Appendix H Bases for the Selection of the US-APWR PAM Variables

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~~The US-APWR PAM list provided in the DCD Table 7.5-3 was developed to be in compliance with the guidance of RG 1.97 Rev. 4 and IEEE 497-2002, which is endorsed by RG 1.97 Rev. 4. The US-APWR PAM variables are utilized by 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 the DCD Table 7.5-3.~~

~~Table 3 of RG 1.97 Rev. 3 prescribes a minimum list of Type B, C, D, and E variables to monitor. However, The US-APWR PAM variables are utilized by the performance-based criteria of RG 1.97 Rev. 4 and IEEE 497-2002 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 and the US-APWR PAM lists for these variable types. Additionally, Type A variables were not included in RG 1.97 Rev. 3, 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 the DCD Table 7.5-3 is provided in Tables H.1-1 through H.5-1 for each variable classification type.~~

~~The variables required by 10 CFR 50.34(f)(2)xvii are included in RG 1.97 Rev.3. Therefore, conformance to 10 CFR 50.34(f)(2)xvii are also shown in Tables H.1-1 through H.5-1~~This appendix describes the selection basis of the US-APWR PAM variables. The US-APWR PAM design is discussed in DCD Section 7.5, and PAM list is provided in DCD Table 7.5-3.

Many current operating plants in the US have developed their PAM lists based on the prescriptive list provided in RG 1.97 Rev. 3. However, the more recent guidance intended for new reactors in RG 1.97 Rev. 4 is based on the performance-based approach identified in IEEE 497-2002 which is endorsed by RG 1.97 Rev.4. MHI developed the PAM list based on the guidance in RG 1.97 Rev. 4 and IEEE 497-2002.

IEEE 497-2002 identifies the plant's emergency operating procedures (EOPs) and abnormal operating procedures (AOPs) as sources for determining the required PAM variables for some types of parameters. Typically, the EOPs are derived from an approved set of emergency response guidelines (ERGs). In the case of MHI, the PAM list was developed prior to the existence of an approved set of ERGs for the US-APWR. Therefore, MHI could not use these documents as sources during the development of the PAM list. Instead, MHI utilized the design information in the US-APWR DCD as the primary source material for developing the PAM list following the performance-based criteria in IEEE 497-2002.

In addition, MHI utilized Japanese domestic and US operational experience and emergency procedures, and known differences between current operating plants and the US-APWR design to further refine the PAM list and to ensure it was a complete PAM list for the US-APWR. Since this method is slightly different than the method that is described in RG 1.97 Rev. 4, it is possible that it will be viewed as an alternate method. The selection basis for the Type A, B, C, D, and E variables for the US-APWR are described in detail in the sections below.

As an additional confirmation of the adequacy of the US-APWR PAM list, MHI has performed a comparison of the US-APWR PAM list to the generic list of PAM instrumentation in RG 1.97 Rev. 3. For each of the PAM variable types, MHI has identified the differences between the two lists and provided an explanation of these differences. DCD_07.05-18 S02

H.1 Type A Variables

~~NUREG-1431 Table 3.3.3-1 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. The US-APWR PAM variables are utilized in this list as an initial starting point for the US-APWR Type A PAM list. Then the US-APWR PAM variables are~~ MHI utilized by the performance-based criteria of RG 1.97 Rev. 4 and IEEE 497-2002 to select the ~~specific~~ Type A accident monitoring variables for the US-APWR. IEEE 497-2002 defines Type A variables as follows. DCD_07.05-18 S02

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-related 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.~~ DCD_07.05-18 S02

~~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 the DCD 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 the DCD Subsection 15.1.5 and Feedwater Line Break (FLB) in the DCD 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 the DCD Table 7.5-3.~~ DCD_07.05-18 S02

~~Table H.5-1 compares all of the Category 1 variables (any Type) functions in NUREG-1431 Table 3.3.3-1 to the US APWR Type A variables currently listed in the DCD Table 7.5-3 and summarizes the bases for differences between the Type A variables in the US-APWR 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 the DCD Table 7.5-3.~~
As described above, the Type A variables are based on manual actions. DCD Chapter 15 (the Accident Analysis Licensing Basis) does credit manual operator actions for some accidents. Therefore, any variables that provide information for the operators to perform those credited manual actions would meet the definition of criterion “a” above. The basis for the manual operator action is discussed in DCD Chapter 15 in the relevant section for each of the events that credit a manual action. These credited manual actions are summarized in DCD Table 7.5-5.

Each of the credited manual actions is listed and described below according to the Chapter 15 event.

Inadvertent Decrease in Boron Concentration in RCS (Subsection 15.4.6)
This event is a boron dilution and is classified as an AOO. After the boron dilution begins, the decrease in boron concentration will result in an increase in reactivity. Depending on the Technical Specification Mode of operation that the plant is in at the time of the event, there are different alarms available to prompt the operator that a boron dilution is occurring. The available alarms are the control rod insertion limit alarm, the reactor makeup water flow rate deviation alarm, the boric acid flow rate deviation alarm, the high primary makeup water flow rate alarm, and neutron flux alarms. After receiving one or more of these alarms, the operator would implement the appropriate alarm response procedure (ARP). The ARP requires that the operator verify the occurrence of a boron dilution by monitoring the neutron flux (wide range). To prevent a return to criticality, the operator would take action to terminate the boron dilution event by performing one (or more) of the following actions: closing the charging flow isolation valve, closing the primary makeup water control valve, or stopping the primary makeup water pump. The wide range neutron flux is the PAM variable to be used by the operator to confirm the alarm and ensure that the appropriate manual action is taken. Therefore, the wide range neutron flux meets the criteria for a Type A PAM variable. Then the wide range neutron flux is selected as a Type A PAM variable. Note that the analysis in DCD Subsection 15.4.6 seems to “credit” several main control room alarms. However, these alarms are only credited in the context of determining the amount of time between the occurrence of an alarm and the return to criticality as required by SRP 15.4.6. This does not mean that these alarms, or the associated parameters, are credited as the prompt for the actual manual action. The primary indication that would provide the prompt for the specific manual action to terminate the boron dilution is the wide range neutron flux indication as discussed above.

The US-APWR has several types of neutron detectors ; therefore, the rationale for selection of wide range neutron flux is as follows. There are two source range neutron flux detectors, two intermediate range neutron flux detectors, four power range neutron flux detectors, and two wide range neutron flux detectors. The detectors are provided individually; the detectors for wide range neutron flux are not shared with those of power range neutron flux, intermediate range neutron flux, or source range neutron flux. Therefore, the input signals to the alarms for power range neutron flux or source range neutron flux are not related to the input signal for wide range neutron flux. DCD Table 15.4.6-1 indicates that high source range neutron flux and high power range neutron flux alarms occur during the boron dilution event for certain modes of operation. The power range neutron flux is useful only in Mode 1 and source range neutron flux is useful in the shutdown Modes. On the other hand, wide range neutron flux has the ability monitor the whole range. Therefore, by selecting wide range neutron flux as the PAM variable, it is not necessary to have separate PAM variables for different Modes. Also, the wide range neutron flux monitors are already qualified for harsh conditions, while the other flux indications may not be. The wide range neutron flux is sufficiently sensitive to detect the flux changes associated with the boron dilution over time and also for monitoring the reactivity critical safety function (see Type B section of this RAI response). The US-APWR ERGs utilize the wide range neutron flux for these purposes. DCD_07.05-18 S02

Since the event is a boron dilution, the RCS boron concentration is an important parameter for the operator to determine during this event. However, RCS boron concentration is only obtained by periodic sampling of the RCS. Sampling requires some time to perform and thus there is some delay between the actual measurement and the time at which the operator would obtain the result. If the operator delays action to confirm the RCS boron concentration, the boron dilution will have progressed even further during the delay. For this reason, monitoring of the RCS boron concentration by the operator is not credited as the basis for terminating the boron dilution event.

Rod Ejection Accidents (Subsection 15.4.8)

As stated in DCD Subsection 15.4.8.2, a rod ejection accident is initiated by the failure of a CRDM housing and results in an increase in core reactivity and a distortion of the local power distribution. The increase in local power near the ejected rod could possibly lead to fuel failure. The event will also result in a loss of reactor coolant to containment. The combination of these effects can result in radiation being released to the containment. If the break flow of RCS coolant caused by the rod ejection is large, the containment pressure will increase to the setpoint for automatic actuation of containment spray. The containment spray will mitigate the radiation levels inside containment. Since containment spray automatically actuates, no manual operator actions are needed. The consequences of this scenario are bounded by the LOCA analyses in DCD Subsection 15.6.5. However, for the case where the break flow rate caused by the rod ejection is smaller, the containment pressure may not reach the setpoint for automatic initiation of containment spray. Since automatic action may not occur, manual action may be required. As a result, the DCD Subsection 15.4.8 analysis assumes manual operator actions to actuate containment spray and the annulus emergency exhaust system within 35 minutes. The operators perform these actions based on the containment high range area radiation indication after the containment high range area radiation alarm is initiated as indicated in DCD Table 7.5-5. In addition, containment radiation is monitored continuously following an accident as part of the plant critical safety functions and containment high range area radiation is selected as a Type B PAM variable (see Type B explanation).

Following the containment high range area radiation alarm, the operator will confirm whether or not the containment high range area radiation is above a certain setpoint. If so, the operator would manually actuate containment spray and the annulus emergency exhaust system. Therefore, containment high range area radiation is selected as a Type A PAM variable.

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CVCS Malfunction that Increases Reactor Coolant Inventory (Subsection 15.5.2)
In DCD Chapter 15, this event credited automatic actions rather than manual actions. For this event, no Type A PAM variables are required.

Failure of Small Lines Carrying Primary Coolant Outside C/V (Subsection 15.6.2)
This event concerns the failure of the RCS sample lines or the CVCS letdown line to the demineralizers that results in RCS coolant leakage outside containment. This is a concern because it is a violation of RCS and containment integrity and can result in radiation release to the environment. In the safety analysis in DCD Chapter 15, the RCS sample line failure is the limiting case. The dose analysis in DCD Subsection 15.6.2 credits a manual operator action to isolate the RCS sample line within 45 minutes. The leakage from this line will reduce the water level in the volume control tank (VCT) and require automatic makeup from the CVCS. The frequent operation of the automatic makeup system may be an early indication to the operator that there may be some leakage in the reactor coolant system. There are also alarms that may provide some early indication of the increase RCS leakage. One such alarm is the low VCT water level alarm. However, the low VCT water level alarm, and its corresponding indication, is not considered to be prompt for manual operator action. Upon receipt of the VCT low water level alarm the operator would begin to investigate a possible RCS leak. One of the ways this may be investigated is by a mass balance of the reactor coolant system. As indicated in the discussion in Subsection 15.6.2, the flow rate due to the sample line failure is such that the CVCS can maintain the pressurizer water level. Therefore, the operator would not notice any decrease in pressurizer level (which is already selected as a Type A variable). However, in order to maintain the pressurizer level, the CVCS flow rate will increase to the point where the high charging flow rate alarm occurs. This alarm, or the elevated charging rate prior to alarm occurrence, along with the absence of a decrease in pressurizer level, would indicate to the operator that there may be a break in one of the small lines, such as the sample line. In this case, the operator would take action to isolate the leakage by closing the containment isolation valves associated with this small RCS lines. Once the containment isolation valves are closed, the break would be isolated. With the RCS now intact, the need for additional makeup would cease and the charging flow would decrease. The operator could verify the success of the actions by this decrease in the charging flow rate. Once the containment isolation valves are closed, the event is terminated from the perspective of the Subsection 15.6.2 analysis. For these reasons, MHI considers that the charging flow rate indication to be the primary indication that the operator would use to take the manual actions credited in the Chapter 15 safety analysis. Therefore, the charging flow rate indication meets the criteria for Type A PAM variables. The high alarm setpoint for charging flow rate is set to detect this event based on the assumed leakage outside containment. The setpoint is significantly higher than the normal operating flow rate range such that the alarm would not be expected to occur during normal operations. This eliminates the potential for spurious alarms that may desensitize operators and degrade their ability to respond to the line break event in DCD Chapter 15. On the other hand, the setpoint is not set so high as to allow the line break event in DCD Chapter 15 to occur without detection.

Note that the Chapter 15 analysis assumes the maximum break flow from the RCS sample line in order to maximize the coolant released and thus the dose. If a smaller break or leak were to occur from the RCS sample line, the change in charging flow rate would be smaller and possibly more difficult for the operator to observe. However, due to the size of the line, this smaller break or leak would result in RCS leakage that was within the maximum flow rate in the Technical Specifications. Technical Specification 3.4.13 sets limits on the allowable RCS leakage and requires the operator to take actions if the leakage is outside of these limits.

Very small leaks that are within the Technical Specification limits do not fall under the scope of the event in Subsection 15.6.2 and do not apply to PAM. The event analyzed in Subsection 15.6.2 results in leakage rates that are above the Technical Specification limits and therefore require the operator actions credited in the analysis. It is also important to note that Technical Specification 3.4.15 requires certain instrumentation to be available to detect RCS leakage. This required instrumentation, which is outside the scope of the PAM criteria, would be available to the operator help determine the cause and location of the leakage. The indications in Technical Specification 3.4.15 are in-containment indications, such as containment sump level. The Subsection 15.6.2 event results in leakage outside the containment. However, these Technical Specification instruments would still be helpful to the operator as an increased charging flow rate coupled with no increase in containment sump level would indicate leakage outside of containment. The operator may then take action to close the containment isolation valves of the sample line and confirm the isolation of the leak by the reduction in charging flow.

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MHI would like to note that the discussion of these other indications (available due to Technical Specifications) that would be helpful to the operator is applicable only to very small breaks. As described previously, detection of these small breaks falls under the Technical Specifications and is outside the scope of the accident analysis in Chapter 15 (and therefore PAM Type A).

In summary, the accident in analysis in Chapter 15 assumes maximum break flow at the time of the sample line rupture outside containment. In this condition, a high charging flow alarm is annunciated. After receiving this alarm, the operator would implement the appropriate ARP. The potential leakage path outside containment is limited to the sample and letdown lines. Therefore, when this charging flow alarm is annunciated, prior to determining the location of the break, the isolation of the sample line is performed by closing containment isolation valves according to the procedure. As a result of this operation, the isolation within 45 minutes as described in DCD Subection 15.6.2 is possible. For these reasons, charging flow is selected as the Type A PAM variable to support the operator action.

Radiological Consequences of a SG Tube Failure (Subsection 15.6.3)
A steam generator tube rupture (SGTR) is a PA that results in leakage of reactor coolant from the primary side to the secondary side of the SG. Due to this primary-to-secondary leakage, the response to an SGTR requires numerous operator actions. The DCD Chapter 15 analysis assumes the specific manual operator actions listed below. For each of these manual actions listed below, the operator needs to monitor various variables prior to performing the action. This meets the criteria for Type A PAM variables. The Type A PAM variables associated with each manually assumed action in the DCD Subsection 15.6.3 analysis are also described below.

- *Manual Reactor Trip* – The primary coolant entering the SG contains radioactive N-16 which is monitored by the high sensitivity main steam line radiation (N-16) monitors which alarm in the main control room. The operators will then enter the steam generator tube leakage ARP. In the DCD Subsection 15.6.3 analysis, this is assumed to occur at 2 minutes after event initiation. In reality, the N-16 alarms are highly sensitive and would occur almost instantly following the SGTR. Therefore, the 2 minute time assumed in Chapter 15 is conservative. The SGTR will result in a decrease in pressurizer water level due to the primary-to-secondary leakage. The amount of the decrease will vary with the size of the tube rupture. The operator will attempt to maintain pressurizer water level using charging flow. However, if pressurizer water level cannot be maintained, then the

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operator will manually trip the reactor (and actuate SI) and enter the SGTR EOP. Thus, the operator must be able to monitor pressurizer water level to determine whether to manually trip the reactor. DCD Subsection 15.6.3.4.2 describes the decrease in pressurizer water level as an indication of the event. The SGTR accident analysis for DCD Chapter 15 conservatively shows that the low pressurizer water level alarm occurs within 5 minutes of event initiation. The timing of the occurrence of the pressurizer water level alarm (within 5 minutes) supports the manual operator action time of 15 minutes assumed in DCD Chapter 15 and the alarm is included in DCD Table 7.5-5 for the SGTR event. Therefore, pressurizer water level is selected as a Type A PAM variable. Although the N-16 alarm is assumed to provide an earlier indication of the occurrence of the SGTR, it is not the primary indication which causes the operator to manually trip the reactor. In addition, although the N-16 alarm is available to alert the operators to an SGTR, the operators do not actually monitor N-16 radiation to perform any action and therefore N-16 radiation is not required to be a Type A PAM variable. Note that the SGTR break size analyzed in DCD Subsection 15.6.3 is selected as a bounding case. Larger tube ruptures may result in automatic reactor trip setpoints being reached (such as low pressurizer water level or low pressurizer pressure) and therefore may not require this manual operator action. On the other hand, small tube ruptures may not require a manual reactor trip if pressurizer water level can be maintained by normal charging. Either of these cases is less limiting than the DCD case and therefore not applicable to PAM variable selection since they are not part of the accident analysis licensing basis.

- *Identify and Isolate Ruptured SG* – The next manual operator actions to perform are to first identify and then isolate the ruptured SG. There are several means available to identify the ruptured SG. The primary-to-secondary leakage will result in an increase in the water level in the ruptured SG. Following reactor trip, the main feedwater to all of the SGs will be automatically isolated (on low T_{avg}). When feedwater is isolated, the water levels in the intact SGs will level off whereas the water level in the ruptured SG will continue to increase due to the primary-to-secondary leakage. The differing response of the intact SGs vs. ruptured SG allows for the operator to easily identify the ruptured SG. Thus, the operator must be able to monitor SG water level to determine which SG is ruptured. Therefore, SG water level (narrow range) is selected as a Type A PAM variable. Note that there are several radiation alarms which are also available to assist the operator in identifying the ruptured SG. The N-16 radiation monitors (or the alarm that occurred previously), the main steam line radiation monitors, or the SG blowdown water radiation monitors can be used. However, for the purpose of the DCD Subsection 15.6.3 analysis, these radiation monitors are considered to be backup means to identify the ruptured SG. Since the DCD Subsection 15.6.3 analysis assumes that the ruptured SG can be identified based on SG water level alone, none of the radiation monitors are required to be Type A PAM variables. After identifying the ruptured SG, the operator then manually isolates the ruptured SG by closing the main steam isolation valve, EFW isolation valve, and various other valves as necessary. It is not necessary to monitor any variables in order for the operators to close the valves. If the operator knows which SG is ruptured, then the operator will know which valves to close. The need for monitoring the valve position indications as part of the system status is addressed in the section describing the Type D PAM variables.

- *Cool Down Primary Coolant System* – Eventually the operator will need to depressurize the RCS to reduce the primary-to-secondary leakage (see next operator action). In order to ensure that subcooling is maintained during the depressurization, the RCS temperature must be reduced. In the DCD Subsection 15.6.3 analysis operator action is credited to open the intact SGs' main steam depressurization valves (MSDVs) in order to cool down the RCS. While no specific variables are required to be monitored to open the valves, the operator will need to monitor the RCS hot leg temperature during the cooldown to assess the progress of the cooldown and to determine when to terminate the cooldown. In addition, the operator will need to monitor the main steam line pressures of the intact SGs to verify that SG pressure is decreasing as expected and to ensure that the SG MSDVs have closed properly when the cooldown is terminated. Therefore, both RCS hot leg temperature (wide range) and main steam line pressure are selected as Type A PAM variables. DCD_07.05-18 S02

- *Depressurize Primary Coolant System to Equalize Pressure between Primary and Secondary* – In order to stop the primary-to-secondary leakage, the primary and secondary pressures in the SG must be equalized. This is done by depressurizing the RCS. The DCD Subsection 15.6.3 analysis credits the manual operator action of opening the pressurizer safety depressurization valve (SDV) to reduce the RCS pressure. The RCS depressurization cannot be initiated until the RCS temperature has been adequately reduced. The operator must first verify the RCS hot leg temperature, which is already selected as a Type A PAM variable. Since the goal of this action is to equalize primary and secondary pressure, the operator must carefully monitor both the RCS pressure and main steam line pressures throughout the depressurization process. Therefore, RCS pressure is also selected as a Type A PAM variable. During the depressurization, RCS subcooling will decrease. The operator must monitor the subcooling to prevent subcooling from being lost. If subcooling is lost, the operator must terminate the RCS depressurization. Therefore, RCS degrees of subcooling is selected as a Type A PAM variable. The RCS depressurization will also result in an increase in pressurizer water level. Since pressurizer overfill would complicate the event recovery and could lead to additional coolant loss through the pressurizer safety valves, the operator must terminate the RCS depressurization if pressurizer level becomes too high. Pressurizer water level is already selected as a Type A PAM variable.

- *Terminate Safety Injection Flow* – Even after the primary and secondary pressures are equalized, the primary-to-secondary leakage will continue due to safety injection (SI) flow. SI must be terminated in order to completely stop the primary-to-secondary leakage and terminate this event. SI is terminated based on very specific conditions that are assumed in DCD Subsection 15.6.3.4.2. The SI termination criteria are based on RCS subcooling, EFW flow, SG water level, pressurizer level, RCS pressure. The operator must monitor these parameters in order to determine when it is appropriate to terminate SI. All of the parameters except EFW flow have already been selected as Type A PAM variables. Therefore, EFW flow is also selected as a Type A PAM variable.

As described above, some variables are monitored by the operator in order to take the manual operator actions assumed in the DCD Subsection 15.6.3 analysis. Therefore these variables are designated as Type A PAM variables.

Loss of Coolant Accidents (Subsection 15.6.5)
Although not described in DCD Table 7.5-5, the post-LOCA long term cooling analysis credits a manual operator action to switch from the RV injection mode to the simultaneous RV and hot-leg injection mode as described in DCD Subsection 15.6.5.3.2.3. The reason that this action is not included in DCD Table 7.5-5 is that DCD Table 7.5-5 lists the operator actions in the context of the associated prompting alarms. The manual operator action credited for post-LOCA long term cooling is not based on any alarm or indication; the action is performed purely

based on time. The analysis credits manual operator action to perform the switchover from the RV injection mode to simultaneous RV and hot-leg injection mode at 4 hours after the LOCA occurs. The timing of the operator action is determined by the solubility limit of boric acid in the core. For cold leg break LOCAs, RV injection mode is not effective in flushing the core and hence the boron concentration in the core may increase. This could result in boron precipitation that could interfere with core cooling. By switching one train of SI to hot-leg injection mode, the core will be flushed and the boron concentration will no longer increase. This switchover does not occur automatically and requires manual operator action at (or before) 4 hours. The operator is not required to monitor any variable to perform this action. Instead, the operator will perform the action at the designated time according to the EOPs. This operator action is included in the LOCA related EOPs of currently operating US and Japanese plants and will be required to be included in the US-APWR ERGs. Therefore, no Type A PAM variable is required to support this manual operator action that is credited in DCD Chapter 15. DCD_07.05-18 S02

An additional consideration for the LOCA event is that the refueling water storage pit (RWSP) level is an important indication in some currently operating plants in order to prompt operator action to realign the suction source 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 SI pumps and the CS/RHR pumps is always from the RWSP. Therefore, it is not necessary to confirm the RWSP level to perform any manual realignment during the LOCA event and RWSP level is not included as a Type A PAM variable for the US-APWR.

Main Steam and Feedwater Line Breaks (Subsections 15.1.5 and 15.2.8)
The analyses of the Steam Line Break (SLB) and Feedwater Line Break (FLB) events assume EFW isolation to the faulted SG. In some operating plants, this action is performed manually. However, in the US-APWR, this action is performed automatically by the low steam line pressure signal EFW isolation function. Therefore, there are no manual actions for these events in DCD Table 7.5-5 and therefore there are no instruments required to be Type A PAM variables.

Mitigation of Consequences of AOOs
For selection criterion “b” for Type A, no explicit operator actions based on primary information from PAM instruments are assumed in any AOO analysis. However, SI termination and long-term core cooling from secondary heat sink are necessary to bring the plant to cold shutdown conditions for some AOOs. Operator actions for SI termination and long-term core cooling are already included in the operator actions assumed in the SGTR analysis. Therefore, almost all of the variables that meet selection criterion “b” are the same as those already selected based on criterion “a”. The only variable that is selected specifically for criterion “b” is reactor coolant cold leg temperature, since it is not explicitly used in the SGTR analysis. Therefore, all of the instruments associated with the mitigation of the consequences of AOOs have been included in the Type A PAM list provided in DCD Tables 7.5-3 and 7.5-6.

Summary for Type A PAM
In summary, MHI has used the performance-based criteria from IEEE 497-2002 to select the Type A PAM variables based on the US-APWR accident analysis assumptions in DCD Chapter 15. The SGTR analysis in DCD Subsection 15.6.3 is the key event that determines almost all of the Type A PAM variables. As discussed previously, ERGs for the US-APWR were not available at the time the PAM list was developed. However, the operator actions assumed in DCD Chapter 15 are the same as those used by domestic Japanese plants and

also by many US operating plants. Thus MHI believes that the Type A PAM variables do take into account operational experience and previously existing procedures. For these reasons, MHI believes that the US-APWR Type A PAM variables have been selected in a manner consistent with the intent of IEEE 497-2002 and RG 1.97 Rev. 4. DCD_07.05-18 S02

NUREG-1431 Table 3.3.3-1 provides a generic list of PAM instrumentation for a Westinghouse NSSS plant based on the guidance in RG 1.97 Rev. 3; however, a reviewer's note in NUREG-1431 requires that this table be amended by individual licensees to add all RG 1.97 Type A and Category 1 non-Type A variables to this generic list in accordance with the plant's RG 1.97 Safety Evaluation Report. Therefore the PAM list provided in NUREG-1431 is a minimal list of Category 1 variables (any Type) for a typical Westinghouse NSSS plant. As a final check of the adequacy of the US-APWR Type A PAM variables, MHI performed a detailed comparison of all of the Category 1 variables (any Type) functions in NUREG 1431 Table 3.3.3-1 to the US APWR Type A PAM variables listed in DCD Table 7.5-3. This comparison is provided in Table H.1-1. MHI has provided the bases for the differences between the Type A variables in the MHI PAM list and the Category 1 PAM for a typical Westinghouse 4 loop PWR plant in Table A.

H.2 Type B Variables

MHI utilized the performance-based criteria of RG 1.97 Rev. 4 and IEEE 497-2002 to select the Type B accident monitoring variables for the US-APWR. IEEE 497-2002 defines Type B variables as follows. DCD_07.05-18 S02

> 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 ~~below~~above shall also be included. DCD_07.05-18 S02

~~The plant critical safety functions are those functions necessary to prevent a direct and immediate threat to public health and safety. The following basic critical safety functions are defined in RG 1.97 Rev. 3 and IEEE 497-2002:~~ DCD_07.05-18 S02

- ~~Reactivity Control,~~
- ~~Core Cooling,~~
- ~~Maintaining Reactor Coolant System Integrity,~~
- ~~Maintaining Containment Integrity (including radioactive effluent control).~~

~~Plant safety is accomplished by ensuring that certain parameters related to the plant critical safety functions are not exceeded. The US-APWR Emergency Response Guidelines (ERGs) provide protection of these plant critical safety functions. The ERGs 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 bounding US-APWR Type B PAM variables are selected ensure availability of the variables needed to implement the functional restoration portion of the ERGs as described above. Table H.2-1 describes the bases for the differences between the Type B variables included in the US-APWR PAM list compared to those included in RG 1.97 Rev. 3 Table 3.~~
As described above, the Type B variables are selected based on what the operator needs to monitor plant critical safety functions. The ultimate goal of the plant safety systems is to

prevent an uncontrolled release of radioactive material in order to protect the health and safety of the public. This is accomplished by ensuring that certain parameters related to the plant critical safety functions are not exceeded. The plant critical safety functions are based on the very general PWR design principles of 1) Shutdown, (2) Cooldown, and (3) Contain, where each of these concepts is defined as follows. DCD_07.05-18 S02

- "Shutdown" means that the plant should be subcritical in order to reduce the thermal energy in the core to decay heat levels during 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 RCS.
- "Contain" refers to the integrity of the RCS and containment vessel. Heat should be removed from the containment to the ultimate heat sink.

IEEE 497-2002 defines five critical safety functions as: reactivity control, reactor core cooling, reactor coolant system integrity, primary reactor containment integrity, and radioactive effluent control. MHI uses the same conceptual critical safety functions, but groups them slightly differently in some cases. In MHI's case, the reactor core cooling critical safety function from IEEE 497-2002 is separated into the critical safety functions of core cooling and secondary heat sink. This is done in order to emphasize the importance of the secondary system for maintaining core cooling. If the secondary heat sinks (SGs) are maintained available following an accident, they can be used to help ensuring core cooling. Therefore, the secondary heat sink critical safety function identified by MHI is one aspect to maintain reactor core cooling. In IEEE 497-2002, primary reactor containment integrity and radioactive effluent control are defined as two separate critical safety functions. However, in RG 1.97 Rev. 3, the description for Type B variables in Table 3 (page 1.97-22) includes radioactive effluent control as part of the critical safety function for maintaining containment integrity. Consistent with MHI operational experience and also with the RG 1.97 Rev. 3 definition, MHI has included radioactive effluent control as part of the containment integrity critical safety function. Although IEEE 497-2002 only defines five critical safety functions, the definition for Type B variables indicates that other critical safety functions in addition to these five may be designated. Based on operational experience, MHI has chosen to add a sixth critical safety function for RCS inventory. The purpose of this critical safety function will be described later. Therefore, the six critical safety functions for the US-APWR are identified as follows.

- Reactivity control
- Core cooling
- Secondary heat sink
- RCS integrity
- Containment integrity
- RCS inventory

Although there are some slight differences in grouping as described above, MHI believes that the US-APWR critical safety functions are consistent with those defined in IEEE 497-2002. In addition, the US-APWR critical safety functions are identical to those used in many Japanese and US operating plants. The critical safety functions are described as part of the US-APWR design information in DCD Subsections 7.5.1.4 and 7.8.3.2, and DCD Tables 7.8-1 and 7.8-2. The selection of the Type B PAM variables related to these safety functions, based on the IEEE 497-2002 criteria, is discussed in the sections below.

Reactivity Control

The reactivity control safety function exists to ensure that the reactor is adequately shutdown and the only source of heat to the RCS is decay heat. The most direct way to determine whether the reactor is shutdown is to measure the neutron flux. Following an accident, the operator can monitor the neutron flux at all times in order to verify that the reactor is shutdown. Neutron flux can be measured over the full scale of reactor power, with some detectors monitoring only certain portions of the scale. The wide range neutron detectors allow the operator to measure the neutron flux from full power all the way to post-trip and shutdown levels. Therefore, wide range neutron flux is selected as a Type B PAM variable. DCD_07.05-18 S02

A boron dilution or an inadequate boron concentration in the RCS following an accident can lead to an increase in the reactivity of the core. If unchecked, this could result in a lack of reactivity control. The increase in reactivity will eventually be noticed by the operator via the wide range neutron flux indication, which is Type B as discussed above. It would also be possible to detect a boron dilution or inadequate boron concentration by monitoring the RCS soluble boron concentration. However, the RCS boron concentration is not monitored continuously in the control room; it is obtained by periodically sampling. Therefore, although it remains important for the operator to be aware of the RCS boron concentration for long term reactivity control, the boron concentration is not the primary means to monitor this critical safety function. RG 1.97 Rev. 3 does classify the RCS soluble boron concentration as a Type B PAM variable, but also denotes it as Category 3, which means that it is a backup variable. RG 1.97 Rev. 4 indicates that backup indications do not need to meet the criteria in the guide.

The critical safety functions are monitored after the reactor is shutdown or expected to be shutdown. In this condition, the control rods are expected to be fully inserted and reactor trip breakers are open. EOPS normally included verification of several different types of indications following a reactor trip, including neutron flux, rod position, and reactor trip breakers. The control rod position indications can be monitored by the operators to confirm that the rods are fully inserted to ensure the reactor is shutdown. Control rod position indications were included in RG 1.97 Rev. 3 as Type B PAM variables. However, they were denoted as Category 3, which means that they are backup variables. According to Section B of RG 1.97 R4, the intent of the PAM variable selection is to select those variables which provide the primary indication to the operators. The RG 1.97 R4 does not require every indication used in the EOPs to be PAM variables, since many of the indications are considered backup, etc. Since the EOPs may include multiple indications for verification of reactor trip, MHI selected wide range neutron flux as the PAM variable because MHI believes it is the most useful, and therefore, primary indication of reactor trip. The reason is that there may be situations where control rod position may be a somewhat misleading indication. If all the control rod positions said that a control rod group was fully inserted, but one of the rods of that group was stuck, it may not be a definitive indication that the reactor is shutdown. Neutron flux would be definitive as to whether the reactor was shutdown. For the plant critical safety functions, the reactivity control safety function may be violated even with control rods fully inserted during certain events, such as a boron dilution or a recriticality caused by an RCS cooldown. Therefore, the operator should monitor neutron flux rather than control rod position. For these reasons, MHI selected wide range neutron flux and considered the control rod position a backup indication. MHI believes this approach is the reason that RG 1.97 R3 classified control rod position as “Category 3”. Therefore, the wide range neutron flux remains the primary indication and control rod position indication is not selected as a Type B PAM variable.

Due to the negative moderator temperature coefficients in PWRs, a decrease in RCS temperature can result in an increase in core reactivity. Normally the effect is small, but if a large decrease in RCS temperature occurs, it may have an effect on the reactivity control critical safety function. In order to prevent an uncontrolled cooldown of the RCS, the operators may monitor RCS cold leg temperatures as supporting information for this critical safety function. RCS cold leg temperature is selected as a Type B PAM variable primarily based on the core cooling critical safety function (see discussion below), but is also available as a backup indication to support the reactivity control safety function. DCD_07.05-18 S02

Core Cooling

The core cooling critical safety function exists to ensure that heat can be removed from the core to protect the fuel cladding. Core cooling can be assessed by monitoring the temperatures in the RCS. There are several ways to measure temperature in the RCS. Both RCS hot leg and cold leg temperatures are very useful for the operator to monitor core cooling. Therefore, RCS hot leg (wide range) and RCS cold leg (wide range) temperatures are both selected as Type B PAM variables.

Typically, the temperatures in the core are higher than those in the hot and cold legs. The core exit temperature indication provides a means to measure the temperatures at a point very close to the top of the core. The core exit temperature can then be used to determine if the core cooling critical safety function is satisfied. Therefore, core exit temperature is also selected as a Type B PAM variable.

Another means to ensure that adequate core cooling exists is to maintain subcooled conditions in the RCS. The operator can directly monitor subcooling via the degrees of subcooling monitor. Therefore, degrees of subcooling is selected as a Type B PAM variable. Note that the subcooling indication is actually a calculated value that is determined based on the RCS temperatures and RCS pressure. Therefore, the temperature and pressure indications that feed into the calculation must also be Type B parameters. For this reason, RCS pressure is selected as a Type B PAM variable (RCS temperatures were already selected as Type B as discussed previously).

The operator needs to monitor whether the inventory in the RCS is adequate to keep the core covered. If the core becomes uncovered during an accident core cooling will be lost and the core will heat up rapidly. The reactor vessel water level indication is the most direct way to ensure that the core is covered and thus core cooling is maintained. Therefore, reactor vessel water level is selected as a Type B PAM variable.

Secondary Heat Sink

As described previously, this critical safety function is not explicitly included in the IEEE 497-2002 definition, but is included by MHI to emphasize the importance of the secondary side in protecting core cooling. The secondary heat sink critical safety function exists to ensure that one or more SGs can safely remove heat from the RCS. If this critical safety function is lost, the RCS will heat up and then eventually the core cooling critical safety function will also be violated.

The effectiveness of the secondary heat sink can be monitored by the operators by observing various parameters related to the SGs. The SG water level, both wide and narrow range, indicates whether the SGs have sufficient inventory to remove heat from the RCS. Therefore, SG water level (narrow range) and SG water level (wide range) are selected as Type B PAM variables. Even if SG water level is currently adequate during an accident, the SGs may dry out if they are not continuously supplied with water. Following an accident, EFW flow is the means by which the SGs are supplied with water. The operators monitor the EFW flow to ensure that the SGs will not dry out and cause a loss of secondary heat sink. Therefore, EFW flow is selected as a Type B PAM variable. The EFW to the SGs is supplied from two EFW pits. If the EFW pit levels become low, EFW to an SG may be lost. Therefore, EFW pit level is selected as a Type B PAM variable. DCD_07.05-18 S02

In addition, the SGs will not function as a secondary heat sink if the pressure in the main steam lines is either too low or too high. The operator must monitor the main steam line pressures continuously to ensure proper SG function. Therefore, main steam line pressure is selected as a Type B PAM variable.

RCS Integrity

The RCS integrity critical safety function exists to ensure that the RCS remains intact following an accident. If the RCS remains intact, adequate RCS inventory will be maintained, which serves to ensure core cooling. An intact RCS will also prevent the release of any radioactivity. The most direct way for the operator to monitor RCS integrity is to monitor the RCS pressure. An unexpected decrease in RCS pressure would indicate to the operator that a break or leak in the RCS may have occurred. Therefore, RCS pressure is selected as a Type B PAM variable.

If there is a leak or break in the RCS inside containment, the flow of RCS coolant into the containment will cause an increase in containment pressure. By also monitoring containment pressure, the operator can detect a break or leak in the RCS. Therefore, containment pressure is also selected as a Type B PAM variable.

If RCS coolant is entering the containment through a leak or break, it will eventually drain into the RWSP which is inside containment and fulfills the role of a containment sump. An increase in the RWSP level would indicate to the operators that RCS integrity may have been lost. This is a much more indirect means of monitoring RCS integrity as compared to RCS pressure and containment pressure. However, RWSP water level (wide range) and RWSP water level (narrow range) are selected as Type B PAM variables.

Containment Integrity

The containment integrity critical safety function exists to ensure that the containment does not fail and remains leak-proof to prevent radiation from being released to the atmosphere. The containment is designed to withstand expected internal pressures following an accident. If the containment pressure exceeds its design limits, the containment may ultimately fail. The operators monitor containment pressure to ensure that it remains safely below the design pressures following an accident. Therefore, containment pressure is selected as a Type B PAM variable.

As discussed previously, the containment integrity critical safety function also encompasses the function of radioactive effluent control. The containment is therefore designed with isolation valves that can be closed to keep radiation inside the containment building. In order to ensure that there are no open pathways that could allow radiation to escape, the operator can monitor the valve position of each of the containment isolation valves. Therefore, containment isolation valve positions are selected as Type B PAM variables. Note that some containment isolation valves are check valves. Since they are passive valves that fail in a safe position, the operators do not need to monitor the check valve positions and therefore check valves are excluded from Type B.

As part of the function of radioactive effluent control, it is important for operators to have a means to monitor the radiation levels inside containment. The containment high range area radiation monitor provides the capability for the operator to monitor containment radiation. Therefore, containment high range area radiation is selected as a Type B PAM variable. DCD_07.05-18 S02

Note that PAM variable selection is based on design basis accidents. The containment analyses in Chapters 6 and 15 show that containment remains intact for the limiting design basis accident. Therefore, a beyond design basis accident would be required to breach containment. The leak-tightness of containment prior to an event is confirmed by the Technical Specifications and also testing as required in 10 CFR 50 Appendix J. Since the initial condition is that containment is operable, containment integrity can be maintained by ensuring proper closure of containment isolation valves and monitoring containment pressure, which are Type B PAM variables as described above. Monitoring for additional variables needed to address leakage from challenges such as degraded seals or penetrations is beyond a design basis event and is therefore outside of the scope of PAM.

The radiation monitors outside of containment are already assigned as Type E PAM variables. Therefore, additional Type B PAM variables for this item are not needed.

RCS Inventory

As described previously, the RCS inventory critical safety function is not defined in IEEE 497-2002 but is added by MHI. The RCS inventory critical safety function exists to complement the core cooling and RCS integrity critical safety functions. If the RCS inventory is too low, it will be difficult to ensure adequate core cooling. For some accidents, safety injection (SI) is used to ensure core cooling initially, but may not be needed at later stages of the accident. In order to terminate SI safely, the operators need to ensure that core cooling will be maintained when SI is terminated. In addition to checking variables related to core cooling and the secondary heat sink, the operators also check RCS inventory by monitoring the pressurizer water level.

An excessive RCS inventory can result in a loss of RCS integrity. If the pressurizer is overfilled, water relief through the pressurizer safety valves may occur. To prevent this loss of RCS integrity, the operators monitor the pressurizer water level. Therefore, pressurizer water level is selected as a Type B PAM variable.

Summary for Type B PAM

In summary, MHI has used the performance-based criteria from IEEE 497-2002 to select the Type B PAM variables based on the US-APWR plant critical safety functions. The US-APWR plant critical safety functions are consistent with those defined in IEEE 497-2002 and are documented in DCD Chapter 7. Note that the selection criteria in IEEE 497-2002 also states that critical safety functions from the plant specific EOPs should also be included. As discussed previously, ERGs for the US-APWR were not available at the time the PAM list was developed. However, the plant critical safety functions are based on conceptual safety principles that apply to all PWR designs. The US-APWR plant critical safety functions are the same as those used by domestic Japanese plants and also by many US operating plants. Thus MHI believes that the Type B PAM variables do take into account operational experience and previously existing procedures. For these reasons, MHI believes that the US-APWR Type B PAM variables have been selected in a manner consistent with the intent of IEEE 497-2002 and RG 1.97 Rev. 4.

The US-APWR Functional Restoration Guidelines (FRGs) are being developed to provide protection of the plant critical safety functions described in DCD Chapter 7. 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 return the parameters used for entry conditions back to values sufficient to ensure protection of the plant critical safety function. Therefore, the FRGs utilize the Type B PAM variables to monitor the critical safety functions. DCD_07.05-18 S02

As a final check of the adequacy of the US-APWR Type B PAM variables, MHI performed a detailed comparison between the Type B variables included in the MHI PAM list and those included in RG 1.97 Rev. 3 Table 3. This comparison is provided in Table H.2-1. For each difference, MHI has provided a detailed explanation of the basis for the difference in Table H.2-1.

As described earlier, the US-APWR critical safety functions are also described in DCD Subsections 7.5.1.4 and 7.8.3.2 as part of the design for the safety parameter display system (SPDS) and the diverse actuation system (DAS), respectively. There are some differences between the variables selected for the SPDS and DAS compared to the Type B PAM variables. The SPDS is intended to display key parameters for monitoring the critical safety functions. In some cases, the SPDS includes additional parameters that are useful for the operator but were not selected as Type B PAM variables. For example, the SPDS includes status of reactor trip breakers as a parameter for monitoring the reactivity control safety function. This parameter is included in the SPDS since it may be useful for the operator to confirm that reactor trip has occurred, but it is not necessary to include as a Type B PAM variable since neutron flux provides adequate indication of reactor trip. Therefore, MHI considers the Type B PAM variables to be the primary parameters for monitoring critical safety functions while the SPDS includes all of the primary parameters and some additional secondary parameters for operator convenience. On the other hand, the DAS design is based on the best-estimate approach consistent with BTP 7-19. Therefore, the DAS design does not need to include all of the Type B PAM variables. Only a subset of the Type B PAM variables that are necessary for the monitoring of critical safety functions on a best-estimate basis are selected to be included for the DAS. The details of the DAS are provided in the D3 Topical Report (MUAP-07006) and the D3 Coping Analysis Technical Report (MUAP-07014).

H.3 Type C Variables

MHI utilized the performance-based criteria of RG 1.97 Rev. 4 and IEEE 497-2002 to select the Type C accident monitoring variables for the US-APWR. IEEE 497-2002 defines Type C variables as follows. DCD_07.05-18 S02

> 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. DCD_07.05-18 S02

~~Table H.3-1 describes the bases for the differences between the Type C variables included in the US-APWR PAM list compared to those included in RG 1.97 Rev. 3 Table 3.~~
As described in the criteria above, the US-APWR (and all PWRs) has three fission product barriers. The fission product barriers contain the highly radioactive fission products and therefore prevent radiological releases during an accident. The failure or potential failure of each fission product barrier can be determined by the operator by monitoring certain variables. Each fission product barrier is discussed below. DCD_07.05-18 S02

Fuel Cladding

The cladding around the fuel keeps the fission products separate from the RCS coolant. The detailed fuel cladding design is described in DCD Section 4.2. The information in DCD Section 4.2 provides the design basis for evaluating the IEEE 497-2002 criteria for Type C variables. The cladding is designed to withstand the high temperatures associated with normal operational conditions. The cladding is also designed to withstand even higher temperatures such as may occur during transients or accidents. However, at extremely high temperatures, cladding failure may occur and fission products could be released into the RCS coolant. If the operator can ensure that temperatures remain below a certain threshold, then the integrity of the fuel cladding will be maintained. Although temperatures in the core cannot be directly measured, the temperatures at the core exit can be used to determine if cladding failure is a possibility. Since the operator monitors core exit temperature in order to ensure that the fuel cladding remains intact, core exit temperature is selected as a Type C PAM variable. This Type C PAM variable allows the operator to monitor for a possible failure of the cladding. DCD_07.05-18 S02

If an actual failure of the cladding has occurred, then fission products will be present in the RCS coolant. Monitoring the RCS coolant for radioactivity allows the operator to detect an actual fuel cladding breach. Therefore, radioactivity concentration or radiation level in circulating primary coolant is selected as a Type C PAM variable. It is also possible to obtain detailed analysis of the primary coolant (gamma spectrum) to monitor for an actual fuel cladding failure by periodic sampling. However, this indication was denoted as a Category 3 Type C PAM variable in RG 1.97 Rev. 3. Category 3 means that it is a backup indication and not the primary source of information. RG 1.97 Rev. 4 states that backup indications do not need to meet the criteria in the guide. Core exit temperature and radioactivity concentration or radiation level in circulating primary coolant are the primary indications. Therefore, analysis of primary coolant (gamma spectrum) is not selected as a Type C PAM variable for the US-APWR.

Reactor Coolant System Pressure Boundary

In the unlikely event that the fuel cladding fails, the RCS pressure boundary is the second fission product barrier that prevents the release of radiation to the environment. The RCS pressure boundary is described in detail in DCD Section 5.2. The information in DCD Section 5.2 provides the design basis for evaluating the IEEE 497-2002 criteria for Type C variables. The RCS is maintained at a high pressure. If the pressure boundary fails, the RCS pressure will decrease at a rate dependent upon the size of the break. By monitoring RCS pressure, the operators will be able to identify any failures of the RCS pressure boundary. Therefore, RCS pressure is selected as a Type C PAM variable.

The entire RCS, with the exception of some small sample and letdown lines, is contained within the containment vessel. A breach in the RCS pressure boundary will thus result in the release of high pressure RCS coolant into the containment vessel which is normally maintained at near-atmospheric pressure conditions. This break flow into the containment will result in an increase in the containment pressure at a rate dependent upon the size of the break. By monitoring containment pressure, the operators have an additional means to detect a failure of the RCS pressure boundary. Therefore, containment pressure is selected as a Type C PAM variable.

The release of RCS coolant into the containment vessel will also result in an increase in radiation inside the containment vessel. The increase in containment radiation levels will be especially large if failure of the fuel cladding has also occurred. Containment radiation

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provides another means to monitor for a failure of the RCS pressure boundary. Therefore, containment high range area radiation is selected as a Type C PAM variable.

As described in the section for Type B, RWSP water level can be used to monitor the RCS integrity. However, the criteria for Type C PAM refer to the primary information to monitor fission product barriers. RWSP water level is considered a backup, not a primary, means to detect a failure of the RCS pressure boundary since RWSP water level does not provide direct information on where the fluid is coming from. Therefore, RWSP level is not selected as a Type C PAM variable.

Note that there are some portions of the RCS that are outside containment, such as small sample lines and the letdown line. Failures of these portions of the RCS pressure boundary outside containment would result in a decrease in RCS pressure but would likely not result in an increase in containment pressure or radiation. However, these lines are automatically isolated by the containment isolation function. Containment isolation may occur based on RCS pressure or containment pressure, which have already been identified as Type C PAM variables. The containment isolation valves are part of the reactor coolant pressure boundary. Similarly, an SGTR is a failure of the RCS pressure boundary that would not cause an increase in containment pressure or radiation.

As a backup to the RCS pressure variable, the operator could detect these scenarios by decreases in pressurizer water level or, in the case of an SGTR, increases in SG water level. Pressurizer water level and SG water level are already Type A PAM variables. As indicated in DCD Table 7.5-2, the requirements for Type A PAM variables already meet all the requirements for Type C variables. Since these other PAM variables are also available as backup indications, no additional Type C PAM variables are needed to detect a failure of the RCS pressure boundary. Note that the Type C variable of RCS pressure and the backup Type A variables of pressurizer water level and SG water level do not detect the full range or RCS leak sizes and locations, particularly for small SGTRs. However, Technical Specification 3.4.13 requires monitoring RCS leakage. For RCS leakage that is less than the Technical Specification limit, although the Type C and backup Type A variables may not be able to detect the leak, no action is required in this case anyway. For RCS leakage in excess of the Technical Specification limits, the impact on the current Type C and Type A backup variables would be large enough to be detectable using these existing variables.

Containment Pressure Boundary

The containment vessel is the last fission product boundary that prevents the release of radiation to the environment. The containment pressure boundary is described in detail in DCD Section 6.2. The information in DCD Section 6.2 provides the design basis for evaluating the IEEE 497-2002 criteria for Type C variables. As described in DCD Section 6.2, the containment is designed to ensure leak-tight integrity under normal operation and during postulated design basis accidents that result in high internal pressures, such as LOCAs. The leak-tightness of the integrity is controlled by Technical Specifications 3.6.1, 3.6.2, and 3.6.3. Together, these Technical Specifications ensure that the containment is intact. For this reason, the design basis accidents, such as LOCAs, assume that the containment is intact at the start of the accident. Technical Specification 3.6.4 also requires the containment pressure to be controlled very close to atmospheric pressure. Note that some leakage is expected to occur though. This leakage is controlled by the containment leakage rate testing program in Technical Specification 5.5.16. The testing program in 5.5.16 indicates that the maximum allowable containment leakage rate is 0.1% of containment air weight per day at a containment internal pressure corresponding to the calculated peak containment internal pressure for design basis LOCA. Since leakage less than this amount is allowed by the Technical Specifications during normal operation and during accidents, the PAM criteria for containment integrity only apply for potential or actual breaches of containment integrity that would result in leakages in excess of that allowed by Technical Specifications. DCD_07.05-18 S02

As indicated by Technical Specification 3.6.4, the containment pressure is kept in a range very close to atmospheric pressure and may in fact be negative. In this case, a potential or even an actual breach in containment would not violate containment integrity since only in-leakage would occur. This means that an elevated containment pressure must occur before a potential or actual breach would violate the containment fission product boundary. Therefore, monitoring containment pressure is necessary. As described previously, the containment vessel is designed to withstand high internal pressures. The analyses in Chapter 6 demonstrate that containment integrity can be maintained even during a LOCA event. However, if the containment pressure becomes too high, the containment will eventually be breached or fail. The operator can monitor the containment pressure to determine if the containment pressure is high enough that a potential for containment breach or failure exists. Therefore, containment pressure is selected as a Type C PAM variable (it was previously selected to support monitoring the RCS pressure boundary as well).

Containment pressure is the primary variable to monitor the potential for containment breach. The criteria for Type C PAM variables also indicate that actual breaches should be addressed as well. As discussed previously, the design of the containment is such that no actual breach of containment can occur during a design basis event. In order for the containment internal pressure to be high enough to cause an actual breach to occur, a beyond design basis event would have to occur. If a breach or failure of containment does occur due to such a beyond design basis event, the breach will result in a decrease in containment pressure. This unexpected decrease in containment pressure would be observable by the operator such that containment pressure would still remain the primary indication even during this beyond design basis accident (i.e., beyond the scope of the PAM requirements). The Technical Specification limits allow for leakage from containment up to certain rates. Any leakage rate less than these limits, regardless of the cause, does not apply the PAM criteria for Type C. Similarly, the other requirement for containment integrity in the Technical Specifications ensures that there can be no breach due to a known leakage path being open, such as an equipment hatch, etc. An inadvertently open containment isolation valve could be considered a small breach that would be difficult to detect by containment pressure. However, the containment isolation valve positions are already monitored as Type B PAM variables for the containment integrity critical safety function. So any failure of a containment isolation valve would be detected by the operator while monitoring the containment isolation valve positions. Therefore, there are no actual breaches of containment that require any additional Type C PAM variables.

Note that RG 1.97 Rev. 3 identified containment effluent radioactivity and effluent radioactivity as two Type C PAM variables for the containment fission product barriers. These variables would allow the operator to detect radiation outside of containment. This might indicate that fission products had escaped from containment and could represent a potential breach of containment. However, it is also possible that this radiation outside containment is not caused by a potential breach of containment. One reason for this is that, as discussed previously, there are some portions of the RCS that are outside of the containment pressure boundary and thus bypass the containment fission product barrier. These pathways are automatically isolated by the containment isolation function which may occur based on containment pressure or RCS pressure. The RCS pressure indication is available to monitor for these situations and is selected as a Type C PAM variable for this reason (although it is already a

Type C PAM variable as described previously). In addition, an SGTR may also allow the containment pressure boundary to be bypassed. In the case of an SGTR, the RCS pressure, pressurizer water level, and SG water level are available as backups to detect the SGTR and are Type A PAM variables (RCS pressure is also Type C). Therefore, the containment effluent radioactivity and effluent radioactivity indications are not required as Type C since they are not the primary means to detect these scenarios. DCD_07.05-18 S02

Based on the above reasons, MHI believes that the potential for breaches in the containment fission product barrier are those that are caused by design basis accidents that result in large increases in containment pressure. Therefore, containment pressure is the primary information utilized by the operator and is selected as Type C. There are some scenarios where containment may be bypassed, such that monitoring of RCS pressure is required as Type C (and some other Type A variables are available as backup). The containment effluent radioactivity and effluent radioactivity indications are not required to be Type C. In fact, RG 1.97 Rev. 3 identified these indications as Category 2 Type C which means that they did not have to meet the same qualification requirements as the Category 1 Type C variables (like containment pressure). Then these indications should be considered backup indications. RG 1.97 Rev. 4 states that backup indications do not need to meet the criteria in the guide.

One final point regarding the radiation indications is that the safety analyses in Chapter 15 do consider the release of radiation from containment for offsite dose calculations even though containment integrity is demonstrated to be maintained. These radiation releases are assumed to occur through the known leakage pathways at the leakage rates consistent with the Technical Specifications described above. Since the leakage rate is governed by Technical Specifications, the monitoring of this radiation is not required per the Type C criteria. However, monitoring of radioactivity releases is required as part of the Type E PAM variables. As a result, some of these radiation indications are selected as Type E PAM variables. The selection of the variables necessary to monitor these releases is discussed in the Type E section.

Note that PAM variable selection is based on design basis accidents. The containment analyses in Chapters 6 and 15 show that containment remains intact for the limiting design basis accident. Therefore, a beyond design basis accident would be required to breach containment. The leak-tightness of containment prior to an event is confirmed by the Technical Specifications and also testing as required in 10 CFR 50 Appendix J. Since the initial condition is that containment is operable, containment integrity can be maintained by monitoring containment pressure, which are Type C PAM variables as described above. Monitoring for additional variables needed to address leakage from challenges such as degraded seals or penetrations is beyond a design basis event and is therefore outside of the scope of PAM. The radiation monitors outside of containment are already assigned as Type E PAM variables. Therefore, additional Type C PAM variables for this item are not needed.

Summary for Type C PAM

In summary, MHI has used the performance-based criteria from IEEE 497-2002 to select the Type C PAM variables based on the three fission product barriers used in the US-APWR design. As discussed previously, ERGs for the US-APWR were not available at the time the PAM list was developed. However, the fission product barriers are a standard principle of PWR design and do not depend on the ERGs. The means of monitoring the US-APWR fission product barriers are the same as those used by domestic Japanese plants and also by many US operating plants. Thus MHI believes that the Type C PAM variables do take into account operational experience and previously existing procedures. For these reasons, MHI believes that the US-APWR Type C PAM variables have been selected in a manner consistent with the intent of IEEE 497-2002 and RG 1.97 Rev. 4. DCD_07.05-18 S02

As a final check of the adequacy of the US-APWR Type C PAM variables, MHI performed a detailed comparison between the Type C variables included in the MHI PAM list and those included in RG 1.97 Rev. 3 Table 3. This comparison is provided in Table H.3-1. For each difference, MHI has provided a detailed explanation of the basis for the difference in Table H.3-1.

H.4 Type D Variables

MHI utilized the performance-based criteria of RG 1.97 Rev. 4 and IEEE 497-2002 to select the Type D accident monitoring variables for the US-APWR. IEEE 497-2002 defines Type D variables as follows. DCD_07.05-18 S02

> Type D variables are those variables that provide primary information to the control room operators and are required in procedures and LBD to: DCD_07.05-18 S02
>
> a) Indicate the performance of those safety-related 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-related 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. 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.~~ DCD_07.05-18 S02

~~Another notable departure from the RG 1.97 Rev.3 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-related 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 US-APWR Type D variable list.~~

~~Table H.4-1 describes the bases for the differences between the Type D variables included in the US-APWR PAM list compared to those included in RG 1.97 Rev. 3 Table 3.~~
As described above, there are three criteria identified in IEEE 497-2002 for selecting Type D variables. Each of the criteria is discussed below.

Criterion "a"
This criterion is related to systems that are credited for the mitigation of design basis events. The accident analysis licensing basis is described in DCD Chapter 15 events. Some of the analyses in DCD Chapter 15 credit engineered safety features (ESF) to mitigate the event.

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The safety injection (SI) system is one of these systems credited in DCD Chapter 15. For example, SI is assumed in the main steam line break analysis in DCD Subsection 15.1.5 and the LOCA analysis in DCD Subsection 15.6.5, as well as a few other analyses. The operators need to be able to monitor the performance of the SI system during an accident, which meets criterion “a” for Type D PAM variables. The SI system has four SI pumps and the operator must be able to verify the proper SI pump flow. Therefore, SI pump discharge flow and SI pump minimum flow are selected as Type D PAM variables.

The SI system also includes four SI accumulators. The accumulators are passive devices that inject water when RCS pressure decreases below the accumulator pressure. The accumulators are assumed in the LOCA analysis in DCD Subsection 15.6.5. In order to determine that the accumulators are operating correctly, the operator can monitor the accumulator water level and accumulator pressure. Therefore, accumulator water level and accumulator pressure are selected as Type D PAM variables.

During the LOCA analysis in DCD Subsection 15.6.5, the accumulators help to ensure adequate coolant inventory in the reactor vessel during the time when the SI pumps have not yet started due to the assumed loss of offsite power. The success of the accumulators can therefore also be monitored by the operators based on the reactor vessel water level. Therefore, reactor vessel water level is selected as a Type D PAM variable.

One notable departure is the variable to monitor flow in the low pressure injection system. The accumulators and high head safety injection system in the US-APWR are designed to replace the entire low head safety injection function. Therefore, the low pressure injection system is not part of the US APWR design and this monitoring variable is not applicable to the US-APWR.

The reactor trip system is credited to mitigate almost all design basis events in DCD Chapter 15. Reactor trip results in the rapid insertion of the control rods. The success of the reactor trip can be determined by monitoring neutron flux. Therefore, wide range neutron flux is selected as a Type D PAM variable.

The safety valves on the primary side (pressurizer safety valves) and the safety and relief valves on the secondary side (main steam safety valves and main steam relief valves) can be considered ESF and are credited in some design basis accidents. For example, the turbine trip event in DCD Subsection 15.2.2 credits the actuation of both the pressurizer and main steam safety valves. Note that the Chapter 15 safety analyses do not actually credit the main steam relief valves to mitigate a design basis accident, but their set pressure is such that they would be expected to open prior to the main steam relief valves in events like the turbine trip. The safety or relief valves open at a specific setpoint to relieve pressure, then close again when pressure has decreased. It is not typically necessary to monitor the safety valve position to ensure proper safety valve opening and closure. Instead, the operators can monitor the safety valve performance by monitoring other parameters such as the RCS pressure, temperature, and subcooling for the pressurizer safety valves and monitoring the main steam line pressure for the main steam safety valves. Therefore, RCS pressure, reactor coolant hot and cold leg temperature, degrees of subcooling, and main steam line pressure are selected as Type D PAM variables. However, the primary and secondary safety and relief valve positions provide a very efficient way for the operators to verify system status as will be discussed in criterion “c” and are therefore also selected as Type D PAM variables.

Automatic isolation of the main steam lines (by closure of the main steam line isolation valves), the main feedwater lines (by closure of the main feedwater isolation valves), and emergency feedwater lines (by closure of the EFW isolation valves) are also included as ESF and are credited in some design basis accidents. For example, the steam line piping failure event in DCD Subsection 15.1.5 credits the automatic isolation of the main steam and feedwater lines and the EFW line to the faulted SG. Similar to the safety valves described above, the closure of these valves can be verified by the operators by checking variables other than the valve positions. Main steam isolation can be monitored by the operators using main steam line pressure. MFW and EFW isolation can be monitored by the operators using SG water level and EFW isolation can also be monitored by EFW flow. Therefore, main steam line pressure, EFW flow, and SG water level (both narrow and wide range) are selected as Type D PAM variables. However, the isolation valve positions provide a very efficient way for the operators to verify system status as will be discussed in criterion "c" and are therefore also selected as Type D PAM variables. DCD_07.05-18 S02

Another ESF used to mitigate design basis events is the containment spray system. The main steam line break (inside containment) and LOCA analysis in DCD Subsections 15.1.5 and 15.6.5, respectively, result in the release of coolant to the containment. The mass and energy release for these events are calculated in DCD Chapter 15 to use for the containment pressure and temperature response analysis in DCD Chapter 6. The analyses credit the containment spray system for mitigating the increase in containment temperature and pressure. Containment spray is provided by the four CS/RHR pumps and the operator must be able to verify the proper CS/RHR pump flow. Therefore, containment temperature, containment pressure, CS/RHR pump discharge flow, and CS/RHR pump minimum flow are selected as Type D PAM variables.

Both the SI pumps and the CS/RHR pumps take suction from the RWSP. In order for the pumps to be able to provide the necessary flow credited in the Chapter 15 analyses, the operators need to monitor the RWSP level while using the SI and containment spray systems. Therefore, RWSP water level (wide range) and RWSP level (narrow range) are selected as Type D PAM variables. Note that it is not necessary for the operator to monitor the RWSP temperature. The SI pumps and CS/RHR pumps were designed to ensure adequate net positive suction head (NPSH) even with a RWSP temperature that bounds the temperatures that may occur during a design basis accident. As discussed previously SI flow and CS/RHR flow are available to monitor the performance of these pumps. Therefore, RWSP temperature is not selected as a Type D PAM variable.

Additionally, the SI pumps and the CS/RHR pumps are cooled by component cooling water (CCW) during their operation. The CCW system is an intermediate system that removes heat from important components and transfers the heat to the essential service water (ESW) system via the CCW heat exchangers. The CCW system also provides cooling to the RCPs. Although the RCPs are not required to be available during any Chapter 15 analysis, cooling of the RCP seals is necessary to prevent a seal leakage LOCA from occurring. As a result, it is necessary for the operator to be able to monitor the CCW and ESW systems following an accident. The CCW system provides cooling to many plant components. The flow to each component can be monitored by the operator. However, this requires the operator to check each component separately. The CCW flow to each component comes from a common header (there are two headers that each supply flow to two trains, although the headers can be cross-connected). Since the valves in the CCW system are normally aligned to cool the components, flow to the component is ensured if there is adequate pressure in the CCW headers. This means that the operator can verify the proper status of the CCW system by

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checking the pressure in the two CCW headers. This is much simpler than verifying the individual flow to each component. Therefore, CCW header pressure is selected as a Type D PAM variable. Additional justification for this selection is as follows:

- Prior to any design basis accident, the CCW system is operable with pressure, temperatures, and flows in their normal ranges
- Following the occurrence of a design basis accident, various CCW valves are automatically repositioned (e.g., to direct cooling flow to CS/RHR heat exchanger during a LOCA)
- The system pressure indicates that the system is operating successfully because the system is a closed loop system (i.e., a pump failure or break in the system would result in a pressure decrease
- The monitoring of flow to individual components receiving CCW flow is not necessary because appropriate flow can be assumed by verification of proper header pressure along with the automatic valve repositioning
- The appropriate functioning of individual components receiving CCW flow is monitored by that component's own indications (e.g., SI flow is selected to monitor the proper functioning of the SI pump, which is cooled by CCW)

Similarly, the proper status of the ESW system can be verified by checking the pressure in the ESW headers. Therefore, ESW header pressure is selected as a Type D PAM variable. Note that there are other parameters that are available for monitoring the performance of the CCW and ESW systems. For example, some operating plants select CCW temperature and flow together, as was done in RG 1.97 Rev. 3. However, MHI has once again selected the Type D PAM variable based on the desire to use the indication that the operators would use as the primary indication of the performance of the system. MHI has selected the header pressure based on an operational perspective after discussing with plant operators. The conclusion is that CCW header pressure and ESW header pressure give the operator the most immediate and accurate indication of the performance of the system of any available indication. For any condition which would degrade the performance of the CCW or ESW systems, such as partial system blockage, the effect on header pressure would be immediately indicated to the operator. A loss of header pressure would indicate to the operator that there is a problem with the system and allow the operator to recognize the potential impact on the safety systems that are supported by CCW and ESW (e.g., SI). Other conditions, such as heat exchanger fouling, are bounded by the design of the systems. Then it is not necessary to include additional parameters, such as temperature and/or flow, to meet criterion "a". Therefore, the CCW and ESW header pressures are the primary indications that should be selected based on this criterion and are selected as Type D PAM variables.

Another ESF used to mitigate design basis events is the emergency feedwater (EFW) system. The loss of AC power analysis in DCD Subsection 15.2.6 is one of several Chapter 15 events that credit the EFW system for event mitigation. The EFW system has four EFW pumps, with two of the pumps being motor-driven and the other two being turbine-driven. In the US-APWR design, one EFW pump is aligned to each SG. Regardless of which type of EFW pump is used, the operator must be able to verify that the pump is supplying the proper EFW flow to the SG. Therefore, EFW flow is selected as a Type D PAM variable. The EFW pumps take suction from two EFW pits. In order for the pumps to be able to provide the necessary flow credited in the Chapter 15 analyses, the operators need to monitor the EFW pit level while using the EFW system. Therefore, EFW pit water level is selected as a Type D PAM variable.

As described in Section 9.1.3.1, the cooling portion of the spent fuel pit cooling and purification system (SFPCS) performs a safety-related function to maintain the spent fuel pit (SFP) temperature within the appropriate range. Following certain design basis accidents, such as a loss of offsite power (LOOP), the operators must monitor the SFP conditions to ensure that the cooling function of the SFPCS is not degraded. In the scenario where the cooling function continues to degrade, the SFP water could eventually boil, resulting in a loss of SFP inventory that could result in fuel uncovery. This could result in the release of radiation. The analyses for the design basis accidents in Chapter 15 do not analyze this scenario as SFP temperature is assumed to be maintained in an acceptable range throughout the analysis period. The operators can monitor the SFP pump discharge flow to determine if cooling flow is being provided. Maintenance of SFP pump flow following a design basis accident is one indication that indicates the success of the SFP cooling function. Therefore, SFP pump discharge flow is selected as a Type D PAM variable. However, following a LOOP, the SFP pumps trip on an undervoltage signal. For this reason, it is also necessary for the operators to directly monitor the SFP temperature when no active cooling flow is provided. Therefore, SFP temperature is selected as a Type D PAM variable. In addition, SFP level may decrease due to evaporation, especially under conditions of elevated SFP temperatures. Therefore, SFP water level (narrow range) is also selected as a Type D PAM variable. DCD_07.05-18 S02

Criterion "b"

This criterion is related to the performance of systems necessary to achieve and maintain safe shutdown conditions. DCD Section 7.4 describes the systems that are used to perform normal and safe shutdown for the US-APWR. The control of these systems allow operators to transition to and maintain hot standby, transition to cold shutdown through hot shutdown, and maintain cold shutdown. In DCD Section 7.4, safe shutdown is achieved using only safety-related I&C systems. For the purpose of this criterion, the safe shutdown systems are used to select the Type D PAM variables. DCD Table 7.4-2 provides a list of systems and associated instruments used for normal and safe shutdown. Note that some of the instruments used for safe shutdown can be considered backup indications. This is because the Type D PAM variables selected using this criterion were considered a subset of the safe shutdown instruments in DCD Table 7.4-2.

The first safe shutdown system is the safety injection system (SIS). The (SIS) can be used to maintain RCS inventory and provide RCS boration in order to ensure sufficient shutdown margin. In the US-APWR, the SIS performs these functions during safe shutdown rather than the CVCS. (These functions are performed by the high head injection system and emergency letdown system, which are parts of the SIS.) DCD Table 7.4-2 indicates several instruments are available to monitor the performance of the SIS. Of these instruments, SI pump discharge flow and SI pump minimum flow are the primary instruments used by the operators. Therefore, SI pump discharge flow and SI pump minimum flow are selected as Type D PAM variables. The SI pump discharge pressure and SI pump suction pressure can be considered backup indications. In addition, the SIS also interfaces with the refueling water system (RWS) since the SI pumps take suction from the RWSP. For this reason, the RWS is included in DCD Table 7.4-2. The RWS can be monitored using the RWSP water level (wide range) instrument. Therefore, RWSP water level (wide range) is selected as a Type D PAM variable. The SIS also includes the SI accumulators. When the accumulators are not required, they must be isolated prior to reaching the low RCS pressures characteristic of shutdown conditions. Isolating the accumulators prevents inadvertent actuation. Successful isolation can be verified by accumulator pressure as indicated in DCD Table 7.4-2. Therefore, accumulator pressure is selected as a Type D PAM variable.

The next safe shutdown system is the nuclear instrumentation system (NIS). The NIS is used by the operators to verify the shutdown margin and ensure the shutdown state by monitoring

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the neutron flux. The primary indication for the NIS is the wide range neutron flux. Therefore, wide range neutron flux is selected as a Type D PAM variable.

Next, the long-term core cooling in the RCS must be maintained. As indicated in DCD Table 7.4-2, the operators can verify long-term core cooling by monitoring the RCS temperatures. Both RCS hot leg and cold leg temperatures are very useful for the operator to monitor long-term core cooling. Therefore, RCS hot leg (wide range) and RCS cold leg (wide range) temperatures are both selected as Type D PAM variables.

The operator also needs to monitor whether the inventory in the RCS is adequate. If necessary, the SI system provides makeup flow to maintain RCS inventory. The pressurizer water level indication provides a means to monitor RCS inventory. Therefore, pressurizer water level is selected as a Type D PAM variable. It is also necessary to ensure RCS integrity during long-term core cooling. RCS pressure and pressurizer pressure can both be used for this purpose, as indicated in DCD Table 7.4-2. However, only one of the indications is needed, since the other will be available as a backup. For this reason, RCS pressure is selected as a Type D PAM variable (pressurizer pressure is considered the backup instrument).

Safe shutdown conditions also require long-term heat removal from the RCS. As discussed in DCD Subsection 7.4.1.6.2.2, long-term heat removal can be accomplished by steam release from the SGs to the atmosphere, by providing EFW to the SGs, and by using the RHR system (when conditions allow). Therefore, DCD Table 7.4-2 includes the RHR system (RHRS), the EFW system (EFWS), the main steam system (MSS), and the condensate and feedwater system (CFS) as safe shutdown systems. Each of these systems is discussed as follows.

For the EFWS, the operator must be able to verify that the EFW pumps are supplying the proper EFW flow to the SGs in order to remove heat from the RCS. Therefore, EFW flow is selected as a Type D PAM variable.

In order for the pumps to be able to provide the necessary flow in the long-term, the operators need to monitor the EFW pit level while using the EFW system. Therefore, EFW pit water level is selected as a Type D PAM variable. EFW pump discharge pressure is included in DCD Table 7.4-2, but is not selected as Type D PAM since it is considered a backup indication to EFW flow.

For the CFS, sufficient water level must be maintained in the SGs in order to efficiently remove heat. The operators can verify the SG water level on the wide range. Therefore, SG water level (wide range) is selected as a Type D PAM variable.

For the MSS, the SGs will only function properly if they are maintained at the appropriate pressure. If the pressure is too low or too high, the SGs will not be able to remove enough heat from the RCS. The operators can monitor the main steam line pressure to ensure proper operation of the SGs. Main steam line pressure also confirms that steam flow from the SGs is occurring as expected. Therefore, main steam line pressure is selected as a Type D PAM variable.

Long-term heat removal may also be provided by the RHRS. RHR cooling is typically used as the plant approaches cold shutdown conditions. The operator must be able to verify the RHR system flow to ensure long-term cooling at cold shutdown. Therefore, CS/RHR pump discharge flow and CS/RHR pump minimum flow are selected as Type D PAM variables. From an operational perspective, the RHR system flow gives the operator the most immediate and accurate indication of the performance of the RHR system. A loss of RHR flow would indicate to the operator that there is a problem with the system and allow the operator to recognize the potential impact on long-term heat removal. In addition, the operator monitors the success of long-term heat removal by monitoring the hot and cold leg RCS temperatures as primary information. Since the hot and cold leg RCS temperatures are already selected as Type D PAM variables to satisfy criterion "b" as discussed above, the RHR heat exchanger outlet temperature is not required to provide primary indication of the performance of the RHR system. For these reasons, RHR heat exchanger outlet temperature and the other RHRS instruments listed in DCD Table 7.4-2, except for the pump discharge flow and minimum flow, are considered backup indications and are not selected as Type D PAM variables. DCD_07.05-18 S02

As described previously in the criterion "a" discussion, the essential service water system (ESWS) and component cooling water system (CCWS) are supporting systems for the SIS. DCD Table 7.4-2 indicates that several instruments are available to monitor the ESWS and CCWS under safe shutdown conditions. However, as described previously, the header pressures for these two systems are the primary means for the operator to monitor these two systems. The other indications, including flow, are considered backup. Note that this is not inconsistent with the previous selection of CS/RHR flow described above. The reason for the difference is that the CS/RHR flow is indicative of system performance since the primary function is to supply containment spray. The header pressure is indicative of CCW and ESW system performance and flow indication would not be sufficient to ensure system function. Therefore, ESW header pressure and CCW header pressure are selected as Type D PAM variables.

Criterion "c"

This criterion is related to checking the status of safety systems. Since many safety systems are credited in the Chapter 15 accident analysis, many of the variables for verifying system status have already been selected as Type D variables as part of criterion "a" above. There are a few additional variables to consider though due to criterion "c".

The US-APWR is designed such that containment isolation valves will automatically close when containment pressure increases to a certain setpoint. This safety feature is described in DCD Subsection 7.3.1.5. The operator verifies this automatic action by checking the position of the containment isolation valves. Therefore, containment isolation valve positions are selected as Type D PAM variables. Note that some containment isolation valves are check valves. Since they are passive valves that fail in a safe position, the operators do not need to monitor the check valve positions and therefore check valves are excluded from Type D. The US-APWR has another set of containment valves known as the containment purge isolation valves. Similar to the containment isolation valves described above, the operator will check the containment purge isolation valve positions to verify the system status. As a result, the containment purge isolation valve position indications are selected as Type D PAM variables.

The US-APWR safety systems are designed to be powered by the Class 1E ac busses. As described in DCD Subsection 8.3.1, the Class 1E ac busses are normally powered by the offsite power supply. In the event where the offsite power supply is lost, the Class 1E ac busses can also be powered by Class 1E gas turbine generators. The US-APWR also has Class 1E dc busses. The Class 1E dc busses are used to provide continuous power for controls, instrumentation, and dc motors as described in DCD Subsection 8.3.2. Since these power sources are necessary for the safety systems to function properly, the operators must be able to monitor the status of the power sources. Therefore, the status of standby power and other energy sources important to safety (Class 1E ac bus voltage and Class 1E dc bus voltage) is selected as a Type D PAM variable. DCD_07.05-18 S02

The US-APWR has a main control room (MCR) isolation function as described in DCD Subsection 7.3.1.5. This function automatically switches the MCR HVAC system to pressurization mode when the radiation levels outside the MCR are high. This design feature reduces the dose to MCR personnel during design basis accidents. The MCR isolation function results in the repositioning of several dampers in the MCR HVAC system. The operator can use the MCR HVAC damper position indications in order to verify the status of the MCR HVAC system. Therefore, these MCR HVAC damper positions meet criterion “c”. As a result, the MCR HVAC damper position indications are selected as Type D PAM variables. Note that these indications are selected to allow the operator to monitor the performance of the MCR HVAC system in limiting the MCR operators exposure to radiation. The actual monitoring of radiation the control room is discussed in the section on Type E PAM variables.

It is noted that the MCR process monitors (i.e., MCR outside air intake radiation detector) is located in the process lines, outside of the MCR. Therefore, when the MCR is isolated by the MCR isolation with high MCR outside air intake radiation, the radiation level of the process monitor will be high. This does not tell the operator whether the MCR HVAC system is operating correctly or not, it would only indicate the need for the MCR HVAC system to operate. In a similar way, the MCR area monitor is located in the MCR. The radiation level in the MCR depends on the specific accident event so that the radiation level inside the MCR will not always be low when the MCR isolation is initiated. Therefore, both the MCR process monitor and area monitor are not reliable for monitoring for the achievement of the safety function. In addition, MCR isolation will occur upon ECCS actuation as described in DCD Chapter 7. ECCS actuation can occur for events where there is no significant radiation. In those cases, the radiation detectors would not provide any information regarding the MCR HVAC status. Instead, MHI added the MCR HVAC Damper Position as the Type D variable for the direct measurement of the system. Therefore, both MCR process monitor and area monitor are not assigned as Type D variables, only Type E.

In similar manner to the MCR explanation above, the TSC process monitor (i.e., TSC outside air intake radiation detector) is located in the process lines, outside of the TSC. Therefore, when the TSC is isolated by the TSC isolation with high TSC outside air intake radiation, the radiation level of process monitor will be high. This does not tell the operator whether the TSC HVAC system is operating correctly or not, it would only indicate the need for the TSC HVAC system to operate. In a similar way, the TSC area monitor is located in the TSC. The radiation level in the TSC depends on the specific accident event so that the radiation level inside the TSC will not always be low when the TSC isolation is initiated. Therefore, both the TSC process monitor and area monitor are not reliable for monitoring for the achievement of the isolation function. In addition, the TSC and the function in the TSC is categorized as non-safety, therefore, there is no safety function to monitor. Therefore, both TSC process monitor and area monitor are not assigned as Type D variables, only Type E.

The US-APWR reactor coolant system is equipped with pressurizer safety valves and safety depressurization valves. These valves provide the capability to relieve excessive pressure in the primary system. The safety valves open and close automatically, so it is not typically necessary to monitor the valve positions. The safety depressurization valves are normally closed and only opened by operator action, so it is expected that the operator would not need to check the valve positions. A pressurizer safety valve or safety depressurization valve that did not properly open or close would result in an impact on RCS pressure and other primary system parameters, such as temperature or subcooling, that could be detected by the operator. However, in order to very quickly and simply verify the system status, the operator could just monitor the valve position indications. Therefore, these position indications meet criterion "c" for Type D PAM variables. As a result, the pressurizer safety valve and safety depressurization valve position indications are selected as Type D PAM variables. DCD_07.05-18 S02

The US-APWR main steam supply system is equipped with main steam safety, relief, and depressurization valves. These valves provide the capability to relieve excessive pressure in the main steam lines. The safety and relief valves open and close automatically, so it is not typically necessary to monitor the valve positions. The depressurization valves are normally closed and only opened by operator action, so it is expected that the operator would not need to check the valve positions. The main steam system also contains main steam isolation valves and other associated valves that are automatically closed on a main steam isolation signal to isolate steam flow from the steam generators as described in DCD Subsection 7.3.1.5. Any of the main steam isolation valves or a main steam safety, relief, or depressurization valve that was in the incorrect position would result in an impact on main steam line pressure and other secondary system parameters that could be detected by the operator. However, in order to very quickly and simply verify the system status, the operator could just monitor the valve position indications. Therefore, these position indications meet criterion "c" for Type D PAM variables. As a result, the main steam isolation valve (and associated valve) position indications, main steam safety valve position indications, main steam relief valve position indications, and main steam depressurization valve position indications are selected as Type D PAM variables.

The US-APWR main feedwater system is equipped with main feedwater isolation valves. These valves and other associated valves are automatically closed on a main feedwater isolation signal to isolate feedwater flow to the steam generators as described in DCD Subsection 7.3.1.5. Any of the main feedwater isolation valves that were in the incorrect position would result in an impact on SG water level and other secondary system parameters that could be detected by the operator. However, in order to very quickly and simply verify the system status, the operator could just monitor the valve position indications. Therefore, these position indications meet criterion "c" for Type D PAM variables. As a result, the main feedwater isolation valve (and associated valve) position indications are selected as Type D PAM variables. In a similar way, the emergency feedwater system is equipped with emergency feedwater valves. These valves are automatically closed on an emergency feedwater isolation signal and are automatically opened on an emergency feedwater actuation signal as described in DCD Subsection 7.3.1.5. The emergency feedwater actuation signal also results in the automatic closure of several associated valves that are not closed on the main feedwater isolation signal but help to ensure the SGs are isolated. Although the impact of an incorrect valve could be detected by secondary system parameters, the operators can verify the system status using the valve positions, thus meeting criterion "c". As a result, the emergency feedwater isolation valve (and associated valve) position indications are selected as Type D PAM variables.

As discussed in criterion "b", the RHR system is used to provide long-term cooling at cold shutdown. Under the temperature and pressure conditions associated with cold shutdown, it is important to protect against overpressure conditions in the RCS. The pressurizer safety valves, described as part of criterion "a", are not available in these conditions. Therefore, the US-APWR has a Low Temperature Overpressure Protection system which consists of CS/RHR pump suction relief valves that are capable of mitigating pressure transients during cold shutdown. Monitoring the valve positions provides a very efficient and simple way for the operators to verify the system status. As a result, the CS/RHR pump suction relief valve position indications are selected as Type D PAM variables. DCD_07.05-18 S02

Summary for Type D PAM

In general, the Type D PAM variables are checked during the performance of EOPs. In many US and Japanese plants, the status of many systems are checked at the very beginning of the EOPs, usually in the first EOP entered. This is based on operating experience. Consistent with this operating experience, MHI has developed the list of Type D PAM variables so that they can be quickly checked during the EOPs to ensure systems are operating as expected, in accordance with the IEEE 497-2002 criteria. For the US-APWR, the majority of these systems are checked in the entry procedure E-0, Reactor Trip or Safety Injection. A few systems related to maintaining long-term safe shutdown conditions are checked during later performance of FRGs. During the audit in February 2012, the NRC staff was able to look at the current US-APWR E-0 and FRGs and verify that the Type D PAM variables were used to monitor the status of safety systems as expected. This provided additional confirmation that the Type D PAM variables were selected in a manner consistent with the intent of IEEE 497-2002 and RG 1.97 Rev. 4.

As a final check of the adequacy of the US-APWR Type D PAM variables, MHI performed a detailed comparison between the Type D variables included in the MHI PAM list and those included in RG 1.97 Rev. 3 Table 3. This comparison is provided in Table H.4-1. For each difference, MHI has provided a detailed explanation of the basis for the difference in Table H.4-1. One notable difference that was discussed previously was that the US-APWR does not have a low pressure injection system and therefore has no Type D PAM variables associated with this system. Another notable departure from the RG 1.97 Rev. 3 Type D variable list involves the chemical volume and control system (CVCS). The SI system (the high head injection system and emergency letdown system) of the US APWR has a required safety function to ensure a means for boration and RCS inventory control if the normal CVCS is unavailable. The ability of the SI system to perform this function is monitored by indications such as SI flow, RCS pressure, pressurizer water level, and RWSP water, which are all selected as Type D PAM variables as described in detail above. 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 are not included in the MHI Type D PAM variable list.

H.5 Type E Variables

MHI utilized the performance-based criteria of RG 1.97 Rev. 4 and IEEE 497-2002 to select the Type E accident monitoring variables for the US-APWR. IEEE 497-2002 defines Type E variables as follows. DCD_07.05-18 S02

> Type E variables are those variables 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 H.5-1 describes the bases for the differences between the Type E variables included in the US-APWR PAM list compared to those included in RG 1.97 Rev. 3 Table 3.~~
As described above, there are four criteria identified in IEEE 497-2002 for selecting variables related to monitoring radiological releases. Each of the criteria is discussed below. DCD_07.05-18 S02

Criterion "a"
The US-APWR is designed to prevent the release of radioactive material in general. However, radioactive material may be released during some accidents. The radioactive materials are released through known pathways, depending on the accident. By monitoring these pathways, the operator will be able to determine if/when radioactive materials have been released.

For some accidents that result in radioactive material release, the radioactive material will be released into the containment vessel. For example, a LOCA will result in the release of primary coolant into containment. Since the primary coolant may be radioactive, especially if fuel damage has occurred, the radiation levels in the containment will increase. Since the containment vessel is one pathway where radiation may be released, the operators monitor the radiation in the containment. The containment high range area radiation indication is the best means available for the operator to monitor containment radiation. Therefore, containment high range area radiation is selected as a Type E PAM variable.

A SGTR results in primary-to-secondary leakage. This allows radioactive primary coolant to enter the main steam system. As a result, the main steam system is another pathway for radioactive material release. The main steam lines are monitored for radiation as described in DCD Subsection 11.5.2.2.4. Therefore, main steam line radiation is selected as a Type E PAM variable.

Steam from the main steam system is condensed in the condenser. Non-condensable gases are removed from the condenser and vented using condenser vacuum pumps. The operators monitor the exhaust from these pumps for radiation as described in DCD Subsection 11.5.2.4.2. In the case of an SGTR, the exhaust radiation may be high due to the primary-to-secondary leakage. Therefore, condenser vacuum pump exhaust line radiation (including high range) is selected as a Type E PAM variable.

Similar to the condenser exhaust fan, the gland seal system (GSS) exhaust fan also removes non-condensable gases from the secondary side. A SGTR may result in high radiation in the GSS due to the primary-to-secondary leakage. The operators monitor the GSS exhaust for radiation as described in DCD Subsection 11.5.2.4.3. Therefore, GSS exhaust fan discharge line radiation (including high range) is selected as a Type E PAM variable.

Another pathway for radioactive material release is the plant vent. The plant vent receives the discharge from the containment purge, auxiliary building, control building, fuel building, and the condenser air removal filtration system. Radioactive materials that are released in any of

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these buildings will be collected by the HVAC system in the building and directed to the plant vent for release. The plant vent is equipped with several different radiation monitors corresponding to different radiation ranges as described in DCD Subsection 11.5.2.4.1. The combination of these monitors allows the operators to monitor radiation during normal operation and following accidents. Therefore, the plant vent radiation gas radiation (including high range) is selected as a Type E PAM variable.

In addition, the plant air vent high concentration sampling system is also selected as a Type E PAM variable.

There are no other release points in the US-APWR.

Criterion "b"
Following an accident that results in radiological releases, the operators and plant staff need to be able to monitor the local environmental conditions. The spread of radioactive materials will very much depend on the wind speed and direction and other meteorological parameters, such as estimation of atmospheric stability. Therefore, these parameters are selected as Type E PAM variables. However, the actual instruments to measure these parameters will be very site specific. The location for placing these monitors will also highly depend on the site specific plant location. Therefore, the description of the instruments to measure these parameters cannot be part of the standard design of the US-APWR. Instead, this information must be provided by the COL applicant as described in DCD Subsection 7.5.1.1. This COL applicant requirement is described in COL Item 7.5(1). As a result, MHI has selected the Type E PAM variables based on criterion "b" for the standard plant design. However, MHI expects that the COL applicants will provide additional information regarding the selection of these meteorological parameters as Type E PAM variables based on criterion "b" in their PAM lists.

Criterion "c"
The operators need to monitor the plant environs to determine radiation levels and radioactivity following an accident. Therefore, plant and environs radiation is selected is a Type E PAM variable. Plant and environs radioactivity is also selected as a Type E PAM variable. Note that this instrumentation is portable.

Another way in which the operators can monitor the plant environs is by indications of airborne radio halogens and particulates. This indication is provided by portable sampling and then is analyzed onsite. Therefore, airborne radio halogens and particulates (portable sampling with onsite analysis capability) is selected as a Type E PAM variable.

Criterion "d"
Operators respond to a plant accident from the main control room (MCR). The US-APWR is designed such that the operators can shutdown the plant to cold shutdown from the MCR as described in DCD Subsection 7.4.1.1. In order to protect the operators from radiation, the US-APWR includes design features to prevent radioactive materials from entering the MCR. (The selection of PAM variables for these design features was discussed in the Type D section.) Although these design features exist, the operators still need to be able to monitor the radiation levels in the MCR in order to ensure their safety. Any radiation monitors for the MCR would thus meet criterion "d".

Radiation within the control room itself is monitored by the MCR area radiation indication. Therefore, MCR area radiation is selected as a Type E PAM variable.

The HVAC system for the MCR has an air intake that could possibly be a pathway for radioactive materials to enter the MCR. This pathway is monitored using the MCR outside air intake radiation indication. Therefore, MCR outside air intake radiation is selected as a Type E PAM variable. DCD_07.05-18 S02

The criterion also identifies other areas where actions for plant recovery may need to be performed. As discussed previously, the US-APWR can be safely shutdown from the MCR (or also the remote shutdown room). However, during an accident, the MCR operators may consult with the staff in the technical support center (TSC). Although actions are not performed in the TSC, it is also necessary to monitor the radiation in the TSC to protect TSC staff from radiation. Therefore, TSC area radiation is selected as a Type E PAM variable.

Similar to the MCR, the TSC also has an air intake where radioactive materials could enter the TSC. Therefore, TSC outside air intake radiation is selected as a Type E PAM variable.

The MCR and TSC are the only areas where actions for plant recovery are expected. There are no other specific areas where planned actions for plant recovery are required. For this reason, no other specific area radiation indications are selected based on criterion “d”. However, depending on the accident scenario, it is possible that plant personnel may have to enter some areas. In those cases, radiation exposure will be monitored by using portable radiation monitors and air sampling. Those portable instruments are already selected as Type E PAM variables based on criterion “c” as discussed above.

Summary for Type E PAM

In summary, MHI has used the performance-based criteria from IEEE 497-2002 to select the Type E PAM variables based on the US-APWR design information in the DCD. As discussed previously, ERGs for the US-APWR were not available at the time the PAM list was developed. However, the method of monitoring radioactive releases in the US-APWR is the same as those used by domestic Japanese plants and also by many US operating plants. Thus MHI believes that the Type E PAM variables do take into account operational experience and previously existing procedures. For these reasons, MHI believes that the US-APWR Type E PAM variables have been selected in a manner consistent with the intent of IEEE 497-2002 and RG 1.97 Rev. 4.

As a final check of the adequacy of the US-APWR Type E PAM variables, MHI performed a detailed comparison between the Type E variables included in the MHI PAM list and those included in RG 1.97 Rev. 3 Table 3. This comparison is provided in Table H.5-1. For each difference, MHI has provided a detailed explanation of the basis for the difference in Table H.5-1.

Miscellaneous Requirements

As discussed previously, RG 1.97 Rev. 3 provided a prescriptive list of PAM variables. Some of the variables included in RG 1.97 Rev. 3 were placed on the PAM list in order to satisfy requirements related to TMI. These TMI-related requirements are defined in 10 CFR 50.34(f). By utilizing the RG 1.97 Rev. 3 PAM list, some of the TMI-related requirements were also met at the same time. For example, 10 CFR 50.34(f)(2)(xvi) requires that containment hydrogen concentration be displayed in the control room and the RG 1.97 Rev. 3 PAM list includes containment hydrogen concentration as a Type C variable. Since MHI utilized the performance-based approach in RG 1.97 Rev. 4 and IEEE 497-2002, some of the variables related to these TMI requirements, such as containment hydrogen concentration, were not

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included in the US-APWR PAM list. However, MHI does address the TMI-related requirements elsewhere in the DCD. Tier 2 Chapter 1 Table 1.9.3-2 provides the reference to the appropriate section in the DCD that addresses each of the TMI-related requirements from 10 CFR 50.34(f). Table 1.9.3-2 indicates that the containment hydrogen concentration requirement is met as described in DCD Subsection 6.2.5. Therefore, MHI is in compliance with all of the TMI-related requirements of 10 CFR 50.34(f).

IEEE 497-2002 and RG 1.97 Rev. 4 also briefly address the concept of a common mode or common cause failure of instrumentation channels. The use of identical software in redundant instrumentation channels is acceptable as long as an analysis has been performed to demonstrate defense-in-depth against common cause failure. MHI has performed a detailed defense-in-depth and diversity (D3) analysis as described in DCD Section 7.8. The review of MHI's D3 design and analysis is being addressed as part of the review of DCD Section 7.8 and is therefore outside the scope of the PAM list.

Summary

MHI has developed the PAM list in DCD Table 7.5-3 based on the performance-based approach identified in IEEE 497-2002 and RG 1.97 Rev. 4. IEEE 497-2002 indicated that the plant procedures (EOPs and AOPs) can be used as source documents for the performance-based approach. However, the plant procedures are not the only source documents. Other source documents identified in IEEE 497-2002 include plant accident analysis licensing basis, plant critical safety functions (which are defined in IEEE 497-2002), and design basis documentation (of fission product barriers). MHI did not have US-APWR specific procedures available at the time the PAM list was developed. Therefore, MHI did not use EOPs or AOPs as source documents. However, MHI did use the other source documents as described in IEEE 497-2002. In the case where IEEE 497-2002 recommended using procedures as the source documents, MHI used other design basis documentation as the source documents. In addition, MHI utilized the operating experience and knowledge of US and Japanese plants during the process. As a result, MHI believes that the PAM list has been selected in a manner consistent with the intent of IEEE 497-2002 and RG 1.97 Rev. 4.

Table H.1-1 Basis for Differences between NUREG-1431 Table 3.3.3-1 and the US-APWR Type A PAM List (Sheet 1 of 12)

RG 1.97 Function	Purpose	NUREG-1431 Table 3.3.3-1 Variable	Corresponding US-APWR Type A PAM Variable	Basis for Difference
Reactivity Control	Indication of subcritical conditions	Power Range Neutron Flux	Wide Range Neutron Flux~~-~~	Wide range neutron flux is used to confirm the appropriate operator action to terminate a boron dilution following an alarm associated with the boron dilution. Therefore, the selection criterion a) of IEEE 497-2002 is applicable. This is a Type A variable for the US-APWR. Note that wide range neutron flux covers the full power range as well as extending below the power range.~~There are no credited manual actions prompted by indications of subcritical conditions and no credited manual actions that require monitoring subcritical conditions. Wide Range Neutron Flux is a Type B and D variable for the US-APWR.~~

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Table H.1-1 Basis for Differences between NUREG-1431 Table 3.3.3-1 and the US-APWR Type A PAM List (Sheet 2 of 12)

RG 1.97 Function	Purpose	NUREG-1431 Table 3.3.3-1 Variable	Corresponding US-APWR Type A PAM Variable	Basis for Difference
Reactivity Control	Indication of subcritical conditions	Source Range Neutron Flux	-	Neither selection criteria a) nor b) of IEE 497-2002 are applicable to this parameter because there are no manual actions based on this parameter in the safety analysis. Therefore, this is not a Type A variable for the US-APWR.~~There are no credited manual actions prompted by indications of subcritical conditions and no credited manual actions that require monitoring subcritical conditions.~~
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 used to determine when to terminate the RCS cooldown and when to initiate RCS depressurization in the SGTR analysis. Therefore, the selection criterion a) of IEE 497-2002 is applicable. This is a Type A variable for the US-APWR.~~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.~~

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Table H.1-1 Basis for Differences between NUREG-1431 Table 3.3.3-1 and the US-APWR Type A PAM List (Sheet 3 of 12)

RG 1.97 Function	Purpose	NUREG-1431 Table 3.3.3-1 Variable	Corresponding US-APWR Type A PAM Variable	Basis for Difference
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 the safety analysis. However, monitoring of this parameter is necessary for cooling down after mitigating an AOO. Therefore, the selection criterion b) of IEE 497-2002is applicable. This is a Type A variable for the US-APWR.~~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. This is a Type A variable for the US-APWR.

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Table H.1-1 Basis for Differences between NUREG-1431 Table 3.3.3-1 and the US-APWR Type A PAM List (Sheet 4 of 12)

RG 1.97 Function	Purpose	NUREG-1431 Table 3.3.3-1 Variable	Corresponding US-APWR Type A PAM Variable	Basis for Difference
Core Cooling	To ensure RCS inventory	Reactor Vessel Water Level	-	Neither selection criteria a) nor b) of IEEE 497-2002 are applicable to this parameter because there are no manual actions based on this parameter in the safety analysis. Therefore, this is not a Type A variable for the US-APWR. Note that RV Water Level is a Type B and D variable for the US-APWR.~~This parameter is not applied in the safety analysis. RV Water Level is a Type B and D variable for the US-APWR.~~

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Table H.1-1 Basis for Differences between NUREG-1431 Table 3.3.3-1 and the US-APWR Type A PAM List (Sheet 5 of 12)

RG 1.97 Function	Purpose	NUREG-1431 Table 3.3.3-1 Variable	Corresponding US-APWR Type A PAM Variable	Basis for Difference
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)	-	Neither selection criteria a) nor b) of IEE 497-2002 are applicable to this parameter since the US-APWR RWSP is located inside containment and does not require manual action to switch over to the recirculation sump. Therefore, this is not a Type A variable for the US-APWR. Note that RWSP level is a Type B and D variable for the US-APWR.~~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.~~

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Table H.1-1 Basis for Differences between NUREG-1431 Table 3.3.3-1 and the US-APWR Type A PAM List (Sheet 6 of 12)

RG 1.97 Function	Purpose	NUREG-1431 Table 3.3.3-1 Variable	Corresponding US-APWR Type A PAM Variable	Basis for Difference
Maintaining Containment and RCS Integrity; RCS Pressure Boundary	Indication of containment integrity function	Containment Pressure	-	Neither selection criteria a) nor b) of IEE 497-2002 are applicable to this parameter because there are no manual actions based on this parameter in the safety analysis. Therefore, this is not a Type A variable for the US-APWR. Note that Containment Pressure is a Type B, C, and D variable for the US-APWR.~~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	-	Neither selection criteria a) nor b) of IEEE 497-2002 are applicable to this parameter because there are no manual actions based on this parameter in the safety analysis. Therefore, this is not a Type A variable for the US-APWR. Note that C/V Isolation Valve Position is a Type B and D variable for the US-APWR.~~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.~~

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Table H.1-1 Basis for Differences between NUREG-1431 Table 3.3.3-1 and the US-APWR Type A PAM List (Sheet 7 of 12)

RG 1.97 Function	Purpose	NUREG-1431 Table 3.3.3-1 Variable	Corresponding US-APWR Type A PAM Variable	Basis for Difference
Containment Radiation; RCS Pressure Boundary	Identify challenge to fission product barrier	Containment Area Radiation (High Range)	Containment High Range Area Radiation~~-~~	No difference. Containment high range area radiation is used to prompt manual operator actions during a rod ejection event. Therefore, the selection criterion a) of IEEE 497-2002 is applicable. This is a Type A variable for the US-APWR. ~~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. Pressurizer level is also used to prompt manual operator actions during a steam generator tube rupture event. Therefore, the selection criterion a) of IEEE 497-2002 is applicable. This is a Type A variable for the US-APWR.

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Table H.1-1 Basis for Differences between NUREG-1431 Table 3.3.3-1 and the US-APWR Type A PAM List (Sheet 8 of 12)

RG 1.97 Function	Purpose	NUREG-1431 Table 3.3.3-1 Variable	Corresponding US-APWR Type A PAM Variable	Basis for Difference
Secondary System; RCS Pressure Boundary	Verification of heat sink availability	Steam Generator Water Level (Wide Range)	-	Neither selection criteria a) nor b) of IEEE 497-2002 are applicable to this parameter because there are no manual actions based on this parameter in the safety analysis. SG narrow range level is applied in the safety analysis instead of this parameter. Therefore, this is not a Type A variable for the US-APWR. Note that SG Wide Range Level is a Type B and D variable for the US-APWR.~~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.~~

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Table H.1-1 Basis for Differences between NUREG-1431 Table 3.3.3-1 and the US-APWR Type A PAM List (Sheet 9 of 12)

RG 1.97 Function	Purpose	NUREG-1431 Table 3.3.3-1 Variable	Corresponding US-APWR Type A PAM Variable	Basis for Difference
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 sufficient water to maintain long-term core cooling to mitigate AOOs and PAs. Therefore, neither selection criteria a) nor b) are applicable to this parameter because there are no manual actions based on this parameter in the safety analysis. Therefore, this is not a Type A variable for the US-APWR. Note that EFW Pit Water Level is a Type B and D variable for the US-APWR.~~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.~~

Table H.1-1 Basis for Differences between NUREG-1431 Table 3.3.3-1 and the US-APWR Type A PAM List (Sheet 10 of 12)

RG 1.97 Function	Purpose	NUREG-1431 Table 3.3.3-1 Variable	Corresponding US-APWR Type A PAM Variable	Basis for Difference
Core Cooling; Fuel Cladding Integrity; Maintain RCS Integrity; RCS Pressure Boundary; Primary Coolant System	Indication of core cooling	Core Exit Temperature – Quadrant [1]-[4]	-	Neither selection criteria a) nor b) are applicable to this parameter because there are no manual actions based on this parameter in the safety analysis. Therefore, this is not a Type A variable for the US-APWR. Note that Core Exit Temperature is a Type B and C variable for the US-APWR.~~This parameter is not applied in the safety analysis. Core Exit Temperature is a Type B and C variable for the US-APWR.~~
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.

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Table H.1-1 Basis for Differences between NUREG-1431 Table 3.3.3-1 and the US-APWR Type A PAM List (Sheet 11 of 12)

RG 1.97 Function	Purpose	NUREG-1431 Table 3.3.3-1 Variable	Corresponding US-APWR Type A PAM Variable	Basis for Difference
Secondary System	Verification of manual action for SGTR termination (along w/ RCS Pressure)	-	Main Steam Line Pressure	This parameter is used to determine when to terminate the RCS cooldown and when to initiate RCS depressurization in the SGTR analysis. Therefore, this is a Type A variable for the US-APWR.~~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 used to identify the ruptured SG in the SGTR analysis. This parameter is also monitored by the operator to determine if the ECCS termination criteria are met in the SGTR analysis. Therefore, this is a Type A variable for the US-APWR.~~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.~~

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Table H.1-1 Basis for Differences between NUREG-1431 Table 3.3.3-1 and the US-APWR Type A PAM List (Sheet 12 of 12)

RG 1.97 Function	Purpose	NUREG-1431 Table 3.3.3-1 Variable	Corresponding US-APWR Type A PAM Variable	Basis for Difference
Core Cooling	Indication of core cooling	-	Degrees of Subcooling	This parameter is monitored by the operator to determine if the criteria for terminating the RCS depressurization or terminating ECCS are met in the SGTR analysis. Therefore, this is a Type A variable for the US-APWR. ~~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.~~
RCS Pressure Boundary	Verification of manual action for isolation of failure of small lines carrying primary coolant outside containment	-	Charging Flow	This parameter is monitored by the operator to determine if isolation of the RCS sample line or CVCS letdown line is necessary in the analysis of the radiological consequences of the failure of small lines carrying primary coolant outside containment. Therefore, this is a Type A variable for the US-APWR.

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Table H.2-1 Basis for Type B Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 1 of 4)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR PAM Variable	Basis for Difference	
Reactivity Control				
Neutron Flux	Function detection; accomplishment of mitigation	Wide Range Neutron Flux	No difference. This is a Type B variable for the US-APWR.	DCD_07.05-18 S02
Control Rod Position	Verification	-	This is considered a Category 3 or backup indication in RG 1.97 Rev. 3. 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 already a PAM variable, control rod indication is not included in the US-APWR PAM list. ~~The primary indication of reactor shutdown is neutron flux (Type B). Therefore, for the US-APWR control rod position is provided, but it is not identified as a PAM variable.~~	DCD_07.05-18 S02
RCS Soluble Boron Concentration	Verification	-~~Reactor Coolant Soluble Boron Concentration~~	This is considered a Category 3 or backup indication in RG 1.97 Rev. 3. Reactivity control is directly monitored by neutron flux. RCS soluble boron concentration is not monitored continuously, but only obtained periodically. Since the primary indication is neutron flux, which is already a PAM variable, RCS soluble boron concentration is not included in the US-APWR PAM list.~~No difference~~	DCD_07.05-18 S02
RCS Cold Leg Water Temperature	Verification	Reactor Coolant Cold Leg Temperature (Wide Range)	No difference. This is a Type B variable for the US-APWR.	
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. This is a Type B variable for the US-APWR.	DCD_07.05-18 S02
RCS Cold Leg Water Temperature	Function detection; accomplishment of mitigation; verification; long-term surveillance	Reactor Coolant Cold Leg Temperature (Wide Range)	No difference. This is a Type B variable for the US-APWR.	DCD_07.05-18 S02

Table H.2-1 Basis for Type B Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 2 of 4)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR PAM Variable	Basis for Difference	
RCS Pressure	Function detection; accomplishment of mitigation; verification; long-term surveillance	Reactor Coolant Pressure	No difference. This is a Type B variable for the US-APWR.	DCD_07.05-18 S02
Core Exit Temperature	Verification	Core Exit Temperature	No difference. This is a Type B variable for the US-APWR.	DCD_07.05-18 S02
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.~~No difference~~	
Degrees of Subcooling	Verification and analysis of plant conditions	Degrees of Subcooling	No difference. This is a Type B variable for the US-APWR.	
Maintaining Reactor Coolant System Integrity				
RCS Pressure	Function detection; accomplishment of mitigation	Reactor Coolant Pressure	No difference. This is a Type B variable for the US-APWR.	DCD_07.05-18 S02
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)	Unlike some current operating plants, the US-APWR RWSP is located inside containment. The US-APWR RWSP essentially combines the functions of the sump and RWSP. Therefore, the 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.~~No difference. The US-APWR RWSP is located inside containment, essentially combining the function of the sump and RWSP.~~	DCD_07.05-18 S02
Containment Pressure	Function detection; accomplishment of mitigation; verification	Containment Pressure	No difference. This is a Type B variable for the US-APWR.	DCD_07.05-18 S02
Maintaining Containment Integrity				
Containment Isolation Valve Position (excluding check valves)	Accomplishment of isolation	Containment Isolation Valve Position (Excluding Check Valves)	No difference. This is a Type B variable for the US-APWR.	DCD_07.05-18 S02
Containment Pressure	Function detection; accomplishment of mitigation; verification	Containment Pressure	No difference. This is a Type B variable for the US-APWR.	DCD_07.05-18 S02

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Table H.2-1 Basis for Type B Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 3 of 4)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR PAM Variable	Basis for Difference
Other			
-	-	Pressurizer Water Level	This parameter is important to monitor because it is related to the SI termination criteria. The SI termination criteria are related to maintaining adequate RCS inventory to assure core cooling. In addition, this parameter is also related to RCS integrity by preventing water relief through pressurizer safety valves.~~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 the secondary heat sink for removing the core decay heat. Adequate secondary heat sink ensures that core cooling can be maintained.~~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 the secondary heat sink availability for removing core decay heat. Adequate secondary heat sink ensures that core cooling can be maintained.~~This parameter provides indication of heat sink availability and is selected to monitor core cooling.~~
-	-	SG Water Level (Narrow Range)	This parameter provides indication of the secondary heat sink availability for removing core decay heat. Adequate secondary heat sink ensures that core cooling can be maintained.~~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 for secondary heat sink availability. Adequate secondary heat sink ensures that core cooling can be maintained.~~This parameter provides verification of the automatic actuation of EFW and is selected to monitor core cooling.~~

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Table H.2-1 Basis for Type B Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 4 of 4)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR PAM Variable	Basis for Difference
-	-	EFW Pit Water Level	This parameter provides indication of the secondary heat sink availability for removing core decay heat. Adequate secondary heat sink ensures that core cooling can be maintained.~~This parameter provides indication of heat sink availability and is selected to monitor core cooling.~~
-	-	Containment High Range Area Radiation	This parameter provides indication of the radiation level in containment. This parameter is related to containment integrity by ensuring containment is isolated when radiation levels are high.

Table H.3-1 Basis for Type C Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 1 of 3)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR 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. This is a Type C variable for the US-APWR.	DCD_07.05-18 S02
Core Exit Temperature	Detection of breach	Core Exit Temperature	No difference. This is a Type C variable for the US-APWR.	DCD_07.05-18 S02
Analysis of Primary Coolant (Gamma Spectrum)	Detail analysis; accomplishment of mitigation; verification; long-term surveillance	-	The concentration of each radioactive nuclide can be derived from periodic RCS sampling. However, this is considered a Category 3 or backup indication in RG 1.97 Rev. 3. The primary indications for monitoring fuel cladding are core exit temperature and radioactivity concentration or radiation level in circulating primary coolant. Therefore, analysis of primary coolant (gamma spectrum) is not included as a Type C PAM variable for the US-APWR.~~In the US-APWR, concentration of each radioactive nuclide is derived from RCS sampling. RG 1.97 Rev.3 allows analysis of primary coolant by sampling.~~	DCD_07.05-18 S02
Reactor Coolant Pressure Boundary				
RCS Pressure	Detection of potential for or actual breach; accomplishment of mitigation; long-term surveillance	Reactor Coolant Pressure	No difference. This is a Type C variable for the US-APWR.	DCD_07.05-18 S02
Containment Pressure	Detection of breach; accomplishment of mitigation; long-term surveillance	Containment Pressure	No difference. This is a Type C variable for the US-APWR.	DCD_07.05-18 S02
Containment Sump Water Level	Detection of breach; accomplishment of mitigation; long-term surveillance	-	Containment pressure is a more direct indication of a potential reactor coolant pressure boundary breach. Therefore, RWSP level is not included as a Type C variable for the US-APWR.~~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.~~	DCD_07.05-18 S02

Table H.3-1 Basis for Type C Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 2 of 3)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR PAM Variable	Basis for Difference
Containment Area Radiation	Detection of breach; verification	Containment High Range Area Radiation	No difference. This is a Type C variable for the US-APWR.
Effluent Radioactivity - Noble Gas Effluent from Condenser Air Removal System Exhaust	Detection of breach; verification	-	Coolant leakage into the secondary system due to an actual breach of the reactor coolant pressure boundary can be detected by RCS pressure, SG water level, and/or pressurizer water level. These variables are already Type A and/or Type C PAM variables. Therefore, it is not necessary to include effluent radioactivity as a Type C variable. ~~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.~~
Containment			
RCS Pressure	Detection of potential for breach; accomplishment of mitigation	Reactor Coolant Pressure	No difference. This is a Type C variable for the US-APWR.
Containment Hydrogen Concentration	Detection of potential for breach; accomplishment of mitigation; long-term surveillance	-	This variable is not used for design basis events (it is only used for beyond design basis accidents). Therefore, it does not need to be a Type C variable. However, the US-APWR does have the ability to monitor containment hydrogen concentration as described in DCD Subsection 6.2.5 in order to fulfill the TMI-related requirements of 10 CFR 50.34(f). ~~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. This is a Type C variable for the US-APWR.

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Table H.3-1 Basis for Type C Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 3 of 3)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR PAM Variable	Basis for Difference
Containment Effluent Radioactivity - Noble Gas Effluent from Identified Release Points	Detection of breach; accomplishment of mitigation; verification	-	Containment effluent radioactivity may be used to detect a containment breach. However, this is considered a Category 2 or backup indication in RG 1.97 Rev. 3. The primary indication for monitoring containment pressure boundary is containment pressure. Therefore, containment effluent radioactivity is not included as a Type C PAM variable for the US-APWR. Note that for the purpose of monitoring the release of radioactivity from pathways controlled by Technical Specifications, 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 the plant vent radiation monitors (including high range) as part of the Type E variables for that purpose. ~~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	-	Coolant leakage outside of containment into secondary containment or the auxiliary or fuel handling buildings due to an actual breach of the reactor coolant pressure boundary can be detected by RCS pressure, SG water level, and/or pressurizer water level. These variables are already Type A and/or Type C PAM variables. In addition, effluent radioactivity is considered a Category 2 or backup indication in RG 1.97 Rev. 3. Therefore, it is not necessary to include effluent radioactivity as a Type C variable. ~~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 H.4-1 Basis for Type D Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 1 of 9)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR 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. This is a Type D variable for the US-APWR.
RHR Heat Exchanger Outlet Temperature	To monitor operation and for analysis	-	Proper operation of the RHR system is verified by the 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.~~Proper operation of the RHR system is verified by CS/RHR flow rate. Additionally, T~~$\cancel{T_{hot}}$ ~~and~~ $\cancel{T_{cold}}$ ~~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. This is a Type D variable for the US-APWR.
Accumulator Isolation Valve Position	Operation status	-	Accumulator water level and accumulator pressure are available to monitor the accumulator operation status. Therefore, it is not necessary to include accumulator isolation valve position as a separate Type D variable in the US-APWR PAM list.~~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.~~

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Table H.4-1 Basis for Type D Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 2 of 9)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR PAM Variable	Basis for Difference
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.~~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. This is a Type D variable for the US-APWR.
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 US-APWR PAM list does not need any variables to indicate low pressure injection (LPI) system performance.~~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 US-APWR PAM list does not need any variables to indicate LPI system performance.~~
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. This is a Type D variable for the US-APWR.

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Table H.4-1 Basis for Type D Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 3 of 9)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR PAM Variable	Basis for Difference
Primary Coolant System			
Reactor Coolant Pump Status	To monitor operation	-	The safety analysis does not rely on the RCPs to mitigate design basis events. The RCPs are also not necessary to achieve and maintain safe shutdown conditions. In addition, CCW header pressure is available to monitor CCW performance related to its function to provide seal cooling 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.~~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	Pressurizer Safety Valve Position, Safety Depressurization Valve Position~~-~~	No difference. This is a Type D variable for the US-APWR.~~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. This is a Type D variable for the US-APWR.
Pressurizer Heater Status	To determine operating status	-	Pressurizer water level and RCS pressure are indicative of the performance of the pressurizer heaters. Therefore it is not necessary to separately include the heater status indications in the PAM list.~~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.~~

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Table H.4-1 Basis for Type D Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 4 of 9)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR PAM Variable	Basis for Difference
Quench Tank Level	To monitor operation	-	This component is not necessary to mitigate design basis events and is also not necessary to achieve and maintain safe shutdown conditions. Therefore, it is not included in the US-APWR PAM list.~~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.
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. This is a Type D variable for the US-APWR.
Steam Generator Pressure	To monitor operation	Main Steam Line Pressure	No difference. This is a Type D variable for the US-APWR.
Safety/Relief Valve Positions or Main Steam Flow	To monitor operation	Main Steam Safety Valve Position, Main Steam Relief Valve Position, Main Steam Depressurization Valve Position~~-~~	No difference. This is a Type D variable for the US-APWR. ~~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. In addition, the EFW system is used to provide flow to the SGs and EFW flow indication is available. Since these variables are already available to monitor SG operation, it is not necessary to separately include MFW flow in the PAM list.~~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. This is a Type D variable for the US-APWR.

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Table H.4-1 Basis for Type D Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 5 of 9)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR PAM Variable	Basis for Difference
Condensate Storage Tank Water Level	To ensure water supply for auxiliary feedwater	EFW Pit Water Level	No difference. This is a Type D variable for the US-APWR.
Containment Cooling Systems			
Containment Spray Flow	To monitor operation	CS/RHR Pump Discharge Flow CS/RHR Pump Minimum Flow	No difference. This is a Type D variable for the US-APWR.
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 cool the containment and maintain containment integrity. Therefore this variable is not included in the PAM list. ~~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.~~
Containment Atmosphere Temperature	To monitor operation	Containment Temperature	No difference. This is a Type D variable for the US-APWR.
Containment Sump Water Temperature	To monitor operation	-	Containment pressure, containment temperature, and CS/RHR pump flow are utilized to monitor containment cooling system performance. In the US-APWR, the RWSP also serves as the normal suction source for the SI pumps. The design of the SI and CS/RHR pumps is such that NPSH is ensured even for RWSP water temperatures that bound accident conditions. Therefore it is not necessary to include this variable in the US-APWR PAM list.~~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)			

Table H.4-1 Basis for Type D Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 6 of 9)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR PAM Variable	Basis for Difference
Makeup Flow - In	To monitor operation	-	Since RCS inventory control and boration are provided 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.~~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 primary indication of the performance of the cooling water system. Monitoring header pressure gives the operator the most immediate and accurate indicate of the performance of the system of any available indication. Therefore it is not necessary to separately include this other variable to monitor CCW system performance in the PAM list.~~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			

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Table H.4-1 Basis for Type D Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 7 of 9)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR PAM Variable	Basis for Difference
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 is also not necessary to achieve and maintain a safe shutdown condition. Further addition of 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.~~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.
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 monitors provides verification of proper automatic ventilation path isolation. Therefore, damper position indication is not included in the US-APWR PAM list. ~~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.~~

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Table H.4-1 Basis for Type D Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 8 of 9)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR PAM Variable	Basis for Difference
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. This is a Type D variable for the US-APWR.
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.
-	-	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.

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Table H.4-1 Basis for Type D Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 9 of 9)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR PAM Variable	Basis for Difference
-	-	Containment Purge Isolation Valve Position.	This variable is used to indicate the system status of the containment purge system.
-	-	Main Steam Isolation Valve Position	This variable is used to indicate the system status of the MSS system. All associated valves closed by the main steam line isolation signal are also included.
-	-	Main Feedwater Isolation Valve Position	This variable is used to indicate the system status of the MFW system. All associated valves closed by the main feedwater isolation signal are also included.
-	-	Emergency Feedwater Isolation Valve Position	This variable is used to indicate the system status of the EFW system. All associated valves repositioned by the emergency feedwater actuation signal or emergency feedwater isolation signal are also included.
-	-	MCR HVAC Damper Position	This variable is used to indicate the system status of the MCR HVAC system.
-	-	CS/RHR Pump Suction Relief Valve Position	This variable is used to indicate the system status of the RHR system.
-	-	SFP Pump Discharge Flow	This variable is used to indicate the system status of the cooling portion of the SFPCS.
-	-	SFP Temperature	This variable is used to indicate the system status of the cooling portion of the SFPCS.
-	-	SFP Water Level (Narrow Range)	This variable is used to indicate the system status of the cooling portion of the SFPCS.

Table H.5-1 Basis for Type E Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 1 of 5)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR 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. This is a Type E variable for the US-APWR.
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	-	The MCR and TSC are the main areas where access is required. The radiation in the MCR and TSC are selected as Type E PAM variables for the US-APWR (described in the "other" section below). If access to other areas is necessary, personnel protection will be provided by the use of portable radiation monitors and air sampling which are selected as Type E PAM variables. No other area radiation monitors are required. Therefore, it is not necessary to include this variable in the US-APWR PAM list. ~~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 the plant vent radiation monitors (including high range) and therefore are not included as separate Type E variables for the US-APWR.~~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	-	

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Table H.5-1 Basis for Type E Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 2 of 5)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR PAM Variable	Basis for Difference	
Condenser Air Removal System Exhaust	Detection of significant releases; release assessment	-		
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 the plant vent radiation monitors (including high range) and therefore is not included as a separate Type E variable for the US-APWR.~~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.~~	DCD_07.05-18 S02
Vent From Steam Generator Safety Relief Valves or Atmospheric Dump Valves	Detection of significant releases; release assessment	-	This variable is measured by the main steam line monitors. Therefore it is not included as a separate Type E variable for the US-APWR.~~This variable is measured by main steam line monitor. Therefore it is not included as a separate Type E variable for the US-APWR.~~	DCD_07.05-18 S02
All Other Identified Release Points	Detection of significant releases; release assessment; long-term surveillance	-	This variable can be measured by the plant vent radiation monitors (including high range) and therefore is not included as a separate Type E variable for the US-APWR.~~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	-	The main release point is the vent stack. This variable can be measured by the plant vent sampler (accident sampler). Therefore it is not included as a separate Type E variable for the US-APWR. Note that the other release points are the main steam safety valves and relief valves which are specifically excluded from this category in RG 1.97 Rev. 3. Release from those points can be determined by the portable instruments which are already identified as Type E variables. ~~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.~~	DCD_07.05-18 S02
Environs Radiation and Radioactivity				

Table H.5-1 Basis for Type E Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 3 of 5)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR PAM Variable	Basis for Difference
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. This is a Type E variable for the US-APWR.
Plant and Environs Radiation (portable instrumentation)	Release assessment; analysis	Plant and Environs Radiation (Portable Instrumentation)	No difference. This is a Type E variable for the US-APWR.
Plant and Environs Radioactivity (portable instrumentation)	Release assessment; analysis	Plant and Environs Radioactivity (Portable Instrumentation)	No difference. This is a Type E variable for the US-APWR.
Meteorology			
Wind Direction	Release assessment	Meteorological Parameters (Wind Direction, Wind Speed, Estimation of Atmospheric Stability)	No difference. This is a Type E variable for the US-APWR. Note that the description of this variable will be provided by the COL applicant since it is site specific.
Wind Speed	Release assessment	Meteorological Parameters (Wind Direction, Wind Speed, Estimation of Atmospheric Stability)	No difference. This is a Type E variable for the US-APWR. Note that the description of this variable will be provided by the COL applicant since it is site specific.
Estimation of Atmospheric Stability	Release assessment	Meteorological Parameters (Wind Direction, Wind Speed, Estimation of Atmospheric Stability)	No difference. This is a Type E variable for the US-APWR. Note that the description of this variable will be provided by the COL applicant since it is site specific.

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Table H.5-1 Basis for Type E Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 4 of 5)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR PAM Variable	Basis for 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.
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 Area Radiation	To monitor radiation and radioactivity levels in the technical support center.
-	-	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.

Table H.5-1 Basis for Type E Differences between RG 1.97 Rev.3 and the US-APWR PAM List (Sheet 5 of 5)

RG 1.97 Rev. 3 Variable	Purpose	US-APWR PAM Variable	Basis for Difference
-	-	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.