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Subject: **Response to Portion of NRC Request for Additional Information Letter No. 25 Related to ESBWR Design Certification Application – Accident Analyses – RAI Numbers 15.0-1 and 15.0-2**

Enclosure 1 contains GE's response to the subject NRC RAIs transmitted via the Reference 1 letter.

If you have any questions about the information provided here, please let me know.

Sincerely,

Kathy Sedney for

David H. Hinds
Manager, ESBWR

D068

Reference:

1. MFN 06-142, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 25 Related to ESBWR Design Certification Application,* May 9, 2006

Enclosure:

1. MFN 06-331 – Response to Portion of NRC Request for Additional Information Letter No. 25 Related to ESBWR Design Certification Application – Accident Analyses – RAI Numbers 15.0-1 and 15.0-2

cc:	AE Cubbage	USNRC (with enclosures)
	GB Stramback	GE/San Jose (with enclosures)
	eDRFs	0054-2158 and 0057-0343

MFN 06-331

ENCLOSURE 1

Response to Portion of NRC Request for

Additional Information Letter No. 25

Related to ESBWR Design Certification Application

Accident Analyses – RAI Numbers 15.0-1 and 15.0-2

NRC RAI 15.0-1

The list of transients and accidents provided in DCD Tier 2, Chapter 15 appears incomplete. There may be new initiating events that must be considered within the scope of design basis accidents and transients, which result from the new and unique design features of the ESBWR. For example, events such as inadvertent actuation of control rod drive (CRD) system in the injection mode to the reactor pressure vessel (RPV) or inadvertent gravity driven cooling system (GDCS) injection into reactor vessel are not included. Identify all possible transients and accidents which may occur due to the unique design features of the ESBWR.

GE Response

A systematic approach has been used to respond to this RAI. All ESBWR systems have been reviewed to determine if credible failures in the system or operator errors of the system could initiate an ESBWR design basis event (accident, infrequent event, or AOO). The event categories used are from Regulatory Guide (RG) 1.70. If an event is the result of a credible failure for a system then it is determined whether or not the event is addressed in the DCD. If the event is addressed in the DCD the Section number is given in the response. If the event is not discussed in the DCD the table is noted as such and further discussion is provided for the event scenario below. The results of this systematic review are given in Table 15.0-1. The results of this table consider normal operating state D (DCD Table 15.1-2).

The event categories from RG 1.70 presume a secondary heat removal system and therefore are more applicable to the PWR design configuration therefore the ESBWR DCD uses event categories more applicable to the BWR design configuration. The ESBWR event categorization is described in DCD Subsection 15.0.1. RG 1.70 event categories are used in this RAI response to ensure that consideration of system effects is consistent with the regulatory guidance.

Event Discussion (See reference number in Table):

1 Pipe breaks in the Isolation Condenser System (ICS) outside of containment could result in a radioactive release from a subsystem or component. IC breaks outside of containment are isolated automatically when either a high radiation level in the ICS pool area is detected or excess flow is detected in the steam supply line or condensate return line. To assure the mass release due to ICS line breaks outside of containment is bounded by the break location analyzed, the IC break mass release will be quantified for comparison to the break location analyzed in a future revision of the DCD.

2 The Control Rod Drive System (CRDS) has the capability to inject via reactor water cleanup/feedwater to the vessel in case feedwater is lost. Inadvertent injection of this system could result in an increase in reactor coolant inventory. The minimum flow for this injection mode is 3920 l/min (DCD Table 4.6-1) if you assume 115% of this flow and 1 kg/l, you get 4508 kg/min or ~75 kg/s. This is approximately 3% of feedwater flow (2451 kg/s DCD Table 4.4.1a). This is a conservative assumption for flow rate because flow control valves will regulate the flow (DCD Section 4.6.1.2.5). This event is similar to the inadvertent IC injection. However the flow rate for High Pressure CRD injection is about 3 times less than IC injection initially. Therefore, the CRDS injection is expected to be bounded by the inadvertent IC injection as an increase in reactor coolant inventory.

However, in a BWR, CRDS injection could also be considered a decrease in core coolant temperature event. The limiting event in this category is a loss of feedwater heating. The loss of feedwater heating assumes a 100°F drop in feedwater temperature which is more than a 100 Btu/lbm drop in feedwater enthalpy. Assuming a CRDS injection enthalpy of 20.0 Btu/lbm (~50°F) injection of CRDS would only result ~11 Btu/lbm drop in feedwater enthalpy. This would result in a much less CPR change than the loss of feedwater heating event. No additional discussion in DCD Chapter 15 is needed.

3 The Feedwater Control System (FWCS) maintains the reactor level by controlling feedwater pump speed. Failure of the system to maximum demand is considered in DCD Subsection 15.3.2. Because of the high reliability of the FWCS the event is considered an Infrequent Event. The failure of the FWCS to minimum demand is not discussed because it is bounded by the loss of all feedwater event (DCD Subsection 15.2.5.3) which is considered an anticipated operational occurrence. Since it is bounded by an event in a higher frequency category there is no need to discuss it in Chapter 15.

4 The Standby Liquid Control System (SLCS) has the capability to inject borated water directly into the core bypass. Inadvertent injection of this system could result in an increase in reactor coolant inventory. Injection of SLCS results in the shutdown of the reactor as discussed in DCD Subsection 9.3.5. No Chapter 15 discussion is needed.

5 The Gravity-Driven Cooling System (GDCS) has the capability to inject water to the vessel in case of a LOCA. GDCS cannot inject into the reactor vessel unless the pressure is below static head of the system. In this case the reactor is shutdown and flow would drain into the vessel until equilibrium is reached. Manual initiation of this system at high pressure is not credible due to pressure permissive logic that inhibits the actuation of the squib valves. Inadvertent actuation of a single squib valve at high pressure would be contained by the check valve in the GDCS line. Discussion of the system is given in DCD Subsection 6.3.2.7. No Chapter 15 discussion is needed.

Table 15.0-1

MPL#	System Name	Increase in heat removal by the secondary system	Decrease in heat removal by the secondary system	Decrease in reactor coolant flow rate	Reactivity and power distribution anomalies	Increase in reactor coolant inventory	Decrease in reactor coolant inventory	Radioactive release from a subsystem or component
B11	REACTOR PRESSURE VESSEL SYSTEM[A]	N	N	N	N	N	N	N
B21	NUCLEAR BOILER SYSTEM	N	15.2.2.6,7	N	N	N	6.3; 15.3.13,14,15	15.3.13,14,15; 15.4.4[H],5[I]
B32	ISOLATION CONDENSER SYSTEM	N	N	N	N	15.2.4.1	6.3[G]	15.4.4[H];Y-1
C11	ROD CONTROL AND INFORMATION SYSTEM	N	N	N	15.3.8,9	N	N	N
C12	CONTROL ROD DRIVE SYSTEM	N	N	N	N	Y-2	N	N
C21	LEAK DETECTION AND ISOLATION SYSTEM[B]	N	N	N	N	N	N	N
C31	FEEDWATER CONTROL SYSTEM	N	N	N	N	15.2.4.2; 15.3.2	Y-3	N
C41	STANDBY LIQUID CONTROL SYSTEM	N	N	N	N	Y-4	6.3[G]	15.4.4[H]
C51	NEUTRON MONITORING SYSTEM	N	N	N	N	N	N	N
C61	REMOTE SHUTDOWN SYSTEM[B]	N	N	N	N	N	N	N
C62	NON-ESSENTIAL DCIS[B]	N	N	N	N	N	N	N
C63	ESSENTIAL DCIS[B]	N	N	N	N	N	N	N
C71	REACTOR PROTECTION SYSTEM	N	N	N	N	N	N	N
C72	DIVERSE PROTECTION SYSTEM	N	N	N	N	N	N	N
C74	SAFETY SYSTEM LOGIC AND CONTROL[B]	N	N	N	N	N	N	N
C82	PLANT AUTOMATION SYSTEM[B]	N	N	N	N	N	N	N
C85	STEAM BYPASS AND PRESS CONTROL SYS	15.2.5.1; 15.3.3	15.2.2.1; 15.3.4	N	N	N	N	N
D11	PROCESS RADIATION MONITORING SYSTEM[B]	N	N	N	N	N	N	N
D21	AREA RADIATION MONITORING SYSTEM	N	N	N	N	N	N	N
E50	GRAVITY-DRIVEN COOLING SYSTEM	N	N	N	N	Y-5	6.3[G]	15.4.4[H]
F11	FUEL SERVICING EQUIPMENT	N	N	N	N	N	N	15.4.1
F12	MISCELLANEOUS SERVICING EQUIPMENT	N	N	N	N	N	N	N

Table 15.0-1

MPL#	System Name	Increase in heat removal by the secondary system	Decrease in heat removal by the secondary system	Decrease in reactor coolant flow rate	Reactivity and power distribution anomalies	Increase in reactor coolant inventory	Decrease in reactor coolant inventory	Radioactive release from a subsystem or component
F13	RPV SERVICING EQUIPMENT	N	N	N	N	N	N	N
F14	RPV INTERNAL SERVICING EQUIPMENT	N	N	N	N	N	N	N
F15	REFUELING EQUIPMENT	N	N	N	15.3.10,11[L]	N	N	15.4.1
F16	FUEL STORAGE RACKS	N	N	N	N	N	N	N
F17	UNDER-RPV SERVICING EQUIPMENT	N	N	N	N	N	N	N
F21	CRD MAINTENANCE EQUIPMENT	N	N	N	N	N	N	N
F32	FUEL CASK CLEANING EQUIPMENT	N	N	N	N	N	N	N
F41	PLANT STARTUP AND TEST EQUIPMENT	N	N	N	N	N	N	N
F42	FUEL TRANSFER SYSTEM	N	N	N	N	N	N	N[C]
F43	LOOSE PARTS MONITORING SYSTEM	N	N	N	N	N	N	N
G21	FUEL AND AUX POOLS COOLING SYSTEM	N	N	N	N	N	N	N
G31	REACTOR WATER CLEANUP AND SHUTDOWN COOLING SYSTEM	15.3.12	15.2.2.9	N	N	N	6.3[G]	15.4.4[H],9
H11	MAIN CONTROL ROOM PANELS[B]	N	N	N	N	N	N	N
H12	MCR BACK ROOM PANELS[B]	N	N	N	N	N	N	N
H14	RADWASTE CONTROL ROOM PANELS[B]	N	N	N	N	N	N	N
H21	LOCAL PANELS AND RACKS[B]	N	N	N	N	N	N	N
J10	CORE AND FUEL SERVICES[D]	N/A	N/A	N/A	N/A	N/A	N/A	N/A
J11	NUCLEAR FUEL	N	N	N	N	N	N	15.4.1
J12	FUEL CHANNEL	N	N	N	N	N	N	N
K10	LIQUID WASTE MANAGEMENT SYSTEM	N	N	N	N	N	N	15.3.16
K20	SOLID WASTE MANAGEMENT SYSTEM	N	N	N	N	N	N	N
K30	OFFGAS SYSTEM	N	N	N	N	N	N	11.3.7

Table 15.0-1									
MPL#	System Name	Increase in heat removal by the secondary system	Decrease in heat removal by the secondary system	Decrease in reactor coolant flow rate	Reactivity and power distribution anomalies	Increase in reactor coolant inventory	Decrease in reactor coolant inventory	Radioactive release from a subsystem or component
N11	TURBINE MAIN STEAM SYSTEM	N	15.2.2.4,5; 15.3.6	N	N	N	6.3[G]	15.4.5[I]
N21	CONDENSATE AND FEEDWATER SYSTEM	15.2.1.1; 15.3.1[J]	N	N	N	15.2.4.2; 15.3.2	6.3	15.4.7
N22	HEATER DRAIN AND VENT SYSTEM	15.2.1.1; 15.3.1[J]	N	N	N	N	N	N
N25	CONDENSATE PURIFICATION SYSTEM	N	N	N	N	N	N	N
N31	MAIN TURBINE	N	15.2.2.2,3,4, 5; 15.3.5,6	N	N	N	6.3[G]	15.4.5[I]
N32	TURBINE CONTROL SYSTEM	N	15.2.2.2,3,4, 5; 15.3.5,6	N	N	N	N	N
N33	TURBINE GLAND SEAL SYSTEM	N	15.2.2.2,3,4, 5; 15.3.5,6	N	N	N	6.3[G]	15.4.5[I]
N34	TURBINE LUBRICATING OIL SYSTEM	N	15.2.2.2,3,4, 5; 15.3.5,6	N	N	N	N	N
N35	MOISTURE SEPARATOR REHEATER SYSTEM	N	N	N	N	N	6.3[G]	15.4.5[I]
N36	EXTRACTION SYSTEM	15.2.1.1; 15.3.1[J]	N	N	N	N	6.3[G]	15.4.5[I]
N37	TURBINE BYPASS SYSTEM	15.2.5.1	N	N	N	N	6.3[G]	15.4.5[I]
N39	TURBINE AUXILIARY STEAM SYSTEM	N	N	N	N	N	6.3[G]	15.4.5[I]
N41	GENERATOR	N	15.2.2.2,3,4, 5; 15.3.5,6	N	N	N	N	N
N42	HYDROGEN GAS COOLING SYSTEM	N	15.2.2.2,3,4, 5; 15.3.5,6	N	N	N	N	N
N43	GENERATOR COOLING SYSTEM	N	15.2.2.2,3,4, 5; 15.3.5,6	N	N	N	N	N
N44	GENERATOR SEALING OIL SYSTEM	N	15.2.2.2,3,4, 5; 15.3.5,6	N	N	N	N	N
N45	H2 AND CO2 BULK STORAGE	N	N	N	N	N	N	N

Table 15.0-1

MPL#	System Name	Increase in heat removal by the secondary system	Decrease in heat removal by the secondary system	Decrease in reactor coolant flow rate	Reactivity and power distribution anomalies	Increase in reactor coolant inventory	Decrease in reactor coolant inventory	Radioactive release from a subsystem or component
N51	EXCITER	N	15.2.2.2,3,4, 5; 15.3.5,6	N	N	N	N	N
N61	MAIN CONDENSER AND AUXILIARIES	N	15.2.2.2,3,4, 5,8; 15.3.5,6	N	N	N	6.3[G]	15.4.5[I]
N71	CIRCULATING WATER SYSTEM	N	15.2.2.2,3,4, 5,8; 15.3.5,6	N	N	N	N	N
P10	MAKE UP WATER SYSTEM	N	N	N	N	N	N	N
P21	REACTOR COMP COOLING WATER SYSTEM	N	N	N	N	N	N	N
P22	TURBINE COMP COOLING WATER SYSTEM	N	N	N	N	N	N	N
P25	CHILLED WATER SYSTEM	N	N	N	N	N	N	N
P30	CONDENSATE STOR AND TRANSFER SYSTEM	N	N	N	N	N	N	N
P32	OXYGEN INJECTION SYSTEM	N	N	N	N	N	N	N
P33	PROCESS SAMPLING SYSTEM	N	N	N	N	N	N	15.4.8
P41	PLANT SERVICE WATER SYSTEM	N	N	N	N	N	N	N
P51	SERVICE AIR SYSTEM	N	N	N	N	N	N	N
P52	INSTRUMENT AIR SYSTEM	N	15.2.2.6,7	N	N	N	N	N
P54	HIGH PRESSURE N2 SUPPLY SYSTEM	N	15.2.2.6,7	N	N	15.2.4[F]	N	N
P62	AUXILIARY BOILER SYSTEM	N	N	N	N	N	N	N
P63	HOT WATER SYSTEM	N	N	N	N	N	N	N
P73	HYDROGEN WATER CHEMISTRY SYSTEM	N	N	N	N	N	N	N
P74	ZINC INJECTION SYSTEM	N	N	N	N	N	N	N
R10	ELECTRIC POWER DISTRIBUTION SYSTEM[E]	N	15.2.5.2	N	N	N	N	N
R11	MEDIUM VOLTAGE DISTRIBUTION SYSTEM[E]	N	N	N	N	N	N	N
R12	LOW VOLTAGE DISTRIBUTION SYSTEM[E]	N	N	N	N	N	N	N
R13	UNINTERRUPTABLE AC POWER SUPPLY[E]	N	N	N	N	N	N	N

Table 15.0-1

MPL#	System Name	Increase in heat removal by the secondary system	Decrease in heat removal by the secondary system	Decrease in reactor coolant flow rate	Reactivity and power distribution anomalies	Increase in reactor coolant inventory	Decrease in reactor coolant inventory	Radioactive release from a subsystem or component
R14	INSTR AND CONTROL POWER SUPPLY[E]	N	N	N	N	N	N	N
R15	LIGHTING AND SERVICING POWER SUPPLY[E]	N	N	N	N	N	N	N
R16	DIRECT CURRENT POWER SUPPLY[E]	N	N	N	N	N	N	N
R21	STANDBY ON SITE AC POWER SUPPLY	N	N	N	N	N	N	N
R31	RACEWAY SYSTEM	N	N	N	N	N	N	N
R41	PLANT GROUNDING SYSTEM	N	N	N	N	N	N	N
R51	COMMUNICATION SYSTEM	N	N	N	N	N	N	N
S21	SWITCH YARD	N	15.2.2.2,3; 15.3.5	N	N	N	N	N
T10	CONTAINMENT SYSTEM	N	N	N	N	N	N	N
T11	CONTAINMENT VESSEL	N	N	N	N	N	N	N
T12	CONTAINMENT INTERNAL STRUCTURES	N	N	N	N	N	N	N
T15	PASSIVE CONTAINMENT COOLING SYSTEM	N	N	N	N	N	N	N
T31	CONTAINMENT INERTING SYSTEM	N	N	N	N	N	N	N
T41	DRYWELL COOLING SYSTEM	N	N	N	N	N	N	N
T62	CONTAINMENT MONITORING SYSTEM	N	N	N	N	N	N	N
T64	ENVIRONMENTAL MONITORING SYSTEM	N	N	N	N	N	N	N
U31	CRANES	N	N	N	N	N	N	N
U36	ELECTRICAL BUILDING HVAC[K]	N	N	N	N	N	N	N
U37	SERVICE BUILDING HVAC[K]	N	N	N	N	N	N	N
U38	RADWASTE BUILDING HVAC[K]	N	N	N	N	N	N	N
U39	TURBINE BUILDING HVAC[K]	N	N	N	N	N	N	N
U40	REACTOR BUILDING HVAC[K]	N	N	N	N	N	N	N
U41	OTHER BUILDING HVAC[K]	N	N	N	N	N	N	N

Table 15.0-1

MPL#	System Name	Increase in heat removal by the secondary system	Decrease in heat removal by the secondary system	Decrease in reactor coolant flow rate	Reactivity and power distribution anomalies	Increase in reactor coolant inventory	Decrease in reactor coolant inventory	Radioactive release from a subsystem or component
U42	POTABLE WATER AND SAN WASTE SYSTEM	N	N	N	N	N	N	N
U43	FIRE PROTECTION SYSTEM	N	N	N	N	N	N	N
U44	SANITARY WASTE DISCHARGE SYSTEM	N	N	N	N	N	N	N
U50	EQUIPMENT AND FLOOR DRAIN SYSTEM	N	N	N	N	N	N	N
U65	OTHER BUILDING STRUCTURES	N	N	N	N	N	N	N
U71	REACTOR BUILDING STRUCTURE	N	N	N	N	N	N	N
U72	TURBINE BUILDING STRUCTURE	N	N	N	N	N	N	N
U73	CONTROL BUILDING STRUCTURE	N	N	N	N	N	N	N
U74	RADWASTE BUILDING STRUCTURE	N	N	N	N	N	N	N
U75	SERVICE BUILDING STRUCTURE	N	N	N	N	N	N	N
U77	CONTROL BUILDING HVAC[K]	N	N	N	N	N	N	N
U78	COLD MACHINE SHOP	N	N	N	N	N	N	N
U80	ELECTRICAL BUILDING STRUCTURE	N	N	N	N	N	N	N
U81	SEISMIC MONITORING SYSTEM	N	N	N	N	N	N	N
U84	SERVICE WATER BUILDING STRUCTURE	N	N	N	N	N	N	N
U85	SERVICE WATER BUILDING HVAC[K]	N	N	N	N	N	N	N
U91	ADMINISTRATION BUILDING STRUCTURE	N	N	N	N	N	N	N
U93	TRAINING CENTER	N	N	N	N	N	N	N
U95	HOT MACHINE SHOP	N	N	N	N	N	N	N
U97	FUEL BUILDING STRUCTURE	N	N	N	N	N	N	N
U98	FUEL BUILDING HVAC[K]	N	N	N	N	N	N	N
U99	STACK	N	N	N	N	N	N	N
W12	INTAKE AND DISCHARGE STRUCTURES	N	N	N	N	N	N	N

Table 15.0-1								
MPL#	System Name	Increase in heat removal by the secondary system	Decrease in heat removal by the secondary system	Decrease in reactor coolant flow rate	Reactivity and power distribution anomalies	Increase in reactor coolant inventory	Decrease in reactor coolant inventory	Radioactive release from a subsystem or component
W24	COOLING TOWER	N	N	N	N	N	N	N
W32	SCREEN CLEANING FACILITY	N	N	N	N	N	N	N
W33	SCREENS	N	N	N	N	N	N	N
W41	INTAKE STRUCTURE POWER SUPPLY	N	N	N	N	N	N	N
Y12	ROADS AND WALKWAYS	N	N	N	N	N	N	N
Y21	TANKS AND EQUIPMENT PADS	N	N	N	N	N	N	N
Y41	STATION WATER SYSTEM	N	N	N	N	N	N	N
Y46	CATHODIC PROTECTION SYSTEM	N	N	N	N	N	N	N
Y47	METEOROLOGICAL OBSERVATION SYSTEM	N	N	N	N	N	N	N
Y51	YARD MISCELLANEOUS DRAIN SYSTEM	N	N	N	N	N	N	N
Y52	OIL STORAGE AND TRANSFER SYSTEM	N	N	N	N	N	N	N
Y53	CHEM STORAGE AND TRANSFER SYSTEM	N	N	N	N	N	N	N
Y71	PIPING DUCT	N	N	N	N	N	N	N
Y72	CABLE DUCT	N	N	N	N	N	N	N
Y86	SITE SECURITY[D]	N	N	N	N	N	N	N

Table Key

N	No credible failure in this system or operator error of this system can result in an event of the category above.
DCD Section	A credible failure in this system or operator error of this system can result in an event of the category above and the event is discussed/analyzed in the DCD Section number shown.
Y-#	A credible failure in this system or operator error of this system can result in the event category above and the event is NOT discussed/analyzed in the DCD. Discussion item # discusses the event.

Notes:

A Failures of the RPV are not considered credible.

B Failures or operator errors in these systems are considered in the systems that they control.

C Discussion in DCD Section 9.1.4.12 addresses failure modes of the FTS.

D Not a physical System.

E Failure in electrical systems can impact various mechanical and control systems. Except for the loss of non-emergency AC power to station auxiliaries, all electrical failures are considered in the system they impact. For example a loss of feedwater can be caused by a postulated failure in the medium voltage electrical system. However, for the purpose of this table that failure is considered in the condensate and feedwater system.

F Failure in the high pressure N2 supply system could cause inadvertent isolation condenser injection on one isolation condenser. This is bounded by the event postulated in the DCD (inadvertent injection of all isolation condensers).

G The LOCA analysis in Section 6.3 considers a spectrum of postulated line breaks to evaluate the ECCS performance. Limiting events are identified but all possible line break scenarios are not identified. For purposes of this table all systems which could potentially result in a decrease in reactor coolant inventory are considered evaluated by DCD section 6.3

H The LOCA radiological analysis in Section 6.3 considers the worst case piping break inside containment based in part on the analysis in Section 6.3. As in 6.3 all possible line break scenarios are not identified. For purposes of this table all systems which could potentially result in a radiological release from a subsystem or component in containment are considered evaluated by DCD section 15.4.4.

I The main steam line break radiological analysis in section 15.4.5 considers a limiting steam line break outside of containment. As stated in the section "This postulated event represents the envelope evaluation of steam line failures outside of containment."

J This event is loss of feedwater heating. This event is located in the "increase in heat removal by the secondary system" for lack of a better category. In the DCD this is considered under the event category "Decrease in Core Coolant Temperature" which is more appropriate for the BWR design.

K Failure in HVAC systems can impact various mechanical, electrical and control systems. All HVAC failures are considered in the system they impact. For example, a failure of an HVAC system that causes a turbine support system to fail and results in a turbine trip before the operators can react is considered in the turbine support system.

L Fuel assembly loading errors are not caused by failures in the refueling equipment but are caused by human errors during operation of the equipment combined with failure of the verification procedures.

NRC RAI 15.0-2

Provide a table in the DCD listing all of the non-safety related systems and components used for mitigating transients and accidents analyzed in DCD Tier 2, Chapter 15.

GE Response

The systems and components used for mitigating transients (AOOs and infrequent events) and accidents are provided below. The systems and components are identified as safety-related or nonsafety-related. This data will be incorporated into DCD Table 15.1-5 in a future revision of the DCD. Further discussion is provided for the nonsafety-related systems below.

System or Component	Safety Design[1]
Reactor Pressure Vessel System[2]	SR
Nuclear Boiler System[2]	SR
Connection valves (Dryer/Separator storage pool to Passive Containment Cooling System pool)	SR
Containment Isolation	SR
Control Rod Drive System - Scram Function	SR
Control Rod Drive System - Makeup Water	NS
Control Rod Drive System - Selected Control Rod Run In Function	NS
Control Building HVAC - Control Room Isolation and Pressurization	SR
Drywell to Wetwell vacuum breaker valves	SR
Drywell to Wetwell vacuum breaker isolation valves	SR
Fuel and Auxiliary Pool Cooling System - Suppression Pool Cooling	NS
Fuel Building HVAC System - Isolation	SR
Drywell to suppression pool float valves to spillover pipes	SR
Main feedwater pump breakers trip	SR
Feedwater Control System	NS
Gravity-Driven Cooling System	SR
Isolation Condenser System	SR
Main Steam Isolation Valves	SR
Neutron Monitoring System	SR
Passive Containment Cooling System	SR
Process Radiation Monitoring System	SR
Reactor Building HVAC - Isolation	SR
Rod Control and Information System	NS
Reactor Protection System	SR
Steam Bypass and Control System	NS
Standby Liquid Control System	SR

Notes:

1 Safety Designation: SR designates that the system or component listed, or the specific function of the system or component listed is safety-related. NS designates that the system or component listed, or the specific function of the system or component listed is nonsafety-related.

2 The Reactor Pressure Vessel System and the Nuclear Boiler System contain several safety-related functions/subsystems used for mitigating the transients and accidents analyzed in DCD Tier 2, Chapter 15.

Control Rod Drive System (CRDS): The high pressure makeup water function of this system is credited in several event scenarios as backup level control to feedwater. This function of CRDS is nonsafety-related. If credit is not taken for the high pressure makeup water function of the CRDS, then the Isolation Condenser System and Gravity-Driven Cooling System would ensure acceptable inventory control. The fine motion CRD subsystem in conjunction with the Rod Control and Information System performs the Selected Control Rod Run In (SCRRI) function. The SCRRI function is credited for the mitigation of the loss of feedwater heating AOO. The SCRRI function is not credited in any accident scenario.

Fuel and Auxiliary Pool Cooling System (FAPCS): The suppression pool cooling mode of FAPCS is credited for long term cooling following an inadvertent opening of an SRV/DPV, or stuck open SRV in Tables 15.1-5 and 15.3-12. This function of FAPCS is not safety-related. However, as discussed in the event sections (DCD 15.3.13,14,15) of these events, FAPCS in not needed. With no operation of the FAPCS the suppression pool would heat up to its scram setpoint and initiate a scram (if scram has not already occurred) and the containment design limits would not be challenged. Therefore, FAPCS is not required to mitigate the consequences of any design basis event. Tables 15.1-5 and 15.3-12 will be changed in a future revision of the DCD to clarify use of FAPCS.

Feedwater Control System (FWCS): Function of this system is modeled or assumed in several AOO and Infrequent Events to control water level by controlling feedwater flow and therefore mitigates the severity of the events. The FWCS also detects a loss of feedwater heating and sends the signal to RC&IS for initiation of the nonsafety-related Selected Control Rod Run In function. The system is not credited in any accident scenario.

Rod Control and Information System (RC&IS): This system is credited to mitigate control rod withdrawal errors during power operations using rod block functions. The system is not credited in any accident scenario.

Steam Bypass and Pressure Control System (SB&PCS): Function of this system is modeled or assumed in several AOO and Infrequent Events to control reactor pressure and therefore mitigates the severity of the events. The system is not credited in any accident scenario.