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March 27, 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-09117

Subject: MHI's Response to US-APWR DCD RAI No. 207-1577

References: 1) "Request for Additional Information No. 207-1577 Revision 0, SRP Section: 03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components, Application Section: 3.9.2.5," dated 2/25/2009.

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. 207-1577 Revision 0."

Enclosed are all of the responses to the questions of the RAI (Reference 1).

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

Sincerely,

Y. Ogata

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

D081
NRO

Enclosures:

1. Response to Request for Additional Information No. 207-1577, Revision 0

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

Contact Information
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Enclosure 1

UAP-HF-09117
Docket No. 52-021

Response to Request for Additional Information No. 207-1577,
Revision 0

March 2009

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

3/27/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 207-1577 REVISION 0**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.5**
DATE OF RAI ISSUE:	**02/25/09**

QUESTION NO.: RAI 3.9.2-50

The applicant stated in DCD Tier 2, Subsection 3.9.2.5.1, that the mathematical model for dynamic system analysis includes representation of reactor vessel (RV) support system, inlet and outlet piping nozzles, control rod drive mechanism (CRDM) system, integrated head support system, in-core instrumentation support system, and fuel assembly nozzles and grids. The applicant further stated that fluid-structure interaction effects were accounted for by matrices developed for that purpose.

The staff reviewed Subsection 3.9.2.5 and found that the applicant did not provide sufficient details. Section 3.9.2 of the SRP states that mathematical model used for dynamic system analysis for LOCA in combination with SSE effects should include fluid-structure effects when applicable. Also, typical diagrams and modeling basis should be described. The applicant is requested to provide the details to explain how the fluid-structure effects are accounted for in the modeling of the reactor internals and dynamically related piping, pipe supports, and components. Alternately, provide a reference document that describes the mathematical models used for the dynamic system analysis for LOCA in combination with SSE. The staff needs this information to assure conformance with GDC 2, 4, 14, and 15. Revise the DCD to include the requested information or provide a reference where this information is available.

ANSWER:

Both seismic and LOCA dynamic analysis models are three dimensional, non-linear finite element (FE) models representing the reactor vessel (RV) and its internals in six degrees of freedom. The modeling is described in MHI Document MUAP-09002, Revision 0 (Ref. 1).

For the seismic dynamic analysis model, the hydrodynamic mass matrices for following three locations are included.

(1) Between the RV and the core barrel (CB) in two horizontal directions

(2) Between the CB and the neutron reflector (NR) in two horizontal directions

(3) Between the upper core support (UCS) and the RV head in vertical direction

The hydrodynamic mass matrices calculated for locations (1) and (2) were determined from the three dimensional solid-fluid FE analyses that are described in MHI document MUAP-07027 (Ref. 3). And the matrix for location (3) was derived from a hand calculation.

For the LOCA dynamic analysis model, the hydrodynamic mass matrices for location (1) are excluded because the hydrodynamic mass effect in this location is accounted for in the blowdown analysis with MULTIFLEX (Ref. 2).

References:

1. Summary of Seismic and Accident Load Conditions for Primary Components and Piping. MUAP-09002-P, Mitsubishi Heavy Industries, January 2009 (DCD Ref. 3.9-58)
2. MULTIFLEX, A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics. WCAP-8709 (proprietary), and WCAP-8709 (nonproprietary), September 1977 (DCD Ref. 3.9-9)
3. Comprehensive Vibration Assessment Program for US-APWR Reactor Internals. MUAP-07027, Mitsubishi Heavy Industries, December 2007 (DCD Ref. 3.9-22)

Impact on DCD

See Attachment 1 for the changes to be incorporated to DCD Section 3.9, Revision 2.

- Add a new paragraph after the first paragraph of Subsection 3.9.2.5:

  "Both seismic and LOCA dynamic analysis models are three dimensional, non-linear finite element (FE) mathematical models representing the reactor vessel and its internals in six degrees of freedom. The general purpose FE computer code, ANSYS (Ref 3.9-7) is used as the basis for the modeling."

- Add a new sentence to the end of the sixth paragraph of Subsection 3.9.2.5.1: "The hydrodynamic masses are calculated for the following locations in the seismic analysis model.

  (1) Between the reactor vessel and the core barrel in two horizontal directions

  (2) Between the core barrel and the neutron reflector in two horizontal directions

  (3) Between the upper core support and the reactor vessel head in vertical direction"

- Add the new sentence to end of the first paragraph in Subsection 3.9.2.5.1:"These models and description are described in Reference 3.9-58."

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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RAI NO.:	**NO. 207-1577 REVISION 0**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.5**
DATE OF RAI ISSUE:	**02/25/09**

QUESTION NO.: RAI 3.9.2-51

In Subsection 3.9.2.5.2 of the DCD the applicant stated that the pipe rupture analysis methodology is similar to the seismic analysis methodology. The reactor internals are represented in the model by beam elements; and the connectivity of the reactor internals and interfacing structures, is represented by mass inertia effect, stiffness and hydrodynamic matrices, springs, and/or impact elements including gap and damping. Dominant frequencies are identified by comparing the frequency response of the reactor internals with the response based on experience and measurements.

The staff's review of Subsection 3.9.2.5 of the DCD indicated that the applicant did not provide any discussion regarding system structural partitioning and directional decoupling employed in the model. Section 3.9.2 of the SRP states that mathematical model used for dynamic system analysis of reactor internals should include a justification regarding any system structural partitioning and directional decoupling employed in the model. The applicant is requested to provide a discussion and justification for any system structural partitioning and directional decoupling employed in the dynamic system modeling of the reactor internals and the reactor pressure vessel. The staff needs this information to assure conformance with GDC 2, 4, 14, and 15. Revise the DCD to include the requested information.

ANSWER:

The nodal point degrees of freedom, and damping coefficients of the reactor internals and surrounding structures are selected such that most dominant frequencies are represented in the seismic-LOCA response. This forms the basis for establishing any directional decoupling and system structural partitioning in the seismic-LOCA model.

Both the seismic and LOCA dynamic analyses models consist of beam elements, linear and non-linear springs, gaps, hydrodynamic mass matrices, and stiffness matrices. The only shell element modeled is for the diffuser plates.

The main structures are modeled by the beam elements. The structures using beam elements are:

- Reactor Vessel (RV)
- Core Barrel (CB)
- Control Rod Drive Mechanism (CRDM)
- Integrated Head Package (IHP)
- Upper Core Support (UCS)
- Upper Core Plate (UCP)
- Lower Core Support Plate (LCSP)
- Guide Tube (GT)
- Upper Support Columns (USC)
- In-core Instrumentation System (ICIS)
- Neutron Reflector (NR)
- Fuel Assembly (FA)

The structural interfaces are modeled by the spring, gap, and impact elements.

The structures using linear spring elements are:

- RV supports in tangential direction (8 locations)
- Holddown Spring in vertical direction
- Fuel Assembly Holddown Spring in vertical direction

The structures modeled by the non-linear spring, gap, and impact elements are:

- RV supports in vertical direction (8 locations)
- Radial keys between the RV and CB in tangential direction (6 locations)
- Upper core plate alignment pins between the UCP and CB in tangential direction (4 locations)
- CB outlet nozzles between the CB and RV in radial direction (4 locations)
- Between CB flange and RV flange in radial direction (16 locations)
- Between UCS flange and RV flange in radial direction (16 locations)
- Between FA and NR in horizontal direction
- Between FAs in horizontal direction
- FA nozzle and UCP and LCSP in vertical direction

The RCS loops are not explicitly modeled in the seismic and LOCA dynamic models; however they are simulated by stiffness matrices connected to the RV nozzle center location. The fluid-structural effects are accounted for between the RV and CB, CB and NR, UCS and RV by hydrodynamic mass matrices. The effects of friction between the CB flange and RV, the UCS flange and RV are accounted for by the friction elements. The shell elements are used for modeling the diffuser plates. The core region is divided to five regions, one center region and four outer regions. The beam elements are used for representing the UCS, UCP and LCSP vertical vibration and out of plane stiffness.

The time history seismic accelerations developed from the reactor coolant loop (RCL) seismic analysis are applied to the RV supports and integrated head package (IHP) locations. The LOCA time history forcing functions from the MULTIFLEX (Ref. 3) analysis are applied to the RV and CB in the horizontal direction and the vertical forces are applied to the RV, UCS, CB, LCSP, UCP and FAs.

The output from the seismic and LOCA dynamic analyses include the element forces, impact forces, and deflections for the static FE stress analysis of the structures.

Impact on DCD

See Attachment 1 for the changes to be incorporated to DCD Section 3.9, Revision 2:

Add a new paragraph at the end of Subsection 3.9.2.5:
"The nodal point degrees of freedom, and damping coefficients of the reactor internals and surrounding structures are selected such that most dominant frequencies are represented in the seismic-LOCA response. This forms the bases for establishing any directional decoupling and system structural partitioning in the seismic-LOCA system models. Detailed discussion of the seismic and LOCA system models are described in Reference 3.9-58."

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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Docket No. 52-021

RAI NO.:	**NO. 207-1577 REVISION 0**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.5**
DATE OF RAI ISSUE:	**02/25/09**

QUESTION NO.: RAI 3.9.2-52

The applicant stated in DCD Tier 2, Subsection 3.9.2.5.2, that the mathematical model for dynamic system analysis includes such structural characteristics as the flexibility, mass inertia effect, geometric configuration, and damping (including possible coexistence of viscous and Coulomb damping).

However, the staff found that the applicant did not include any justification that the model is representative of the system structural characteristics, or provide a reference document where such information is available. Section 3.9.2 of the SRP states that mathematical model used for dynamic system analysis of reactor internals under faulted conditions should typify such system structural characteristics as flexibility, mass inertia effect, geometry geometric configuration, and damping (including possible coexistence of viscous and Coulomb damping). The applicant is requested to provide a discussion to justify that the dynamic reactor internals models are representative of system structural characteristics, such as the flexibility, mass inertia effect, geometric configuration, and damping (including possible coexistence of viscous and Coulomb damping). Alternately, provide a reference document where this information is available. The staff needs this information to assure conformance with GDC 2, 4, 14, and 15. Revise the DCD to include the requested information or provide a reference where this information is available.

ANSWER:

The dynamic system structural characteristics are discussed in MHI document MUAP-09002, Rev. 0, (Ref 1, see RAI 3.9.2-50). Also refer to the summary of the dynamic system structural characteristics in responses to RAI 3.9.2-50 and 3.9.2-51.

Validation that the reactor internals dynamic models are representative is made by the comparison of a simulation analysis of 1/5 scale model test and the test results as discussed in Reference 1 (below).

Reference:

1. Comprehensive Vibration Assessment Program for US-APWR Reactor Internals. MUAP-07027, Mitsubishi Heavy Industries, December 2007 (DCD Ref.3.9-22), Subsection 3.2.1, Validation of Structure Models

Impact on DCD

See Attachment 1 for the changes to be incorporated into DCD Section 3.9, Revision 2.

- Add the following text to the end of the forth paragraph of Subsection 3.9.2.5.2: "These models are described in References 3.9-22 and 3.9-58."

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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RAI NO.:	**NO. 207-1577 REVISION 0**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.5**
DATE OF RAI ISSUE:	**02/25/09**

QUESTION NO.: RAI 3.9.2-53

The applicant stated in DCD Tier 2, Subsection 3.9.2.5.2, that the mathematical model for dynamic system analysis includes such structural characteristics as the flexibility, mass inertia effect, geometric configuration, and damping (including possible coexistence of viscous and Coulomb damping). In addition, the effects of flow upon the mass and flexibility properties of the system are accounted for in the model.

The staff reviewed the DCD and found that the applicant did not provide sufficient details regarding the mathematical model. Section 3.9.2 of the SRP states that mathematical model used for dynamic system analysis for LOCA in combination with SSE effects should address the effects of flow upon the lumped-mass and flexibility properties of the system. The applicant is therefore requested to provide the details to explain how the effects of flow upon the lumped-mass and flexibility properties of the system are accounted for in the LOCA dynamic system analysis model. Alternately, provide a reference document where this information is available. The requested information is needed to complete the review of the mathematical model for dynamic system analysis and to assure conformance with GDC 2, 4, 14, and 15. Revise the DCD to include the requested information or provide a reference where this information is available.

ANSWER:

The effects of flow upon both the lumped mass and flexibility properties in the LOCA dynamic system model are accounted for in the model because the MULTIFLEX (Ref. 3) results used as input to the LOCA dynamic system model included fluid-structural interaction. Refer to response to RAI 3.9.2-50 and 51 for additional discussion of the modeling.

The hydraulic forces acting on the reactor internal structures during normal operation are accounted for in the seismic dynamic analysis by applying the loads as steady-state uniform loadings.

Impact on DCD

See Attachment 1 for the changes to be incorporated to DCD Section 3.9.

- Add the following two paragraphs after the modified sixth paragraph of Subsection 3.9.2.5.1:

  "In the LOCA dynamic analysis, the hydrodynamic mass matrices between the reactor vessel and core barrel were deleted because the hydrodynamics mass effect were included in the pressure force as the output of blow-down analysis code, MULTIFLEX.

  The effects of flow upon both the lumped mass and flexibility properties in the LOCA dynamic system model were also accounted because they were included in the MULTIFLEX results which were used as input to the LOCA dynamic system model."

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.5**
DATE OF RAI ISSUE:	**02/25/09**

QUESTION NO.: RAI 3.9.2-54

The applicant stated in DCD Tier 2, Subsection 3.9.2.5.2 that the design input for the dynamic system analyses is defined by postulated pipe rupture, as discussed in Subsection 3.6.2 of the DCD. The time-history forcing function on the reactor internals is determined from pipe rupture events that are enveloped by the most limiting blow-down hydraulic loads.

The DCD, however, does not provide any discussion regarding the basis for developing the forcing function. Section 3.9.2 of the SRP states that evaluation of the dynamic effects on reactor internals associated with postulated pipe rupture should include a description of the governing hydrodynamic equations and the assumptions used for flow path geometries, tests for determining flow coefficients, and any semi-empirical formulations and scaled model testing for determining pressure differentials or velocity distributions. Typical diagrams and the basis for postulating the LOCA-induced forcing function should also be provided. The applicant is requested to provide (a) typical diagrams and the basis for postulating the pipe break-induced forcing function, including a description of the governing hydrodynamic equations and the assumptions used for flow path geometries, and (b) tests to determine flow coefficients, and any semi-empirical formulations and scaled model tests to determine pressure differentials or velocity distributions. The staff needs this information to assure conformance with GDC 2, 4, 14, and 15. Revise the DCD to include the requested information.

ANSWER:

In response to (a): the forcing functions, in the case of LOCA events, are developed by blowdown analysis using the MULTIFLEX computer code. The typical diagrams, analysis conditions, and the assumptions of the flow path geometry and postulated pipe break are described in section 4.2.1.1 of Reference 1 (below, DCD Ref. 3.9-58). The summary of MULTIFLEX code is described in section 9 of Reference 1 (DCD Ref. 3.9-58). The descriptions of the governing hydrodynamic equation of MULTIFLEX code are described in section 2.0 of Reference 2 (DCD Ref. 3.9-9).

In response to (b): the test to determine the flow coefficient and semi-empirical formulations, and the verification methods are described in Reference 2 (DCD Ref. 3.9-9). The MULTIFLEX code

has been approved by NRC (reference 3). The analysis model described in Reference 1 (DCD Ref. 3.9-58) is the same as the US conventional plants evaluated by Westinghouse.

References:

1. Summary of Seismic and Accident Load Conditions for Primary Components and Piping, MUAP-09002, Rev. 0, Mitsubishi Heavy Industry, January 2009 (DCD Ref. 3.9-58)
2. MULTIFLEX, A Fortran-IV Computer Program for Analyzing Thermal- Hydraulic-Structure System Dynamics, WCAP-8709 (proprietary and nonproprietary), September 1977 (DCD Ref. 3.9-9).
3. Evaluation of Westinghouse Topical Reports WCAP-8709 (proprietary/ nonproprietary), Letter from John F. Stolz (NRC) to C. Elicheldinger (Westinghouse) dated June 17, 1977.

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.

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RAI NO.:	**NO. 207-1577 REVISION 0**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.5**
DATE OF RAI ISSUE:	**02/25/09**

QUESTION NO.: RAI 3.9.2-55

In Subsection 3.9.2.5.2 of the DCD the applicant stated that methods and procedures used for the LOCA dynamic system analysis are based on the computer code used in the LOCA analysis. The computer code incorporates the governing equations of motion and the computational scheme for deriving results. Asymmetric LOCA loads for the reactor internals are considered for the LOCA dynamic system analysis.

However, staff's review indicated that the DCD did not provide sufficient details. Section 3.9.2 of the SRP states that evaluation of the dynamic effects on reactor internals associated with postulated pipe rupture should include a description of the methods and procedures for dynamic system analyses, including the governing equations of motion and the computational scheme for deriving results. The applicant is requested to describe in detail the methods and procedures used for dynamic system analyses including the governing equations of motion and the computational scheme used for deriving results. Alternately, provide a reference document where this information is available. The staff needs this information to assure conformance with GDC 2, 4, 14, and 15. Revise the DCD to include the requested information or provide a reference where this information is available.

ANSWER:

The general purpose finite element (FE) computer code ANSYS (Ref. 1) is used for the modeling. The methods and procedures used for seismic and LOCA dynamic system analyses, including the governing equations of motion and the computational scheme for deriving results, are provided in MHI document MUAP-09002, Revision 0 (Ref. 2). Refer to responses in RAI 3.9.2-50 and 3.9.2-51 for additional discussion on the subject.

References:

1. ANSYS, Finite Element Structural Analysis Program, Release 11.0, ANSYS, Inc., Canonsburg, PA, 2007 (DCD Ref.3.9-7).

2 Summary of Seismic and Accident Load Conditions for Primary Components and Piping, MUAP-09002, Rev. 0, Mitsubishi Heavy Industry, January 2009 (DCD Ref. 3.9-58)

Impact on DCD

See Attachment 1 for the changes to be incorporated into DCD Section 3.9, Revision 2.

- Add the following text to the end of the forth paragraph of Subsection 3.9.2.5.2: "These models are described in References 3.9-22 and 3.9-58."

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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APPLICATION SECTION:	**3.9.2.5**
DATE OF RAI ISSUE:	**02/25/09**

QUESTION NO.: RAI 3.9.2-56

The applicant stated in Subsection 3.9.2.5.2 of the DCD that methods and procedures used for the LOCA dynamic system analysis are based on the computer code used in the LOCA analysis. The computer code incorporates the governing equations of motion and the computational scheme for deriving results. Asymmetric LOCA loads for the reactor internals are considered for the LOCA dynamic system analysis.

The applicant did not include a discussion to assure that there is no significant dynamic amplification of the load on reactor internals as a result of the oscillatory nature of the blow-down forces during a postulated LOCA. Section 3.9.2 of the SRP states that evaluation of the dynamic effects on reactor internals associated with postulated pipe rupture should include a description of the methods and procedures for dynamic system analyses, including the governing equations of motion and the computational scheme for deriving results. The applicant is requested to provide the analytical results to demonstrate that there is no significant amplification of the loads on reactor internals and core support structures as a result of postulated pipe rupture. The staff needs this information to assure conformance with GDC 2, 4, 14, and 15. Revise the DCD to include the requested information.

ANSWER:

The methods and procedures used for the seismic and LOCA dynamic system analyses including the governing equations of motion and computational scheme for deriving results are provided in MHI document MUAP-09002, Revision 0 (Ref. 1). The dynamic system analysis results for LOCA in combination with SSE are also provided in MHI document MUAP-09002, Revision 0 (Ref. 1).

These dynamic output parameters are used into the detailed structures component static FE model and the maximum stress intensities are calculated. The results are compared to the ASME Code, Section III (DCD Reference 3.9-1) Level D service limits.

The stress report of the core support structures inducing above results will be submitted to the NRC in March, 2009.

Impact on DCD

See Attachment 1 for the changes to be incorporated into DCD Section 3.9, Revision 2.

- Add the following sentence to the end of the eighth paragraph of Subsection 3.9.2.5.2: "More details about the methods of LOCA dynamic system analysis is described in Section 6 of Reference 3.9-58."

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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RAI NO.:	**NO. 207-1577 REVISION 0**
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APPLICATION SECTION:	**3.9.2.5**
DATE OF RAI ISSUE:	**02/25/09**

QUESTION NO.: RAI 3.9.2-57

The applicant stated in Subsections 3.9.2.5.2 and 3.9.2.5.3 of the DCD that the maximum stress intensities and displacements obtained from the LOCA dynamic system analyses are compared with the ASME Code, Section III, stress limits, and the allowable interface load and displacement limits given in Table 3.9-2 of the DCD. The applicant further stated that the LOCA dynamic system analyses results confirm that the structural design adequacy of the reactor internals can withstand the dynamic loadings of the most severe LOCA in combination with the SSE.

The staff's review of the DCD showed that the applicant did not provide any details regarding the dynamic systems analyses. Section 3.9.2 of the SRP states that the dynamic system analyses should confirm the design adequacy of the reactor internals and unbroken loops of the reactor coolant piping, to withstand the dynamic loadings of the most severe LOCA in combination with SSE. The applicant is requested to identify the locations in the reactor internals where the stress deformation and fatigue are determined to be the highest. Also identify the corresponding loading combination. The staff needs this information to assure conformance with GDC 2, 4, 14, and 15. Revise the DCD to include the requested information.

ANSWER:

Locations that are likely to be sources of high stresses during a Level D event are structures that have to transmit high loads and have limiting minimum thicknesses. Critical core support structure locations identified in MUAP-09002-P (R0), Figures 8-12 and 8-13 include:

- Core Barrel Flange Discontinuity
- Upper/Lower Core Barrel Discontinuity
- Lower Core Barrel/Lower Core Support Plate Discontinuity
- Radial Support Key
- Lower Core Support Plate
- Upper Core Support (UCS)

- UCS Flange/Skirt Discontinuity
- Top Slotted Column Extension
- Top Slotted Column
- Top Slotted Column Fastener
- Upper Core Support Column Extension
- Upper Core Support Column
- Upper Core Support Column Fastener

The critical location of the deformation is shown in Table 3.9-2 in the DCD.

The stress report of the core support structure will be submitted to the NRC in March, 2009.

Impact on DCD

See Attachment 1 for the changes to be incorporated into DCD Section 3.9, Revision 2.

- Add the following sentence after the first paragraph of Subsection 3.9.2.5.3:

  "The locations to evaluate the stress and fatigue are described in Reference 3.9-58."

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.5**
DATE OF RAI ISSUE:	**02/25/09**

QUESTION NO.: RAI 3.9.2-58

In Subsections 3.9.2.5.2 and 3.9.2.5.3 of the DCD the applicant stated that the maximum stress intensities and displacements obtained from the LOCA dynamic system analyses are compared with the ASME Code, Section III, stress limits, and the allowable interface load and displacement limits given in Table 3.9-2 of the DCD. In addition, the functional requirements that need to be met include the following:

(a) Allowable horizontal load of the guide tube should not impede insertion of the control rod after a LOCA,
(b) Upper core barrel displacement is not to impede the down comer emergency core cooling flow after a LOCA,
(c) Reaction loads at the RV connections are not to exceed allowable values of the interface load,
(d) Maximum vertical displacement of the upper core plate relative to the upper support plate should preclude buckling of the guide tube, and
(e) Upper core barrel permanent displacement should not prevent loss of function of the control rod assembly by radial inwardly deforming the upper guide tube.

The staff reviewed the DCD and found that the applicant did not address the stability of the core barrel in compression. Section 3.9.2 of the SRP states that the dynamic system analyses of the reactor internals under pipe rupture loadings should investigate the stability of the elements in compression such as the core barrel and control rod guide tubes. The applicant is requested to describe how the stability of the elements in compression such as the core barrel and control rod guide tubes, under pipe rupture loadings, was investigated. The staff needs this information to assure conformance with GDC 2, 4, 14, and 15. Revise the DCD to include the requested information.

ANSWER:

The stability of the core barrel and control rod guide tubes during a pipe rupture are evaluated with the pressure loadings from the blow-down analysis using MULTIFLEX code. The design limit for the evaluation is determined in according to Ref.1.

DCD Tier 2, Subsection 3.9.2.5.2 will be revised to state the stability of the core barrel and control rod guide tubes, during pipe rupture event, must be assured.

Reference

1. ASME Boiler and Pressure Vessel Code Section III, Division 1 , NONMANDATRY APPENDIX F , ARTICLE F-1000, RULES FOR EVALUATION OF SERVICE LOADINFS WITH LEVEL D SERVICE LINITS, F-1331.5 Requirements for Compressive Loads (Included in DCD Ref.3.9-1)

Impact on DCD

See Attachment 1 for the changes to be incorporated into DCD Section 3.9, Revision 2.

- Add the following item to the list at the end of the second paragraph of Subsection 3.9.2.5.3:

  "(f) The core barrel and upper guide tube are stable in compression during a pipe rupture break."

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

3/27/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 207-1577 REVISION 0**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.5**
DATE OF RAI ISSUE:	**02/25/09**

QUESTION NO.: RAI 3.9.2-59

The applicant stated in DCD Tier 2, Subsections 3.9.2.5.3, that the pipe break sizes of current 4-loop plants were based on the largest LOCA loads that resulted from either a 0.093 m2 (1.0 ft^2) single-ended cold leg break or a double-ended hot leg break, whereas, the leak before break (LBB) criteria is applied to determine the break condition for the US-APWR design input. The magnitude of blow-down hydraulic loads applying LBB is smaller than either the loads for the large cold leg or hot leg breaks. Thus, maximum stresses and displacements of the reactor internals under faulted conditions meet the ASME Code Section III Subsection NG stress and deflection limits.

The staff's review indicated that the DCD did not provide sufficient details. The applicant is requested to:

(a) Confirm that to eliminate the dynamic effects of pipe rupture from the design basis, leak before break (LBB) evaluation was performed in accordance with SRP Section 3.6.3 to demonstrate that the probability of pipe rupture is extremely low for the applied loading resulting from normal conditions, anticipated transients, and postulated SSE.
(b) Identify the piping systems that were included in the evaluations.
(c) What were the nominal pipe diameter and postulated pipe break flow area for the limiting design basis pipe size used to determine the pipe rupture dynamic response.

The staff needs this information to assure conformance with GDC 2, 4, 14, and 15.
Revise the DCD to include the requested information.

ANSWER:

MHI identified the postulated pipe ruptures and provided the results of the dynamic analysis in Technical Report, MUAP-09002-P, Rev. 0, Summary of Seismic and Accident Load Conditions for Primary Components and Piping. Subsection 4.1 (page 4-1) provides the cases of postulated pipe ruptures as shown below.

MHI applied LBB criteria for main coolant piping and Main Steam Line. The detailed information of the LBB evaluation will be provided in Technical Reports summarizing the stress analyses for

the main steam line and class 1 piping. As a result of the LBB evaluation, the following postulated pipe break events have been selected for the reactor coolant system (RCS) loop components.

1) Hot Leg branch line break at the 10 in. Schedule 160 Residual Heat Removal (RHR)/ Safety Injection (SI) line nozzle
   Nominal pipe diameter: NPS 10 in. Flow area: 56.75 in^2
2) Cold Leg branch line break at the 14 in. Schedule 160 accumulator line nozzle
   Nominal pipe diameter: NPS 14 in. Flow area: 98.31 in^2
3) Feedwater line break at the SG feedwater nozzle
   Nominal pipe diameter: NPS 16 in. Flow area: 160.88 in^2
4) Main Steam line break at the SG main steam nozzle
   Nominal pipe diameter: NPS 32 in. Flow area: 660.88 in^2

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.

This completes MHI's responses to the NRC's questions.

- Secondary core support assembly: Vibration responses will be measured by strain gages mounted on the diffuser plate support columns.
- RCCA guide tubes and upper support columns: Beam mode responses due to the cross-flow in the upper plenum will be measured by strain gages and accelerometers.
- Upper core support: The vertical response will be measured by an accelerometer mounted near the center of the upper core support plate. Horizontal responses will be measured by strain gages installed on the upper core support skirt.

3.9.2.3.1 Inspection Program

The internal components of all US-APWR plants will be inspected before and after the hot functional test. The reactor internals will not be considered adequate and pass the comprehensive vibration assessment program unless no indication of harmful sign, abnormally large vibration amplitudes or excessive wear is detected.

3.9.2.3.2 Acceptance Criteria

The acceptance criteria of the pre-operational flow-induced vibration testing for reactor internals are as follows.

- Vibration measurement

  The measured rms vibration amplitudes will be multiplied by 4.5 to convert them into 0-peak values. The corresponding 0-peak stresses in key connecting components will be calculated from the measure vibration amplitudes or strains. These stresses must show sufficient safety margins based on the design fatigue curves in the ASME Code, Section III, Appendix-I.

- Inspection

  Broken components and/or excessive wear or deformation is not observed in the post-hot functional test inspection.

3.9.2.5 Dynamic System Analysis of the Reactor Internals under Faulted Conditions

The NUREG-0800, SRP 3.9.2, Rev. 3 (Ref. 3.9-26), requires that the Design Control Document (DCD) provide a detailed discussion of the reactor internals, design criteria and dynamic analyses methodology for the combined seismic and postulated pipe rupture events under ASME Level D (faulted) service conditions. The results of the analyses are required to meet the stress limits of the ASME Code, Section III, Subsection NG (Ref. 3.9-1) for Core Support Structures (CSSs), and the functional requirements of the reactor internals design specification. Meeting the requirements of the ASME Code, Section III (Ref. 3.9-1) and the design specification should provide assurance of the structural and functional integrity of the reactor internals under ASME Level D service condition combined the loads of seismic and pipe rupture events.

Both seismic and LOCA dynamic analysis models are three dimensional, non-linear finite element (FE) mathematical models representing the reactor vessel and its internals

in six degrees of freedom. The general purpose FE computer code, ANSYS (Ref 3.9-7) is used as the basis for the modeling.

The nodal point degrees of freedom, and damping coefficients of the reactor internals and surrounding structures are selected such that most dominant frequencies are represented in the seismic-LOCA response. This forms the bases for establishing any directional decoupling and system structural partitioning in the seismic-LOCA system models. Detailed discussion of the seismic and LOCA system models are described in Reference 3.9-58."

3.9.2.5.1 Seismic Analysis Methodology and Acceptance Criteria

The seismic analysis methodology is based on two separate mathematical models and uses general purpose FE computer code. The first model is a three-dimensional non-linear dynamic FE computer model representing the reactor internals and the support system and is used to determine the maximum accelerations, displacements, and loadings that are used as input to the second model. The second computer model or models are three-dimensional static FE computer programs that are used to determine the maximum seismic stress intensities and displacements. These models are described in Reference 3.9-58.

The maximum stresses from the static FE model for both SSE and LOCA events are combined by the SRSS and the results are compared to the service limit stress intensities in the ASME Code, Section III, Subsection NG (Ref. 3.9-1). The maximum displacements and loads are compared to the allowable limits in the design specification. The details of the seismic dynamic computer model are discussed below.

The pre-processing input of the seismic mathematical computer model comes from the design drawings and is the bases of the geometrical and material representation and connectivity of the reactor internals components and interfacing components. This includes the representation of the RV support system, inlet and outlet piping nozzles, Control Rod Drive Mechanism (CRDM) system, integrated head support system, in-core instrumentation support system, and fuel assembly nozzles and grids.

Figures 3.9-3 and 3.9-4 represent a typical mathematical model of the reactor internals used for seismic analysis. The physical geometry and material properties (density, modulus of elasticity, Poisson's ratio) of the reactor internals are represented by beams elements. The reactor internals and interfacing structures are connected or represented by mass inertia effect, stiffness matrices, and hydro-dynamic matrices, springs, and/or impact elements including gap and damping (including coexistence of viscous and Coulomb damping).

The nodal point degrees of freedom, and damping coefficients of the reactor internals and surrounding structures are selected such that the most dominant frequencies are represented in the seismic response. Dominant frequencies are identified by comparing the frequency response of the reactor internals with the expected responses based on experience and measurements.

Fluid-structure interaction effects are accounted for by the hydrodynamic masses are calculated for the following locations in the seismic analysis model.

(1) Between the reactor vessel and core barrel in two horizontal directions

(2) Between the core barrel and neutron reflector in two horizontal directions

(3) Between the upper support plate and reactor vessel head in vertical direction

In the LOCA dynamic analysis, the hydrodynamic mass matrices between the reactor vessel and core barrel were deleted because the hydrodynamics mass effect were included in the pressure force as the output of blow-down analysis code, MULTIFLEX.

The effects of flow upon both the lumped mass and flexibility properties in the LOCA dynamic system model were also accounted because they were included in the MULTIFLEX results which were used as input to the LOCA dynamic system model.

The reactor internals seismic input can either be from in-structure response spectra or in-structure time-history accelerations which is obtained from the analysis results described in Subsection 3.8.3. This model employs the design response spectra of the building-RCL coupled model based on modified input from RG 1.60, Rev. 1 (Ref. 3.9-27) as described in Section 3.7. This model is used in determining the effect of vibratory motion for SSE and 1/3 SSE seismic conditions.

Additional loading input to the seismic analysis are vertical pressure loadings converted to nodal point external loads, and the vertical weights of the reactor internals and interfacing components by input of density on the beams with spring effects or mass nodal points.

The reactor internals static computer models are used to determine the reactor internals component stresses and displacements. The structural design adequacy of the reactor internals can withstand the dynamic loadings of the most severe SSE in combination with the LOCA events.

3.9.2.5.2 Pipe Rupture Analysis Methodology and Acceptance Criteria

The pipe rupture design basis methodology is similar to the seismic methodology, wherein, a dynamic computer model is used to determine the maximum accelerations, displacements, and loadings, and the reactor internals static computer models are used to determine the reactor internals component stresses and displacements. However, instead of a response spectra or time-history for the seismic input, a time-history computer code is used to determine the pipe rupture loads-time history on the dynamic computer model nodes and elements. The details of the pipe rupture dynamic model and pressure input loads are discussed below.

The LOCA dynamic computer model is a three-dimensional FE model that defines the geometry, material properties, and nodal point connections and elements.

The mathematical model for LOCA dynamic effects includes reactor internals and dynamically-related piping stiffness, RV supports, interfacing components, and fluid-structure interaction effects.

The mathematical models in Figures 3.9-3 and Figure 3.9-4 are used for the LOCA dynamic system analysis, which include such structural characteristics as the flexibility, mass inertia effects, geometric configuration, spring, and impact elements including gap and damping (including coexistence of viscous and Coulomb damping). The effects of flow upon the mass and flexibility properties of the system are accounted for in the model. These models are described in References 3.9-22, and 3.9-58.

The design input for the pipe rupture event is defined by the postulated Leak-Before-Break (LBB) pipe rupture as discussed in Subsection 3.6.2 of the DCD. A time-history forcing function on the reactor internals comes from pipe rupture that are enveloped by the most limiting blow-down hydraulic loads.

The MULTIFLEX computer code is used for the blowdown analysis in the hydraulic load evaluation of the postulated LOCA accident. The MULTIFLEX is a computer program which calculates the transient of pressure, flow rate and density during the initial phase of the blowdown in a complex system such as the primary coolant system of a Pressurized Water Reactor (PWR). The MULTIFLEX code includes mechanical structure models and their interactions with thermal-hydraulic system (Ref.3.9-9).

The general characteristics of the MULTIFLEX code are shown in the following.

- The complex system is modeled with one-dimensional hydraulic piping.
- The flow conditions within the system are calculated by solving the one-dimensional equations of mass, momentum and energy conservation using the method of characteristics.
- The MULTIFLEX code includes heat transfer models of the core and the SG, and also simulates various boundary conditions of the PWR system including the core.
- The calculated results of the MULTIFLEX code (pressure, flow rate, and so on) are used in the RV internals load evaluation and the RCL mechanical load evaluation.

The methods and procedures for the LOCA dynamic system analysis is based on the computer program code used in the LOCA analysis. The computer code incorporates the governing equations of motion and the computational scheme for deriving results. Asymmetric LOCA loads for the reactor internals are considered for the LOCA dynamic system analysis. More details about the methods of LOCA dynamic system analysis is described in Section 6 of Reference 3.9-58.

The outputs of the LOCA response analysis are time-history accelerations, displacement (absolute and relative), and loadings (forces and moments). The maximum loadings and displacements are input into reactor internals component static FE models and the maximum stress intensities and displacements are compared to the ASME Code, Section III (Ref. 3.9-1) and the allowable interface load and displacement limits (Ref. Table 3.9-2). The criteria for acceptance of the LOCA loads and displacements are discussed in Section 3.9.2.5.

The LOCA dynamic system analyses results confirm that the structural design adequacy of the reactor internals can withstand the dynamic loadings of the most severe LOCA in combination with the SSE.

3.9.2.5.3 Structural Design Adequacy Criteria for Level D Combined Loadings

The most severe dynamic loadings and displacements of the pipe rupture event are combined with the SSE event and the resulting stresses are compared with the limits of the ASME Level D service limits for acceptability.
The locations to evaluate the stress and fatigue are described in Reference 3.9-58.

In addition to the ASME Code, Section III (Ref. 3.9-1) stress criteria, there are functional requirements as listed in Table 3.9-2 to be met as follows:

(a) The allowable horizontal load of the guide tube should not impede insertion of the control rod assemblies after the LOCA event.

(b) The upper core barrel displacement is not to impede the down comer emergency core cooling flow after the LOCA event.

(c) The reaction loads at the RV connections are not to exceed allowable values of the interface load.

(d) The maximum vertical displacement of the upper core plate relative to the upper support plate should preclude buckling of the guide tube.

(e) The upper core barrel permanent displacement should not prevent loss of function of the control rod assembly by radial inwardly deforming the upper guide tube.

(f) The core barrel and upper guide tube are stable during a pipe rupture break.