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Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. Jeffrey A. Ciocco,

Project No.0751
MHI Ref: UAP-HF-08037

Subject: Responses to NRC's Questions for NRC's Acceptance Review of the US-APWR Design Certification Application

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the documents entitled "MHI's Response to the NRC's Questions for NRC's Acceptance Review of the US-APWR Design Certification Application" and "MHI's Responses to NRC's Questions on Design of the Sump Strainers". In the enclosed documents, MHI provides responses to NRC's questions to support the NRC's acceptance review of the US-APWR Design Certification Application. The responses both provide additional information in response to the NRC's questions and provide commitments to submit additional information to the NRC at a later date.

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

Sincerely,

M. Kaneda

Masahiko Kaneda,
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

Enclosures:
Enclosure1 - MHI's Responses to NRC's Questions for NRC's Acceptance Review of the US-APWR Design Certification Application
Enclosure2 - MHI's Responses to NRC's Questions on Design of the Sump Strainers

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Enclosure 1

US-APWR

MHI's Responses to NRC's Questions for NRC's Acceptance Review of the US-APWR Design Certification Application

February 2008

MHI's Responses to NRC's Questions for NRC's Acceptance Review of the US-APWR Design Certification Application

February 2008

INTRODUCTION

This report documents MHI's responses to NRC's questions to support the NRC's acceptance review of the US-APWR Design Certification Application on Chapters 4, 6 and 15.

NRC QUESTION-1:
NRC requests MHI to provide Emergency Response Guidelines (ERGs) for operator actions credited in the Chapter 15 safety analyses of the US-APWR Design so that the NRC can verify that future plant Emergency Operating Procedures (EOPs) will correspond to operator actions assumed in the Chapter 15 safety analyses.

MHI RESPONSE:
The key features of accident prevention and mitigation for the US-APWR design are essentially the same as those utilized in currently operating reactors. Therefore, the operator actions assumed in the Chapter 15 analyses are similar to the assumptions made for currently operating reactors. DCD Subsection 15.0.0.6 lists the specific events where operator actions are assumed as part of the event analysis and provides a cross-reference to the event specific analysis section for additional details regarding the specific assumed operator actions. Table 1-1 provides a summary of all of the operator actions that are applied to the analysis of Chapter 15 events.

Table 1-1 Operator Actions

Accident	Manual Action
Inadvertent Decrease in Boron Concentration in RCS (Subsection 15.4.6)	Closure of Charging Flow Isolation Valve or Closure of Primary Makeup Water Control Valve or Stop of Primary Makeup Water Pump
CVCS Malfunction that Increases Reactor Coolant Inventory (Subsection 15.5.2)	Closure of Charging Line Isolation Valve or Charging Line Containment Isolation Valve
Radiological Consequences of a SG Tube Failure (Subsection 15.6.3)	- Manual reactor trip - Isolation of Affected SG - Cooldown of Primary Coolant System by using Main Steam Depressurization Valve - Equilibrium of Pressure between Primary and Secondary Coolant System by using Safety Depressurization Valve - Stop of Injection from ECCS
Control Rod Assembly Ejection Accidents (Subsection 15.4.8)	-Manual C/V Spray System Operation -Manual Annulus Emergency Exhaust System Operation
Failure of Small Lines Carrying Primary Coolant Outside C/V (Subsection 15.6.2)	RCS Sample Lines or CVCS Letdown Line Isolation
Post-LOCA Long Term Cooling (Subsection 15.6.5)	Manual switchover to the simultaneous RV and hot leg injection mode

MHI will provide by the end of February 2008 additional information for operator actions assumed in the Chapter 15 safety analyses that is contained in the ERG, such as the operator action criteria in terms of parameter and values as well as the source of the information (instrument or channel).

NRC QUESTION-2:
NRC requests MHI to confirm that the cases presented in the Chapter 15 safety analyses of the DCD bound all operating conditions (mode and power level).

MHI RESPONSE:
Most of the events evaluated in Chapter 15 assume that the event occurs while the reactor is operating at rated full power. This initial condition is assumed because it ensures a conservatively high initial primary coolant system temperature and minimum initial margin to several fuel design limits, resulting in the most limiting analysis results. However, the rated operating condition is not always the limiting initial condition. Some events are analyzed using initial conditions that are specifically selected to assure a conservative analysis result. An example of this situation is reactivity initiated events, whose transients can be more severe under low power conditions due to the delay of the Doppler feedback effect. Therefore, hot zero power conditions are assumed as the initial conditions for certain reactivity initiated events. Certain reactivity events credit different reactor trip functions for transients initiated at different power levels or reactivity insertion rates, such as the RCCA withdrawal at power. The DCD presents the results of this event as a function of these assumptions so that the limiting combination of assumptions can be identified. Similarly, hot shutdown conditions are considered more severe initial conditions for the main steam line break analysis because the lower coolant temperature results in a limiting reactivity insertion.

Additional information such as a qualitative basis (power level and mode) for the selection of the limiting cases presented in the DCD will be provided by the end of February 2008.

NRC QUESTION-3:
NRC requests a grid stability analysis that justifies the assumed 3-second delay for loss of offsite power.

MHI RESPONSE:
During normal operation, reactor coolant pumps (RCPs) are supplied from the main generator via the generator load break switch (GLBS), unit auxiliary transformer (UAT) and medium voltage bus. The power supply to the RCPs following turbine trip is maintained 3 second (or more) as described below:

If a turbine trip occurs, the GLBS will open after a time delay of at least 15 seconds. During this delay time the generator will be able to provide voltage support to the grid if needed. The reactor coolant pumps will receive power from the grid for at least 3 seconds following the turbine trip. To confirm that the grid will remain stable and the medium voltage buses will remain at the required voltage in the Chapter 15 analysis for a minimum of 3 seconds, a grid stability analysis is provided for as a COLA information item in section 8.2.

If the initiating event is an electrical system failure (such as failure of the isophase bus, generator etc), the main transformer circuit breaker will be opened and the medium voltage bus voltage will continuously maintain via reserve auxiliary transformer (RAT).

If the turbine trip occurs when the grid is not connected (including loss of offsite power), the energy stored in the rotational inertia of the main turbine-generator is used to power-supply the medium voltage buses (including the RCPs).

NRC QUESTION-4:
In Table 6.3-3 of the US-APWR DCD, Response of US APWR to Generic Safety Issues (Sheet 2 of 3), No. 105 and in Table 6.3-4 of the US-APWR DCD, Response of US APWR to Generic Letters and Bulletins (Sheet 1 of 12), No. GL-80-014 of DCD (as shown in page 7 and 8), there is no mention of the RHRS, which is a low pressure system connected to the RCS and of low pressure design. It should be mentioned consistent with SECY-93-087.

MHI RESPONSE:
Since the US-APWR configuration of systems that are connected to the RCS and extended outside the containment is similar to the conventional PWR, the object system is the RHRS.

The RHRS is a low pressure system that is connected to the RCS and extended outside the containment. The RHR system is designed to prevent an interfacing system LOCA by the design rating of 900 lb. The RHR system with 900 lb design rating withstands the full RCS pressure. Additionally, the two motor operated valves located in series between the RCS and the RHR suction line are designed with power lockout capability during normal power operation. Even if both these valves are opened during normal power operation, the RHR system is designed to discharge the RCS inventory to the in-containment RWSP. Therefore, the design of the RHR system is consistent with SECY-93-087.

The design of the RHRS to prevent an interfacing system LOCA is described in subsection 5.4.7 of Chapter 5.

NRC QUESTION-5:
NRC expressed concern that the beyond design basis multiple SG tube rupture event described in SECY-93-087 has not been adequately addressed in the US-APWR DCD.

MHI RESPONSE:
Section II Item R of SECY-93-087 (Apr 2, 1993) and the associated SRM require the consideration and analysis of multiple steam generator tube ruptures for passive PWRs. Since the US-APWR does not have the same passive safety system as the AP600 and AP1000, the US-APWR has no possibility of secondary-to-primary leakage caused by the automatic depressurization system, and the associated secondary water flashing could not disrupt or degrade emergency core coolant injection as described in SECY-93-087.

Section II Item R also recommends that the applicant for a design certification for an evolutionary PWR assess design features to mitigate the amount of containment bypass leakage that could result from [multiple] steam generator tube ruptures. In responding to the SGTR event in the US-APWR, operators isolate the ruptured steam generator, reduce the reactor coolant system pressure by opening the safety depressurization valve, and stop primary-to-secondary leakage in the same manner as operators do in a conventional US operating PWR plant. SRP 15.6.3 acknowledges that multiple steam generator tube ruptures are beyond the design basis, and therefore, the DCD does not present analysis of multiple tube ruptures as part of the SGTR analysis. As part of the detailed review, the response of the US-APWR to a multiple tube rupture can be provided to show the sensitivity of the response to this parameter. This analysis, using best estimate models and assumptions, consistent with other studies of events that are beyond the design basis, will confirm that the recovery strategy for a single tube rupture will also be effective for a multiple steam generator tube rupture.

NRC QUESTION-6:
US-APWR DCD Section 4.3: The NRC requests MHI to provide a graphical or tabular depiction of Gadolinia depletion as a function of burnup.

MHI RESPONSE:
Among Gd isotopes, ^{155}Gd and ^{157}Gd have the largest absorption cross sections and are therefore of primary interest. The depletion of these isotopes as a function of fuel burnup is described below for 4.15wt% fuel assemblies with 6wt% and 10wt% Gadolinia fuel rods, which belong to regions R3GC and R3GB of the US-APWR initial core shown in Figure 4.3-2 of the US-APWR DCD. The burnups of these fuel regions are shown in Table-1 for BOC, MOC and EOC.
Figure-1 shows ^{155}Gd and ^{157}Gd isotopic depletion, where the sum of initial number density of ^{155}Gd and ^{157}Gd of 10wt% Gadolinia fuel is normalized as a unit. ^{155}Gd and ^{157}Gd concentration decrease monotonically with increase of the burnup.
Figure-2 shows k_{inf} (infinite neutron multiplication factor) values as a function of burnup for the fuel assembly types shown in Table-1. The k_{inf} of the fuel regions containing Gadolinia increase monotonically as the Gadolinia depletes, showing a peak in k_{inf} at a burnup of approximately 15 to 20 GWD/MTU. After the peak, the k_{inf} decrease with burnup is similar to that of Uranium fuel without Gadolinia fuel rods. Comparing the burnups of Table-1 with Figure-1 and Figure-2, it can be concluded that ^{155}Gd and ^{157}Gd have been entirely depleted by the EOC of the initial core.

Table-1 Region average burnup for regions containing Gadolinia

Cycle life time (Core average burnup, MWD/MTU)		BOC (0)	MOC (11000)	EOC (23000)
Region average burnup (MWD/MTU)	Region R3GB(1)	0	12200	26700
	Region R3GC(2)	0	14000	28900

(1) Region R3GB: UO2 –only rods: 4.15wt% UO_2; Gd-bearing rods:2.55wt% UO_2 , 10wt% Gd_2O_3
(2) Region R3GC: UO2 –only rods: 4.15wt% UO_2; Gd-bearing rods:2.55wt% UO_2 , 6wt% Gd_2O_3

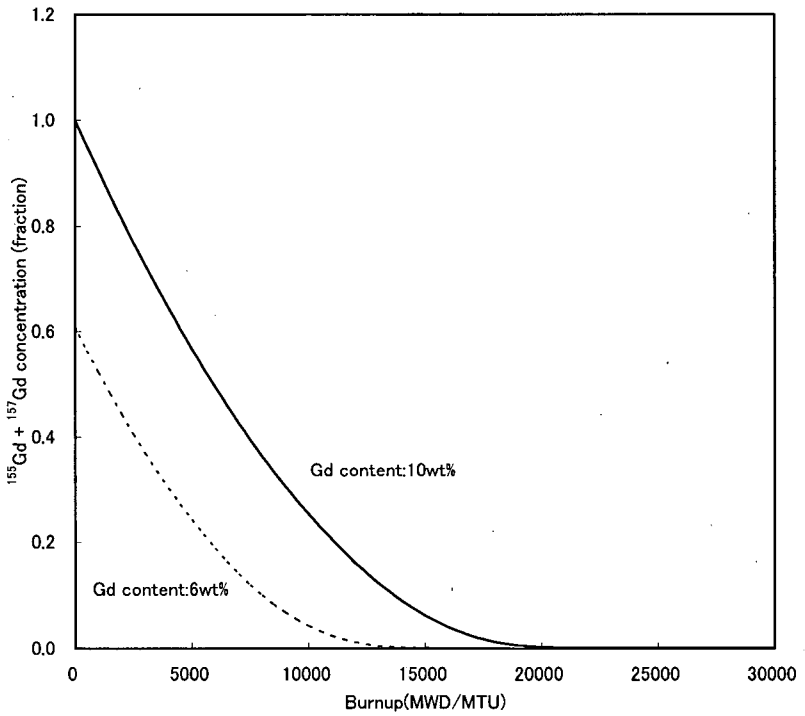

Figure-1 ^{155}Gd and ^{157}Gd isotopic depletion

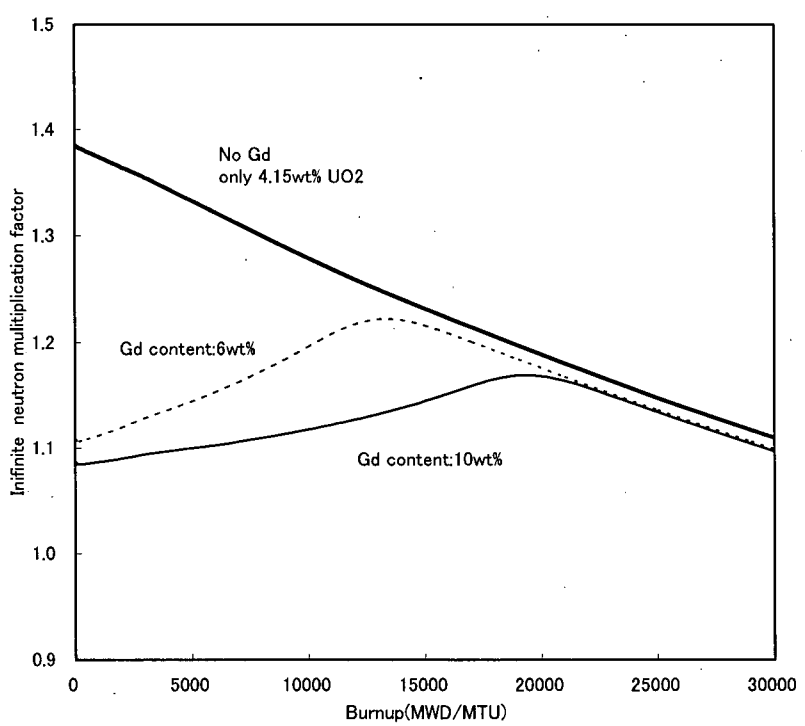

Figure-2 Infinite neutron multiplication factor versus fuel depletion

NRC QUESTION-7:
MUAP-07019-P (R0), Qualification of Nuclear Design Methodology using PARAGON/ANC: The NRC requests MHI to confirm that the codes used in this report have not been modified from the NRC approved methodology.

MHI RESPONSE:
The PARAGON/ANC code system has been used by MHI for the nuclear design of the US-APWR. The methodology of both PARAGON and ANC are described respectively in References (1) and (2), which were both approved by the NRC. MHI has not modified the methodology of PARAGON/ANC for the US-APWR nuclear design. In addition, the applicability of PARAGON/ANC to PWR nuclear design, as already approved by the NRC, is described in References (1) and (3).

References:
(1) Ouisloumen, M. et al., Qualification of the Two-Dimensional Transport Code PARAGON, WCAP-16045-P-A (Proprietary), and WCAP-16045-NP-A (Non-Proprietary), August, 2004.
(2) Liu, Y. S., et al., ANC – A Westinghouse Advanced Nodal Computer Code, WCAP-10965-P-A (Proprietary), and WCAP-10966-A (Non-Proprietary), September, 1986.
(3) Nguyen, T. Q., et al., Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores, WCAP-11596-P-A (Proprietary), and WCAP-11597-A (Non-Proprietary), June, 1988.

NRC QUESTION-8:
MUAP-07020 (R0), Validation of the MHI Criticality Safety Methodology: The NRC requests MHI to confirm the MCNP version of the code used in this report and to provide justification for the NRC approval of the code.

MHI RESPONSE:
The MCNP version used in MUAP-07020 (R0) is the most current version generally available. The version is 5.1.40. (It is also referred to as version 1.40 of MCNP5.) The version number given in MUAP-07020 (R0) is correct.

MCNP and SCALE/KENO are the most commonly used codes for criticality analysis. MCNP utilizes continuous energy cross sections and therefore is free of the approximations needed to make the multi-group cross sections used in SCALE/KENO. The justification for the use of this code is provided in MUAP-07020 (R0) by comparison of analytical results produced by this code to criticality experiments. The mean bias of the 120 experiments analyzed is less than 0.3% in k_{eff}. This report confirms that the 120 experiments covers the range of the applications expected and analyzes for trends in the calculated results as a function of key parameters. No large trends which would challenge the use of MCNP are identified.

Enclosure 2

US-APWR

MHI's Responses to NRC's Questions on Design of the Sump Strainers

February 2008

MHI's Responses to NRC's Questions

on

Design of the Sump Strainers

February 2008

INTRODUCTION

This report documents MHI's responses to the NRC's questions concerning the design of the sump strainers.

BACKGROUND

In the US-APWR DCD, the design of the sump strainers is described in Chapter 6, Subsection 6.2 "Containment Systems (CS)", and Subsection 6.3 "Emergency Core Cooling Systems (ECCS)". The design features of the ECC/CS strainers for the US-APWR as described in the DCD are as follows:

- Passive type, fully submerged strainer system is applied.
- Disc, fin, or cassette-type strainers with a large surface area are installed.
- NPSH evaluation summary is described in Table 6.2.2-1.
- System design parameters are summarized in Table6.3-5. (Surface area, mesh size, etc.)

In addition, Table 6.2.2-2 of the DCD provides a comparison of the ECC/CS strainers design to each of the Regulatory Positions set forth in RG 1.82 Rev.3.

The NRC has requested MHI to provide the detailed design and the evaluation of the ECC/CS strainers for its review. The information requested by the NRC is summarized as follows:

- ECC/CS strainer design specifications
- ECC/CS strainer design compliance with RG 1.82 Rev.3

The NRC has requested the following specific information for the sump strainer for its review:

- Break selection
- Debris generation
- Debris characteristics
- Head loss
- Net positive suction head (NPSH)
- Downstream effects (core, ex-vessel)
- Upstream effects
- Chemical effect

NRC has also requested MHI either to select a specific strainer type for it standard design, i.e., passive disc type, fin-type, or cassette-type, or if the standardized design permits all types of strainer design, to provide the necessary information specified above for each type of strainer design.

MHI' S RESPONSE

In response to NRC requests, MHI commits to provide the additional information described below for the NRC review.

1. Technical report

MHI will provide a technical report that describes the design and the evaluation of the US-APWR ECC/CS strainers. The report will cover the design areas, as described in Table-1 below, which are required to specify the strainer characteristics. The standard US-APWR design will utilize a passive disc layer type of strainer systems and the requested information is provided for this strainer design. MHI will submit the technical report by the end of February, 2008.

Regarding the chemical effect, MHI has so far evaluated the influence of chemical effect on the debris head loss as follows:

- The strainer head loss, defined as "$h_{ECCS/CS\ strainer}$" described in DCD Table 6.2.2-1, resulted from a calculation based on the plant design defined in the US-APWR Design Control Document. The strainer was sized to minimize the expected debris head loss, and to install in the practical space inside containment.
- The strainer head loss consists of a) head loss due to the debris, and b) increasing the head loss due to chemical effect.
- Head loss due to the debris was estimated applying the correlation identified in NUREG-CR6224.
- Increasing head loss due to chemical effect is uncertain, but it was assumed conservatively and considered in the strainer head loss.
- The actual head loss due to the debris including the chemical effect will be verified by hydraulic test, as described in the Section 2 of this paper.

The evaluation of downstream effects will also be discussed in the February 2008 technical report with additional information to be supplied in a subsequent amendment to the report. The date for the submittal of this amendment will be provided in the February technical report.

2) Hydraulic test

MHI will perform confirmatory hydraulic tests that verify the capability of the US-APWR ECC/CS strainer design to meet the NRC's regulatory requirements. These tests will measure the actual debris head loss of the strainer, as well as the influence on the head loss due to chemical effect.

The confirmatory hydraulic tests are planned to be conducted in 2009, because the test facilities of the U.S strainer vender are fully occupied by planned testing for existing PWR plants of their replacement strainers as required by Generic Letter 2004-02.

3) Structural analysis

MHI will provide the structural analysis of the US-APWR ECC/CS strainer design in the same time frame as the US-APWR ASME Class-II Stress Report submittal, September, 2009. This schedule has already been proposed to NRC by the reference letter, "UAP-HF-07170 Application for Design Certification of the US-APWR Standard Design".

The following table summarizes the information to be supplied by MHI

Table-1 ECC/CS strainer design and evaluation areas

Summary of Evaluation	Information Available
1) Description of Strainer - Design features - Specifications	First submittal report (February, 2008)
2) Break Selection - Break selection criteria - Break size and location	First submittal report
3) Debris Generation - Zone of influence (ZOI) - Estimation quantity of debris	First submittal report
4) Debris Characteristics - Insulation, coating, and latent debris - Size, physical properties (densities, etc.) - Debris transport	First submittal report
5) Debris Head loss - Evaluation debris head loss - Thin bed effect	First submittal report (Confirmatory hydraulic test in 2009)
6) Net Positive Suction Head - Water head (submerged level) - Specifications of safety related pumps - Head loss of piping, valves - NPSH margin	First submittal report
7) Downstream Effect - Description downstream components - Core cool ability	First submittal report with additional information to be supplied in the amendment report
8) Upstream Effect - Flow paths upstream the strainer - Ineffective water volume - Submerged water level	First submittal report
9) Chemical Effect - Identify chemical precipitates - Influence debris head loss	First submittal report (Confirmatory hydraulic test in 2009)
10) Structural analysis - Structural analysis result	Provided at same timeframe of US-APWR ASME Class-II Stress Report submittal. (September, 2009)