GE Energy

**James C. Kinsey**
Project Manager, ESBWR Licensing

PO Box 780 M/C J-70
Wilmington, NC 28402-0780
USA

T 910 675 5057
F 910 362 5057
jim.kinsey@ge.com

MFN 07-022

Docket No. 52-010

January 19, 2007

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

Subject: **Response to Portion of NRC Request for Additional Information Letter No. 63 Related to ESBWR Design Certification Application – Technical Specifications – RAI Numbers 16.2-23, 16.2-30, 16.2-45, 16.2-50, 16.2-54, 16.2-73, 16.2-74, and 16.2-76**

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

If you have any questions or require additional information regarding the information provided here, please contact me.

Sincerely,

Kathy Sedney for

James C. Kinsey
Project Manager, ESBWR Licensing

D068

Reference:

1. MFN 06-375, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 63 Related to ESBWR Design Certification Application,* October 4, 2006

Enclosure:

1. MFN 07-022 – Response to Portion of NRC Request for Additional Information Letter No. 63 Related to ESBWR Design Certification Application – Technical Specifications – RAI Numbers 16.2-23, 16.2-30, 16.2-45, 16.2-50, 16.2-54, 16.2-73, 16.2-74, and 16.2-76

cc: AE Cubbage USNRC (with enclosures)
GB Stramback GE/San Jose (with enclosures)
eDRFs 59-5537, 60-4196, 60-4198, 60-4199

# Enclosure 1

# MFN 07-022

# Response to Portion of NRC Request for

# Additional Information Letter No. 63

# Related to ESBWR Design Certification Application

# - Technical Specifications -

# RAI Numbers 16.2-23, 16.2-30, 16.2-45, 16.2-50, 16.2-54, 16.2-73, 16.2-74, and 16.2-76

**NRC RAI 16.2-23**

*Proposed SR 3.9.1.1 requires performing a CHANNEL FUNCTIONAL TEST on the required refueling equipment interlock inputs. Verify that the appropriate definition of CFT will be used in view of the likelihood that the digital I&C will have a different definition. This comment may be related to RAI 16.2-26.*

**GE Response**

The definition for CHANNEL FUNCTIONAL TEST (CFT) used for protection instrumentation is also applicable to the refueling equipment interlocks. GE explained the application of the term CHANNEL FUNCTIONAL TEST (CFT) to specific components in the Reactor Protection System (RPS), Main Steam Isolation Valve (MSIV) isolation, and Engineered Safety Features (ESF) functions in the response to RAI 16.2-26, in letter MFN 06-285 dated August 21, 2006.

For the refueling equipment interlocks, the detailed design necessary to explain the application of the term CFT to specific components has not been completed. However, as described in Revision 2 of the Design Control Document (DCD) Tier 2, Subsection 7.7.2.2.1, the refueling interlocks include signals for refueling machine position and refueling machine main hoist loading. These signals are provided to the redundant rod action and position information (RAPI) panels of the Rod Control and Information System (RC&IS). The electronic components comprising the RC&IS, including the RAPI panels, include digital instrumentation similar to the RPS, MSIV isolation, and ESF instrumentation systems. The RAPI panels consist of a RAPI-A and a RAPI-B panel. The RAPI panels include the logic for comparing the refueling equipment position and loading input signals with control rod position indication input signals to determine if a control rod block is necessary, or if limiting refueling machine operation is needed.

Similar to the other digital instrumentation systems, Limiting Condition for Operation 3.9.1 includes the instrument sensors (limit switches, load switches, and control rod position indication electronic signals), any associating multiplexing equipment for transmission of the input signals to the RAPI panels, and the logic of the RAPI panels that result in control rod blocks or blocking of refueling machine operations in the definition of the sensing channels. Therefore, a CFT would include verification that these components properly function from input sensors through the output of the logic for each of the RAPI panels.

**DCD Impact**

No DCD changes will be made in response to this RAI.

**NRC RAI 16.2-30**

*The staff proposes the following additions the ESBWR TS Section 3.7.1, Emergency Breathing Air System (EBAS):*

a. *Add an "Action Item" as follows: ACTION CONDITION: Control room temperature not within limit REQUIRED ACTION: Restore control room air temperature to within limit COMPLETION TIME: "XX" hours (Applicant should provide completion time)*

b. *Provide a new SURVEILLANCE REQUIREMENTS (SR) 3.7.1.X1 for air quality of the air storage tanks to meet the requirements of Appendix C, Table C-1 of ASHRAE Standard 62 and associated "FREQUENCY" of "7 days."*

c. *Restate SR 3.7.1.4 to state "Verify that each EBAS train maintains the control room boundary at a positive pressure of 31Pascals (0.125 inches of W.G.) relative to the adjacent areas at the required air addition flow rate of 9.5 liter/second (100 scfm) ± tolerance limit (i.e 0.5 liter/second ( 5 scfm) using the safety related EBAS air storage tanks." Also, provide SR FREQUENCY as "24 months."*

d. *Provide a new SR 3.7.1.X2 for EBAS pressure relief isolation valves to state "Verify that each EBAS pressure relief isolation valve within the control room boundary is OPERABLE." Also, provide SR FREQUENCY as "In accordance with the Inservice Testing Program."*

e. *Provide a new SR 3.7.1.X3 for EBAS pressure relief dampers to state "Verify that each EBAS pressure relief damper is OPERABLE." Also, provide SR FREQUENCY as "24 months."*

f. *Provide a new SR 3.7.1.X4 for EBAS pressure regulating valves to state "Verify that pressure regulating valve in each EBAS train is OPERABLE." Also, provide "SR FREQUENCY" as "In accordance with the Inservice Testing Program."*

g. *Provide the associated BASES information in details for the above LCO Action Items and SRs in DCD Tier 2 Section B.3.7.1, "Emergency Breathing Air System (EBAS)."*

h. *In DCD Tier 2 Chapter 16, Technical Specifications 3.7.1, "Emergency Breathing Air System," the applicant did not provide a list of Codes and Standards used in Technical Specifications 3.7.1. The Technical Specifications Bases typically reference ASTM Standards, ASHRAE Standards, Regulatory Guides, the Code of Federal Regulations, and others. Therefore, the applicant should provide (as references) a list of Codes and Standards used in the Bases of Technical Specifications 3.7.1. The NRC staff expects a commitment to the latest revisions of the applicable Codes and Standards included in the DCD.*

**GE Response**

GE is currently evaluating the Control Room Habitability Area Heating and Ventilation System (CRHAHVS) and Emergency Breathing Air System (EBAS) functional requirements and design as a result of continuing development of the control room habitability dose analyses. Because of these ongoing activities, it is anticipated that the CRHAHVS and EBAS design requirements may change in the near future affecting the response to each of the specific issues contained in this RAI. It is the intent of GE to address each of the requested changes in this RAI in

Revision 3 of DCD Chapter 16 and Chapter 16B, and to provide a supplemental response to this RAI at that time.

**DCD Impact**

No changes will be made at this time to DCD Tier 2, Chapter 16 or Chapter 16B as a result of this RAI.

**NRC RAI 16.2-45**

*Justify deviations from NUREG-1434, Rev.3, STS 3.6.1.3 Applicability - are no ESBWR containment isolation instrumentation functions required in Mode 5?*

**GE Response**

NUREG-1434, Revision 3, Standard Technical Specification (STS) Limiting Condition for Operation (LCO) 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," specifies the requirements for PCIVs. LCO 3.6.1.3 is applicable in Modes 1, 2, and 3, and in other Modes when the associated instrumentation is required to be Operable per LCO 3.3.6.1, "Primary Containment Isolation Valve Instrumentation." A review of NUREG-1434, LCO 3.3.6.1 shows that the following Primary Containment Isolation Instrumentation Functions are required to be Operable in Modes other than Modes 1, 2, or 3:

- Primary Containment Isolation due to Containment and Drywell Ventilation Exhaust Radiation – High (required during movement of recently irradiated fuel assemblies in primary or secondary containment, or operations with a potential for draining the reactor vessel); and
- Shutdown Cooling System Isolation due to Reactor Vessel Water Level – Low, Level 3 (required in Modes 4 and 5).

ESBWR Design Control Document (DCD), Tier 2, Revision 1, Chapter 16 LCO 3.6.1.3, "Containment Isolation Valves (CIVs)," specifies the requirements for ESBWR CIVs. LCO 3.6.1.3 is applicable in Modes 1, 2, 3, and 4. With respect to isolation instrumentation, the requirements of NUREG-1434, LCO 3.3.6.1 were reformatted into four specifications. LCO 3.3.6.1, "Main Steam Isolation Valve (MSIV) Instrumentation," specifies the requirements for instrumentation associated with MSIV isolation and LCO 3.3.6.3, "Isolation Instrumentation," specifies the requirements for instrumentation associated with CIVs and the reactor building boundary isolation dampers. In addition, the actuation logic associated with these instrumentation specifications is separately specified in LCO 3.3.6.2, "Main Steam Isolation Valve (MSIV) Actuation," and in LCO 3.3.6.4, "Isolation Actuation," respectively. With respect to containment isolation, a review of the related specifications (i.e., LCO 3.3.6.1, LCO 3.3.6.2, LCO 3.3.6.3, and LCO 3.3.6.4 shows that none of the specified Functions are applicable in Modes other than Modes 1, 2, 3, and 4. Therefore, the applicability of LCO 3.6.1.3 is consistent with the requirements LCO 3.3.6.1, LCO 3.3.6.2, LCO 3.3.6.3, and LCO 3.3.6.4.

The isolation instrumentation provides the capability to generate isolation signals to the containment isolation valves and the reactor building boundary isolation dampers. The function of the isolation valves and dampers, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs). NUREG-1434, LCO 3.3.6.1 requires the Primary Containment Isolation due to Containment and Drywell Ventilation Exhaust Radiation – High Function to be Operable during movement of recently irradiated fuel assemblies in primary or secondary containment, or operations with a potential for draining the reactor vessel. A review of the NUREG-1434, LCO 3.3.6.1 Bases shows that this function is required to be Operable during movement of recently irradiated fuel assemblies in primary or secondary containment, or operations with a potential for draining the reactor vessel because the Function is assumed to initiate isolation of the primary containment during a fuel handling accident. The Fuel Handling Accident described in the ESBWR DCD,

Revision 1 Section 15.4.1, "Fuel Handling Accident," does not credit automatic isolation of CIVs or the reactor building boundary isolation dampers while showing acceptable dose consequences. Therefore, this function is not required to be Operable during movement of irradiated fuel or operations with a potential for draining the reactor vessel.

NUREG-1434, LCO 3.3.6.1 requires the Shutdown Cooling System Isolation due to Reactor Vessel Water Level – Low, Level 3 Function to be Operable in Modes 3, 4, and 5. A review of the NUREG-1434, LCO 3.3.6.1 Bases shows that this function is required to be Operable in Modes 3, 4, and 5 to support actions to ensure that the reactor pressure vessel water level does not drop below the top of the active fuel during a vessel draindown event caused by a leak (e.g., pipe break or inadvertent valve opening) in the Residual Heat Removal (RHR) Shutdown Cooling System. This Function is not required, in Modes 3, 4, and 5, to support containment isolation, but is required to support system isolation in these Modes.

The ESBWR is a passive plant and does not have the traditional RHR system. For normal shutdown and cooldown, residual and decay heat is removed via the main condenser and the Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) system, described in DCD, Tier 2, Revision 1, Section 5.4.8. The RWCU/SDC CIVs are instrumented to isolate the RPV upon detection of a leak from the RWCU/SDC system. This isolation feature protects the reactor core by minimizing the potential loss of RPV coolant inventory. Because this isolation function is not required to close CIVs to limit fission product release during and following postulated DBAs, RWCU/SDC isolation on it was not included as a required Function in LCO 3.3.6.3. GE proposes to revise DCD, Tier 2, Chapter 16 and 16B to include a new specification in Section 3.4, "Reactor Coolant System (RCS)," to address the RWCU/SDC system isolation valves and reactor vessel isolation function in shutdown modes.

**DCD Impact**

DCD Tier 2, Chapter 16 and Chapter 16 B will be revised in a future update as noted in the above discussion.

**NRC RAI 16.2-50**

*TS SR 3.6.3.1.4 has no analogous surveillance in the BWR/4 secondary containment and secondary containment isolation valve specifications in NUREG-1433, Rev 3.1. Discuss the flow paths to be isolated and the method to be used for conducting the test and the justification for the 60-month Frequency.*

**GE Response**

ESBWR Design Control Document (DCD), Tier 2, Revision 1, Chapter 16, Technical Specification (TS) Surveillance Requirement (SR) 3.6.3.1.4, requires verification that the Reactor Building exfiltration rate is within limits, once every 60 months. DCD, Tier 2, Revision 1, Section 6.2.3.1 states that the Reactor Building is capable of periodic testing to assure that the leakage rates assumed in the radiological analyses are met.

DCD, Tier 2, Revision 1, Chapter 16, SR 3.6.3.1.4 is consistent with this intent to perform testing to assure that the containment fission product removal evaluation model assumptions for Reactor Building leakage are met. The construction level design details necessary to fully describe the flow paths to be isolated and the methods to be used to conduct the testing are not available at this time. Actual plant procedure development program is described in DCD Section 13.5. Final plant surveillance testing procedures will be required to be in place and satisfactorily completed prior to operations crediting the Reactor Building operability.

The Reactor Building provides an added holdup volume for fission products released from the containment in the event of an accident. Unlike the comparison to secondary containment designs represented by NUREG-1434 testing, there is no active ventilation nor significant differential pressure generated in response to an accident condition. Any leakage would simply be a result of adiabatic building heatup, and minimal out leakage from primary containment. The vast majority of adiabatic heatup is via the Passive Containment Cooling (PCC) pools, where heat rejection by boiling results in vapor release to the steam space above each pool where it is released to the atmosphere through large-diameter discharge vents. The remainder of Reactor building leakage would not be expected to be significant, and general building integrity inspections along with a 60-month in-depth confirmatory evaluation is expected to provide adequate assurance that the assumptions remain valid.

**DCD Impact**

No DCD changes will be made in response to this RAI.

**NRC RAI 16.2-54**

*Staff & industry have reached agreement on control room envelope habitability STS improvements with action, surveillance, and administrative requirements for the CRE boundary. Adopt TSTF-448, Rev 3 in the next ESBWR DCD revision.*

**GE Response**

GE is currently evaluating the Control Room Habitability Area Heating and Ventilation System (CRHAHVS) and Emergency Breathing Air System (EBAS) functional requirements and design as a result of continuing development of the control room habitability dose analyses. Because of these ongoing activities, it is anticipated that the CRHAHVS and EBAS design requirements may change in the near future affecting the implementation of TSTF-448, Revision 3, as requested by this RAI. Therefore, GE is evaluating the proposed changes to NUREG-1434, Revision 3.1, in TSTF-448, Revision 3, and the resulting industry adoption of those changes, against the evolving CRHAHVS and EBAS design and will adopt TSTF-448, Revision 3, to the extent applicable to the ESBWR design, in Revision 3 of DCD Chapter 16 and Chapter 16B.

**DCD Impact**

GE will adopt TSTF-448, Revision 3, to the extent applicable to the ESBWR design, in Revision 3 of DCD Chapter 16 and Chapter 16B.

**NRC RAI 16.2-73**

*DCD 16B, TS 3.5.3 states that, in MODES 5 and 6, GDCS is used to provide additional water inventory inside the containment to respond to a loss of decay heat removal capability or a loss of reactor coolant inventory. The Applicability basis states that operability in mode 6 is not required when the new fuel pool gate is removed and water level is above the specified level over the top of the reactor pressure vessel flange because of the additional inventory available when in this configuration. However, this inventory is not protected in that it may be lost through various paths including failure of the non-seismic and nonsafety refueling seal or fuel transfer system.*

*The Bases for the Residual Heat Removal-High Water Level TS in NUREG-1434, under Actions, describe that the residual heat removal system provides reliable heat removal for loss of cooling water inventory conditions initiating from the high water level conditions. However, the proposed ESBWR Bases for TS 3.5.5 states that RWCU/SDC is a nonsafety-related system [that cannot be assumed to remain available following an equipment failure or a loss of offsite power] and that, once the reactor vessel head is removed, loss of the normal decay heat removal method could result in boiling in the vessel. The ESBWR Bases go on to state that water in the GDCS pools is a source of reactor coolant inventory for this mode of decay heat removal.*

*Provide justification for limiting the applicability of GDCS injection operability to operational Mode 6, "refueling," with water level less than 23 feet above the reactor vessel flange.*

**GE Response**

ESBWR Technical Specifications (TSs) establish operability requirements for safety-related decay heat removal (DHR) capability only. The safety-related DHR capability provides adequate cooling following the unlikely loss of both of the redundant trains of the reactor water cleanup/shutdown cooling (RWCU/SDC) system, which provide the nonsafety-related decay heat removal capability.

Design Control Document (DCD) Tier 2, Revision 1, Chapter 16, LCO 3.5.3, "Gravity-Driven Cooling System (GDCS) - Shutdown," provides safety-related DHR capability when in Mode 6 but prior to water level being increased greater than 7.01 meters (23.0 feet) over the top of the reactor pressure vessel flange with the new fuel pool gate removed (i.e., reactor pressure vessel (RPV) not flooded).

When in Mode 6 with the water level greater than 7.01 meters (23.0 feet) over the top of the reactor pressure vessel flange and the new fuel pool gate removed (i.e., RPV flooded), the large amount of water stored above the core provides sufficient DHR capability. This approach is consistent with BWR/6 Standard Technical Specifications (STS), NUREG-1434, Revision 3.1, which also use the volume of water stored above the core as the safety-related DHR capability when the RPV is flooded.

As explained in sections 13 and 16 of NEDO-33201, Revision 1, "Licensing Topical Report: ESBWR Certification Probabilistic Risk Assessment," dated September 2006, the risk associated with loss of decay heat removal in Mode 6 is negligible. When in Mode 6 with the RPV flooded, the Fuel and Auxiliary Pool Cooling System (FAPCS) can be aligned to cool water in the reactor well, which provides an additional alternative to the RWCU/SDC. Additionally, the cooling provided by the water stored above the core provides a substantial amount of time to establish

makeup from alternate sources including control rod drive pumps, FAPCS pumps, condensate pumps, or firewater pumps.

Loss of coolant accidents (LOCAs) in Mode 6 (RPV flooded and not flooded) are addressed in section 16 of NEDO-33201, Revision 1, which addresses LOCAs at four break locations: the GDCS injection line, feedwater line, break above top of active fuel (TAF) other than GDCS or feedwater lines, and breaks below the TAF. Only break elevations below the level that automatically isolates RWCU/SDC are analyzed because RWCU/SDC continues removing the decay heat and no safety function is directly challenged for breaks above this level. A loss of inventory via a break in the refueling seal or fuel transfer system, as postulated in this RAI, is not considered a significant risk because a very substantial amount of coolant inventory would remain below the break location but above the TAF. In the unlikely event of a coincident loss of RWCU/SDC, the cooling provided by the water stored above the core provides a substantial amount of time to establish makeup from alternate sources including control rod drive pumps, FAPCS pumps, condensate pumps, or firewater pumps.

**DCD Impact**

No DCD changes will be made in response to this RAI.

**NRC RAI 16.2-74**

*Provide basis for not including an operability requirement for a decay heat removal method in the refueling mode with the head fully detensioned or removed.*

*DCD Tier 2, Rev. 1, Chapter 16, TS 3.5.5 states that use of the ICS as an emergency backup for decay heat removal in MODE 6 requires the reactor vessel head to be in place. Once the reactor vessel head is removed, loss of the normal decay heat removal method could result in boiling in the vessel. NUREG 1434, Rev. 3.1, specifies one or more heat removal paths operable, depending on water level, and one in operation. If the operating loop fails, an alternate residual heat removal loop must be placed in operation. If no alternate heat removal path is available at high water level, the required action specifies operation of the standby gas treatment system and establishment of secondary containment.*

*Provide the basis for not including an Operability requirement for a decay heat removal method in the refueling mode with the head fully detensioned or removed. Since the basis for TS 3.5.5 describes boiling within the vessel, describe how the heat would be transferred to an ultimate heat sink, and how the potential effects of boiling would be managed.*

**GE Response**

ESBWR Technical Specifications (TSs) do establish Operability requirements for safety-related decay heat removal (DHR) capability when in the refueling mode with the head fully detensioned or removed. The safety-related DHR capability provides adequate cooling following the unlikely loss of both of the redundant trains of the reactor water cleanup/shutdown cooling (RWCU/SDC) system, which provide the nonsafety-related decay heat removal capability in all Modes, including Mode 6 with head fully detensioned or removed.

Design Control Document (DCD) Tier 2, Revision 1, Chapter 16, LCO 3.5.3, "Gravity-Driven Cooling System (GDCS - Shutdown," provides safety-related DHR capability when in Mode 6 prior to water level being increased greater than 7.01 meters (23.0 feet) over the top of the reactor pressure vessel flange with the new fuel pool gate removed (i.e., reactor pressure vessel (RPV) not flooded). LCO 3.5.3 requires operability of four branch lines of the GDCS injection subsystem capable of injecting a combined volume equal to the volume of the two smaller GDCS pools when each is filled to the normal operating level. Prior to the removal of the head, GDCS operability requires sufficient RPV venting capacity to maintain the RPV depressurized following loss of the normal DHR capability. If the Automatic Depressurization System (ADS) is selected as the available vent path, decay heat is released to the containment atmosphere. If requirements for GDCS are not met, LCO 3.5.3, Actions, require establishing an operable Reactor Building boundary, similar to the actions in NUREG-1434 for establishing secondary containment.

When in Mode 6 with the water level greater than 7.01 meters (23.0 feet) over the top of the reactor pressure vessel flange and the new fuel pool gate removed (i.e., RPV flooded), the large amount of water stored above the core provides the safety-related DHR capability. Decay heat is released to the containment atmosphere. This approach is consistent with BWR/6 Standard Technical Specifications (STS), NUREG-1434, Revision 3.1, which also use the volume of water stored above the core as a safety-related DHR capability when the RPV is flooded.

General Electric (GE) has revised DCD Tier 2, Revision 1, Chapter 16, LCO 3.5.5, Isolation Condenser System (ICS) – Shutdown," to change the Applicability to eliminate "Mode 6 when the reactor head is in place" because adequate safety-related DHR capability when in this configuration is provided by GDCS in LCO 3.5.3. As described above, GDCS-DHR capability provides the required cooling function following the unlikely loss of both of the redundant trains of the RWCU/SDC system, which provide the normal decay heat removal capability in all Modes, including Mode 6 with the head fully detensioned or removed.

**DCD Impact**

No additional DCD changes will be made in response to this RAI.

**NRC RAI 16.2-76**

*DCD Tier 2, section 9.1.2.7, states that on a complete loss of the FAPCS active cooling capability and under the condition of maximum heat load, sufficient quantity of water is available in the Spent Fuel Pool above the top of active fuel (TAF) level to allow boiling for 72 hours and still have the TAF at least 3.0 m (10 ft) submerged under water. The water level necessary to provide this heat removal capacity constitutes an initial condition of a transient analysis for a loss of forced cooling. The loss of inventory presents a challenge to a fission product barrier in that water cooling is necessary to assure protection of the fuel cladding.*

*Describe how the water level necessary to satisfy this transient analysis is included in a Limiting Condition for Operation consistent with the requirements of 10 CFR 50.36(c)(2)(ii), Criterion 2.*

**GE Response**

The complete loss of the Fuel and Auxiliary Pools Cooling System (FAPCS) is not currently analyzed as an Anticipated Operational Occurrence (AOO), Infrequent Event, or Design Basis Accident (DBA) in Revision 2 of the Design Control Document (DCD) Tier 2, Chapter 15. However, an analysis to ensure that the design features and assumptions necessary to maintain adequate cooling for this event is described in DCD Tier 2, Subsection 9.1.2.7, which states: "On a complete loss of the FAPCS active cooling capability and under the condition of maximum heat load, sufficient quantity of water is available in the Spent Fuel Pool above the top of active fuel (TAF) level to allow boiling for 72 hours and still have the TAF submerged under water." The water level assumed in this analysis is already bounded by the water level required in DCD Tier 2, Chapter 16, Technical Specification 3.7.5, which is included in the Technical Specifications as an initial condition for the fuel handling accident safety analyses described in DCD Tier 2, Section 15.1.4, "Fuel Handling Accident." However, since the complete loss of the FAPCS is not an analyzed AOO or DBA, the initial conditions assumed in the evaluation of that event do not meet Criterion 2 of 10 CFR 50.36(c)(2)(ii).

**DCD Impact**

No DCD changes will be made in response to this RAI.