Response to questions of PAM

The following is the response to staff questions related to RAI 07.05 -18 from e-mail.

RAI 07.05-18; Need clarifications on the MHI response to on following items:

a. Is the comparison of US-APWR PAM variables to NUREG-1431 & RG 1.97 Rev. 3 documented in any of the Technical Reports or DCD?

MHI: The comparison of US-APWR PAM variables to NUREG-1431 & RG 1.97 Rev. 3 is not addressed in the current revision of the Technical Reports or the DCD. However MHI will added the comparison tables to the DCD Section 7.5 after RAI 07.05-18 issues are closed. MHI has the preparation which is the markup of the adding the comparison tables.

b. It appears that US-APWR Functional Restoration Guidelines (FRGs) form the basis for most of the US-APWR PAM variables. Has this FRG been referenced or documented in the DCD?

MHI: PAMs in US-APWR are mainly based on FRGs. The development plan for the FRGs are addressed in the Technical Report “US-APWR Procedure Development Implementation Plan” (MUAP-10010), the FRGs are being developed in accordance with the MUAP-10010 now.

c. Where in the DCD is the compliance to 10CFR50.34(f)(2)xvii demonstrated? It appears that not all of the instrumentation requirements of this regulation are satisfied by the proposed US-APWR PAM variables.

MHI: The compliance to 10CFR50.34(f)(2)xvii is demonstrated in the Response to RAI 07.05-18 as the comparison tables. As described in response to question a., MHI will added the table to DCD Section 7.5. Some variables addressed in 10CFR50.34(f)(2)xvii is considered to be backup variables in the comparison tables, so the variables are not addressed on the PAM list of US-APWR in DCD Section 7.5.

d. Has this proposed list of PAM variables been reviewed and accepted by the respective technical branches of the NRC? Such as, lack of “control rod position indication,” “effluent radioactivity”, “RHR HX outlet Temp”, “accumulator tank valve position”, etc. from the list of PAM variable.

MHI: These variables are addressed in RG1.97 Rev.3 and it is not necessary to assign these variables to PAM of US-APWR in accordance with RG1.97 Rev.4. The basis of the difference between RG1.97 Rev.3 and PAMs of US-APWR is described in comparison tables of Response to RAI 07.05-18.

7.5.2.1 Post Accident Monitoring

The US-APWR PAM list provided in Table 7.5-3 was developed to be in compliance with the guidance of RG 1.97 Rev. 4 (Reference 7.5-1) and IEEE 497-2002 (Reference 7.5-2), which is endorsed by RG 1.97 Rev. 4 (Reference 7.5-1). MHI utilized a combination of previous versions of RG 1.97, Japanese domestic and US operational experience and emergency procedures, and known differences between current operating plants and the US-APWR design to develop a bounding and complete PAM list for the US-APWR. The following subsections describe the selection basis for the variables included in Table 7.5-3.

Table 3 of RG 1.97 Rev. 3 (Reference 7.5-24) prescribes a minimum list of Type B, C, D, and E variables to monitor. However, MHI utilized the performance-based criteria of RG 1.97 Rev. 4 (Reference 7.5-1) and IEEE 497-2002 (Reference 7.5-2) to select the Type B, C, D, and E accident monitoring variables for the US-APWR. Therefore, there are some differences between the RG 1.97 Rev. 3 (Reference 7.5-24) and MHI variable lists for these variable types. Additionally, Type A variables were not included in RG 1.97 Rev. 3 (Reference 7.5-24), so a slightly different methodology was utilized to select the bounding list of Type A variables for the US APWR. A discussion of the variable selection basis for each type of PAM variable is described below. The specific basis for the inclusion or exclusion of a specific variable in Table 7.5-3 is provided in Tables 7.5-11 through 7.5-15 for each variable classification type.

Type A Variables

NUREG-1431 Table 3.3.3-1 (Reference 7.5-23) provides a minimal list of Category 1 variables (any Type) for a typical Westinghouse NSSS plant based on the guidance in RG 1.97 Rev. 3 (Reference 7.5-24). MHI utilized this list as an initial starting point for the US-APWR Type A PAM list. Then MHI utilized the performance-based criteria of RG 1.97 Rev. 4 (Reference 7.5-1) and IEEE 497-2002 (Reference 7.5-2) to select the specific Type A accident monitoring variables for the US-APWR. IEEE 497-2002 (Reference 7.5-2) defines Type A variables as follows.

> Type A variables are those variables that provide the primary information required to permit the control room operating staff to:
>
> a) Take specific planned manually-controlled actions for which no automatic control is provided and that are required for safety systems to perform their safety-related functions as assumed in the plant Accident Analysis Licensing Basis.
>
> b) Take specific planned manually-controlled actions for which no automatic control is provided and that are required to mitigate the consequences of an AOO.

The SGTR is the only event that assumes planned operator actions using the Type A variables listed in Table 7.5-3. Planned operator actions required for other events are initiated by an alarm or they are based on a time limit.

In the event of an SGTR, the DBA analysis in Subsection 15.6.3 assumes the following specific operator actions:

- Identify and Isolate Ruptured SG
- Cool Down Primary Coolant System
- Depressurize Primary Coolant System to Equalize Pressure between Primary and Secondary
- Terminate Safety Injection Flow

Some Type A variables are monitored before the operator takes the above manual actions. These Type A variables are shown in Table 7.5-11.

Regarding the LOCA event, RWSP level is an important indication in some currently operating plants because operator action is needed to realign the injection of ECCS from the RWSP to the containment sump before the RWSP becomes empty. In the US-APWR, the RWSP is located at the bottom of the containment and the suction of both the SIP and CS/RHRP is the RWSP from the beginning. Therefore, it is not necessary to confirm the RWSP level during the LOCA event and this variable is not included as a Type A variable for the US-APWR.

The analyses of the Steam Line Break (SLB) in Subsection 15.1.5 and Feedwater Line Break (FLB) in Subsection 15.2.8 assume EFW isolation from a faulted SG. However, this action is performed automatically by the low main steam line pressure signal EFW isolation function. Therefore, there are no PAM instruments related to operator actions assumed in the SLB and FLB analyses.

In all DBA analysis, except for the SGTR previously discussed, explicit operator actions are not assumed based on primary information from PAM instruments. However, SI termination and long-term core cooling from secondary heat sink are necessary to bring the plant to cold shut down conditions. Operator actions for SI termination and core cooling are already included in the operator actions assumed in the SGTR analysis. Therefore, the instruments associated with these functions have already been included in the bounding PAM list provided in DCD Table 7.5-3.

Table 7.5-11 compares all of the Category 1 variables (any Type) functions in NUREG-1431 (Reference 7.5-23) Table 3.3.3-1 to the US APWR Type A variables currently listed in Table 7.5-3 and summarizes the bases for differences between the Type A variables in the MHI PAM list and the Category 1 PAM for a typical Westinghouse 4 loop PWR plant. The above described methodology serves as the basis for the selection of the US-APWR Type A PAM variables included in Table 7.5-3.

Type B Variables

IEEE 497-2002 (Reference 7.5-2) defines Type B variables as follows.

> Type B variables are those variables that provide primary information to the control room operators to assess the plant critical safety functions. Any plant critical safety functions addressed in the EPGs or the plant specific EOPs that are in addition to those identified above shall also be included.

The ultimate goal of the plant safety systems is to prevent an uncontrolled release of radioactive material. This is accomplished by ensuring that certain parameters related to plant critical safety functions are not exceeded. The US-APWR Functional Restoration Guidelines (FRGs) provide protection of these plant critical safety functions. The FRGs establish predefined function-related restoration strategies for responding to emergency transients where the initiating event is unknown and the transient is not predefined. The restoration strategies utilize available plant equipment to restore the parameters used for entry conditions to values sufficient to ensure protection of the plant critical safety function.

The most essential and important methods of protecting the plant critical safety functions are the concepts of (1) Shutdown, (2) Cooldown, and (3) Contain, where each of these concepts is defined as follows.

- “Shutdown” means that the plant should be subcritical in order to reduce the thermal energy in the core to as low as the decay heat level during the emergency conditions.

- “Cooldown” means that the heat should be removed from the core (fuel rods) to protect the integrity of the cladding. Decay heat should be removed from the Reactor Coolant System (RCS).

- “Contain” refers to the integrity of the RCS and containment vessel. Heat should be removed from the containment to the ultimate heat sink.

The bounding US-APWR Type-B PAM variables are selected from the concept of the FRGs

described above. The Type B functional category of “Reactivity Control” is related to the FRG concept of “Shutdown”. The functional categories of “Core Cooling” and “Reactor Coolant System Integrity” are related to the FRG concept of “Cooldown”. And the Type B functional category “Containment Integrity” is related to the FRG concept “Contain”.
Table 7.5-12 describes the bases for the differences between the Type B variables included in the MHI PAM list compared to those included in RG 1.97 Rev. 3 (Reference 7.5-24) Table 3.

Type C Variables
IEEE 497-2002 (Reference 7.5-2) defines Type C variables as follows.

> Type C variables are those variables that provide primary information to the control room operators to indicate the potential for breach or the actual breach of the three fission product barriers (extended range): fuel cladding, reactor coolant system pressure boundary, and containment pressure boundary.

Table 7.5-13 describes the bases for the differences between the Type C variables included in the MHI PAM list compared to those included in RG 1.97 Rev. 3 (Reference 7.5-24) Table 3.

Type D Variables
IEEE 497-2002 (Reference 7.5-2) defines Type D variables as follows.

> Type D variables are those variables that provide primary information to the control room operators and are required in procedures and LBD to:
> a) Indicate the performance of those safety systems and auxiliary supporting features necessary for the mitigation of design basis events.
> b) Indicate the performance of other systems necessary to achieve and maintain a safe shutdown condition.
> c) Verify safety system status.

The US-APWR Type D variable list is almost identical to the Type D variables included in Table 3 of RG 1.97 Rev.3 (Reference 7.5-24). One notable departure is the variable to monitor flow in the low pressure injection system. The accumulators and high head safety injection system in US-APWR are designed to replace the entire low head safety injection function; therefore, this system is not part of the US-APWR design and this monitoring variable is not applicable to the US-APWR.
Another notable departure from the RG 1.97 Rev.3 (Reference 7.5-24) Type D variable list involves the chemical volume and control system (CVCS). The high head injection system and emergency letdown system of the US APWR has a required safety function to ensure a means for feed and bleed for boration and make up water for compensation of shrinkage if the normal CVCS is unavailable. Since the US-APWR SI system performs the necessary RCS inventory and boration functions, the CVCS-related monitoring variables are not necessary for the US-APWR design and thus not included in the MHI Type D variable list.
Table 7.5-14 describes the bases for the differences between the Type D variables included in the MHI PAM list compared to those included in RG 1.97 Rev. 3 (Reference 7.5-24) Table 3.

Type E Variables
IEEE 497-2002 (Reference 7.5-2) defines Type E variables as follows.

> Type E variables are those variables that provide primary information to the control room operators and are required for use in determining the magnitude of the release of radioactive materials and continually assessing such releases.
> The selection of these variables shall include, but not be limited to, the following:
> a) Monitor the magnitude of releases of radioactive materials through identified pathways (e.g., secondary safety valves, and condenser air ejector).
> b) Monitor the environmental conditions used to determine the impact of releases of radioactive materials through identified pathways (e.g., wind speed, wind direction, and air temperature).
> c) Monitor radiation levels and radioactivity in the plant environs.

d) Monitor radiation levels and radioactivity in the control room and selected plant areas where access may be required for plant recovery.

Table 7.5-15 describes the bases for the differences between the Type E variables included in the MHI PAM list compared to those included in RG 1.97 Rev. 3 (Reference 7.5-24) Table 3.

Table 7.5-11: Basis for Differences between NUREG-1431 Table 3.3.3-1 and the MHI Type A PAM List

RG 1.97 Function	Purpose	NUREG-1431 Table 3.3.3-1 Variable	Corresponding MHI Type A PAM Variable	Basis for Difference
Reactivity Control	Indication of subcritical conditions	Power Range Neutron Flux	-	This parameter is not applied in the safety analysis. Wide Range Neutron Flux is a Type B and D variable for the US-APWR.
Reactivity Control	Indication of subcritical conditions	Source Range Neutron Flux	-	This parameter is not applied in the safety analysis.
Core Cooling	Indication of core cooling; Manual action; Long-term core cooling	RCS Hot Leg Temperature	Reactor Coolant Hot Leg Temperature (Wide Range)	Intact loop hot leg temperature is applied for determining the termination of RCS cooldown and initiation of RCS depressurization in the SGTR analysis. Therefore, this is a Type A variable for the US-APWR.
Core Cooling	Indication of core cooling; Long-term core cooling	RCS Cold Leg Temperature	Reactor Coolant Cold Leg Temperature (Wide Range)	This parameter is not explicitly assumed in safety analysis; however, monitoring of this parameter is necessary for cooling down after mitigating a PA or AOO. Therefore, this is a Type A parameter for the US-APWR.
Core Cooling; Maintaining RCS Integrity; RCS Pressure Boundary; Primary Coolant System	-SGTR Safety Analysis Manual Action -RCS Depressurization based on EOPs for SGTR event	RCS Pressure (Wide Range)	Reactor Coolant Pressure	No difference.
Core Cooling	To ensure RCS inventory	Reactor Vessel Water Level	-	This parameter is not applied in the safety analysis. RV Water Level is a Type B and D variable for the US-APWR.
Core cooling; Maintaining RCS Integrity; RCS Pressure Boundary	Indication of core cooling function for RWSP switchover and status of ECCS recirculation delivery	Containment Sump Water Level (Wide Range)	-	This parameter is not applied in safety analysis since the US-APWR RWSP is located inside containment and does not require switchover to the recirculation sump. RWSP level is a Type B and D variable for the US-APWR.

Table 7.5-11: Basis for Differences between NUREG-1431 Table 3.3.3-1 and the MHI Type A PAM List

RG 1.97 Function	Purpose	NUREG-1431 Table 3.3.3-1 Variable	Corresponding MHI Type A PAM Variable	Basis for Difference
Maintaining Containment and RCS Integrity; RCS Pressure Boundary	Indication of containment integrity function	Containment Pressure	-	This parameter is not applied in the safety analysis. Containment Pressure is a Type B and D variable for the US-APWR.
Containment Isolation/Integrity	Indication of containment integrity function	Penetration Flow Path Containment Isolation Valve Position	-	This parameter is not applied in the safety analysis. C/V Isolation Valve Position is a Type B and D variable for the US-APWR.
Containment Radiation; RCS Pressure Boundary	Identify challenge to fission product barrier	Containment Area Radiation (High Range)	-	This parameter is not applied in the safety analysis. Containment Area Radiation is a Type C and E variable for the US-APWR.
Primary Coolant System; RCS Pressure Boundary	To ensure proper operation of the pressurizer	Pressurizer Level	Pressurizer Water Level	No difference. This is a Type A variable for the US-APWR.
Secondary System; RCS Pressure Boundary	Verification of heat sink availability	Steam Generator Water Level (Wide Range)	-	This parameter is not applied in the safety analysis. SG narrow range level is applied in safety analysis and US-APWR ERG instead of this parameter. SG Wide Range Level is a Type B and D variable for the US-APWR.
Auxiliary Feedwater System	Indication of ability to maintain SG heat sink and indication of long-term AFW operation	Condensate Storage Tank Level	-	The EFW pit has enough water to maintain long-term core cooling; therefore, this variable is not applied in the safety analysis. This is a Type B and D variable for the US-APWR.
Core Cooling; Fuel Cladding Integrity; Maintain RCS Integrity; RCS Pressure Boundary; Primary Coolant System	Indication of core cooling	Core Exit Temperature – Quadrant [1]-[4]	-	This parameter is not applied in the safety analysis. Core Exit Temperature is a Type B and C variable for the US-APWR.

Table 7.5-11: Basis for Differences between NUREG-1431 Table 3.3.3-1 and the MHI Type A PAM List

RG 1.97 Function	Purpose	NUREG-1431 Table 3.3.3-1 Variable	Corresponding MHI Type A PAM Variable	Basis for Difference
Auxiliary Feedwater System	Verification of automatic actuation and ability to satisfy heat sink requirements	Auxiliary Feedwater Flow	EFW Flow	No difference. This parameter is used to determine if the ECCS termination criteria are met in the SGTR analysis. EFW Flow is a Type A parameter for the US-APWR.
Secondary System	Verification of manual action for SGTR termination (along w/ RCS Pressure)	-	Main Steam Line Pressure	This parameter is applied for determining the termination of RCS cooldown and initiation of RCS depressurization in the SGTR analysis. Therefore, this is a Type A variable for the US-APWR.
Secondary System; RCS Pressure Boundary	Verification of heat sink availability	-	SG Water Level (Narrow Range)	This parameter is monitored for the operator to determine if the ECCS termination criteria are met in the SGTR analysis. This parameter is also used in the ERGs to identify ruptured SG(s). Therefore, this is a Type A variable for the US-APWR.
Core Cooling	Indication of core cooling	-	Degrees of Subcooling	This parameter is monitored for the operator to determine if the terminating RCS depressurization criteria or ECCS termination criteria are met in the SGTR analysis. Therefore, this is a Type A variable for the US-APWR.

Table 7.5-12: Basis for Type B Differences between RG 1.97 Rev.3 and the MHI PAM List

RG 1.97 Rev. 3 Variable	Purpose	MHI PAM Variable	Basis for Difference
Reactivity Control			
Neutron Flux	Function detection; accomplishment of mitigation	Wide Range Neutron Flux	No difference.
Control Rod Position	Verification	-	Reactivity control is directly monitored by Neutron Flux. Control rod position provides back-up indication of reactor shutdown. Since the primary indication is neutron flux, which is a PAM variable, control rod indication is not included in the US-APWR PAM list.
RCS Soluble Boron Concentration	Verification	Reactor Coolant Soluble Boron Concentration	No difference.
RCS Cold Leg Water Temperature	Verification	Reactor Coolant Cold Leg Temperature (Wide Range)	No difference.
Core Cooling			
RCS Hot Leg Water Temperature	Function detection; accomplishment of mitigation; verification; long-term surveillance	Reactor Coolant Hot Leg Temperature (Wide Range)	No difference.
RCS Cold Leg Water Temperature	Function detection; accomplishment of mitigation; verification; long-term surveillance	Reactor Coolant Cold Leg Temperature (Wide Range)	No difference.
RCS Pressure	Function detection; accomplishment of mitigation; verification; long-term surveillance	Reactor Coolant Pressure	No difference.
Core Exit Temperature	Verification	Core Exit Temperature	No difference.
Coolant Inventory	Verification; accomplishment of mitigation	RV Water Level	Reactor vessel water level is a key indication of adequate inventory for core cooling. There is no difference in the intent of these two variables.

Table 7.5-12: Basis for Type B Differences between RG 1.97 Rev.3 and the MHI PAM List

RG 1.97 Rev. 3 Variable	Purpose	MHI PAM Variable	Basis for Difference
Degrees of Subcooling	Verification and analysis of plant conditions	Degrees of Subcooling	No difference.
Maintaining Reactor Coolant System Integrity			
RCS Pressure	Function detection; accomplishment of mitigation	Reactor Coolant Pressure	No difference.
Containment Sump Water Level	Function detection; accomplishment of mitigation; verification	Refueling Water Storage Pit Water Level (Wide Range) Refueling Water Storage Pit Water Level (Narrow Range)	The US-APWR RWSP is located inside containment, essentially combining the function of the sump and RWSP. Therefore, RWSP water level meets the intent of this monitoring variable and there is no difference between RG 1.97 Rev 3 and the US-APWR PAM list.
Containment Pressure	Function detection; accomplishment of mitigation; verification	Containment Pressure	No difference.
Maintaining Containment Integrity			
Containment Isolation Valve Position (excluding check valves)	Accomplishment of isolation	Containment Isolation Valve Position (Excluding Check Valves)	No difference.
Containment Pressure	Function detection; accomplishment of mitigation; verification	Containment Pressure	No difference.
Other			
-	-	Pressurizer Water Level	This parameter is important to monitor because it is related to the SI termination criteria, which is related to maintaining adequate RCS inventory to assure core cooling.
-	-	Main Steam Line Pressure	This parameter is important to monitor the efficiency of removing the decay heat of core, which is related to core cooling.
-	-	SG Water Level (Wide Range)	This parameter provides indication of heat sink availability and is selected to monitor core cooling.

Table 7.5-12: Basis for Type B Differences between RG 1.97 Rev.3 and the MHI PAM List

RG 1.97 Rev. 3 Variable	Purpose	MHI PAM Variable	Basis for Difference
-	-	SG Water Level (Narrow Range)	This parameter provides indication of heat sink availability and is selected to monitor core cooling.
-	-	EFW Flow	This parameter provides verification of the automatic actuation of EFW and is selected to monitor core cooling.
-	-	EFW Pit Water Level	This parameter provides indication of heat sink availability and is selected to monitor core cooling.

Table 7.5-13: Basis for Type C Differences between RG 1.97 Rev.3 and the MHI PAM List

RG 1.97 Rev. 3 Variable	Purpose	MHI PAM Variable	Basis for Difference
Fuel Cladding			
Radioactivity Concentration or Radiation Level in Circulating Primary Coolant	Detection of breach	Radioactivity Concentration or Radiation Level in Circulating Primary Coolant	No difference.
Core Exit Temperature	Detection of breach	Core Exit Temperature	No difference.
Analysis of Primary Coolant (Gamma Spectrum)	Detail analysis; accomplishment of mitigation; verification; long-term surveillance	-	Concentration of each radioactive nuclide can be derived from RCS sampling.
Reactor Coolant Pressure Boundary			
RCS Pressure	Detection of potential for or actual breach; accomplishment of mitigation; long-term surveillance	Reactor Coolant Pressure	No difference.
Containment Pressure	Detection of breach; accomplishment of mitigation; long-term surveillance	Containment Pressure	No difference.
Containment Sump Water Level	Detection of breach; accomplishment of mitigation; long-term surveillance	-	Containment Pressure is a more direct indication of a potential containment breach. Therefore, RWSP level is not included as a Type C variable for the US-APWR.
Containment Area Radiation	Detection of breach; verification	Containment High Range Area Radiation	No difference.
Effluent Radioactivity - Noble Gas Effluent from Condenser Air Removal System Exhaust	Detection of breach; verification	-	Coolant leakage outside containment to secondary system due to an actual breach of the reactor coolant pressure boundary can be detected by RCS pressure, SG water level, and pressurizer water level. These variables are PAM variables. Therefore, it is not necessary to include effluent radioactivity as a Type C variable.

Table 7.5-13: Basis for Type C Differences between RG 1.97 Rev.3 and the MHI PAM List

RG 1.97 Rev. 3 Variable	Purpose	MHI PAM Variable	Basis for Difference
Containment			
RCS Pressure	Detection of potential for breach; accomplishment of mitigation	Reactor Coolant Pressure	No difference.
Containment Hydrogen Concentration	Detection of potential for breach; accomplishment of mitigation; long-term surveillance	-	This instrumentation is used for monitoring severe accidents. Therefore, it does not need to be a Type C variable.
Containment Pressure	Detection of potential for or actual breach; accomplishment of mitigation	Containment Pressure	No difference.
Containment Effluent Radioactivity - Noble Gas Effluent from Identified Release Points	Detection of breach; accomplishment of mitigation; verification	-	The plant vent receives the discharge from the containment purge, auxiliary building, control building, fuel building, and the condenser air removal filtration system. This variable can be measured by plant vent radiation monitor (including high range) and therefore is not included as a separate Type C variable for the US-APWR.
Effluent Radioactivity - Noble Gases (from buildings or areas where penetrations and hatches are located, e.g., secondary containment and auxiliary buildings and fuel handling buildings that are in direct contact with primary containment)	Indication of breach	-	The plant vent receives the discharge from the containment purge, auxiliary building, control building, fuel building, and the condenser air removal filtration system. This variable can be measured by plant vent radiation monitor (including high range) and therefore is not included as a separate Type C variable for the US-APWR.

Table 7.5-14: Basis for Type D Differences between RG 1.97 Rev.3 and the MHI PAM List

RG 1.97 Rev. 3 Variable	Purpose	MHI PAM Variable	Basis for Difference
Residual Heat Removal (RHR) or Decay Heat Removal System			
RHR System Flow	To monitor operation	CS/RHR Pump Discharge Flow CS/RHR Pump Minimum Flow	No difference.
RHR Heat Exchanger Outlet Temperature	To monitor operation and for analysis	-	Proper operation of the RHR system is verified by CS/RHR flow rate. Additionally, T_{hot} and T_{cold} are available to monitor RHR system performance with respect to decay heat removal. Therefore, it is not necessary to include the RHR heat exchanger outlet temperature as a Type D variable in the US-APWR PAM list.
Safety Injection System			
Accumulator Tank Level and Pressure	To monitor operation	Accumulator Water Level, Accumulator Pressure	No difference.
Accumulator Isolation Valve Position	Operation status	-	Accumulator water level and accumulator pressure are available to monitor operation status. Therefore, it is not necessary to include isolation valve position as a separate Type D variable in the US-APWR PAM list.
Boric Acid Charging Flow	To monitor operation	-	The safety injection system delivers boric acid water to the RCS in the US-APWR. Safety Injection Pump Discharge Flow and Safety Injection Pump Minimum Flow are available to monitor the flow. Therefore it is not necessary to include this as a Type D variable in the US-APWR PAM list.
Flow in HPI System	To monitor operation	Safety Injection Pump Discharge Flow Safety Injection Pump Minimum Flow	No difference.
Flow in LPI System	To monitor operation	-	The US-APWR design allows the accumulators and high head safety injection system to fully replace the safety function associated with the low head safety injection system. Therefore, the MHI PAM list does not need any variables to indicate LPI system performance.

Table 7.5-14: Basis for Type D Differences between RG 1.97 Rev.3 and the MHI PAM List

RG 1.97 Rev. 3 Variable	Purpose	MHI PAM Variable	Basis for Difference
Refueling Water Storage Tank Level	To monitor operation	Refueling Water Storage Pit Water Level (Wide Range) Refueling Water Storage Pit Water Level (Narrow Range)	No difference.
Primary Coolant System			
Reactor Coolant Pump Status	To monitor operation	-	The safety analysis does not rely on the RCP to mitigate design basis events. The RCPs are also not necessary to achieve and maintain a safe shutdown condition. CCW header pressure is available to monitor CCW performance related to its function to deliver seal flow to the RCP in order to maintain its RCS pressure boundary function. Therefore, RCP status is not included as a PAM variable for the US-APWR.
Primary System Safety Relief Valve Positions (including PORV and code valves) or Flow Through or Pressure in Relief Valve Lines	Operation status; to monitor for loss of coolant	-	RCS pressure, Reactor Coolant Hot Leg Temperature, and Reactor Coolant Cold Leg Temperature are available to monitor operation status of the primary coolant system. Consistent trends in changes to the values of these variables are indicative of a loss of coolant. Therefore, it is not necessary to include position indication or flow indication for the primary relief valves in the PAM list.
Pressurizer Level	To ensure proper operation of pressure	Pressurizer Water Level	No difference.
Pressurizer Heater Status	To determine operating status	-	Pressurizer water level and RCS pressure are indicative of the performance of the pressurizer heater. Therefore it is not necessary to separately include heater status in the PAM list.
Quench Tank Level	To monitor operation	-	This component is not necessary to mitigate design basis events, and not necessary to achieve and maintain a safe shutdown condition. Therefore, it is not included in the US-APWR PAM list.
Quench Tank Temperature	To monitor operation	-	Same as above.

Table 7.5-14: Basis for Type D Differences between RG 1.97 Rev.3 and the MHI PAM List

RG 1.97 Rev. 3 Variable	Purpose	MHI PAM Variable	Basis for Difference
Quench Tank Pressure	To monitor operation	-	Same as above.
Secondary System (Steam Generator)			
Steam Generator Level	To monitor operation	SG Water Level (Wide Range), SG Water Level (Narrow Range)	No difference.
Steam Generator Pressure	To monitor operation	Main Steam Line Pressure	No difference.
Safety/Relief Valve Positions or Main Steam Flow	To monitor operation	-	Main steam line pressure is indicative of main steam flow and is available to monitor its SG operation. Therefore it is not necessary to separately include this variable in the PAM list.
Main Feedwater Flow	To monitor operation	-	SG water level and main steam line pressure are indicative of adequate feedwater flow. Since these variables are available to monitor SG operation, it is not necessary to separately include MFW flow in the PAM list.
Auxiliary Feedwater or Emergency Feedwater System			
Auxiliary or Emergency Feedwater Flow	To monitor operation	EFW Flow	No difference.
Condensate Storage Tank Water Level	To ensure water supply for auxiliary feedwater	EFW Pit Water Level	No difference.
Containment Cooling Systems			
Containment Spray Flow	To monitor operation	CS/RHR Pump Discharge Flow CS/RHR Pump Minimum Flow	No difference.
Heat Removal by the Containment Fan Heat Removal System	To indicate accomplishment of cooling	-	The containment fan heat removal system is not credited in design basis events since containment spray is credited to maintain containment integrity. Therefore this variable is not included in the PAM list.

Table 7.5-14: Basis for Type D Differences between RG 1.97 Rev.3 and the MHI PAM List

RG 1.97 Rev. 3 Variable	Purpose	MHI PAM Variable	Basis for Difference
Containment Atmosphere Temperature	To monitor operation	Containment Temperature	No difference.
Containment Sump Water Temperature	To monitor operation	-	Containment pressure, containment temperature, and containment spray flow are utilized to monitor containment cooling system performance. Therefore it is not necessary to include this variable in the US-APWR PAM list.
Chemical and Volume Control System (CVCS)			
Makeup Flow - In	To monitor operation	-	Since RCS inventory and boration are achieved by the safety injection system in the US-APWR, the monitoring variables related to CVCS are not necessary PAM variables for the US-APWR design.
Letdown Flow - Out	To monitor operation	-	Same as above.
Volume Control Tank Level	To monitor operation	-	Same as above.
Cooling Water System (CCW)			
Component Cooling Water Temperature to ESF System	To monitor operation	-	CCW header pressure provides indication of the performance of the cooling water system. Therefore it is not necessary to separately include this variable in the PAM list.
Component Cooling Water Flow to ESF System	To monitor operation	-	Same as above.
Radwaste Systems			
High-Level Radioactive Liquid Tank Level	To indicate storage volume	-	The US-APWR design precludes the need for this variable. This component is not necessary to mitigate design basis events and not necessary to achieve and maintain a safe shutdown condition. Addition of additional radioactive waste to the liquid or gaseous radwaste system following an accident is precluded by design and is not postulated. Therefore, this variable is not included in the US-APWR PAM list.
Radioactive Gas Holdup Tank Pressure	To indicate storage capacity	-	Same as above.

Table 7.5-14: Basis for Type D Differences between RG 1.97 Rev.3 and the MHI PAM List

RG 1.97 Rev. 3 Variable	Purpose	MHI PAM Variable	Basis for Difference
Ventilation Systems			
Emergency Ventilation Damper Position	To indicate damper status	-	Containment Isolation Valve Position provides indication of containment integrity. The combination of isolation valve position status and a lack of radioactive release as indicated by the plant vent monitor provides verification of proper automatic ventilation path isolation. Therefore, damper position indication is not included in the US-APWR PAM list.
Power Supplies			
Status of Standby Power and Other Energy Sources Important to Safety (electric, hydraulic, pneumatic) (voltages, currents, pressures)	To indicate system status	Status of Standby Power and Other Energy Sources Important to Safety Class 1E ac Bus Voltage Class 1E dc Bus Voltage	No difference.
Other			
-	-	Reactor Coolant Hot Leg Temperature (Wide Range)	This variable indicates the performance of the primary coolant system for maintaining core cooling.
-	-	Reactor Coolant Cold Leg Temperature (Wide Range)	Same as above.
-	-	Reactor Coolant Pressure	This variable indicates the performance of the primary coolant system for maintaining core cooling and RCS integrity.
-	-	Degrees of Subcooling	This variable is used to indicate the performance of the primary coolant system for core cooling.
-	-	RV Water Level	This variable provides direct indication of inventory available for maintaining core cooling.
-	-	Wide Range Neutron Flux	This variable directly indicates reactivity control and allows for the monitoring of the performance of the control rod assemblies.
-	-	Containment Pressure	This variable is used to indicate the containment integrity status.

Table 7.5-14: Basis for Type D Differences between RG 1.97 Rev.3 and the MHI PAM List

RG 1.97 Rev. 3 Variable	Purpose	MHI PAM Variable	Basis for Difference
-	-	Containment Isolation Valve Position (Excluding Check Valves)	This variable is used to indicate the containment integrity status.
-	-	CCW Header Pressure	This variable is used to indicate the performance of the CCW system.
-	-	ESW Header Pressure	This variable is used to indicate the performance of the ESW system.

Table 7.5-15: Basis for Type E Differences between RG 1.97 Rev.3 and the MHI PAM List

RG 1.97 Rev. 3 Variable	Purpose	MHI PAM Variable	Basis for Difference
Containment Radiation			
Containment Area Radiation - High Range	Detection of significant releases; release assessment; long-term surveillance; emergency plan actuation	Containment High Range Area Radiation	No difference.
Area Radiation			
Radiation Exposure Rate (inside buildings or areas where access is required to service equipment important to safety)	Detection of significant releases; release assessment; long-term surveillance	-	This parameter can be measured by area monitors located where personnel enter areas after the accident. Additional personnel protection will be provided by the use of portable radiation monitors and air sampling. Therefore, it is not necessary to include this variable in the US-APWR PAM list.
Airborne Radioactive Materials Released from Plant			
Noble Gases and Vent Flow Rate			
Containment or Purge Effluent	Detection of significant releases; release assessment	-	The plant vent receives the discharge from the containment purge, auxiliary building, control building, fuel building, and the condenser air removal filtration system. These variables can be measured by plant vent radiation monitor (including high range) and therefore are not included as separate Type E variables for the US-APWR.
Reactor Shield Building (if in design)	Detection of significant releases; release assessment	-	
Auxiliary Building (including any building containing primary system gases, e.g., waste gas decay tank)	Detection of significant releases; release assessment; long-term surveillance	-	
Condenser Air Removal System Exhaust	Detection of significant releases; release assessment	-	

Table 7.5-15: Basis for Type E Differences between RG 1.97 Rev.3 and the MHI PAM List

RG 1.97 Rev. 3 Variable	Purpose	MHI PAM Variable	Basis for Difference
Common Plant Vent or Multipurpose Vent Discharging Any of Above Releases (if containment purge is included)	Detection of significant releases; release assessment; long-term surveillance	-	This variable can be measured by plant vent radiation monitor (including high range) and therefore is not included as a separate Type E variable for the US-APWR.
Vent From Steam Generator Safety Relief Valves or Atmospheric Dump Valves	Detection of significant releases; release assessment	-	This variable is measured by main steam line monitor. Therefore it is not included as a separate Type E variable for the US-APWR.
All Other Identified Release Points	Detection of significant releases; release assessment; long-term surveillance	-	This variable can be measured by plant vent radiation monitor (including high range) and therefore is not included as a separate Type E variable for the US-APWR.
Particulates and Halogens			
All Identified Plant Release Points (except steam generator safety relief valves or atmospheric steam dump valves and condenser air removal system exhaust). Sampling with Onsite Analysis Capability	Detection of significant releases; release assessment; long-term surveillance	-	This can be measured by plant vent sampler (accident sampler). Therefore it is not included as a separate Type E variable for the US-APWR.
Environs Radiation and Radioactivity			
Airborne Radiohalogens and Particulates (portable sampling with onsite analysis capability)	Release assessment; analysis	Airborne Radio Halogens and Particulates (Portable Sampling with Onsite Analysis Capability)	No difference.

Table 7.5-15: Basis for Type E Differences between RG 1.97 Rev.3 and the MHI PAM List

RG 1.97 Rev. 3 Variable	Purpose	MHI PAM Variable	Basis for Difference
Plant and Environs Radiation (portable instrumentation)	Release assessment; analysis	Plant and Environs Radiation (Portable Instrumentation)	No difference.
Plant and Environs Radioactivity (portable instrumentation)	Release assessment; analysis	Plant and Environs Radioactivity (Portable Instrumentation)	No difference.
Meteorology			
Wind Direction	Release assessment	Meteorological Parameters (Wind Direction, Wind Speed, Estimation of Atmospheric Stability)	No difference.
Wind Speed	Release assessment	Meteorological Parameters (Wind Direction, Wind Speed, Estimation of Atmospheric Stability)	No difference.
Estimation of Atmospheric Stability	Release assessment	Meteorological Parameters (Wind Direction, Wind Speed, Estimation of Atmospheric Stability)	No difference.
Accident Sampling Capability (Analysis Capability On Site)			
Primary Coolant and Sump ・Gross Activity ・Gamma Spectrum ・Boron Content ・Chloride Content ・Dissolved Hydrogen or Total Gas ・Dissolved Oxygen ・pH	Release assessment; verification analysis	-	These parameters can be measured by sampling. Many operating plants have received NRC approval for eliminating the PASS requirements specified in RG 1.97 Rev. 3. Therefore, these parameters are also not included in the US-APWR Type E PAM list.

Table 7.5-15: Basis for Type E Differences between RG 1.97 Rev.3 and the MHI PAM List

RG 1.97 Rev. 3 Variable	Purpose	MHI PAM Variable	Basis for Difference
Containment Air ・Hydrogen Content ・Oxygen Content ・Gamma Spectrum	Release assessment; verification analysis	-	These parameters can be measured by sampling. Many operating plants have received NRC approval for eliminating the PASS requirements specified in RG 1.97 Rev. 3. Therefore, these parameters are also not included in the US-APWR Type E PAM list.
Other			
-	-	MCR Area Radiation	To monitor radiation and radioactivity levels in the control room.
-	-	MCR Outside Air Intake Radiation	To monitor radiation and radioactivity levels in the control room.
-	-	TSC Outside Air Intake Radiation	To monitor radiation and radioactivity levels in the technical support center.
-	-	Plant Vent Radiation Gas Radiation (Including High Range)	To monitor the magnitude of releases of radioactive materials through identified pathways.
-	-	Main Steam Line Radiation	To monitor the magnitude of releases of radioactive materials through identified pathways.
-	-	GSS Exhaust Fan Discharge Line Radiation (Including High Range)	To monitor the magnitude of releases of radioactive materials through identified pathways.
-	-	Condenser Vacuum Pump Exhaust Line Radiation (Including High Range)	To monitor the magnitude of releases of radioactive materials through identified pathways.
-	-	Plant Air Vent High Concentration Sampling System	To monitor the magnitude of releases of radioactive materials through identified pathways.

Response to Open RAIs

07.01-11

QUESTION NO.: 07.01-11 (ML090570395)	MHI Response (ML091250290)	Additional Information from the NRC Meeting on 1/21/2011	Amendment MHI Response
MHI is requested to address in Section 7.1.3.4, Independence, conformance with Clause 6.3 of SRP Appendix 7.1-C and IEEE 603-1991 for those systems used to achieve and maintain safe shutdown. Clause 6.3 of SRP Appendix 7.1-C and IEEE 603-1991 address the interaction between the sense and command features and other systems. The objective of this review is to ensure that non-safety system interactions with safety systems are limited such that the requirements of 10 CFR 50 Appendix A, GDC 24 are met. The event of concern is simple failure of a sensing channel shared between control and protection functions. Provisions shall be included so that these requirements can be met in conjunction with the requirements of a safety system still being able to accomplish its safety function while sense and command features equipment is in maintenance bypass. During such operation, the sense and command features shall continue to meet the requirements of the single failure criteria and one of the two sense and command requirements listed above. These provisions include reducing the required coincidence, defeating the non-safety system signals taken from the redundant channels, or initiating a protective action from the bypassed channel.	**ANSWER:** Conformance to Clause 6.3 of IEEE 603-1991, interaction between the sense and command features and other systems, for sensors shared between the PSMS and PCMS are addressed in Subsection 7.1.3.16. MHI will add the reference. **Impact on DCD:** Following sentence will be added after the fifth paragraph in Subsection 7.1.3.4. The independence between the PSMS and PCMS for shared sensors is described in Subsection 7.2.3.6. **Current Status:** Based on above response, MHI already added above sentence in Section 7.1.3.4 of the DCD rev.2.	Staff Status: No, The staff's concern is related to independence between the Protection System and the Control System. An issue related to conformance with GDC 24. DCD Section 7.1.3.16 states that in some cases, it is advantageous to employ signal derived from instrumentation that are also used in the protection trains. The staff disagrees with that approach. From defense-in-depth principal, the protection and the control systems should be independent to the maximum extend. For few occasions for minimizing the number of penetrations.) reduce risk of small LOCA, the applicant may justify to share sensors between the protection system and the control system. Therefore, in response to this question, the applicant should identify which parameter has share sensors, or share sensing line and provide justification for that sharing. DCD Section 7.1.3.1.6 mentions the signal selection algorithm (SSA). The SSA design and qualification details should be addressed (or provide a pointer to the technical report). The SSA system should also have ITAAC to verify its intended function.	**ANSWER:** All safety sensor signals are transmitted from the RPS to the PCMS via the unit bus for monitoring functions. The selected safety sensor signals are used for the control functions in the PCMS almost same as the standard Westinghouse PWR. All shared sensor between the protection system and the control system, except for the DAS, are listed in Table 2.5.1-5 of the DCD Tier 1. And, all control systems which uses control signal from the shared sensors in Table 2.5.1-5 of the DCD Tier-1 are described in Section 7.7 of the DCD Chapter 7. For instance; ・ Section 7.7.1.1.5 (eighth paragraph) Pressurizer pressure input signals for pressurizer pressure control are interfaced from the RPS to the PCMS via the unit bus. Signals from each of the four RPS trains are processed through the SSA (signal selection algorism) within the PCMS before being used for pressurizer pressure control function. ・ Section 7.7.1.1.9 (ninth paragraph) The SG water level input signals for the SG water level control function are interfaced from the RPS to the PCMS via the unit bus. Signals from each of the four RPS trains are processed through the SSA (signal selection algorism) within the PCMS before being used for the SG water level control. As described above, all control signals from the shared sensors are processed through the SSA, so a malfunctioning shared sensor in the PSMS does not cause the control system to take erroneous control actions that would result in a challenge to the PSMS. Therefore, where protection signals are used for control, functional isolation is provided between the control and protection systems, and this design conforms to GDC 24 and IEEE 603-1991, Clause 6.3 as described in Section A.6.3 of the Safety I&C Technical Report (MUAP-07004). All shared sensors between the protection system and the DAS are listed in Table 2.5.3-1 (Tier 1) and Table 7.8-2 (Chapter 7) for the monitoring functions, and Table 2.5.3-3 (Tier 1) and Table 7.8-4 (Chapter 7) for the automatic actuation functions of the DCD. There is no shared sensing line between the safety sensor for the protection system and the non-safety sensors for the control system. The SSA function in the PCMS is considered important to safety, so the augmented quality is required for the SSA function as described in Section A.6.3 of the Safety I&C Technical Report (MUAP-07004). The qualification requirements for the SSA function in the PCMS are will be added in Section 3.2.2 of the DCD Chapter 3 and the software life cycle requirements will be added as Appendix D of the US-APWR Software Program Manual (MUAP-07017). There are two types SSA, the 2nd highest SSA and the average SSA, and the designs and functions of the both SSA are developed and verified as Class 1E basic software described in the MELTAC Platform Technical Report (JEXU-1-12-1002). All application software of the control systems in the PCMS use this Class 1E SSA basic software which is already verified, so ITAAC to verify the intended function of the SSA is not needed. **Impact on DCD:** Following sentence will be added in the thirteenth paragraph in Section 2.5.1.1 of the DCD Tier 1. **All safety sensor signals are transmitted from the RPS to the PCMS via the isolation devises for control and monitoring functions as described in Figure 2.5.1-1 and 2.5.1-2. The selected safety sensor signals are used for the control systems in the PCMS. Also, the selected safety sensor signals are transmitted from the RPS to the DAS as described in Section 2.5.3. The selected safety sensor signals from the PSMS to the control systems of the PCMS are processed through the signal selector algorithm (SSA). The SSA** of the PCMS ensures that the PCMS …………to monitored variables which are commonly used in the PSMS and PCMS as listed in Table 2.5.1-5. In Section 7.7.2.9 of the DCD Tier 2, Table 7.7-5 which identifies monitored signals using SSA will be added to keep consistency with the Table 2.5.1-5 of the DCD Tier 1.

07.02-03

QUESTION NO.: 07.02-03 (ML)	MHI Response (ML)	Additional Information from the NRC Meeting on 2/23/2011	Amendment MHI Response
The Equations in DCD Chapter 7, Sections 7.2.1.4.3.1, 7.2.1.4.3.2, and DCD Chapter 16, Table 3.3.1-1, Notes 1&2 were revised in Revision 2 and are now different from the equations in NUREGG 1431, Table 3.3.1-1, Notes 1&2. Responses to RAI 167-1769, questions 16-212 (#6500) and 16-213 (#6501), did not provide the justification for making those changes. The response was more of consistency between Chapter 7 and 16. Either submit a previously approved reference supporting the changes to the OTΔT and OPΔT trip functions or submit a reference that supports the changes from DCD Revision 1 to Revision 2. Provide a description of the equations in DCD Tier 2 Sections 7.2.1.4.3.1 and 7.2.1.4.3.2 for the over-temperature and over-power delta-T algorithms regarding the lead-lag processing of core heat removal protection trips, including the purpose of the processing. The algorithms shown in DCD Tier 2 Sections 7.2.1.4.3.1 and 7.2.1.4.3.2 for the over-temperature and over-power delta-T calculations only describe the calculation of the trip setpoints under normal conditions. Discuss the limiting events for the core heat removal trip response. The lead-lag signal processing, described in Sections 7.2.1.4.3.1 and 7.2.1.4.3.2 and shown in Figure 7.2.2, depends on the time history of the input signal and so presents a more complicated effect for timing and errors. Discuss the special operation conditions for lead-lag signal processing such startup of the signal processing module, lost data, restart of the module while operating, and any other special operating modes for lead-lag signal processing modules. The response should be coordinated with Chapter 16 PM and SRSB. Relates to question numbers, 6500 (16-212), 6501 (16-213), 6502 (16-214), and 6533 (16-236) of RAI 167-1769. Reference: MHI's Amended Response to US-APWRD DCD RAI 167-1769; MHI Ref: UAP-HF-09354; dated July 3, 2009; ML091890964.	**ANSWER:** The equation of over-temperature ΔT in DCD Chapter 16 Table 3.3.1-1 Note 1 was corrected from DCD Rev. 1 to DCD Rev. 2 by multiplying the right-hand side of each equation by ΔT_0. This typographical revision made this equation consistent with the equation provided in NUREG-1431. The inequality sign in the equation in DCD Chapter 16 Table 3.3.1-1 was revised as described in the RAI response to Question No. 16-212 of RAI No. 167-1769. The revised equation describes a reactor trip plant condition. Although the equation in Rev. 1 DCD Chapter 16 Table 3.3.1-1 Note 1 and NUREG-1431 shows a plant condition of reactor not tripped, there is no technical difference. Although the equation of over-power ΔT in DCD Chapter 16 Table 3.3.1-1 Note 2 should be corrected similar to the correction for the equation in DCD Chapter 16 Table 3.3.1-1 Note 1; Notes 1 and 2, including the equations, will be deleted from Tech Spec Section 3.3.1. These notes were originally intended to explain how to calculate allowable values, however, this information was removed from US-APWR DCD Chapter 16. This deletion was agreed upon with the NRC Chapter 16 reviewer at the NRC meeting held on 12/14/2010 and 12/15/2010. The equation of over-power ΔT in DCD Section 7.2.1.4.3.2 was corrected between DCD Rev. 1 and DCD Rev. 2 by multiplying the right-hand side of each equation by ΔT_0. For the measurement of primary coolant temperature, the lag processing of the measured RCS average temperature is deleted since the US-APWR will utilize a well-type resistance temperature detector (RTD) instead of the RTD with bypass-line used in some operating plants. Lag processing which is also included in the equations in NUREG-1431, for the measured RCS average temperature was previously used for noise filtering because the RTD with bypass-line responds relatively quickly to changes in temperature. On the other hand, Lag processing for noise filtering of the measured RCS average temperature in the US-APWR is no longer necessary because the installed RTD responds slower due to the thermal mass associated with the RTD enclosure. Therefore, the equations for the US-APWR contain no lag processing of the RCS average temperature measurement since this processing is not necessary for the US-APWR primary coolant temperature measurement system. Over-temperature ΔT provides primary protection for both the departure from nucleate boiling (DNB) limit and core exit boiling limit. In NUREG-1431, a single setpoint covers both the DNB limit and core exit boiling limit. However, US-APWR DCD Chapter 16 Rev. 1 / Rev. 2 Table 3.3.1-1 Note 1 and DCD Tier 2 Section 7.2.1.4.3.1, utilize two separate setpoints (ΔT_{sp1}, ΔT_{sp2}) to protect the DNB limit and core exit boiling limit, respectively. The cold leg and hot leg temperature measurement systems are located on the cold leg and hot leg, respectively. Cold leg temperature and hot leg temperature measurements include system piping delay from the reactor core and the temperature measurement systems. Therefore, the corresponding equations of over-temperature ΔT and over-power ΔT include lead/lag processing to compensate for the system piping delay. The limiting condition of the lead-lag processing is system startup since the output of the processing module which has dynamic characteristics generally differs from a stationary value at the time of the CPU-restart including initial CPU startup. Since RPS is bypassed at the time of system startup and then shifted to normal operation after ensuring that all outputs of processing have achieved a stable condition, the limiting condition does not impact the reactor trip functions. In addition, the abnormality of the processing output due to loss of data from memory can be detected by the self-diagnosis for the RPS CPU module. This is considered as a system failure, and it is not the limiting condition of the lead-lag processing. **Impact on DCD** The following description will be the last paragraph of DCD Chapter 7 Section 7.2.1. <u>RPS should be shifted to normal operation from bypass mode, after ensuring that all outputs of the processing have achieved a stable condition, in order to eliminate the influence of dynamic characteristics.</u> **Impact on COLA** There is no impact on the COLA **Impact on PRA** There is no impact on the PRA	Staff Status: No, 1. Response deletes equations from Chapter 16 TS 2. Chapter 16 Bases points to equations 3. Identify design bases for equations Items: 1. Chapter 16, Section B 3.3.1, Item 6. Overtemperature dT, states "The Overtemperature ΔT trip Function is calculated for each loop as described in Note 1 of Table 3.3.1-1." 2. Chapter 16, Section B 3.3.1, Item 7. Overpower 6T, states "The Overpower dT trip Function is calculated for each loop as per Note 2 of Table 3.3.1-1." 3. Chapter 16, Section B 3.3.1, Item 6 & 7 still has discussion on "dynamic compensation." Equations did not remove the lead/lag formulas, MUAP-09022, Figures 6.1 and 6.2 and surrounding discussion still has lead/lag. 4. Time constants shown in Chapter 7 are not denoted "T2≥[*]sec" or where they are developed. Chapter 16 notes state "These values denoted with [*] are specified in the COLR." 5. Is there a design base document that generates and ties these equations with MUAP-07008 & 07009? 6. Will DCD change be a condition for/during operations?	Response: No 1 to 3 The equations of ΔT in Chapter 16 will be deleted in the DCD Rev. 4. Only one lag processing has been deleted from the equation of NUREG-1431. We have no plan to delete another lead/lag from current equation of the DCD Chapter 7. No 4 All constants are controlled in accordance with Setpoint Control Program. No 5 Design bases of the equations are based on MUAP-07008 & 07009. No6 We have no plan to change the DCD description.

07.05-19

QUESTION NO.: 07.05-19 (ML)	**MHI Response (ML)**	**Additional Information from the NRC Meeting on 1/21/2011**	**Amendment MHI Response**
SRP Section 7.5, Item 1.C of Part III, Review Procedures, states that a basis should be provided for EOP action points that accounts for measurement uncertainties. The staff could not find any discussion of EOP action points nor the process that will be used to develop the EOP action points in the DCD. The staff requests that MHI provide a discussion and a basis for EOP action points and the process that will be used to develop the EOP action points, taking into account any measurement uncertainties.	**ANSWER:** It is the intention of MHI to develop a sound technical basis for every action point (setpoint) in the standard EOPs as part of the ERG/EOP development program. As part of the technical basis for each EOP setpoint, consideration to apply instrument uncertainties will be made. The determination of whether or not to include margin within the setpoint value to account for instrument uncertainties will be based on impact of the operator actions and the recovery strategy in the EOPs where the setpoint is used. EOP action setpoint instrument uncertainty calculations will be consistent with the methodology outlined in MUAP-09022, “US-APWR Instrument Setpoint Methodology”. MHI will provide this technical basis information in the EOP Setpoint Basis Methodology companion document to the completed standard EOP. The plant specific EOPs and therefore, the plant specific EOP setpoints are developed by the COL applicant. Please see the US-APWR DCD Subsection 13.5.2.1. **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	Staff Status: No, The finds the response acceptable but the answer needs to be reflect in the DCD. Section 7.5 needs to mention EOP action points even if it is just to point to the appropriate DCD section.	Response: MHI will reflect the answer for the RAI to the DCD Section 7.5. Revised RAI response to add the answer to DCD Section 7.5 will be submitted after discussion with the NRC on April 6th and 7th meeting. The DCD markup will be submitted by March 31. Section 7.5.1.1.4

07.05-20

QUESTION NO.: 07.05-20 (ML)	MHI Response (ML)	Additional Information from the NRC Meeting on 1/21/2011	Amendment MHI Response
In Section 7.5.1.3, "Plant Annunciator (Alarm) System," of the US-APWR DCD Revision 2, states "As for all PCMS components, the alarm system is powered by redundant UPSs. The alarm system is designed and tested to a similar environmental, seismic, and EMI/RFI requirement as the PSMS." The staff requests MHI to provide a clarification and additional details on what the actual difference is between the PCMS and the PSMS systems in terms of their software V&V, seismic qualifications and environmental testing in accordance with GDC 1, "Quality standards and records".	**ANSWER:** GDC-1 does not apply to the PCMS, since PCMS is not safety-related system. PCMS is, however, required to be designed and tested in accordance with similar specifications to that of PSMS for environmental testing, seismic qualifications, and software V&V to ensure its high integrity. Table 1 shows the differences between PSMS and PCMS regarding seismic qualification, environmental testing and software V&V. **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	Staff Status: No, The staff finds this response unacceptable. 1. GDC 1 applies to systems important to safety not just safety-related systems. 2. Contradicts response to RAI 07.07-2 ANSWER:GDC 1 is also applicable to the PCMS. Table 7.1-2 will be updated. 3. Need to identify systems that are not safety-related but still important to safety and how are those systems handled.	Response: The response to the comments from the NRC are as follows; 1. GDC 1 applies to the PCMS. The original response to the RAI will be revised to correct the GDC-1 applicability. 2. GDC 1 applies to the PCMS as stated in above response. 3. Scope of non safety systems but required augmented quality will be identified in Table 7.1-4 of Section 7.1. Quality assurance requirement and equipment qualification requirement will be provided in Section 3.2.2. Especially, software life cycle requirements, including V&V, will be provided in Appendix D of Software Program Manual. Revised RAI response will be submitted after discussion with the NRC on April 6th and 7th meeting. Section 7.1.3

Table.1 Difference between the PCMS and the PSMS

		PSMS	PCMS (Plant Annunciator (Alarm) System)
Seismic qualification		Complies with RG 1.100 Specification is referenced in Safety System Digital Platform -MELTAC- Proprietary Version (MUAP-07005-P (R6)) Table4.1-2 (Table4.1-2 is Non-Proprietary)	Seismic qualification for physical integrity is evaluated by numerical analysis to the same Class 1E requirement level as the PSMS. As for component level of computer system, Seismic qualification for functional integrity is evaluated by actual seismic tests to the same Class 1E requirement level as the PSMS. Complies with JEAG4601・JEAC4601・JSME S NC1 (JEAG: Japan Electric Association Guide)
Environmental testing	EMC (EMS/EMI)	Complies with RG 1.180 Specification is referenced in Safety System Digital Platform -MELTAC- Proprietary Version(MUAP-07005-P(R6)) Table4.1-2 (Table4.1-2 is Non-Proprietary)	Complies with RG 1.180.
	Room Ambient Temperature	Recommended 68 to78.8°F (20 to 26°C) This temperature range is expected within a heated/ air-conditioned instrumentation and control room of the nuclear power plant. The controller should be mounted in a cabinet with no more than 18°F (10°C) heat rise. Operating within this range will maximize the life of the equipment. Operation guarantee 32 to122°F (0 to 50°C) This temperature range is expected during heat/air conditioning failure conditions. The controller should be mounted in a cabinet with no more than 18°F (10°C) heat rise. Requirements are referenced in Safety System Digital Platform -MELTAC- Proprietary Version (MUAP-07005-P (R6)) Table4.1-2 (Table4.1-2 is Non–Proprietary)	Operation guarantee 41 to 104 °F (5 to 40°C) Complies with JEITA IT 1004 Class B (JEITA: Japan Electronics and Information Technology Industries Association)
	Relative Humidity	10 to 95%Rh (No condensation) Requirement is referenced in Safety System Digital Platform -MELTAC- Proprietary Version (MUAP-07005-P (R6)) Table4.1-2 (Table4.1-2 is Non-Proprietary)	20 to 80%Rh (No condensation) Complies with JEITA IT 1004 Class B
	Withstand Voltage	Complies with JIS-C0704-1995 (IEC664/947) Specification is referenced in Safety System Digital Platform -MELTAC- Proprietary Version (MUAP-07005-P (R6)) Table4.1-2 (Table4.1-2 is Non-Proprietary)	Complies with JIS-C0704-1995 (IEC664/947)
Software V&V		Complies with IEEE1012（endorsed by RG 1.168） As described in MELTAC TR Table 6.1-2, V&V is performed by the independent V&V Team for each software development phase, from Platform Design to Integration Test. Details are referenced in Safety System Digital Platform -MELTAC- Proprietary Version (MUAP-07005-P (R6)) Table 6.1-2 (Table 6.1-2 is Proprietary)	Complies with ISO 9001 As written in the Safety I&C System Description and Design Process (MUAP-07004 R5) Appendix C ‘Software Quality Program’, the software life cycle is managed according to a Quality Assurance Plan for high integrity components. That Quality Assurance Plan is referenced in plant licensing documentation. For the US-APWR this QA plan is referenced in the US-APWR DCD Chapter 17.

07.06-3

QUESTION NO.: 07.06-3 (ML090620206)	MHI Response (ML091250290)	Additional Information from the NRC Meeting on 1/21/2011	Amendment MHI Response
Discuss how GDCs 20, 21, 22, 23, and 29 are applied to the design of interlock systems important to safety. Update Table 7.1-2 if necessary. Though not listed in SRP Table 7-1 as applicable to information systems required for safety, DCD Table 7.1-2 cites compliance with GDCs 20, 21, 22, 23 and 29 for the PSMS in Section 7.6. Section 7.6 indicates that detailed compliance to the GDC is described (in general, not specifically related to interlock systems) in TR MUAP-07004-P(R1) Section 3. It is unknown how GDCs 20, 21, 22, 23, and 29 are applied to the design of interlock systems important to safety.	**ANSWER:** Interlock systems important to safety are implemented within the PSMS safety related software and hardware. Therefore, the PSMS is credited for compliance to these GDCs and these GDCs are listed for Section 7.6 in Table 7.1-2. Requirements met by the PSMS itself, such as equipment qualification, are described in DCD Subsection 7.1.3. For conformance to the single failure criterion, these interlocks are redundantly controlled from at least two trains of the PSMS, except for CS/RHR discharge valves. Justification for the single train CS/RHR discharge valve interlock design is discussed in RAI 07.06-15. **Impact on DCD:** There is no impact on the DCD.	Staff Status: No, Single failure criterion for CS/RHR discharge valve interlock is not met. MHI's Response: "Justification for the single train CS/RHR discharge valve interlock design is discussed in RAI 07.06-15." However, RAI 07.06-15 does not address the issue.	Response: The statement in the original response to the RAI “For conformance to the single failure criterion, these interlocks are redundantly controlled from at least two trains of the PSMS, except for CS/RHR discharge valves. Justification for the single train CS/RHR discharge valve interlock design is discussed in RAI 07.06-15.” was incorrect. This description should be corrected as follows: For conformance to the single failure criterion, these interlocks are redundantly controlled from at least two trains of the PSMS, except for CS/RHR **pump hot leg isolation** ~~**discharge**~~ valves. Justification for the single train CS/RHR **pump hot leg isolation** ~~**discharge**~~ valve **open permissive** interlock design is discussed in RAI 07.06-15. Revised RAI response will be submitted after discussion with the NRC on April 6th and 7th meeting. No impact of DCD

07.06-16

QUESTION NO.: 07.06-16 (ML090620206)	MHI Response (ML091250290)	Additional Information from the NRC Meeting on 2/23/2011	Amendment MHI Response
Describe in detail the "Pull Lock" feature of the motor operated isolation valve (MOIV), the conditions under which this feature could be used and, assuming this feature of the MOIV, how the accumulator discharge design meets position 4 of BTP 7-2, "Guidance on Requirements of Motor-Operated Valves in the Emergency Core Cooling System Accumulator Lines." In Section 7.6.1.4, which describes the ECCS accumulator interlock system, it is stated that if the (MOIV) was closed in the "Pull Lock" mode, the accumulator discharge valves will not automatically open, therefore the affected accumulator will be un-available for its designed ESF function. This appears to violate Position 4 of BTP 7-2, which requires "utilization of a safety injection signal to remove automatically (override) any bypass feature that may be provided to allow an isolation valve to be closed for short periods of time…" The DCD indicates that the "Pull Lock" function is described in Topical Report MUAP-07007 Section 4.5.3.a. However, the staff's review of this document for the referenced section showed that Section 4.5.3.a of the Topical Report MUAP-07007, "HSI System Description and HFE Process," only discusses operation-related information display features of ON/OFF switches. The only reference to the "Pull Lock" feature is a display button in Figure 4.5-4, "Soft Operation Switch Moving Feature."	<u>**ANSWER:**</u> As described in Subsection 7.6.1.4, "the ECCS actuation signal will automatically open the valve and make the accumulator system available", except when the valve is manually closed and manually put in the Pull Lock condition. This requires two distinct and deliberate manual operator actions. The pull lock condition for the accumulator discharge valve is applied only when the associated accumulator is re-charged with gas. Recharging is a maintenance activity, which occurs only when the accumulator pressure is lower than required. Under this condition, the accumulator itself is inoperable, therefore automatically opening the accumulator discharge valve would have no safety benefit. The accumulator bypass or inoperable condition is managed by Technical Specification in DCD Chapter 16 Section 3.5.1. In addition, interlock systems important to safety, including the accumulator discharge valve interlock, are indicated by BISI, as described in DCD Subsection 7.5.1.2.2. <u>**Impact on DCD:**</u> There is no impact on the DCD.	Staff Status: No, The detail of "Pull Lock" in the response is acceptable. However, this important information needs to be incorporated into Section 7.6.1.4 to fully resolve the issue raised by the RAI question. The following information should be included in the DCD: "The ECCS actuation signal will automatically open the valve and make the accumulator system available, except when the valve is manually closed and manually put in the Lock condition. The Lock condition for the accumulator discharge valve is applied only when the associated accumulator is re-charged with gas. Recharging is a maintenance activity, which occurs only when the accumulator pressure is lower than required. Under this condition, the accumulator itself is inoperable; therefore, automatically opening the accumulator discharge valve does not provide the accumulator design function. The accumulator discharge valve interlock is indicated by BISI, and the accumulator bypass or inoperable condition is managed by Technical Specifications in DCD Chapter 16 Section 3.5.1. "	Response: The response to the RAI will be revised after discussion with the NRC on April 6th and 7th meeting as follows: <u>**ANSWER:**</u> As described in Subsection 7.6.1.4, "the ECCS actuation signal will automatically open the valve and make the accumulator system available", except when the valve is manually closed and manually put in the Pull Lock condition. This requires two distinct and deliberate manual operator actions. The pull lock condition for the accumulator discharge valve is applied only when the associated accumulator is re-charged with gas <u>**or water**</u>. Recharging is a maintenance activity, which occurs only when the accumulator pressure <u>**or water level**</u> is lower than required. Under this condition, the accumulator itself is inoperable, therefore automatically opening the accumulator discharge valve would have no safety benefit. The accumulator bypass or inoperable condition is managed by Technical Specification in DCD Chapter 16 Section 3.5.1. In addition, interlock systems important to safety, including the accumulator discharge valve interlock, are indicated by BISI, as described in DCD Subsection 7.5.1.2.2. <u>**Impact on DCD:**</u> The following description will be added to Section 7.6.1.4. <u>**The ECCS actuation signal will automatically open the valve and make the accumulator system available, except when the valve is manually closed and manually put in the Lock condition. The Lock condition for the accumulator discharge valve is applied only when the associated accumulator is re-charged with gas or water. Recharging is a maintenance activity, which occurs only when the accumulator pressure or water level is lower than required. Under this condition, the accumulator itself is inoperable; therefore, automatically opening the accumulator discharge valve does not provide the accumulator design function. The accumulator discharge valve interlock is indicated by BISI, and the accumulator bypass or inoperable condition is managed by Technical Specifications in DCD Chapter 16 Section 3.5.1.**</u> Section 7.6.1.4

07.06-21

QUESTION NO.: 07.06-21 (ML)	MHI Response (ML)	Additional Information from the NRC Meeting on 2/23/2011	Amendment MHI Response
MHI is requested to effectively demonstrate how to conform to guidance RG 1.206, "Combined License Applications for Nuclear Power Plants," with regard to interlock to prevent over pressurization of the primary coolant system during low-temperature operations of the reactor vessel. Subsection C.I.5.2.2.2 of RG 1.206 states that "Applicants should describe the design of overpressure protection during low-temperature operations, including the capability to relieve pressure during all overpressure events during startup and shutdown conditions at low temperatures, particularly during water-solid conditions. Applicants should provide the analysis that demonstrates how overpressure protection is achieved, assuming any single active component failure. This section should identify all overpressure events and, as a subset, identify the events that can be prevented by preventive interlocks or locking-out power. Applicants should describe how the overpressure protection system is enabled, the alarms and indications associated with the system, and the power source for the system." Subsection 7.6.3 of the DCD states that "There are no interlocks necessary to prevent over pressurization of the RCS during low-temperature operations of the RV. Refer to Subsection 5.2.2." Subsection 5.2.2 identifies overpressure events but instead of identifying the events that can be prevented by preventive interlocks or locking-out power as described in RG 1.206 above, this subsection further states that "An important aspect of RCS overpressure protection at low temperatures is the use of administrative controls which are discussed in paragraph 5.2.2.2.2.2, Administrative Controls. Although specific alarms do not exist to invoke specific administrative procedures, annunciation is provided to alert the operator to arm the cold overpressure mitigation system." It is not clear how the guidance in RG 1.206 with regard to interlock to prevent overpressurization of the primary coolant system during low-temperature operations of the reactor vessel is met.	**ANSWER:** The LTOP system for US-APWR consists of CS/RHR pump suction relief valves, which are spring-loaded relief valves. Therefore, preventive interlock to activate the LTOP system is not needed. These valves are equipped with direct position indication in accordance with a requirement of Section II.D.3 of the TMI Action Plan. When LTOP event occurs, these relief valves operate reactor coolant pressure and a valve position alarm alerts the operator. In order to ensure the LTOP system is operable status at the correct plant condition during cooldown, the technical specifications require surveillances of the following status (Reference DCD Chapter 16, SR 3.4.12.1 through 3.4.12.7.). - Number of available Safety Injection (SI) pump - Number of available Charging pump - Accumulators are isolated - RHR suction motor-operated valves are open - RHR suction motor-operated valves are locked open with operator power removed Please refer to responses provided to RAI No.103, UAP-HF-08303, which pertain to the LTOP system. Therefore, since the LTOP system does not need preventive interlock, there is no interlock provided to prevent overpressurization of the primary coolant system during low-temperature operation of the reactor vessel. **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	Staff Status: No, The justification for not having interlocks to prevent overpressurization of the RCS during low-temperature operations of the reactor vessel is acceptable. However, this justification needs to be documented in Section 7.6.3 of the DCD. The following information should be included 'in the DCD: "There are no interlocks necessary to prevent overpressurization of the RCS during low temperature operations of the reactor vessel since the spring-loaded CS/RHR pump suction relief valves provide low-temperature overpressure protection for the RCS. When an LTOP event occurs, these relief valves discharge the RCS inventory to the refueling water storage pit in the containment, and a valve position alarm alerts the operator."	Response: "Impact on DCD" of the response to the RAI will be revised after discussion with the NRC on April 6th and 7th meeting as follows: Impact on DCD The following description will be added to Section 7.6.3. **There are no interlocks necessary to prevent overpressurization of the RCS during low temperature operations of the reactor vessel since the spring-loaded CS/RHR pump suction relief valves provide low-temperature overpressure protection for the RCS. When an LTOP event occurs, these relief valves discharge the RCS inventory to the refueling water storage pit in the containment, and a valve position alarm alerts the operator.** Section 7.6.3

07.09-1

QUESTION NO.: 07.09-1	MHI Response	Additional Information from the NRC Meeting on 1/21/2011	Amendment MHI Response
MHI is required to comply with 10 CFR 50.34(f)(2)(v) and 50.62 in relation to the DCSs. MHI is requested to discuss this in Section 7.9 and Table 7.1-2 should be updated to reflect this requirement. Table 7.1-2 in the DC-FSAR cites compliance with various regulations applicable to the DCS with the exception of §50.34(f)(2)(v) and §50.62. §50.34(f)(2)(v) requires licensees to provide for automatic indication of the bypassed and operable status of safety systems. The DCSs support ATWS mitigation functions and RTS functions. The staff cannot determine if the DCS adequately supports RTS and ESFAS functions as necessary to sense accident conditions and AOOs in order to initiate protective actions consistent with the accident analysis presented in Chapter 15 of the DC-FSAR, without compliance with the above regulations known.	**ANSWER:** Section 7.9 will be added to Table 7.1-2 as the conformance to §50.34(f)(2)(v). Refer to response to RAI 7.6-1. Also the column "Safety DCS" will be added in Table 7.1-2 as one of I&C system. **Impact on DCD** The column "Safety DCS" will be added in Table 7.1-2 as Attachment 1. **Impact on COLA** There is no impact on the COLA. **Impact on PRA** There is no impact on the PRA.	Staff Status: No, 7.9 was not added to Table 7.1-2 (10 CFR 50.34 (f)(2)(v) as committed.	Response: Section 7.9 will be added to Table 7.1-2 10 CFR 50.34 (f)(2)(v) as indicated with bold and underlined text in the Attachment 1. Revised RAI response will be submitted after discussion with the NRC on April 6th and 7th meeting. Table 7.1-2

Attachment 1 Revised Table 7.1-2

Table 7.1-2 Regulatory Requirements Applicability Matrix
(per NUREG-0800 Standard Review Plan (SRP) Sec. 7.1 Rev. 5)
(Sheet 1 of 8)

Applicable Criteria		Title	I&C System							Related Section in US-APWR DCD
			RPS	ESFAS	SLS	Safety HSI	Safety DCS	PCMS	DAS	
		1. 10 CFR 50 and 52								
a.	50.55a(a)(1)	Quality Standards for Systems Important to Safety	X	X	X	X	X			7.2 to 7.6, 7.9
b.	50.55a(h)(2)	Protection Systems (IEEE Std 603-1991 or IEEE Std 279-1971)	X	X	X	X	X			7.2 to 7.6, 7.9
c.	50.55a(h)(3)	Safety Systems (IEEE Std 603-1991)	X	X	X	X	X			7.2 to 7.6, 7.9
d.	50.34(f)(2)(v) [I.D.3]	Bypass and Inoperable Status Indication	X	X	X	X	X	X		7.2, 7.3, 7.5, 7.6, **7.9**
e.	50.34(f)(2)(xi) [II.D.3]	Direct Indication of Relief and Safety Valve Position			X		X	X		7.5
f.	50.34(f)(2)(xii) [II.E.1.2]	Auxiliary Feedwater System Automatic Initiation and Flow Indication	X	X	X	X	X			7.3, 7.5
g.	50.34(f)(2)(xvii) [II.F.1]	Accident Monitoring Instrumentation	X		X	X	X	X		7.5
h.	50.34(f)(2)(xviii) [II.F.2]	Instrumentation for the Detection of Inadequate Core Cooling	X			X	X			7.5
i.	50.34(f)(2)(xiv) [II.E.4.2]	Containment Isolation Systems	X	X	X	X	X			7.3
j.	50.34(f)(2)(xix) [II.F.3]	Instruments for Monitoring Plant Conditions Following Core Damage	X			X	X			7.5
k.	50.34(f)(2)(xx) [II.G.1]	Power for Pressurizer Level Indication and Controls for Pressurizer Relief and Block Valves	X		X	X	X			7.4, 7.5
l.	50.34(f)(2)(xxii) [II.K.2.9]	Failure Mode and Effect Analysis of Integrated Control System								N/A to US-APWR
m.	50.34(f)(2)(xxiii) [II.K.2.10]	Anticipatory Trip on Loss of Main Feedwater or Turbine Trip								N/A to US-APWR
n.	50.34(f)(2)(xxiv) [II.K.3.23]	Central Reactor Vessel Water Level Recording								N/A to US-APWR

07.09-12

QUESTION NO.: 07.09-12 (ML090570395)	MHI Response (ML091250290)	Additional Information from the NRC Meeting on 1/21/2011	Amendment MHI Response
MHI is requested to identify how the DCS meets the single failure criterion in the DCFSAR, preferably in Section 7.9.2.4. The US-APWR DC-FSAR briefly discusses (in one short paragraph) potential hazards and how the DCS addresses single failures in Section 7.9.2.4. The DC-FSAR states that "self-diagnostic features described in Topical Report MUAP-07004 Section 4.3, detect DCS errors or failures. All DCS errors and failures are analyzed in the FMEA, which demonstrates that there are no single failures that can result in loss of the safety function." In numerous instances, the TRs refer to "credible" single failures rather than single failures. The purpose and what the single failure analysis shows are not discussed.	**ANSWER:** Within the DCS, there are independent safety busses, maintenance networks, data links and I/O busses for each division. In addition, the non-safety unit bus is isolated from the safety system. In all cases independence includes electrical independence and communications independence. Therefore, safety divisions are independent of each other and independent of non-safety divisions. Per IEEE 379, once independence is established between redundant divisions, the single failure criteria are satisfied. **Impact on DCD:** There is no impact on the DCD.	Staff Status: No, MHI's response to the question with respect to independence is not satisfied, especially the statement "the non-safety unit bus is isolated from the safety system."	Response: MHI has already submitted drafts of the additional summary descriptions and detailed descriptions on the data communication issue to the NRC. Those descriptions will be incorporated into DCD chapter 7 and Safety I&C Technical Report (MUAP-07004), respectively. The modified descriptions will be included in the mark-up version of the DCD Chapter 7 and the Safety I&C Technical Report will be submitted by March 31. Section 7.1.4 Appendix F of Safety I&C Technical Report

07.14-42

QUESTION NO.: 07-14 BTP-42	MHI Response	Additional Information from the NRC Meeting on 2/23/2011	Amendment MHI Response
Criterion III, Design Control, of 10 CFR 50, Appendix B, requires measures to ensure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures and instructions. Also, RG 1.173 specifies additional activities beyond those identified by IEEE Std 1074-1995, which it endorses, to ensure safety system development is consistent with defined system safety analyses. The MELTAC Basic Software Safety Report, JEXU-1015-1009-P(R3), is found to not be the type of document described by this regulatory guidance, or staff guidance, for a software safety analysis. This is because it does not describe, per RG 1.173, the types of software safety analyses, by inputs, outputs or activity description, or how this analysis was done in each phase of the software development life cycle. Also, this document does not describe the types of analyses performed as cited in NUREG/CR-6101 by BTP 7-14, Section B.3.1.9. The safety analyses that were done for the MELTAC Basic Software should be explained in the commercial grade dedication report in each phase, with the differences to all staff guidance noted and adequately explained. Also, for future MELTAC Basic Software development activites, the software safety plan should adequately present the software safety analyses which will be done and how each attribute in staff guidance is met. Therefore, this document should be retitled “Analysis of the MELTAC Basic Platform to Guidance of ISG-04” as ISG-04 is the only identified guidance within the document, which is not consistent with a software safety analysis, and will only be used by the staff for compliance to ISG-04.	**ANSWER:** MELCO has identified three specific requests in the RAI as below. Our response to each request is as follows. 1) Evaluation of the past safety analysis of the MELTAC platform should be in the CGD Report. MELCO will develop a Software Safety Analysis Re-evaluation (SSAR) Report to evaluate whether past safety analysis of the MELTAC platform conforms to staff guidance (R.G. 1.173 and NUREG/CR-6101). This report will be separate from the current MELTAC Re-evaluation Program (MRP) Report, JEXU-1022-6301. The current version of the MRP Report includes the results of the evaluation in accordance with the commercial grade dedication guidelines of EPRI-TR106439 and EPRI-TR107330, which do not require an evaluation of conformance to R.G. 1.173 and NUREG/CR-6101. 2) The software safety plan (SSP) should be established to ensure that software safety analysis for future MELTAC development will be performed in accordance with staff guidance: • R.G. 1.173; • BTP 7-14 B3.1; • NUREG/CR-6101. The current SSP documented in Section 3.9 of the MELTAC Platform Basic Software Program Manual (SPM), JEXU-1012-1132 Rev 1, was written to conform to the guidance of BTP 7-14 and R.G. 1.173. But it does not specifically describe conformance to NUREG/CR-6101. MELCO will revise the MELTAC SPM to clearly describe compliance with NUREG/CR-6101. This addition will ensure future MELTAC development activities meet applicable staff guidance. 3) The analysis in the current SSR is not in accordance with staff guidance (see No. 2). The document should be retitled to "Analysis of the MELTAC Basic Platform to Guidance of ISG-04" as any other applicable guidelines are not covered. MELCO will retitle the document “MELTAC Platform Basic Analysis of Software Safety Hazards (including hazards in ISG-04).” With this title change, other documents that reference this document will need to be updated.	Staff Status: No, "Mitsubishi RAI response dated “2010/12/20” is not acceptable."	Response: 1) CGD (MRP) for the legacy MELTAC Platform has been performed in accordance with EPRI-TR107330 and EPRI-TR106439. EPRI requires only the hazard analysis described in IEEE 1012, which has already been assessed in MRP. MELTAC Platform Software Safety Analysis Report will be renamed and will be focused to the ISG-04 Conformance as described in 3), therefore no additional assessment of Software Safety Analysis for legacy MELTAC Platform is considered to be necessary. 2) The description that states software safety analysis will be performed in accordance with the requirements of RG 1.173, BTP7-14 B3.1, and NUREEG/CR6101 has been added in Section 3.9 of the MELTAC Platform Basic Software Program Manual (SPM), JEXU-1012-1132 Rev2 submitted on January 31, 2011. A specific operation procedure for analysis has also been included in accordance with the above standards. 3) The document title will be renamed to “MELTAC Platform ISG-04 Conformance Analysis”. This report will be revised to exclude all sections other than those pertinent to the ISG-04 conformance assessment (Sections 3.2 through 3.5) and will be submitted to the NRC on March 31. This analysis will exclude discussion of ISG-04 issues related to Functional Independence. Conformance to the Functional Independence issues of ISG-04 are addressed at the application level (ie. in MUAP-07004 for the US-APWR). In addition, the remaining portions of the current Software Safety Report, including Sections 3.1 (Detectability of Input, Operation, and Output hazards) and 3.6 (Analysis of Self-Diagnosis Function), will be described in a new Appendix that will be attached to the MELTAC Technical Report entitled MELTAC Basis Software Critical Function Analysis. This analysis will be retained because the US-APWR application SPM Software Safety Plan defines this section as part of the preliminary hazard analysis. The A-SPM will be revised to refer to JEXU-1015-1009 “MELTAC Platform ISG-04 Conformance Analysis” and this new Appendix “MELTAC Basis Software Critical Function Analysis”.

	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.		

14.03.05-12

QUESTION NO.: 14.03.05-12	MHI Response	Additional Information from the NRC Meeting on 1/21/2011	Amendment MHI Response
Address the applicability of IEEE Std. 603-1991, Section 4.6 with respect to an ITAAC to verify the number and locations of sensors in the RT and ESF safety systems that have a spatial dependence. Based on the requirements of IEEE Std 603-1991, Section 4.6, the ITAAC should include identification in the as-built design of the minimum number and locations of sensors having spatial dependence that are required for protective actions. The staff conducted a review of the DCD Tier 1 and Tier 2 as well as the ITAAC in Table 2.5.1-5 and concluded that no information is given on the minimum number and locations of spatially dependent sensors. Provide as-built information that establishes the minimum number and locations of the spatially dependent sensors that the RT and ESF systems required for protective actions (i.e., revise the ITAAC in Table 2.5.1-5 to address the requirements of Section 4.6 of IEEE Std. 603-1991).	**ANSWER:** MHI Topical Report entitled "Safety I&C System Description and Design Process," MUAP-07004, addresses IEEE-603-1991 requirements for spatially dependent sensors. MUAP-07004 is referenced in DCD Tier 2, Chapter 7 (e.g., Reference 7.9-2), and includes the following description typical of spatially dependent sensors: "Thermowell-mounted resistance temperature detectors (RTDs) installed in each reactor coolant loop provide the hot and cold leg temperature signals required for input to the protection and control functions. The hot leg temperature measurement in each loop is accomplished using three fast-response, dualelement, narrow-range RTDs. The three thermowells in each hot leg are mounted approximately 120 degrees apart in the cross sectional plane of the piping, to obtain a representative temperature sample. The temperatures measured by the three RTDs are different due to hot leg temperature streaming and vary as a function of thermal power. The PSMS averages these signals to generate a hot leg average temperature. Radially varying cold leg temperature is not a concern because the RTDs are located downstream of the reactor coolant pumps. The pumps provide mixing of the coolant so that radial temperature variations do not exist. Radial neutron flux is not a spatially dependent concern because of core radial symmetry. Calculations involving overtemperature and overpower delta T use axial variation in neutron flux. Excore detectors furnish this axially-dependent information to the overtemperature and overpower calculations in the RPS." DCD Tier 1 Subsection 2.5.1 will be revised to identify the RTS and ESFAS monitored variables that have spatial dependency, and include an ITAAC item to verify their consistency with design requirements. **Impact on DCD** See Attachment I for a mark-up of Tier 1 Section 2.5 with the changes as shown below. The Design Description of Tier 1 Subsection 2.5.1, Table 2.5.1-2 Reactor Trip and Monitored Variables, Table 2.5.1-3 ESF Actuations and Monitored Parameters (Sheet 2 of 3), and Table 2.5.1-3 ESF Actuations and Monitored Parameters (Sheet 3 of 3) are revised to identify spatially dependent variables. Only the parameters impacted are marked up below. Tier 1 Subsection 2.5.1.1, Design Description, will be revised to add the following: Spatially dependent sensors that are required for protective actions are identified in Table 2.5.1-2 and Table 2.5.1-3, and have the minimum number of sensors and locations to perform the protective action.	Staff Status: No, 1, Provide minimum number and location of spatially dependent sensors 2. Provide consistency b/w DCD Tier 1 & 2 and MUAP-07004, on what are the spatially dependent sensors	**ANSWER:** Spatially dependent variables that are required for protective actions are as follows, and numbers and location of the spatially dependent variables are same as the standard Westinghouse PWR. • Reactor Coolant System hot leg temperature (3 sensors per loop) • Power Range Neutron flux The numbers and location of the spatially dependent variables will be added in the DCD chapter 7. And, all related description in the DCD Tier 1, the DCD Chapter 7 and the Safety I&C Technical Report (MUAP-07004) will be modified to keep consistency. **Impact on DCD:** Following sentence will be added after the first paragraph in Section 7.2.1.3 of the DCD Chapter 7. Spatially dependent sensors that are required for the reactor trip functions are describes as follows and identified in Table 7.2-3. • The reactor coolant hot leg temperature in Table 7.2-3 is measured by the themowell-mounted RTDs installed in each reactor coolant hot leg. The hot leg temperature measurement in each loop is accomplished using three fast-response, dual-element, narrow-range RTDs. The three thermowells in each hot leg are mounted approximately 120 degrees apart in the cross-sectional plane of the reactor coolant piping, to obtain a representative temperature sample. The temperatures measured by the three RTDs are deferent due to hot leg temperature streaming and very as a function of thermal power. The PSMS averages these signals to generate a hot leg average temperature. The hot leg temperature streaming uncertainty is evaluated in the Instrument Setpoint Methodology Technical Report (MUAP-09002). • The high power range neutron flux in Table 7.2.3 is measured by the four power range nuclear instrumentation detectors are installed vertically at the four corners of the core. Each detector assembly consists of an upper half detector and a lower half detector. The average nuclear power and axial core difference can be monitored by using signals form the upper and lower detectors. The average nuclear power signals for the reactor protection functions are dependent on the axial power distributions, but the uncertainty of this effect is only for a conservative direction (increase the average nuclear power output from the detector). Also, the average nuclear power signals are dependent on the radial neutron flux distributions for anomalies occurring in one core quadrant. These anomalies can be detected by the neutron flux detector in that quadrant and by the detectors in the two adjacent quadrants, but may not be detected by the detector in the opposite quadrant. Therefore, to ensure event detection and accommodate, the neutron flux detectors must be operable in all four quadrants. Following sentence will be added after the first paragraph in Section 7.3.1.4 of the DCD Chapter 7. Spatially dependent sensors that are required for the ESF actuation functions are described in Section 7.2.1.3 and identified in Table 7.3-4. Table 7.2-3 and 7.3-4 of DCD Chapter 7 will be changed to keep consistency with the Table 2.5.1-2 and 2.5.1-3 of the DCD Tier 1. The revised Table 7.2-3 and 7.3-4, including same description with Table 2.5.1-2 of the DCD Tier 1, will be included in the mark-up version of the DCD Chapter 7 which will be submitted by the end of March. Section 7.1.2.3 & 7.3.1.4 / A 4.6 of Safety I&C Technical Report

Table 2.5.1-2 Reactor Trip and Monitored Variables

Actuation Signal	Monitored Variables
High Power Range Neutron Flux (Low Setpoint)	Neutron Flux **(1)**
High Power Range Neutron Flux (High Setpoint)	Neutron Flux **(1)**
High Power Range Neutron Flux Positive Rate	Neutron Flux **(1)**
High Power Range Neutron Flux Negative Rate	Neutron Flux **(1)**
Over Temperature ΔT	Reactor Coolant Temperature **(2)**
	Pressurizer Pressure
	Neutron Flux **(1)**
Over Power ΔT	Reactor Coolant Temperature **(2)**
	Neutron Flux **(1)**

Notes:
1: Power Range Neutron flux is a spatially dependent variable due to axial variations.
2. Reactor Coolant System hot leg (3 sensors) are spatially dependent variables.

Table 2.5.1-3 ESF Actuations and Monitored Parameters (Sheet 2 of 3)

ESF Function	Actuation Signal	Monitored Variables
Main Feedwater Regulation Valve Closure	Low Tavg coincident with RT (P-4)	Reactor Coolant Temperature **(2)**
		Reactor Trip (RTB Open)

Note1: Loop A isolation is initiated by steam generator water level signal and main steam line pressure signal from loop A. All loops are identical (e.g., loop B isolation is initiated by the signal from loop B).

Note 2: Reactor Coolant System hot leg (3 sensors) are spatially dependent variables.

Table 2.5.1-3 ESF Actuations and Monitored Parameters (Sheet 3 of 3)

ESF Function	Actuation Signal	Monitored Variables
Block Turbine Bypass and Cooldown Turbine Bypass Valves	Low-Low Tavg	Reactor Coolant Temperature **(2)**
	Manual Actuation	Manual Switch Position (Turbine Bypass Block Switch)

Note 2: Reactor Coolant System hot leg (3 sensors) are spatially dependent variables.

Revise Table 2.5.1-6 (re-numbered from Table 2.5.1-5) to add new ITAAC Item 28 below.

Table 2.5.1-6 RT System and ESF System Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
28. The spatially dependent sensors that are required for protective actions are identified in Table 2.5.1-2 and Table 2.5.1-3.	**28. An inspection of the as-built spatially dependent sensors required for protective actions will be performed.**	**28. The as-built PSMS includes the minimum number and locations of spatially dependent sensors that are required for protective actions as identified in Table 2.5.1-2 and Table 2.5.1-3.**

Impact on COLA
There is no impact on the COLA.

Impact on PRA
There is no impact on the PRA.

14.03.05-31

QUESTION NO.: 14.03.05-31 (ML)	MHI Response (ML)	Additional Information from the NRC Meeting on 2/23/2011	Amendment MHI Response
Address the applicability of GDC 19 to the DCS with respect to an ITAAC to verifying that communications exist that support instruments and controls within the control room to allow actions to be taken to maintain the nuclear power unit in a safe condition during shutdown, including shutdown following an accident. Based on the requirements of GDC 19, the ITAAC should verify that (1) actions can be taken in the control room to safely operate the nuclear power unit under normal conditions, and maintain it in a safe condition under accident conditions, including LOCAs, and (2) adequate radiation protection has been provided to permit access to, and occupancy of, the control room under accident conditions, for the duration of the accident, without personnel receiving radiation exposures in excess of the total effective dose equivalent (TEDE) of 0.05 Sv (5 rem) specified in 10 CFR 50.2. GDC 19 is applicable to the DCS in the US-APWR in that the DCSs have been provided to support instruments and controls within the control room to allow actions to be taken to maintain the nuclear power unit in a safe condition during shutdown, including shutdown following an accident. An ITAAC verifying that the plant can be maintained in a safe condition under accident conditions, including LOCAs, and that adequate radiation protection has been provided to permit access to, and occupancy of, the control room under accident conditions was not found in the Tier 1 documentation.	**ANSWER:** The Control Room is provided with Safety Related HVAC as described in Tier 1 Section 2.7.5.1. The MCR HVAC system is designed to provide conditioning air to maintain the proper environmental condition of the MCR during all plant conditions, including abnormal and accident conditions. Table 2.7.5.1-3 includes ITAAC Item 4 which ensures that the MCR HVAC system will meet its design basis. The Data Communication System (DCS) includes both safety related communications and non safety related communications. The DCS safety related communications hardware is environmentally qualified to withstand the mild environment of the Control Room. MHI concludes that the existing ITAAC are sufficient to verify that the requirements of GDC 19 are met. **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.	Staff Status: No,	Response: MHI will revise the Tier 1 Section 2.5.2 and Tier 2 Section 7.4 for safe shutdown to clarify the applicability of GDC 19. The revised Tier1 Section 2.5.2 and DCD Section 7.4 will be included in the mark-up version of the DCD which will be submitted by March 31. Section 7.4.1.1 & Section 7.4.1.6.2.1

BTP07.21-1

QUESTION NO.: BTP07.21-1 (ML)	MHI Response (ML)	Additional Information from the NRC Meeting on 2/23/2011	Amendment MHI Response
Clearly identify the performance requirements for the US-APWR safety Instrumentation and Control (I&C) system. 10 CFR 52.47 states in part, that the information submitted for a design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC, and procurement specifications and construction and installation specifications by an applicant. It appears within Technical Report MUAP-09021, "US-APWR Response Time of Safety I&C System," the values within Section 3.4 should be changed to state "basis" for the time response requirements, not "assumptions". This should include how each of the following factors was determined, what estimates were made, and what facts ensure that these are the bounding requirements. 1. The values identified in Tables 4.0-1 and 4.0-2 as T1 through T4, and T1, T2, respectively. Clearly explain estimates madeand how they are deterministic. Provide justification that shows that the allocations can be reasonably expected to be satisfied by the plant design. Section 3.4 states "The response time allocated to the individual components (i.e., response time of sensor, digital controller) are based on MHI experience of digital I&C system in Japanese PWR plant" is not acceptable. 2. Using the Figure 4.4-1 and Figure 4.4-2, of MUAP-09005, MELTAC Topical Report, show the calculation of the maximum, and minimum, response time which is equal to each of the digital controller times, T2, by safety function presented in MUAP-09021. a) Identify each of the values, t1 through t10, in Figure 4.4-1 for the MELTAC Fundamental Cycle and how this can vary for each of the safety functions. b) Identify the differences between the typical MELTAC hardware configuration in Figure 4.4-2 and that used to determine each of the response time calculations.	**ANSWER:** MHI agrees to change the word "assumptions" to "basis". Section 3.4 of MUAP-09021 will be revised as the follows. ●The detailed design is required to meet the response time requirement of the digital controller with taken into account the delay time caused by the processor loading, number loading, number of I/O modules, number of controller nodes ●~~The basis of the The response time allocated to the individual components (i.e., response time of sensor, digital controller) in Section 4.1 is provide in Section 4.2. are based on MHI experience of digital I&C system in Japanese PWR plant, where the processor and the I/O loading of the MELTAC platform has been considered. Base on the experience, response time is established with conservative assumptions. Then response time of the platform with actual load will be verified against response time requirement in ITAAC phase.~~ ●The satisfaction of the allocated response time ~~based on MHI experience~~ is verified in water fall design process during basic design and detail design phase through MHI QA program / process, also these design process is verified through V&V process. Then response time of the platform with actual load will be verified against response time requirement in ITAAC phase. ●The basis of the response time allocated to the individual components (i.e., response times of sensors, digital controllers) in Section 4.1 is provide in Section 4.2. ●Some of the expected component response times listed in Tables 4.2-1 and 4.2-2 are based on typical equipment procured for use in the US nuclear power industry. The expected response times will be specified in the applicable procurement documents prior to making the final determination of equipment make and model, and response times will be verified prior to commercial operation. The following description and Figures 4.2-1, 4.2-2 and 4.2-3 will be added in MUAP-09021 Section 4.2. Allocated response times from T1 to T4 in Tables 4.1-1, 4.1-2 are determined as values which envelop the values based on the specifications of the individual equipments that applied to US-APWR. The values based on specifications are addressed in a column "Expected Response Times Based on Specification" in Tables 4.2-1 and 4.2-2. Equipments assigned to sensor part in Tables 4.2-1 and 4.2-2 of MUAP-09021 are transmitters and RTDs of the general vendors in U.S. The response times of these sensors are addressed in the vender specifications and	Staff Status: No, 1. Proposed changes to MUAP-09021, "US-APWR Response Time of Safety I&C System" make references to other TopicalfTechnical reports. Please assure that next revision to MUAP-09021 correctly identifies Topical or Technical designations. 2. Verify that system configurations depicted in Figures 4.2-1 to 4.2-3 conform to the proposed revisions being made to the data communication architecture, such as, addition of priority functions, etc. 3. Would text on page 2-7 and Tables 1 & 2 on subsequent pages be added in MUAP-09021 4. This RAI response proposes significant changes to MUAP-09021. Review of this technical report can be performed expeditiously if all of the information is consolidated in one document. When can we expect next revision of MUAP-09021, which incorporates all of these proposed changes and additions	Response: 1. MUAP-07005 will be changed to Technical Report in the next revision. The description of "MELTAC Topical Report" will be changed to "MELTAC Technical Report" in the next revision of MUAP-09021. 2. Figures 4.2-1 to 4.2-3 in the RAI response is the same system configurations with Figure 4.4-2 in MUAP-07005. 3. MHI will agree to add the text on page 2-7 and Tables 1 & 2 to MUAP-09021. These will be added to the next revision of MUAP-09021. 4. MHI will add the text on page 2-7 and Table 1 & 2 to the next revision of MUAP-09021. MHI will revise the RAI response to add the text on page 2-7 and Tables 1 & 2 to MUAP-09021. The next revision of MUAP-09021 will be reflected the above amended RAI response.

	we can find the vender specifications in each vendor web site. The response times T1 of NIS and RCP Speed are negligible and the bases are described in MUAP-09021 Sections 5.2 and 5.3 respectively. Response time T2 of digital controller based on a specification is determined by adding up response times of individual components based on configurations of equipments utilized for each safety function. The configurations corresponding to each safety function are provided in Figures 4.2-1, 4.2-2 and 4.2-3. The maximum and minimum response times (T1, T2, T3, T4 and T7) of individual equipments are calculated by the method described in Table 4.4-1 of the Topical Report MUAP-07005, "Safety System Digital Platform –MELTAC-,". Response time T3 (0.1 sec) is in accordance with the specification of RTBs. The response time of RTBs applied to Japanese PWR is less than 0.1 sec. RTBs of the same specification will be applied to US-APWR. Also, response time T4 (0.15 sec) is in accordance with the specification of CRDM as addressed in DCD Section 3.9.4.2.1. .... **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.		

BTP07.21-2

QUESTION NO.: BTP07.21-2 (ML)	MHI Response (ML)	Additional Information from the NRC Meeting on 2/23/2011	Amendment MHI Response
With regards to response time guidance of BTP 7-21, a basis should be provided for systems, particularly, that have not been implemented and tested on a full scale basis. In Technical Report MUAP-09021, "Time Response of Safety I&C System," the basis should include, but not be limited to, 1. A description of the effects of adding sensors, divisions, communication links, controller, computer nodes, or actuation devices required to scale the system to full scale or that which was previously built. 2. A description of the cycle which demonstrates that the watch-dog timer is correctly implemented. The description of the WDT cycle time in MUAP-09021 is not that identified in MUAP-09005 nor that identified as typical in BTP 7-21. 3. The time required for the application modules does not exceed the allotted time given in the architecture timing budget, and diagnostics and other support modules will not cause the allotted time to be exceeded. 10 CFR 52.47 states in part, that the information submitted for a design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC, and procurement specifications and construction and installation specifications by an applicant. MHI is requested to provide a basis for systems that have not been implemented and tested on a full scale basis.	**ANSWER:** The following description will be added to the next revision of MUAP-09021 Section 4.2 to address system scalling. The scaling affecting on response time is based on adding I/O card, increasing the number of function symbols in the application software or increasing the number of data communication signals between controllers. Adding I/O modules increases response time T2 of the Bus Master Modules (I/O), as shown in Figures 4.2-1, 4.2-2 and 4.2-3 in the RAI response to QUESTION NO.: 07-21 BTP-1. Increasing the number of function symbols increases response time T3 of the CPU Modules shown in these same figures. Increasing the number of data communication signals increases response time T7 of the Bus Master Modules (Data Link) and/or response time T4 of the Control Network I/F modules shown in these same figures. The exact affect on response time cannot be determined without knowing the exact changes to the system. However, since changes to the licensed configuration of safety systems are typically infrequent and minor, the margin between the expected maximum calculated response times shown in the RAI response to QUESTION NO.: 07-21 BTP-1 and the required response time (ie. minimum 20% margin) is sufficient to accommodate these changes. The description of the watch dog timer in MUAP-09021 Section A.6 will be revised as shown below. ~~"the watch-dog timer is set at the beginning of each cycle and reset after the completion of application module"~~ will be changed to be consistent with the description in the MELTAC Topical Report, MUAP-07005, as follows: "As described in Section 4.4.1 of MUAP-07005, the CPU Module includes a hardware timer and a hardware watchdog circuit which detects overtime if the timer is not initialized within a certain time limit. The timer is initialized by a software process once every cycle of the CPU Module." The hardware timer is set based on the calculation of T3, which is described in the RAI response above. The following will be added to MUAP-09021 Section 4.2. The digital controller MELTAC Nplus, applied to the US-APWR, is a single-tasking system with no interrupt processing. As described in Section 4.4.1 of MUAP-07005, diagnostics and other support processes are included in the periodic cyclic process. All diagnostic and support processes are included in the calculation of T3 described in the RAI response to QUESTION NO.: 07-21 BTP-1.These diagnostic and support processes cannot cause the allotted time for T3 to be exceeded.	Staff Status: No, 1. Proposed changes to MUPA-09021 should include references to RAI responses, e.g., "Adding 1/0 modules increase response time T2 of the Bus Master Modules (I/O), as shown in Figures 4.2-1,4.2-2 and 4.2-3 in the RAI response to QUESTION NO.: 07 21 BTP 1." 2. Statement justifying changes to the licensed configuration of safety systems is not acceptable. Please note that any changes to the licensed configuration requires a formal change process, which would include a new response time calculation assuring that the new response times are bounded by the maximum allocated times. Also, any changes to a safety-related digital I&C system would require repeating of some of the software development lifecycle process that should assure that the design requirements are satisfied.	Response: 1. Primary changes based on RAI responses are identified in the Revision History of the next revision of MUAP-09021. 2. The description in RAI Response to QUESTION NO.: 07-21 BTP-2 will be amended. Statement justifying changes to the licensed configuration of safety systems will be removed. MHI will revise the RAI response to add the text on page 2-7 and Tables 1 & 2 to MUAP-09021. The next revision of MUAP-09021 will be reflected the above amended RAI response.

BTP07.21-4

<table>
<tr><th>QUESTION NO.: BTP07.21-4
(ML)</th><th>MHI Response
(ML)</th><th>Additional Information from the NRC Meeting on 2/23/2011</th><th>Amendment MHI Response</th></tr>
<tr>
<td>In Technical Report MUAP-09021, "US-APWR Response Time of Safety I&C System," MHI is to correct the list of variables in Table 4.0-1 or the list of variables in Table 7.2-3 of the DCD so that they are specifically consistent.

Table 7.2-3 of the DCD provides a list of reactor trip variables, ranges, accuracies, response times, and setpoints. Similarly, Table 4.0-1 in MUAP-09021 provides a list of reactor trip variables and their response times. However, the list of variables in Table 4.0-1 in the MUAP-09021 do not agree with the list of variables in Table 7.2-3 of the DCD. Also, Table 4.0-1 in Technical Report MUAP-09021 includes the ECCS signal, which is not listed in Table 7.2-3. On the other hand, Table 7.2-3 includes high source range and high intermediate range neutron signal, and high positive and high negative rate of change of the power range flux signal. None of these signals are listed in Table 4.0-1 of Technical Report MUAP-09021. In the response to the RAI, MHI is to explain why these were different and which table will be revised.</td>
<td>ANSWER:
Table 7.2-3 of the DCD provides the response times of all RT variables. So, the item ECCS actuation will be added to Table 7.2-3 of the DCD. On the other hand, Table 4.1-1 in MUAP-09021 provides response times of typical RT variables assumed for the transient analyses in Table 15.0-4 of the DCD as described in MUAP-09021 Section 1.2. ECCS is included because it is a reactor trip initiator included in Table 15.0-4. The response times of the variables which are addressed in Table 7.2-3 of the DCD and not addressed in Table 15.0-4 of the DCD, are determined based on the response times of the variables in Table 4.1-1 of MUAP-09021 and Table 15.0-4 of the DCD. The response times T2, T3 and T4 are common to the variables only addressed in Table 7.2-3 of the DCD and the variables addressed in Table 15.0-4 in the DCD. The response time T1 of the variables addressed only in Table 7.2-3 of the DCD also can be decided from response time T1 of the variables which are addressed in Table 15.0-4 in the DCD and have sensors of the same type. Therefore, it should not pose any problem even if list of variables in Table 4.1-1 in MUAP-09021 does not agree with list of variables in Table 7.2-3 of the DCD.

Impact on DCD
The item ECCS actuation will be added to Table 7.2-3 of the DCD as follows.

Table 7.2-3 Reactor Trip Variables, Ranges, Accuracies, Response Times, and Setpoints (Nominal)
(Sheet 2 of 2)
<table>
<tr><td>ECCS Actuation</td><td>Pressurizer Pressure</td><td>1700 to 2500 psig</td><td>2.5% of span</td><td>3.3 sec</td><td>1765 psig</td></tr>
<tr><td></td><td>Main Steam Line Pressure</td><td>0 to 1400 psig</td><td>3% of span</td><td>3.3 sec</td><td>525 psig</td></tr>
<tr><td></td><td>Containment Pressure</td><td>-7 to 80 psig</td><td>2.8% of span</td><td>3.3 sec</td><td>6.8 psig</td></tr>
</table>

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.</td>
<td>Staff Status: No,

1. Proposed changes to the DCD Table 7.2-3 provided 3.3 sec response time for the
ECCS actuation variables. Whereas, for these same variables DCD Table 15.0-4 and
MUAP-09021 Table 4.2-2 allocate 3.0 se.£. response time.
Applicant is being asked to resolve this discrepancy.</td>
<td>Response:

1. The DCD Table 7.2-3 shows the Reactor Trip Variables, Ranges, Accuracies, Response Times, and Setpoints. ECCS Actuation row in the proposed changes to the DCD Table 7.2.-3 means a reactor trip on ECCS actuation. Therefore, proposed to the Table 7.2-3 is consistent with time delay of “ECCS Signal Reactor Trip” in the DCD Table 15.0-4.

MHI believe the proposed change on the DCD Table 7.2-3 is appropriate.</td>
</tr>
</table>

30-Safety I&C

RAI-30 (ML081530754)	MHI Response (ML082390261)	Additional Information from the NRC Meeting on 2/23/2011	Amendment MHI Response
The response time analysis method is presented in Sect. 6.5.3, Response Time Analysis Method. The response time of the safety functions is used in the plant safety analysis. The response time of each safety function is calculated by adding the response time of each component that makes up the system, from the process measurement to the actuation of the final component. • What is the basis for selecting the response times? • What are the uncertainties of the response times? • Any standard or guideline used as a basis for performing the response time analysis? • What is the basis of the response time value used in the plant Safety Analysis Report? • Which statistical value is used for validation of the time response?	**Response:** For sensors, the response time is based on vender specifications with uncertainties added based on operating experience. For MELTAC components, the response time is based on the processing times and the calculation method defined in Section 4.4 of MUAP-07005. This method accounts for all processing time uncertainties. As stated in Section 3.4, the real time performance for the PSMS conforms to BTP 7-21. The response time value used in the Safety Analysis is determined based on historical precedence and engineering judgment. As stated in Section 6.5.3, the actual response time calculation, described in Section 6.5.3, confirms that the Safety Analysis value bounds the actual response time of the PSMS. The statistical methods used during response time validation testing, are described in V&V procedures. These procedures are plant specific documents. For the US-APWR V&V procedures are within the life cycle process, which is covered by an ITAAC.	Staff Status: No, 1. For MEL TAC response time calculation methodology, the RAI response and MUAP-07004-P(R 1) refer to MUAP-07005-P. The staff believes that for the safety systems (including the MEL TAC modules) response time calculations the appropriate methodology document reference is MUAP-09021, "US-APWR Response Time of Safety I&C System." The applicant is asked to consider revising MUAP-07004, "Safety I&C System Description and Design Process" to change reference to MUAP-09021 for safety system response time calculation method.	Response: 1. MUAP-09021 describes the allocation of "Time Delay" for RTS and ESF functions in Table DCD 15.0-4, not the response time calculation methodology. Therefore, MUAP-07005 is the appropriate reference document for the response time calculation for T2. MHI believes that It is not necessary to change reference. No impact on DCD

36-Safety I&C

RAI-36 (ML081530754)	MHI Response (ML082390261)	Additional Information from the NRC Meeting on 2/23/2011	Amendment MHI Response
Section A.5.6.3.3 discusses "The Effects of a Single Random Failure." Does the safety system design preclude the use of components that are common to redundant portions of the safety system, such as common switches for actuation, reset, mode, or test; common sensing lines. And are there any other features which could compromise the independence of redundant portions of the safety system?	**Response:** There are no electrical components that are common to redundant portions of the safety system. Each train is completely electrically independent from each other train. The only shared component that is common to redundant portions of the safety system is the instrument sensing line for reactor coolant flow measurement used for the low reactor coolant flow reactor trip signal. This common instrument sensing line is used for all four flow instruments (i.e., there is a separate flow instrument for each PSMS train). The instrument sensing line extends reactor coolant system pressure to the flow transmitters. A common instrument sensing line is used to obtain accurate pressure for the flow transmitters. In addition the common sensing line is used to minimize penetrations into the reactor coolant system pressure boundary and thereby reduce the potential for small breaks compared to using four separate instrument sensing lines.	Staff Status: No, 1. Provide consistency b/w DCD Tier 1 & 2 and MUAP-07004, on use of common taps. a. This may required discussion throughout documents that detail the impact on sensors, safety functions, restrictions (restrictive setpoints), any deviations from regulations. 2. DCD specifies conformance to RG 1.151-1983 a. Identify any non conformance "if any" in appropriate DCD sections b. Demonstrate compliance to RG 1.151, Position C, Items 1-6 c. Any justifications cited, should have sufficient, substantiated documentation to support justification 3. Provide detail figure that reflects design (common tap, sensor, PSMS, PCMS, DAS, diagnostics and any other impact to design) 4. Clarify use of bypass functions, what type, show on schematics and discuss any operational constrictions that may apply Additional Information: 1. DCD Tier 2, Table 1.9.1-1 states conformance to RG 1.151 Instrument Sensing Lines (Rev. 0, July 1983) with no exceptions identified and references DCD Tier 2, Section 7.1.3.7. Section 7.1.3.7 does not describe the use of common sense line for flow instruments. 2. RG 1.151-1983, C. REGULATORY POSITION The requirements of ISA-S67.02, "Nuclear-Safety Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants," 1980, provide a basis acceptable to the NRC staff for the design and installation of safety-related instrument sensing lines in nuclear power plants subject to the following: ... 3. DCD, Tier 2, Section 7.1.3.7. states "Instrument sensing lines are specified to be protected in compliance with RG 1.151 (Reference 7.1-11) which endorses ISA-S67-02, including freeze protection." 4. Reference 7.1-11 Instrument Sensing Lines, Regulatory Guide 1.151 Revision 0, July 1983. 5. In MUAP-07004, Section 3.3, (12) RG 1.151 Instrument Sensing Lines –endorses ISA-S67.02 Compliance is described in the US-APWR DCD Chapter 7." 6. At the 1/21/11 MHI meeting, MHI (Ken) stated "DCD takes precedence." 7. See items 1,3 & 5 above 8. MUAP-07004, Appendix A, A.5.S.1, MHI states" ANSIIISA S67 .02-1980 endorsed by RG1.151 describes that a single process pipe tap to connect process signals to redundant instruments shall not be used. However, the latest version of the ANSI, ANSIIISA S67.02.01-1999 describes that if a single process connection cannot be avoided, justification shall be provided to permit its use. The common instrument tap on reactor coolant flow measurement of the US-APWR is justified as follows ... " a. MHI proprietary justification does not demonstrate compliance to RG 1.151-1983, items under Section C (e.g., item 1) b. All OCD sections, tables and MUAP-07004 should be consistent with description of common sense lines. Exceptions should be justified, including all analysis, calculations, and operating experience to support justification. 9. 7.4.3.2 Restrictive Setpoints For the US-APWR, the reactor will not be permitted to operate at power, even when one RCS loop is unavailable as evidenced by; low reactor coolant flow conditions, therefore there are no restrictive setpoints. Is there one? Schematic sheet 5 What does operate at power mean? Clarify RCS channel bypass and impact to restrictive setpoint Clarify P-7 and impact to restrictive setpoint	**Response:** The design basis of the US-APWR to relate the comments from the NRC are as follows; 1. There is no shared sensing line, including tap, between the safety sensor for the protection system and the non-safety sensors for the control system. 2. All instrument sensing lines that are connected to ASME Class 1 or 2 process piping or vessels are designed as ASME Class 2 Seismic Category I from their connection to the process piping or vessel to the sensing instrumentation. 3. All instrument sensing lines for the safety sensors are installed in the building area which is controlled by the safety-related HVAC, and the instrument sensing lines for the safety sensors cannot be exposed to freezing temperature environment. 4. All signals from safety sensors of the PSMS to the non-safety systems (PCMS and DAS) are transmitted via isolation devices in the PSMS. 5. There are no electrical components that are common to redundant portions of the safety system. 6. The only shared mechanical component that is common to redundant portions of the safety system is the instrument tap for reactor coolant flow measurement used for the low reactor coolant flow reactor trip signal. 7. The instrument sensing lines after the tap portion for the four train reactor coolant flow sensors are separated each other train. 8. All other mechanical components, including instrument tap and sensing line, in each train safety system is completely physically independent from each other. 9. All setpoints for the reactor trip and the ESF actuation functions are fixed for all operating conditions from the start-up mode to the full power operation mode, and there are no restrictive setpoints. Note) The standard Westinghouse PWR has the P-7 (low RCS flow of 2/4 per loop) and P-8 (low RCS flow of 1/4 per loop) permissive signals. And the setpoint of the P-8 permissive will be changed at the N-1 loop operation condition (normally 10% power, and change to 50% power to allow N-1 loop operation less than 50% power). And, the P-8 permissive is categorized as the restrictive setpoint. But, the US-APWR only has the P-7 permissive signal that applies to the low RCS flow of 1/4 per loop reactor trip logic as describe in Sheet 5, and the setpoint (10% power) of the P-7 permissive signal is fixed for all operating conditions. All related description in the DCD Tier 1 & Chapter 7 and the Safety I&C Technical Report will be modified based on above design basis and will keep consistency. The modified descriptions will be included in the mark-up version of the DCD Tier 1 & Chapter 7 and the Safety I&C Technical Report will be submitted by the end of March. Section 7.1.3.4, 7.1.3.7 & 7.4.3.2 / A 5.6.1 and A 5.6.3.1 of Safety I&C Technical Report

60-Safety I&C

RAI-60	**MHI Response**	**Additional Information from the NRC e-mail on 3/25/2011**	**Amendment MHI Response**
Section 5.1.3, p.54, Operation under Degraded Conditions, discusses the potential failure of all Operational VDUs. How can the operability of Operational and Safety VDUs be verified? At the January 22, 23 meeting with the staff, MHI agreed to add justification for no periodic manual surveillance testing or offer some type of periodic surveillance to confirm Operational VDU is operating correctly. Per IEEE Std 603, Criterion 5.7, Capability for Test and Calibration, "The capability for testing and calibration of safety system equipment shall be provided during power operation." This requirement does allow exceptions under certain conditions. MHI is requested to identify conformance of the Safety VDUs to this requirement.	**Response:** Processors of the Safety and Operational VDUs and their communication capabilities are checked continuously by self-diagnosis. In addition, the integrity of the Safety VDU panel is manually verified periodically by the test shown in Section 4.4.1 of Safety I&C TR MUAP-07004. The following will be added to Section 5.1.3: In the event of complete failure … very infrequent events. Failure of an individual Operational VDU is easily detected by operators, because the Operational VDU is continuously used for plant operation. The ability to detect individual Operational VDU failures and complete failure of all PCMS VDUs is confirmed during HSI validation testing.	**Staff Evaluation:** The NRC staff does not agree with the response. The response does not fully and directly answer the staff question. With regards to the necessity for periodic testing: GDC 18 of 10 CFR 50, Appendix A, requires, in part, that electric power systems important to safety be designed to permit periodic testing, including periodic testing of the performance of the components of the system and the system as a whole. In addition, by staff guidance of Standard Review Plan (SRP) Section 7.7, Control Systems, such as the Operational VDU, should limit the potential for inadvertent actuation and challenges to the safety systems. To limit this potential, the staff believes periodic manual testing, in addition to the self testing of the Operational VDU, should be designed and identified in the TR or, preferably, the DCD. MHI should fully describe its response with sufficient details.	**Response to Staff Evaluation:** <u>GDC 18</u> In case that periodic testing of electric power systems is performed by operational VDU, expected testing operation and indication from operational VDU is confirmed by the test itself. <u>SRP 7.7</u> Inadvertent actuation cannot be tested by periodic manual testing. Potential for inadvertent actuation and challenges to the safety systems can be limited as described in Section 5.1.3 of Safety I&C Technical Report. Also, if operational VDU spurious commands are generated, the priority logic within the application software of the PSMS, ensures that an automatic safety actuation signal generated from within the PSMS has higher priority than any manual control commands received from the O-VDU (See Appendix D of Safety I&C Technical Report). In addition, the operational VDU spurious commands are analyzed in Appendix D of Safety I&C Technical Report. Therefore, MHI believes that periodic manual testing in addition to the self testing of the Operational VDU is not needed. **Impact on DCD** There is no impact on DCD.

Response to Open RAIs

64-Safety I&C

RAI-64 (ML091600322)	MHI Response (ML091751090)	Additional Information from the NRC Meeting on 1/21/2011	Amendment MHI Response
Section 5.1.10, Unrestricted Bypass of One Safety Instrument Channel, states that "The PSMS remains fully functional with the remaining two trains.." The plant may have one channel out of service but not two or they would not be able to have single failure protection and meet GDC 21. This should be explained in the topical report. MHI is requested to provide this explanation in the topical report. GDC 21, Protection system reliability and testability, states "redundancy and independence designed in to the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. "	**Response:** The words "The PSMS remains fully functional with the remaining two trains", means that: (1) When one of four channels is bypassed, the normal 2/4 voting logic will be automatically changed to 2/3 voting logic. (2) If a single failure occurs during the above 2/3 condition, the remaining 2 (two) operable channels are sufficient to achieve the safety function. The following will be added to Section 5.1.10: The PSMS remains fully functional with the remaining two trains, since two channels are sufficient to satisfy the 2-out-of-N voting logic. **Current Status:** Section 5.1.10 was already revised based on above response in the Safety I&C TR, rev.3, September 2009.	Staff Status: No, 1. Provide a list of functions that can be unlimited bypass and identify on schematics 2. Provide clarifying remarks in DCD when 2004 are made concerning using unlimited bypass 3. Provide clarification in both Chapter 7 and 16 of the impact when using unlimited bypass 4. Provide consistency blw descriptions and use of unlimited bypass in DCD Tier 1 & 2 and MUAP-07004 Additional Information: DCD Tier 1, Tier 2 & MUAP-07004 conflicts. Clarify which functions can be in unlimited bypass (also show on schematics). Clarify where DCD states 2004 is met when one channel is in unlimited bypass. DCD Tier 1 2.5.1.1 Design Description A single channel or division can be bypassed to allow on-line testing, maintenance or repair during the plant operation and this capability does not prevent the PSMS from performing its safety function. For many measurement channels and many division level functions, the PSMS can perform its safety function with a single failure and with one channel or division bypass.ed, or with two channels or divisions bypas'sed (but without an additional s.ingle failure). The technical specifications distinguish the functions for which these capabilities are applicable. MUAP-07004, 5.1.10 Unrestricted Bypass of One Safety Instrument Channel Priority portion (last se'ntence) is not clear Clarify impact of unlimited bypass to Chapter 16.	**Response:** The design basis of the US-APWR to relate the comments from the NRC are as follows; 1. The power range neutron flux trip channels must be operable all four quadrants (four channels) as described in the amendment MHI response on for the RAI 07.01-11, so the unlimited bypass cannot be allowed by the T-Spec. 2. Also, the source range neutron flux trip function and the intermediate range neutron flux function consists of two channel sensors, so the unlimited bypass cannot be allowed by the T-Spec. 3. All other trip and ESF actuation functions are consisted by four channel 2-out-of-4 logic, and the single failure criterion can be satisfied in one channel in the bypass condition, so the unlimited bypass of the one channel of each safety function can be allowed by the T-Spec. 4. When one of four channels is bypassed, the 2-ou-of-4 bypass logic will be automatically changed to the 2-out-of-3 logic for reaming three channels (not bypassed channels), and if a single failure occurs, the remaining two operable channel can perform the safety function. 5. If any failure detected in three operable channels by the self-diagnostic function of the RPS at the one channel bypass condition, the 2-ou-of-3 logic will be automatically changed to the 1-out-of-2 logic for remaining two channels (not bypassed and mot failed channels). 6. The automatic bypass management logic in the 2-out-of-4 bypass logic in Figure 7.2-2 continuously checks for multiple bypassed conditions to ensure the minimum redundancy required by the T-Spec. is always maintained as described in Section 7.1.3.11 of the DCD Chapter 7. 7. Additional bypass (second channel bypass) is allowed by the manual administrative control for the several safety channels which are not used for the control system as described in the T-Spec., and the second channel bypass time is limited by the T-Spec. 8. The unlimited bypass described above is only applied for each safety instrument (sensor) channel level to consist the 2-out-of-4 bypass logic. 9. The train level bypass for the RPS, ESFAS or SLS is restricted by T-Spec. LCO. 10. The PRA are analyzed based on the unlimited bypass are allowed for the safety instrument channel, and the bypass time of the RPS, ESFAS and SLS are limited by the T-Spec. LCO. All related description in the DCD Tier 1 & Chapter 7 and the Safety I&C Technical Report will be modified based on above design basis and will keep consistency. The modified descriptions will be included in the mark-up version of the DCD Tier 1 & Chapter 7 and the Safety I&C Technical Report will be submitted by the end of March. Section 7.1.3.11 / Section 5.1.10 of Safety I&C Technical Report

Proprietary

1-MELTAC

QUESTION NO.: 1-MELTAC (ML)	MHI Response (ML)	Additional Information from the NRC on 1/21/2011	Amendment MHI Response
Identify the specific differences in the MELTAC equipment applied for non-safety applications vs. the equipment applied to safety applications. Section 1.0 briefly mentions this as differences "in Quality Assurance methods for design and other software life cycle processes." This difference is also described in compliance to Branch Technical Position 7-19, "Guidance on Evaluation of Defense-in-Depth and Diversity in Digital Computer-Based Instrumentation and Control Systems."	**Response** The MELTAC Basic Software for the non-safety applications was developed according to the Original Quality Assurance Program (QAP), which is based on the Japanese standards, described in Section 6.0. **[**The MELTAC Basic Software for the safety applications includes additional quality assurance activities defined as the U.S. Conformance Program (UCP), described in Section 6.1.7. As a result of the UCP, functions that are not used in safety applications were removed from the Basic Software of the MELTAC platform for safety applications. The differences between the safety and non-safety platform are primarily in system configuration and application software. The Basic Software of the two platforms is essentially the same. Therefore, in the D3 Topical Report, MUAP-07006, it is assumed that a software defect that results in CCF of the MELTAC safety platform also results in a CCF of the MELTAC non-safety platform. **]**	Staff Status: No, MELCO received the comment on RAI-1 and RAI-5 for the MELTAC TR from the NRC, as described below. "MELTAC Technical Report RAI 1 and 5, which pertain to PCMS quality and identification remain open." (The NRC said that "Staff will clarify what they expect from MHI for these items" but we have not yet received any additional information. So we plan to submit a draft response to the above NRC comment, based on our own interpretation.)	Response: Since the QAP and the identification method for the safety MELTAC system and the non-safety MELTAC system are different, the safety platform is not applied to the non-safety platform, and the non-safety platform is not applied to the safety platform. The differences in quality between MELTAC PSMS and PCMS are as below: **[** ・PSMS: App.B-based QAP ・PCMS:MELCO QAP for non-safety items (Complies with ISO 9001) ・PCMS portions with Augmented Quality: 10 CFR 50 Appendix B-pertinent QAP. Details will be provided in the response to Augmented Quality Issue (No5 of Action Items List to resolve overall I&C Issues). **]** **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

Proprietary

5-MELTAC

<table>
<tr><th>QUESTION NO.: 5-MELTAC
(ML)</th><th>MHI Response
(ML)</th><th>Additional Information from the NRC on 1/21/2011</th><th>Amendment MHI Response</th></tr>
<tr><td>Identify how the MELCO internal design documents are marked for the safety and non-safety MELTAC systems.

Section 3.0, Applicable Code, Standards and Regulatory Guidance, (item 62), referencing IEEE 494 1974 (this is also required by IEEE Std 603-1991, Criterion 5.11) states that documents used for internal use do not contain the “Nuclear Safety Related” designation. Also discuss how documents for the non-safety MELTAC system are differentiated from the safety related system.</td><td><u>Response</u>
[The software document titles for the Safety MELTAC contain “MELTAC Nplus-S”, where S means Safety, while the titles for the non-safety (conventional) MELTAC is “MELTAC Nplus.” These titles are applicable to all MELTAC software documents used internally by MELCO.
The hardware components are common for the safety and non-safety MELTAC.
Therefore, there is no distinct identification for the hardware documents inside MELCO.
]</td><td>Staff Status: No,

MELCO received the comment on RAI-1 and RAI-5 for the MELTAC TR from the NRC, as described below.
“MELTAC Technical Report RAI 1 and 5, which pertain to PCMS quality and identification remain open.”

(The NRC said that “Staff will clarify what they expect from MHI for these items” but we have not yet received any additional information. So we plan to submit a draft response to the above NRC comment, based on our own interpretation.)</td><td>Related Document:
(1) Safety System Digital Platform MELTAC (MUAP-07005)

Response:
Since the response to RAI-5 previously submitted has a misleading description, we will revise the response to RAI-5 as described below.

Distinction in designation will be made for all components (hardware, software, documents) of the safety MELTAC system to provide clear identification.
[The following measures will be taken as committed in the previous response.
a) Documents of safety MELTAC system
・The “MELTAC-S” mark that designates the safety system will be included in the title.
・The “Nuclear safety related” designation will be added.

The description related to the measures of document identification in Section 6.2.4 will be revised in the next revision as follows. The underlined portions are added.
The document titles for the safety MELTAC contain "MELTAC Nplus S", where S means safety, while the titles for the non-safety (conventional) MELTAC contain "MELTAC Nplus." These titles are applicable to all MELTAC <u>hardware and</u> software documents used internally by MELCO.

<u>In addition, the unique safety marking of “nuclear safety related” is saliently and prominently put on all document cover sheets for the safety MELTAC platform.</u>

Application specific documentation (eg. cabinet layout and wiring diagrams, technical manuals, etc) for MELTAC Nplus S systems are also marked “nuclear safety related”.

b) Products of safety MELTAC system
・Safety related hardware will be marked as such.
・For safety related software, measures will be taken such as including characters that designate safety in the name of executable files.
The description related to the measures of product identification in Section 6.2.4 will be revised in the next revision as follows. The underlined portions are added.

The unique safety marking is salient and prominently located on all <u>hardware</u> components to avoid human performance errors during all phases of the product life cycle, including design, production, testing, spare parts ordering, stocking and replacement, etc. The unique safety marking is in addition to other less salient and less prominent component markings that are generically applicable to all MELTAC <u>hardware</u> components, safety and non-safety.

<u>For software products of the safety MELTAC platform, the unique identification information is attached to electronic files. The exact identification method is described in MELTAC Platform Basic Software Program Manual (JEXU-1012-1132).</u>

The above identification means will be applied only to the safety MELTAC system and is not applied to the non-safety MELTAC system. For the non-safety MELTAC system, the identification means below will be applied to distinct it from the safety MELTAC system.

c) Documents of non-safety MELTAC system
・The “Nuclear safety related” designation will not be attached.
・The “MELTAC-S” designation will not be included in the document title.</td></tr>
</table>

			・A different document number system from that for the safety MELTAC system will be used. d) Products of non-safety MELTAC system ・The safety designation will not be applied to non-safety hardware. ・For software products of the non-safety MELTAC Platform, the unique identification information will be attached to electronic files.] **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

03/28/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO.700-5406 REVISION 2**
SRP SECTION:	**07.08 – DIVERSE INSTRUMENTATION AND CONTROL SYSTEMS**
APPLICATION SECTION:	**07.08 – DIVERSE INSTRUMENTATION AND CONTROL SYSTEMS**
DATE OF RAI ISSUE:	**02/28/2011**

QUESTION NO. : 07.08-16

In the Technical Report, "Defense-in-Depth and Diversity Coping Analysis," (MUAP-07014) Revision 2, MHI states that they will be adding additional information identified in the technical report as part of a future revision of the DCD. MHI has identified the following changes to be incorporated in a future DCD revision:

Section 3.5.3 "Erroneous Signals," for "(1) Reactor Trip, Turbine Trip and Main Feedwater Isolation," states that the DCD will be revised to reflect a change made from earlier descriptions of the DAS blocking logic for reactor trip, turbine trip and main feedwater isolation that appear in the current DCD revision and MUAP-07006.

Section 3.5.3 "Erroneous Signals," for "(3) Main Steam Line Radiation (N-16) Alarm," states that the blocking logic for the N-16 alarm is not described in the DCD or MUAP-07006 and that the DCD will be revised to add a description.

Section 3.5.3 "Erroneous Signals," for "(5) Low-Low Pressurizer Pressure Alarm," states that this alarm and blocking logic are not described in the DCD or MUAP-07006 and the DCD will be revised to add the alarm details.

The staff requests that the applicant address the above missing information and incorporate or identify where this additional information is located in the DCD.

ANSWER:

The DAS blocking logic for reactor trip, turbine trip and main feedwater isolation has been added to Figure 7.2-2 (Sheet 14 of 21) in DCD Rev. 3.

The description of the blocking logic for the N-16 alarm, and of the low-low pressurizer pressure alarm and blocking logic will be added to DCD Section 7.8.1.1.2.

Impact on DCD

DCD Section 7.8.1.1.2 will be revised as follows:
[MHI will submit the markup of DCD Chapter 7 by the end of March, 2011.]

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

04/06/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO.710-5493 REVISION 2**
SRP SECTION:	**07.09 – DATA COMMUNICATION SYSTEMS**
APPLICATION SECTION:	**07.01, 07.09**
DATE OF RAI ISSUE:	**03/07/2011**

QUESTION NO. : 07-09-23

The staff's 10 CFR 50 review of Chapter 7 is focused on addressing the Secure Development and Operational Environment (SDOE) per RG 1.152. RG 1.152 has been in a process of revision for the past year, with the latest draft (DG-1249 on the NRC's website, ML100490539) having been proposed in June 2010 and presented to the ACRS on February 23, 2011. This revision, along with RG 5.71, will make changes in how 'cyber security' is handled in nuclear power plant safety systems. Specifically, with the issuance of 10 CFR 73.54 and its companion staff guidance, RG 5.71, 'cyber security' is reviewed under Chapter 13 during COL reviews. RG 1.152, Revision 3, and RG 5.71 were discussed at the public meeting on February 23, 2011. MHI currently is committed to Revision 2 of RG 1.152. Staff requests MHI to consider following the updated guidance of the future Revision 3. If MHI agrees, the NRC staff requests MHI to remove all references to cyber security in Chapter 7 DCD and technical reports. Some examples from MHI's submittals for Chapter 7 that references cyber security include: US APWR DCD, Rev 2, Sections 7.1.3.17, 7.7.2.10, and 7.9.2.6; MUAP-07005-P(R6), Section 6.1.6.

ANSWER:

MHI agrees with staff's requests and will follow the updated guidance of RG 1.152, Revision 3.

MHI has removed references to cyber security from DCD Tier 2 Chapter 7 (Sections 7.1.3.1.7, 7.7.2.10 and 7.9.2.6) and has included a new COL item 7.9 (1) in DCD Rev.3. These changes have been submitted as UTR Rev.7.
Also, references to cyber security from Tier 1 Section 2.5.1.1 (Design Description and Table 2.5.1-6 #24) has been removed in DCD Rev.3.

In addition, all references to cyber security in Technical Reports will be removed or the term "cyber security" will be replaced with other words.

Impact on DCD
There is no impact on the DCD.

Impact on COLA
Corresponding change to adding a new COL item 7.9 (1) will be incorporated to R-COLA and S-COLA.

Impact on PRA
There is no impact on the PRA.

REQUEST FOR ADDITIONAL INFORMATION 722-5597 REVISION 5

3/21/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

SRP Section: 07.01 - Instrumentation and Controls - Introduction
Application Section: 07.01 - Instrumentation and Controls - Introduction

QUESTIONS for Instrumentation, Controls and Electrical Engineering 2 (ESBWR/ABWR Projects) (ICE2)

07.01-29

Title 10 CFR 52.47(a)(2) requires, in part, that the application for a design certification contain a final safety analysis report (FSAR) that includes, “A description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished.” NRC staff guidance in Chapter 7 of Standard Review Plan (SRP) specifically states that the design basis should not contain contradictory requirements and the information provided should have one and only one interpretation (e.g.. unambiguous). The language in technical reports, in support of the application-specific US-APWR design approval, should be evaluated and enhanced. There are a number of cases the technical reports use terms, such as *typical, similar to, in general and equivalent*, that are not specific enough to judge applicability to the US-APWR design certification.

For example, there are 43 instances of “typical” descriptions of design features in MUAP-07004. Some do not provide, if or where, the specificity of the feature can be found in Chapter 7 of the US-APWR DCD. Examples would be: hardwired functions on the operator console, non-safety related functions of the PSMS, duplication of controllers for MSI valves, and priority logic.

There are 29 instances of “typical” descriptions of design features in MUAP-07005. Some do not provide, if or where, the specificity is described for the feature in Chapter 7 of the DCD. An example is: Types of inter-divisional communication between safety and non-safety.

The use of the phrase “in general” causes confusion when prefaced to statements such as: “no manual controlled actions in the plant safety analysis” or “complete plant process systems are assigned to one controller” but for US-APWR this is not the case and those cases are specifically identified. Also, configurations are identified that may or may not be applicable to US-APWR. Example from MUAP-07005, section 4.3.2: “The Control Network can also be used to communicate non-safety related data between different divisions including the non-safety system. **This may be between multiple Controllers in different divisions. Or it may be between Operational VDU Processors and multiple Controllers in different divisions**.”

MHI is requested to implement a thorough review of the documents (TRs and DCD) for specificity to the US-APWR Design Certification in descriptions of applications, if and how they are applicable to US-APWR as well as do they provide reference to the specific section of the DCD that describes this feature. Use of terminology should also be reviewed within the individual documents as well as among all documents.

07.01-30

Appendix B to 10 CFR Part 50 provides detailed quality assurance criteria, including criteria for administrative control, design documentation, design interface control, design change control, and most importantly, document control. NRC staff guidance in Chapter 7 of Standard Review Plan (SRP) specifically states that the design basis should not contain contradictory statements, definitions or requirements and the information provided should have one and only one interpretation (e.g., unambiguous). Given multiple technical reports in addition to the DCD, their consistency through rigorous configuration control is important for more effective and efficient staff review.

MHI is requested to implement a thorough review of the documents (TRs and DCD) for consistency in descriptions of applications, how and if they are applicable to US-APWR as well as do they provide reference to the specific section of the DCD. Use of terminology should also be reviewed within the individual documents as well as among all documents.
Examples:

1) Engineering Tool: The definition of this item should be the same through all documents:
   - i) MUAP-07005, Rev. 6, identifies it as “MELENS” which is the “Mitsubishi Electric Total Advanced Controller Engineering Station,” and "MELENS is installed on a non-safety Personal Computer running the Windows Operating System."
   - ii) In JEXU-1012-1132, R2, MELTAC Platform Basic Software Program Manual, the engineering tool is not listed in the definitions but the body of procedure identifies it as “software (what software?) operating on a computer.”
   - iii) MUAP-07004, R.5, P.22, identifies the engineering tool as a personal computer.
   - iv) MUAP-07017, R.3, Definitions, doesn’t commit to it being a PC or software, only that it “has functions aimed at steadier and more efficient software”.

2) Engineering Tool/ Maintenance Network connection
   - i) MUAP-07004, R.5, states “PSMS controllers are normally disconnected from the Maintenance Network, which is the interface between the controllers and the Engineering Tool.”
   - ii) MUAP-07005, R.6, states “The Maintenance Network is permanently or temporarily connected to the controllers in the same safety division.” “The permanent or temporary connection of the Maintenance network and the Engineering Tool is application dependent.” The application is US-APWR; this is an application specific document.

3) In MUAP-07005, R.6, Table 6.1-8, under software loading, the fifth paragraph, first sentence appears to contain an inaccurate statement regarding when software can be loaded. Additionally, this paragraph seems to contradict the sixth paragraph.

4) By letter dated July 10, 2009 (ML091770212) the NRC Office of Nuclear Reactor Regulation (NRR), decided to discontinue review of report, MUAP-07005, due to the quality and technical issues. Therefore, references to Operating Reactors should be removed from this report.

5) If MUAP-07005 is to be changed to a Technical Report, and applicable only to the US-APWR design certification, the title and content should be changed accordingly with all due specificity and references to each applicable section of the US-APWR DCD as well as any topical or technical reports which may reference MUAP-07005.

6) The following terms should be clarified to identify one and only one term to be used consistently throughout the docketed documents to the extent practical:
i) Train vs division
ii) Safety vs safety grade vs safety-related vs Class 1E vs important safety
iii) 2-port, 2 port, 2port-memory or two-port
iv) FROM, F-ROM, ROM
v) Interface or IF, I/F

RAI Response Status

RAI No	Question No	Subject / Issue	Face to face meeting	Draft Submittal
07-01-25	692-5433	Design summary	Discussed at February 22, 23 meeting	March 18
07-01-26	698-5490	Self-diagnostics and TS	Discussed at April 6 meeting	March 18
07-01-27	705-5495	Important to safety and augmented quality	Discussed at April 6 meeting	March 18
07-01-28	720-5539	Embedded digital component	Discussed at April 6 meeting	March 18
07-01-29	722-5597	Specificity and Consistency		
07-06-25	702-5518	Interlock important to safety	Discussed at April 6 meeting	March 18
07-06-26	702-5518	Electro mechanical interlock	Discussed at April 6 meeting	March 18
07-08-16	700-5406	D3 coping analysis	Discussed at April 7 meeting	March 18
07-09-19	701-5229	ISG-04 1.8	Discussed at February 22, 23 meeting	March 18
07-09-20	701-5229	ISG-04 1.8	Discussed at February 22, 23 meeting	March 18
07-09-21	701-5229	ISG-04 1.3	Discussed at February 22, 23 meeting	March 18
07-09-22	701-5229	ISG-04 1.12	Discussed at February 22, 23 meeting	March 18
07-09-23	710-5493	Cyber security	Discussed at April 7 meeting	March 18

Draft RAIs

SPM: XXX-5619, XXX-5624, XXX-5627, XXX-5650, XXX-5659

FMEA: XXX-5662