements from the vibration analysis are used as input to the structural analysis of the reactor internals.

3.9.5.3.7 Seismic Design Basis

The seismic analysis methodology is based on static and dynamic mathematical models and uses general purpose FE computer code. Refer to Subsection 3.9.2.5 for further discussion on the seismic design basis.

3.9.5.3.8 LOCA Design Basis

The LOCA design basis input is discussed in Subsection 3.9.2.5.

3.9.5.3.9 Interface Components Design Basis

The interface components design basis are those design parameters and design requirements that affect the design of the core support and internal structures. The parameters and requirements for interface components such as the reactor vessel, fuel assemblies, CRDM, and RCCA drive line system, thermocouple instrumentation, and ICIS are included in the design specification.

3.9.5.3.10 Reactor Internals Computational Methods and Verification of Input

Computational methods (e.g., the FE method) are used to determine stresses and displacements in the reactor internal components. Validation of the modeling includes the comparison of results with similar designs or testing for the natural frequencies, mode shapes, and frequency response functions with experimental or plant results.

3.9.5.3.11 Mechanical Design Criteria for the Reactor Internals

The mechanical design criteria for the reactor internals is included in the design specification.

3.9.5.3.12 PSI and ISI Plans

The PSI and ISI plans for the reactor internals is discussed below.

3.9.5.3.12.1 PSI Plan

The PSI plan follows the rules of ASME Code, Section XI (Reference 3.9-43). Visual inspection of parts subject to wear and galling are examined before and after hot functional testing. In addition, critical welds are also examined for any evidence of cracks.

3.9.5.3.12.2 ISI Plan

The ISI plan follows the rules of ASME Code, Section XI (Reference 3.9-43).

3.9.6 Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints

ASME Code, Section III, Class 1, 2 and 3 safety-related pumps, valves and dynamic restraints that are required to perform a specific function in shutting down the reactor to a safe-shutdown condition, in maintaining the safe-shutdown condition, or in mitigating the consequence of an accident, are subjected to IST to assess and verify operational readiness as set forth in 10 CFR 50.55a(f) (Reference 3.9-29) and ASME OM Code (Reference 3.9-13).

The pumps covered in the IST Program are those pumps that are provided with an emergency power source and required to perform a specific function in shutting down a reactor to a safe-shutdown condition, in maintaining the safe-shutdown condition, or in mitigating the consequence of an accident.

The US-APWR utilizes ASME OM Code (Reference 3.9-13) for developing the IST Program for ASME Code, Section III, Class 1, 2 and 3 safety-related pumps, valves and dynamic restraints. The COL Applicant is to administratively control the edition and addenda to be used for the IST program plan for pumps, valves, and dynamic restraints.

3.9.6.1 Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints

The requirement for ISI for ASME Code, Section III, Class 1, 2 and 3, safety-related pumps, valves and dynamic restraints IST assesses and verifies operational readiness included in various sections of the ASME OM Code as follow:

- Requirements for IST of pumps are incorporated in ISTB.
- Requirements for IST of valves are incorporated in ISTC.
- Requirements for IST of Motor-Operated Valve (MOV) are incorporated in ISTC 4.2.
- Requirements for IST of pressure relief valves are incorporated in Appendix I.
- Requirements for IST of dynamic restraints are incorporated in ISTD.

The various provisions for testing pumps, valves, and dynamic restraints are incorporated into the design of the US-APWR. These provisions and requirements are discussed in the respective sections of this DCD where the specific system is described.

It should be noted that the requirements of system pressure test per ASME Code, Section XI, Section IWA 5000 (Reference 3.9-43) that verify the system pressure boundary integrity are part of the ISI Program and are not part of this IST Program.

As required by the 10 CFR 50.55a(f) (Reference 3.9-29), ASME Code, Section III, Class 1, 2 and 3 safety-related pumps, valves and dynamic restraints are incorporated in 120-month interval IST Program Plan that is in compliance with the requirements of the latest edition and addenda of the OM Code, 12 months before the date of issuance of the operating license and, in compliance with Plant, Technical Specification and this DCD.

The IST Program Plan is also used for the required preservice (base line) testing of ASME Code, Section III, Class 1, 2, and 3 safety-related pumps, valves and dynamic restraints.

Relief requests from any of the applicable ASME OM Code test requirements are documented in the IST Program Plan, including justification and proposed alternative of test(s)/examination(s) that assess operation readiness of the impacted pumps, valves, or dynamic restraints.

3.9.6.2 IST Program for Pumps

IST of pumps is to determine the operational readiness of the pumps. The IST of pumps is performed at the required frequency as stated in the IST Program Plan. Test results are compared to the established and accepted preservice reference values, including the use of instrument range and accuracy. Table 3