Enclosure 3

US-APWR

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

February 2008
(Non-Proprietary)

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INTRODUCTION

This report provides supplemental responses to MHI's initial responses to two questions related to the NRC's acceptance review of Chapter 15 of the US-APWR Design Certification Application. The initial responses were submitted to the NRC by letter dated February 8, 2008 (MHI Reference UAP-HF-08037). The following responses are complete responses to the NRC questions and supersede the initial responses.

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

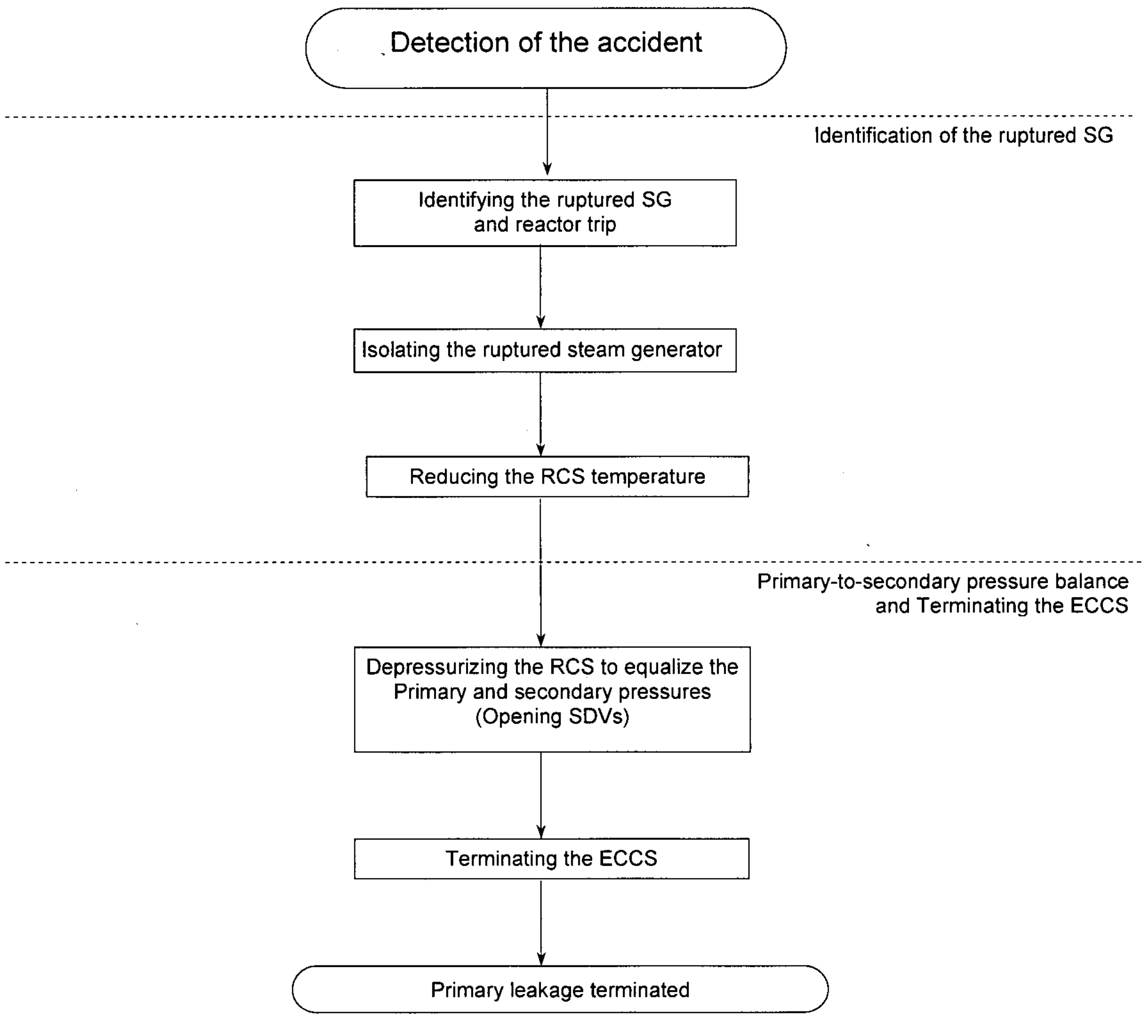
MHI RESPONSE:
The key features of accident prevention and mitigation for the US-APWR are essentially the same as those provided in currently operating 4-loop PWRs in the US. Therefore, the operator actions assumed in the Chapter 15 analyses are similar to the actions made for currently operating reactors. DCD Subsection 15.0.0.6 lists the specific events where operator actions are assumed as part of the event analysis and provides a cross-reference to the event specific analysis section for additional details regarding the assumed operator actions.

It should be noted that Chapter 7 of the US-APWR DCD also contains information related to instrumentation and controls used to support operator actions in Chapter 15 events. DCD Table 7.5-5 further refines the US-APWR design approach for supporting operator actions following specific events by defining the alarm credited to either identify the event is in progress or define the operator response. In addition, DCD Table 7.5-3 identifies instrumentation available to the operator to assist in defining post-event responses or in determining the effectiveness of such actions. DCD Table 7.5-1 defines the RG 1.97 Variable Type, which in turn defines how the instrument or parameter is used to aid the operator in recognizing the condition and/or verifying the adequacy of the response.

Table 1-1 provides a summary of all of the operator actions that are assumed as part of the analysis of Chapter 15 events. The table is organized by applicable Chapter 15 events, which are each further subdivided by the manual actions in the order they occur. Although MHI is not submitting the US-APWR ERG in its entirety at this time, selected ERG content is provided to support the operator actions in response to Chapter 15 events. Specifically, for each operator action, the source of information such as an indicated parameter or alarm used by the operator in identifying the event or performing the responsive action is listed. In the last column, criteria applied to the available information being assessed with respect to that action is also listed.

The chronology of each event requiring operator actions is similar in that after the event occurs, the operator becomes aware that the event has occurred and in most cases, performs a simple (and many times single) action that terminates the event. Because of this similarity, flow charts of the event progression from initiation and cause identification through termination are not provided for each event. The Steam Generator Tube Rupture event, however, is more complex. The flow chart presented on the following page is provided to illustrate the sequence of manual operator actions for that event. It should be noted that information related to the timing of the actions and the acceptability of the response time is described in the DCD, not in this response. The purpose of this submittal is to provide the plant conditions (as defined by combinations of parameters and values, as well as the source of the information) under which specific operator actions can or cannot be taken as they relate to the events analyzed in Chapter 15 crediting operator actions.

Step Sequence for SGTR Operator Actions



In summary, the DCD identifies the Chapter 15 events that assume operator actions, describes the actions (and their time sequence) in each applicable Chapter 15 subsection, and a description of the instrumentation and controls provided to assist the operator in successfully completing these actions is provided in Chapter 7 of the DCD. The information in this response provides the subset of information normally included in the Emergency Response Guidelines that defines the conditions or constraints under which these specific operator actions can be taken.

As described in Section 13.5 of the DCD, MHI plans to develop an ERG document for use in developing plant-specific emergency operating procedures to be used by COL holders. MHI plans to submit this information to the NRC as soon as it is available.

Table 1-1
Summary of Operator Actions and ERG Guidance Information

Accident	Manual Action	Source of Information	Operator Action Criteria
Inadvertent Decrease in Boron Concentration in RCS Subcritical Modes (Subsection 15.4.6)	Terminate event following detection • Close Charging Flow Isolation Valve <u>or</u> • Close Primary Makeup Water Control Valve <u>or</u> • Stop Primary Makeup Water Pump	• Source Range Neutron Flux Alarm – High	Terminate cause following identification
Inadvertent Decrease in Boron Concentration in RCS Critical Modes (Subsection 15.4.6)	Terminate event following detection • Close Charging Flow Isolation Valve <u>or</u> • Close Primary Makeup Water Control Valve <u>or</u> • Stop Primary Makeup Water Pump	• Control Rod Insertion Limit Alarm	Terminate cause following identification
CVCS Malfunction that Increases Reactor Coolant Inventory (Subsection 15.5.2)	Terminate event following detection • Close Charging Line Isolation Valve <u>or</u> • Close Charging Line Containment Isolation Valve	• Pressurizer Water Level Alarm – High	Terminate cause following identification
Radiological Consequences of a SG Tube Failure (SGTR) (Subsection 15.6.3)	Manual reactor trip	• Pressurizer Water Level Alarm – Low • High Sensitivity Main Steam Line Radiation (N-16)	If no automatic Reactor Trip or no ESFAS and one or more of the following: • Alarm - Pressurizer water level more than[ ] below program value <u>or</u> • Alarm – High main steam line radiation (N-16) in one or more SGs

Table 1-1
Summary of Operator Actions and ERG Guidance Information

Accident	Manual Action	Source of Information	Operator Action Criteria
	Identify and Isolate Affected SG • Isolate SG Relief Valve • Isolate EFW Turbine supply • Isolate SG Blowdown • Close Main Steam Line Valves	• High Sensitivity Main Steam Line Radiation (N-16)	Identify affected SG based on comparison of • High Sensitivity Main Steam Line Radiation (N-16) • Narrow Range SG Level
	Cool Down Primary Coolant System using Main Steam Depressurization Valves on Intact SGs • Maintain subcooling	• RCS subcooling (Wide Range RCS pressure , hot leg temperature, secondary side temperature based on pressure, secondary side pressure)	Continue to cool down Primary Coolant System until achieving the following: a) if affected SG pressure > saturation pressure at no-load temperature • Intact loop hot leg temperature equals no-load temperature minus[ ] b) if affected SG pressure < saturation pressure at no-load temperature • Intact loop hot leg temperature equals affected SG saturation temperature minus[ ] After achieving these conditions, maintain them.

Table 1-1
Summary of Operator Actions and ERG Guidance Information

Accident	Manual Action	Source of Information	Operator Action Criteria
	Equalize Pressure between Primary and Secondary using Safety Depressurization Valve (SDV) • Maintain subcooling	• Wide Range Reactor Coolant Pressure • Main Steam Line Pressure • RCS subcooling (Wide Range RCS pressure , hot leg temperature, secondary side temperature based on pressure, secondary side pressure) • Pressurizer Water Level	Open SDV when the following occurs: a) if affected SG pressure > saturation pressure at no-load temperature • Intact loop hot leg temperature < no-load temperature minus[ ] (allowance for uncertainty) b) if affected SG pressure < saturation pressure at no-load temperature • Intact loop hot leg temperature < affected SG saturation temperature minus[ ] Terminate SDV depressurization when one of the following occurs: • RCS pressure < ruptured SG pressure <u>and</u> pressurizer level >[ ](allowance for uncertainty), <u>or</u> • Pressurizer Water Level >[ ] (allowance for uncertainty) to prevent overfill • Subcooling margin <[ ] (allowance for uncertainty)
	Stop Safety Injection Flow	• EFW flow rate • SG level (narrow range) • Pressurizer water level • RCS pressure (wide range)	Terminate SI if one of the following are satisfied: • RCS subcooling is greater than[ ] (allowance for uncertainty) • Minimum EFW is available, or the water level in at least one steam generator is in the narrow range • Pressurizer level is greater than[ ] (allowance for uncertainty) • The RCS pressure is increasing (increases of [ ] from its minimum pressure)
Control Rod Assembly Ejection Accidents (Subsection 15.4.8)	• Operate C/V Spray System • Operate Annulus Emergency Exhaust System	• Containment High Range Area Radiation Alarm	Operate systems if not automatically started after event identification

Table 1-1
Summary of Operator Actions and ERG Guidance Information

Accident	Manual Action	Source of Information	Operator Action Criteria
Failure of Small Lines Carrying Primary Coolant Outside C/V (Subsection 15.6.2)	• Isolate Broken Lines (CVCS Letdown Line or RCS Sample Lines)	• Low Volume Control Tank Water Level Alarm	Terminate cause following identification
Post-LOCA Long Term Cooling (Subsection 15.6.5)	Manual switchover to the simultaneous RV and hot leg injection mode	Not applicable	Long-term action not based on process parameters to prevent boron precipitation

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

MHI RESPONSE:

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

The following table has been prepared to document the rationale as to how and why the Chapter 15 events presented in the US-APWR DCD encompass or bound the plant responses for other cases not presented at different modes and/or initial power levels. Additional discussion and rationale for the analyses presented in Chapter 15 of the DCD is provided as part of each event. The following table expands or clarifies the existing DCD with respect to the initial conditions assumed for each event.

Table 2-1 DCD Limiting Case Selection Matrix

Event	Mode-6 Refueling	Mode-5 Cold Shutdown	Mode-4 Hot Shutdown	Mode-3 Hot Standby		Mode-2 Startup	Mode-1 Power Operation	
				T < no load	No Load		Partial Power	HFP
Reactivity (K_{eff})	NA	<0.99	<0.99	<0.99		≥0.99	≥0.99	
%Rated Thermal Power	NA	NA	NA	NA		≤5	>5	
Avg. Reactor Coolant Temp., T_{avg} (F)	NA	≤200	350>T_{avg}>200	≥350		NA (No Load)	NA (from No Load to HFP-T_{avg})	
				~350 to No Load	No Load		Partial Power	HFP
Decrease in Feedwater Temperature as a Result of Feedwater System Malfunctions (15.1.1)	Non-limiting due to low T_{avg} and margin to criticality.	Same as Mode 6	Same as Mode 6	Same as Mode 6	Under no-load conditions, the rate of energy change is reduced as feedwater flow decreases, making the no-load case less severe than the HFP case.		Same as Mode 2	Discussed in the DCD
Increase in Feedwater Flow as a Result of Feedwater System Malfunctions (15.1.2)	Non-limiting due to low T_{avg} and margin to criticality.	Same as Mode 6	Same as Mode 6	Same as Mode 6	Core reactivity is more severe than at HFP due to increased feedwater supplied at a lower temperature and later Doppler feedback. However, reactivity insertion is bounded by the 15.4.1 event.		Same as Mode 2	Discussed in the DCD

Table 2-1 DCD Limiting Case Selection Matrix

Event	Mode-6 Refueling	Mode-5 Cold Shutdown	Mode-4 Hot Shutdown	Mode-3 Hot Standby		Mode-2 Startup	Mode-1 Power Operation	
				T < no load	No Load		Partial Power	HFP
Increase in Steam Flow as a Result of Steam Pressure Regulator Malfunction (15.1.3)	Same as 15.1.4 event	Same as 15.1.4 event	Same as 15.1.4 event	Same as 15.1.4 event		The peak power is much less than at HFP because of the neutron flux high trip (low setpoint).	Reactivity insertion is higher in this case due to the higher secondary pressure. However, lower initial power and coolant temperature ensure that the minimum DNBR is bounded by the HFP case and 15.1.4 event.	Discussed in the DCD
Inadvertent Opening of a Steam generator Relief or Safety Valve (15.1.4)	Non-limiting because secondary system steam release cannot reduce T_{avg} below T_{sat} at atmospheric pressure.	Same as Mode 6	High shutdown margin keeps core subcritical or significantly limits power increase when critical.		Discussed in the DCD	Bounded by 15.1.3 event	Bounded by 15.1.3 event	Bounded by 15.1.3 event
Steam System Piping Failures Inside and Outside of Containment (15.1.5)	Non-limiting because secondary system steam release cannot reduce T_{avg} below T_{sat} at atmospheric pressure.	Same as Mode 6	High shutdown margin keeps core subcritical or significantly limits power increase when critical.		Discussed in the DCD	Bounded by the at-power and hot standby DCD cases	Discussed in the DCD (100%, 75%)	
Loss of External Load (15.2.1)	Non-limiting since the core is in RHR cooling.	Same as Mode 6	Non-limiting since the reactor is at zero power.			Non-limiting due to smaller load loss and greater margin to DNB and overpower trips relative to HFP case.		Discussed in the DCD

Table 2-1 DCD Limiting Case Selection Matrix

Event	Mode-6 Refueling	Mode-5 Cold Shutdown	Mode-4 Hot Shutdown	Mode-3 Hot Standby		Mode-2 Startup	Mode-1 Power Operation	
				T < no load	No Load		Partial Power	HFP
Loss of Non-Emergency AC Power to the Station Auxiliaries (15.2.6)	Non-limiting since the core is in RHR cooling.	Same as Mode 6	Same as Mode 6	Non-limiting since the reactor is at zero power.		When the reactor power is greater than the P-7 setpoint (10%), the same R/T as for HFP occurs from a lower initial power. Therefore, this case is bounded by the HFP case. When the reactor power is less than the P-7 setpoint, low RCS flow, low RCP speed, and TT R/T signals are blocked. However, the loss of non-emergency AC power results in the loss of power to the MG set, which, in turn, causes a R/T. Additionally, the initial power for the partial power cases is sufficiently low so that the pressure response is bounded by the HFP case. The ability to establish sufficient natural circulation to remove nuclear power or decay heat at these power levels is demonstrated for the HFP case.		Discussed in the DCD
Loss of Normal Feedwater (15.2.7)	Non-limiting since the core is in RHR cooling.	Same as Mode 6	Same as Mode 6	Non-limiting since the reactor is at zero power.		Same as Mode 1 Partial Power	Bounded by HFP case since initial power is less than at HFP and R/T is initiated by low SG water level.	Discussed in the DCD

Table 2-1 DCD Limiting Case Selection Matrix

Event	Mode-6 Refueling	Mode-5 Cold Shutdown	Mode-4 Hot Shutdown	Mode-3 Hot Standby		Mode-2 Startup	Mode-1 Power Operation	
				T < no load	No Load		Partial Power	HFP
Feedwater System Pipe Break Inside and Outside Containment (15.2.8)	Non-limiting since the core is in RHR cooling.	Same as Mode 6	Same as Mode 6	A FLB at zero power is not limiting.		Same as Mode 1 Partial Power.	Bounded by HFP case since initial power is less than at HFP and R/T is initiated by low SG water level.	Discussed in the DCD
Partial Loss of Forced Reactor Coolant Flow (15.3.1.1)	RCPs are in stand-by; therefore, there is no event initiation.	Non-limiting since the reactor is at zero power.	Same as Mode 5	Same as Mode 5		When the reactor power is less than the P-7 setpoint, the low RCS flow R/T signal is blocked. However, the low initial power and minimum of two RCPs running results in significant DNB margin to the nominal operating conditions. For initial power levels greater than the P-7 setpoint, the flow transient and time of trip is the same as HFP, but the higher HFP power level is bounding..		Discussed in the DCD
Complete Loss of Forced Reactor Coolant Flow (15.3.1.2)	RCPs are in stand-by; therefore, there is no event initiation.	Non-limiting since the reactor is at zero power.	Same as Mode 5	Same as Mode 5		For initial power above the P-7 setpoint, the trip and flow response is the same as for HFP and the lower initial power is less limiting than HFP. For initial conditions below the P-7 setpoint where the low flow and low RCP speed trips are blocked, the combination of low core power and the associated natural circulation flow is sufficient to protect the fuel design limits. Therefore, these cases are not limiting.		Discussed in the DCD

Table 2-1 DCD Limiting Case Selection Matrix

<table>
<tr><th rowspan="2">Event</th><th rowspan="2">Mode-6
Refueling</th><th rowspan="2">Mode-5
Cold Shutdown</th><th rowspan="2">Mode-4
Hot
Shutdown</th><th colspan="2">Mode-3
Hot Standby</th><th rowspan="2">Mode-2
Startup</th><th colspan="2">Mode-1
Power Operation</th></tr>
<tr><th>T < no load</th><th>No Load</th><th>Partial Power</th><th>HFP</th></tr>
<tr><td>Reactor Coolant Pump Rotor Seizure & Shaft Break (15.3.3 & 15.3.4)</td><td>RCPs are in stand-by; therefore, there is no event initiation.</td><td>Non-limiting since the reactor is at zero power.</td><td>Same as Mode 5</td><td colspan="2">Same as Mode 5</td><td colspan="2">When the reactor power is less than the P-7 setpoint, low RCS flow R/T signal is blocked. However, the initial power is sufficiently low that this case is not limiting. Above the P-7 setpoint, flow and trip response is the same for all initial power levels as HFP, which is bounding because of the higher initial power.</td><td>Discussed in the DCD</td></tr>
<tr><td>Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition (15.4.1)</td><td>CRDM is in stand-by; therefore, there is no event initiation.</td><td>RCCA operation is administratively restricted. (Core power cannot be used for the RCS heating)</td><td>Same as Mode 5</td><td>Same as Mode5</td><td>Discussed in the DCD</td><td>Bounded by the DCD case</td><td>Bounded by 15.4.2 event</td><td>Bounded by 15.4.2 event</td></tr>
<tr><td>Uncontrolled Control Rod Assembly Withdrawal at Power (15.4.2)</td><td>Bounded by 15.4.1 event</td><td>Bounded by 15.4.1 event</td><td>Bounded by 15.4.1 event</td><td colspan="2">Bounded by 15.4.1 event</td><td>Bounded by 15.4.1 event and Mode 1 spectrum of cases</td><td colspan="2">Discussed in the DCD (100%, 75%, 10%)</td></tr>
<tr><td>One or More Dropped RCCAs within a Group or Bank (15.4.3)</td><td>All RCCAs are fully inserted; therefore, there is no event initiation.</td><td>Non-limiting since the reactor is at zero power.</td><td>Same as Mode 5</td><td colspan="2">Same as Mode 5</td><td>Same as Mode 1 Partial Power</td><td>Non-limiting because the initial power is lower than HFP case.</td><td>Discussed in the DCD</td></tr>
</table>

Table 2-1 DCD Limiting Case Selection Matrix

Event	Mode-6 Refueling	Mode-5 Cold Shutdown	Mode-4 Hot Shutdown	Mode-3 Hot Standby		Mode-2 Startup	Mode-1 Power Operation	
				T < no load	No Load		Partial Power	HFP
Uncontrolled Withdrawal of a Single RCCA (15.4.3)	The core has sufficient criticality margin to prevent an occurrence of criticality caused by a single RCCA withdrawal event.	Same as Mode 6	Same as Mode 6	Same as Mode 6		Same as Mode 1 Partial Power.	Non-limiting because the initial power is lower than HFP case.	Discussed in the DCD
Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (15.4.6)	Discussed in the DCD	Discussed in the DCD	Discussed in the DCD	Discussed in the DCD		Discussed in the DCD	Discussed in the DCD	Discussed in the DCD
Spectrum of Rod Ejection Accidents (15.4.8)	The core has sufficient criticality margin to prevent an occurrence of criticality caused by a rod ejection event.	Same as Mode 6	Same as Mode 6	Same as Mode 6		Discussed in the DCD (HZP)	The analysis conditions such as reactivity assumed in the HFP case and the HZP case are considered to bound partial power cases.	Discussed in the DCD (HFP)

Table 2-1 DCD Limiting Case Selection Matrix

Event	Mode-6 Refueling	Mode-5 Cold Shutdown	Mode-4 Hot Shutdown	Mode-3 Hot Standby	Mode-2 Startup	Mode-1 Power Operation	
				T < no load / No Load		Partial Power	HFP
Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory (15.5.2)	The reactor vessel head is detached; therefore, there is no pressure increase.	Pressurizer is water-solid. Control of the pressurizer water level and LTOP system prevent RCS and RHR system overpressurization due to the addition of reactor coolant inventory caused by a CVCS malfunction.	Same as Mode 5	Bounded by the HFP case because pressurizer water level is lower than at HFP, resulting in more time to terminate the event.	Same as Mode 3	Same as Mode 3	Discussed in the DCD
Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve (15.6.1)	Non-limiting due to low initial RCS pressure.	Non-limiting since the reactor is at zero power.	Same as Mode 5	Same as Mode 5	Same as Mode 1 Partial Power.	When the reactor power is less than the P-7 setpoint, the low RCS pressure R/T signal is blocked. However, a R/T on low RCS pressure SI signal occurs; therefore, this case is not limiting.	Discussed in the DCD
Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment (15.6.2)	Same as Mode-1 Partial Power.	Same as Mode-1 Partial Power.	Same as Mode-1 Partial Power.	Same as Mode-1 Partial Power.	Same as Mode-1 Partial Power.	Bounded by HFP case because release due to flashing of leaked coolant is smaller than at HFP.	Discussed in the DCD

Table 2-1 DCD Limiting Case Selection Matrix

Event	Mode-6 Refueling	Mode-5 Cold Shutdown	Mode-4 Hot Shutdown	Mode-3 Hot Standby		Mode-2 Startup	Mode-1 Power Operation	
				T < no load	No Load		Partial Power	HFP
Radiological Consequences of Steam Generator Tube Failure (15.6.3)	There is no primary-to-secondary coolant leakage because the primary system is not pressurized.	RCS pressure and temperature are low enough to prevent opening of MSRV and/or MSSV.	Same as Mode 5	Same as Mode 5	The heat storage in the primary system is less than at HFP conditions, thus time needed to attain primary-to secondary pressure balance is shorter than at HFP. This reduces the primary-to-secondary coolant leakage and the steam release. Therefore, this is not a limiting case.		Same as Mode 2	Discussed in the DCD
Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary (15.6.5)	Bounded by Mode 5	The decrease of inventory is slow, since the RCS pressure is low. Therefore, non-limiting since there is time for manual start of SI.	Bounded by Mode 3	When the RCS pressure is less than the P-11 setpoint, ECCS actuation signal is blocked. However, non-limiting since safety injection and accumulator injection start automatically upon receipt of the High containment pressure ECCS actuation signal.		Bounded by the DCD case, since initial power is lower than HFP case.		Discussed in the DCD
Fuel Handling Accident (15.7.4)	Discussed in the DCD	This accident is not expected to occur during this mode, because in-vessel fuel handling is not carried out during this mode.	Same as Mode 5	Same as Mode 5		Same as Mode 5	Same as Mode 5	