ENCLOSURE 1

WOG CLARIFICATION OF BASIC SER ITEMS

This enclosure provides WOG clarifications as to how the following NRC findings contained in the Safety Evaluation Report (SER) on the BASIC ERGs have been addressed in the Revision 1 ERGs.

Section 4.1 Items 4 and 5
Section 4.2 Items 2, 4, 6, 20, 26 and 27
Section 4.3 Items 1 through 12

Unless otherwise specified, specific guideline step numbers cited in examples are for the high pressure (HP) version of the guidelines. These step numbers in general also apply to the low pressure (LP) version. Where step numbers differ for the LP version, they are identified with brackets following the HP step number.

Based on the review performed to develop the enclosed information, only the Section 4.3 Items 8, 9 and 11 are evaluated to merit changes to the ERGs. To clarify these items in the ERGs, the Reference Plant Description documents in the Executive Volume and the Background Documents for the Containment Status Tree and Containment Function Restoration Guidelines will be revised to more clearly identify plant design features of the reference plant containment and containment systems. This will facilitate utility procedure writing efforts to identify plant specific differences from the generic reference plant so that the differences can be addressed in the procedure writing activity. All remaining BASIC SER items are adequately addressed in the ERGs.

SECTION 4.1

Section 4.1, Conformance with NUREG-0737 Requirements, discusses the degree of conformance of the BASIC ERGs with NUREG-0737 requirements. In general, the WOG interpreted Section 4.1 to not require specific responses. However, letter OG-105 did state that Items 4 and 5 of this section will be appropriately addressed in the Revision 1 version of the ERGs. Items 4 and 5 of the SER are provided below, followed by the NRC requested information and the WOG clarification to the items.

SER Item 4. Guidelines should address availability of systems under expected plant conditions and corrective or alternative actions that should be performed to mitigate the event should these systems or components fail. The WOG guidelines, being generic, do not list specific plant systems. Specific actions are identified for loss of offsite power and loss of all ac power.

In some instances the plant response and alternative actions to be taken if control grade systems fail to function are not provided. The staff has noted these deficiencies elsewhere; the WOG has agreed to address them in Revision 1 to the guidelines.

SER Item 5. Multiple and consequential failures should be considered. Multiple failures are considered by the ORGs and FRGs. These include multiple steam generator tube rupture, loss of all feedwater, and anticipated transients without scram. Operator error is accounted for by the monitoring of the safety status trees and by checks of plant status within the event oriented procedures.

The staff notes that the inclusion of multiple failure events into the event oriented guidelines has resulted in a long and complex package. The owners recognized that it was impossible to include all combinations of multiple failures with guidelines for each event and therefore developed the status tree and function restoration guideline package.

The operator would only be instructed to utilize the FRGs in the event that a safety function were challenged. The direct treatment of abnormal plant symptoms by the operator might lead to earlier responses to events which develop in an unexpected manner.

Specific instructions are not provided for loss of high pressure makeup before the occurrence of inadequate core cooling. This was required by NUREG-0737, page I.C.1-2, item (3). This guidance should be provided in the next guideline revision.

NRC Request 4. & 5. Identify how and where Items 4 and 5 have been addressed in the Revision 1 version of the ERGs.

4. Provide examples that show alternative actions to be taken if control grade systems fail to function.

5. Provide examples that show actions to be taken should multiple failures occur. For example, show how and where the following are addressed:

   a) loss of all high pressure SI makeup prior to onset of inadequate core cooling condition, and

   b) loss of RVLIS.

WOG Clarification 4. & 5. As stated in letter OG-105, the WOG has addressed the areas of control grade failures and multiple and consequential failures in development of the Revision 1 ERGs. The failure combinations explicitly addressed in the Revision 1 ERGs and the guidelines that are used to address specific failure combinations are identified in the document entitled Summary of Probabilistic Evaluation for Emergency Response Guideline Development Program. This document is contained in the Executive Volume. The functions and functional failures explicitly evaluated as part of this probabilistic evaluation are identified in Table 1 of the subject document. The functional failure combinations and the guidelines that provide coverage for the combinations are identified in Tables 2 through 5 of the subject document.

Examples of failures of non-safety related systems that are addressed, where appropriate, in the ERGs include:

a) Failure of condenser dump system capability (failure to open, failure to close)

b) Failure of atmospheric steam dump capability

c) Failure of normal pressurizer spray capability

d) Failure of auxiliary pressurizer spray capability

e) Failure of pressurizer PORVs (failure to open, failure to close)

f) Failure of normal charging flow capability

g) Failure of normal letdown flow capability

h) Failure of RCPs to operate

Examples of multiple and consequential failures of safety-related systems that are addressed, where appropriate, in the ERGs include:

a) Failure of emergency ac power capability

b) Failure of reactor trip capability

c) Failure of AFW flow capability

d) Failure of high pressure SI flow capability

e) Failure of switchover to recirculation capability for SI system

f) Failure of main steamline isolation capability

In general, such functional and system failures are treated in both the Optimal Recovery Guidelines (i.e., E-series) and the Function Restoration Guidelines (i.e., F-series).

An example of how control systems failures are addressed in the ERGs is illustrated by the treatment of normal pressurizer spray, auxiliary pressurizer spray and pressurizer PORVs to depressurize the reactor coolant system (RCS) during a plant cooldown and depressurization. These three alternative means to depressurize the RCS are typically treated and prioritized in one or two guideline step(s) using the inherent logic of the two column guideline format. For example, Guideline E-3 Steps 18 and 19 provide guidance to depressurize the RCS using (in order of priority) normal spray, pressurizer PORV and auxiliary spray. For this example, if none of the alternative pressurizer pressure control systems are available, Step 18 directs the operator to Guideline ECA-3.3 for response to a steam generator tube rupture transient without pressurizer pressure control. Another example of how control systems failures are addressed is illustrated by the treatment of letdown and excess letdown capability (e.g., E-3 Step 27) in which the two column format is again utilized to prioritize alternative means to accomplish a systems function.

An example of how safety-related system failures are addressed is illustrated by the treatment of the loss of all high pressure SI flow, which is specifically described in Item 26 of Section 4.2. Item 27 of Section 4.2 describes the treatment of RVLIS in the Revision 1 ERGs.

SECTION 4.2

Section 4.2, Generic Items That Must be Addressed in Revision 1, items were addressed in detail in Enclosure 1 to OG-105. SER items that require clarification are provided below, followed by the WOG response per OG-105, the NRC requested information and the WOG clarification.

SER Item 2. Certain system failures are not addressed within the E-1 and ES-1 LOCA guidelines. The operator is instructed to proceed from one step to the next, even when failure to accomplish the prior step is encountered. For events which require alternative operator actions [such as those which would occur from multiple failure events, such as a steamline break occurring inside containment and leading to failure of the reactor coolant pump (RCP) seal (LOCA) upon containment isolation], guidance is provided in the ECAs and FRGs. WOG should review the ERGs to ensure that appropriate instructions are provided to mitigate these events and review the status trees to ensure that appropriate symptoms are provided to direct one to the FRGs without undue delay. Necessary modification should be made within Revision 1 to the guidelines.

WOG Response 2. Revision 1 of the ERGs improves the coverage for the NRC concern expressed in this item. Both the guidelines and the status trees have been reviewed to ensure that the appropriate interfaces are provided to and from the FRGs. The E-series, ES-series and ECA-series guidelines provide for operator actions and for transitions between the ORGs in the presence of single failures and multiple failures which were shown to exceed a cutoff probability in the probabilistic evaluation. Should other types or combinations of system failures occur which pose a threat to the critical safety functions, the use of the status trees will provide the required, direct entry to the appropriate FRG without undue delay.

Revision 1 of the status trees contains modifications which help assure that the necessary interfaces with the FRGs are made in an accurate, timely manner.

NRC Request 2. The specific area of concern is that the loss of RCP seal injection flow is not adequately addressed in Guidelines E-1 or ES-1.1.

WOG Clarification 2. The WOG acknowledges the importance of RCP seal cooling and has addressed its maintenance in the Revision 1 ERGs. The RCP Trip/Restart Generic Issue Background Document contained in the Executive Volume discusses RCP operational considerations, including RCP seal cooling.

It is important to note that adequate RCP seal cooling is provided by either RCP seal injection flow or RCP thermal barrier component cooling water (CCW) flow. The Revision 1 ERGs address both cooling sources.

For the high pressure (HP) ERG plant, seal injection flow is provided by the charging/SI pumps and is not isolated during any transient, except for the loss of all ac power transient which is treated as a special case since RCP seal injection is loss due to the loss of all charging/SI pumps. As long as the charging/SI pumps are operating, seal injection flow and associated RCP seal cooling is provided. For the low pressure (LP) ERG plant, seal injection flow is provided by the charging pumps. Since the charging pumps are not SI pumps, operation of the charging pumps is not assured following safety injection initiation. Although the charging pumps may continue to operate following the safety injection actuation, they are tripped if a loss of offsite power condition occurs in combination with the safety injection signal. In both HP and LP plants, thermal barrier cooling is provided by the CCW pumps. Although these pumps are safeguards pumps that operate following SI actuation, the CCW supply lines to containment are automatically isolated on a containment high-3 pressure condition, thereby terminating thermal barrier cooling.

The approach used for RCP seal cooling in the HP ERGs is to verify charging/SI pump operation which ensures RCP seal injection flow for RCP seal cooling. This verification is performed in E-0 Step 8 for all emergency transients. CCW pump operation is also verified in E-0 Step 9 to ensure thermal barrier cooling for those emergency transients that are not accompanied by a containment high-3 pressure condition. Subsequently, all steps (e.g., E-1 Step 1) that stop the RCPs are preceded by the caution "Seal Injection flow should be maintained to all RCPs" to alert the operator that seal injection flow should not be isolated if the RCPs are tripped. All steps (e.g., ES-1.2 Step 12) that start an RCP are preceded by the caution "If seal cooling had previously been lost, the affected RCP(s) should not be started prior to a status evaluation".

The approach used for RCP seal cooling in the LP ERGs differs since the charging pumps are not SI pumps and will not be running if offsite power is lost. Similar to the HP ERGs, CCW pump operation is verified in LP Guideline E-0 Step 9 for all emergency transients. This will ensure RCP thermal barrier flow and associated seal cooling for the majority of emergency transients, i.e., those not accompanied by a containment high-3 pressure condition. Recognizing that a containment high-3

pressure condition will accompany some emergency transients, Step 19 has been added to the LP version of E-0 to explicitly check RCP seal cooling. If seal cooling via CCW flow to the RCP thermal barriers does not exist, the operator is provided with guidance in the Response Not Obtained column of Step 19 to trip the affected RCP(s) and take appropriate actions to start a charging pump, thereby reestablishing RCP seal injection flow. Again, this LP step applies to all emergency transients and is intended to ensure RCP seal cooling. Subsequently, all LP guideline steps that start the charging pumps (e.g., E-0 Step 38, E-1 Step 10 and ES-1.2 Step 3) are structured to protect the RCP seals and shaft by first checking whether CCW cooling to the RCP thermal barriers had been lost. If cooling had been lost, seal injection flow to the affected RCPs is isolated before starting a charging pump.

In both the HP and LP ERGs, guidelines steps (e.g., ES-1.2 Step 21 [LP 19], and E-3 Step 33) are included to check that RCP seal cooling is normal and to establish normal cooling per plant specific procedure, if necessary. These steps are included in guidelines for transients in which RCP seal cooling is a concern (e.g., small loss of reactor coolant, loss of secondary coolant, steam generator tube rupture, etc.).

Recovery from these transients also employ strategies which cooldown and depressurize the RCS, thereby reducing the temperature and pressure to which the RCP seals are subjected. Recovery for small loss of coolant accidents is provided in Guideline ES-1.2 and recovery for steam generator tube ruptures is provided in Guideline E-3 and Subguidelines ES-3.1, ES-3.2 and ES-3.3. The recovery strategies in these guidelines not only mitigate the initiating transient but also mitigate potential for seal damage and leakage resulting from the loss of RCP seal cooling should it occur in combination with the initiating transient.

In addition to the above features which have been developed to protect the RCP seals from overheating and possible seal damage, the ERGs have been developed with guidance for response to transients that are accompanied by seal damage. This guidance is included in the ERGs due to the symptom-based ERG structure that addresses multiple events and failures. Should RCP seal damage and excessive seal leakage occur by itself or in combination with an initiating event, the seal leakage will result in symptoms of a loss of reactor coolant. The ERGs have been developed to address the spectrum of loss of reactor coolant accidents by themselves or in combination with other events such as secondary break or steam generator

tube rupture. Consequently should RCP seal damage and excessive leakage occur, the ERGs address it as a loss of reactor coolant by itself or in combination with other events as appropriate.

SER Item 4.

Step 21 within the E-1 low pressure guidelines could potentially result in an inadvertent misdiagnosis of the need for containment sprays. It is desirable that, during a LOCA, the operator assess the need for continued operation of the containment sprays rather than assume proper system operation should the spray pumps not be operating. WOG agreed to modify the guideline in Revision 1 to have the operator review the need for reinstating containment spray.

WOG Response 4.

During the development of Revision 1 of the ERGs, a review of this item was conducted. In both BASIC and Revision 1, the need for containment spray is assessed in the Immediate Actions of E-0, "Reactor Trip or Safety Injection", and in the Containment Critical Safety Function. In Revision 1 Step 8 [LP 13] of E-1 "Loss of Reactor or Secondary Coolant" has been modified to make the step clearer. In this guideline if the containment spray is not running the operator is instructed to start spray if the criteria are met. The purpose of BASIC Step 14 [LP 21] of E-1 is to terminate containment spray operation.

NRC Request 4.

Provide additional clarification as to how this item has been resolved in Revision 1.

WOG Clarification 4.

Prior to clarification of this item, note that the WOG response to this item includes incorrect wording. The next to last sentence in the WOG response to this item should be "In this guideline if the containment spray is running the operator is instructed to stop spray if the criteria are met".

For the ERG reference plant, the containment spray system is automatically initiated upon actuation of the containment high-3 pressure signal. The ERGs have separated the actions to check and manually actuate the containment spray system, if required but not automatically actuated, from the manual actions to check and stop the containment spray system, if running but no longer required.

The need for containment spray system operation is checked in both the event-related and function-related guidelines. Guideline E-0 Step 14 checks if containment spray is required and verifies or manually actuates containment spray if required. The step is located before the event-related diagnostic steps so that it is

performed for all emergency transients that are accompanied by SI actuation. The need for containment spray is also checked in the function related sense by CSF Status Tree F-0.5, CONTAINMENT. Containment pressure above the high-2 setpoint results in a high containment pressure (Orange) condition the requires the operator to implement Guideline FR-Z.1. Guideline FR-Z.1 Step 3 checks if containment spray is required and verifies or manually actuates containment spray if required. In summary, Guideline E-0 Step 14 checks the need for containment spray in an event-related sense whereas Status Tree F-0.5 checks the need for containment spray in a function-related sense, independent of initiating event. The check in E-0 is intended to address the need for containment spray due to an initiating event or failures. The check in Status Tree F-0.5 is intended to address the need for containment spray due to subsequent events or failures. It should be noted that in plant specific EOPs, the high containment pressure setpoint can be used in the status tree in lieu of the high-2 setpoint as the symptom of a high containment pressure condition. Entry into FR-Z.1 would then occur on high-3 pressure, which corresponds to spray actuation set pressure. This acceptable plant specific change is discussed in the Background Document for Status Tree F-0.5.

Termination of containment spray in the ERGs is addressed only in the event-related guidelines. Guidelines E-1 Step 8 [LP 13], ECA-2.1 Step 7, ECA-3.1 Step 5 provide checks if containment spray is in operation (i.e., spray pump running) and if it should be stopped (i.e., containment pressure) less than plant specific value for resetting the spray signal. If the pumps are running and pressure is less than the reset value, the step instructs the operator to reset the spray signal and stop the spray pumps. If pumps are running but pressure is above the reset value, the operator is instructed to continue to the next step while continuing to monitor containment pressure. When containment pressure subsequently is decreased to less than the reset value, the operator is instructed to complete the subject step, resetting the spray signal and stopping the spray pumps. These steps which stop containment spray have been reworded from BASIC to highlight the continuous monitoring aspect of the step. The wording "<u>WHEN</u> containment pressure less than (reset value), <u>THEN</u> do steps (that reset spray signal and stop pumps)" has been incorporated into the Response Not Obtained column for substep b to inform the operator of the need to continue to monitor containment pressure and stop the spray pumps when appropriate. In addition to the detailed step wording, the high level step wording has also been modified to clarify the step intent, i.e., "Check If Containment Spray Should Be Stopped".

The WOG has concluded that the step should remain as is and that CSF Status Tree F-0.5 and Function Restoration Guideline FR-Z.1 should continue to be used to start the containment spray system if containment pressure increases above the high-3 setpoint. The basis for this conclusion is that the CSF Status Tree check on containment pressure is continuous (whereas the Optimal Recovery Guideline step is not) and the priority of the associated CSF challenge is sufficiently high (orange) that transition out of the Optimal Recovery Guidelines is immediately required (if no higher priority CSF challenges exist) to implement FR-Z.1 whenever containment pressure is above the high-2 setpoint. Further, if containment pressure is above the high-2 setpoint, it is appropriate to perform other actions related to containment integrity, including (1) verification that containment isolation Phase A and B valves are closed, containment ventilation isolation dampers are closed, containment fan coolers are running, main steamline isolation and bypass valves are closed, (2) isolation of feed flow to any faulted steam generator and (3) evaluation of hydrogen concentration in containment. Considering the spectrum of actions that the operator must perform if containment pressure is above the high-2 setpoint, the WOG considers it to be a necessity to retain the current guidance in F-0.5 and FR-Z.1. Having made this decision, it is redundant and unnecessary to include a check for high containment pressure in the spray system step. Further, if a high containment pressure check was included, it would be inappropriate to only use it to start the containment spray system if needed without performing the other verification and evaluation actions presently contained in FR-Z.1 and itemized above. Including all these actions in the subject containment spray step would make it unnecessarily cumbersome when the guidance already exists elsewhere in the ERGs. Not including these other actions would make the step technically incomplete.

SER Item 6. For LOCA events which do not lead to pressurizer level recovery or reactor coolant system (RCS) subcooling (as a result of break size, SI failures or other postulated LOCA events), the present E-1 and ES-1.2 guidelines could potentially lead to inappropriate accumulator venting (i.e., Step 9 within the high pressure SI plants and Step 11 within the low pressure SI plants). This logic should either be modified to address events which do not restore pressurizer level or should otherwise assure that events requiring accumulator injection are adequately protected against. Any modifications should be incorporated within Revision 1 to the guidelines.

WOG Response 6. This concern was evaluated for Revision 1. As a result of the evaluation, a criterion for isolating or venting of the accumulators has been added to Step 23 [LP 17] of ES-1.2 "Post-LOCA Cooldown and Depressurization".

NRC Request 6. Provide the basis for use of the 50°F subcooling criterion for isolation of the SI accumulators.

WOG Clarification 6. The parameter chosen for accumulator isolation in Guideline ES-1.2 is RCS subcooling, consistent with the use of RCS subcooling elsewhere in ES-1.2 for SI flow reduction. The RCS subcooling value chosen for isolation of the accumulators is 50°F plus instrumentation errors. This value of RCS subcooling can only be achieved for LOCA transients after the pressurizer is refilled. For LOCA transients for which the pressurizer cannot be refilled, the RCS hot legs will remain at saturation conditions, the SI accumulator isolation criteria will not be satisfied and the SI accumulators will remain available to deliver their water contents to the RCS. Consequently, the 50°F subcooling criteria added to the Revision 1 ERGs for accumulator isolation ensures that adequate RCS inventory exists for core cooling and that accumulator inventory is not needed for core cooling. Further, the SI accumulator isolation step is located in the guideline following steps which reduce pumped SI flow based on RCS subcooling. In combination, the subcooling criterion for accumulator isolation and the step location prevent premature isolation of the SI accumulators.

To address the possibility that a subsequent event or failure may occur after the SI accumulators are isolated, Guidelines FR-C.2 and FR-C.1 include steps which check the status of SI accumulators and unisolate the SI accumulators if they were isolated prior to delivery of their water contents. This permits their water contents to be delivered to the RCS if needed for restoration of core cooling.

The Response Not Obtained column of Step 23 [LP 17] includes an alternative criteria which if met permits the accumulators to be isolated.. This alternative criteria is that RCS hot leg temperatures are less than 400°F. As described in the Background Document Step Description Table for this step, this criterion is utilized to isolate the accumulators prior to nitrogen injection into the RCS. Based on the RCS being at saturated conditions, the 400°F criteria will ensure sufficient pressure in the RCS to prevent delivery of accumulator nitrogen after the water has been injected.

In summary, RCS subcooling is used to permit accumulator isolation when accumulator water delivery is not needed for core cooling and RCS hot leg temperature is used to permit accumulator isolation when necessary to prevent accumulator nitrogen delivery.

SER Item 20. Step 9 of FR-C.2 directs the operator to assure pressurizer power operated relief valve (PORV) readiness and closure. This step neglects some leakage paths such as the letdown line. This guideline should be modified to verify isolation of all potential leakage paths from the primary system. The WOG has agreed to include this instruction in Revision 1.

WOG Response 20. Step 3 [LP 3] of FR-C.2, "Response to Degraded Core Cooling", Revision 1 addresses this concern.

NRC Request 20. It does not appear that FR-C.2 has been revised to address all potential RCS leakage paths (such as the letdown line).

WOG Clarification 20. The BASIC version of FR-C.2 included Step 9 to ensure that the pressurizer PORVs and block valves were properly positioned such that a steam space loss of coolant through the subject valves did not exist. Per the NRC comment, this step was modified to ensure that RCS vent paths are closed, similar to Step 12 of the BASIC version of FR-C.1 which was not questioned in the BASIC SER.

In the Revision 1 version of FR-C.2, this step to ensure that RCS vent paths are closed was moved forward to Step 3. It was also revised to include the words "Enter plant specific list" under the substep that checks the closure status of other RCS vent paths. This modification was intended to highlight additional potential vent paths to be added at the plant specific level.

From the generic perspective, letdown is not considered a path that needs to be included in the subject step. The basis for this exclusion is that the letdown line will most likely be isolated prior to entry into FR-C.2 or FR-C.1 and that if not isolated, letdown flow loss will most likely be insignificant relative to other guideline actions intended to increase RCS inventory.

For the ERG reference plants, the letdown line is automatically isolated on pressurizer low level (via RCS letdown isolation valves and letdown orifice isolation valves) and on Phase A containment isolation (via containment isolation valves). The isolation of Phase A containment isolation valves is verified in Guideline E-0, Reactor Trip or Safety Injection, and in Guideline

FR-Z.1, Response to High Containment Pressure. The isolation of the letdown line on low pressurizer level is verified in Guideline FR-Z.2, Response to Low Pressurizer Level. Consequently, the letdown line is most probably isolated prior to the onset of a degraded or inadequate core cooling condition.

If the letdown line is not isolated at the loop, it is most probably intact and isolated at containment with letdown flow loss out the letdown line relief valve downstream of the letdown orifices. At the system conditions that will exist during an inadequate core cooling transient, inventory loss through the relief path should be insignificant. For example, if the RCS is at a pressure of 1000 psig when either FR-C.2 or FR-C.1 is entered, letdown flow loss will be limited to a maximum of about 50 gpm water. Actual inventory loss during a degraded or inadequate core cooling condition would be significantly less since steam would exist in the RCS at that time. Both Guidelines FR-C.2 and FR-C.1 include guidance to depressurize all steam generators to approximately 200 psig (actual number is plant specific and includes allowances for instrument accuracy). During the secondary depressurization, letdown flow loss will decrease with decreasing RCS pressure. When RCS pressure is reduced to less than 600 psig, the letdown line relief valve will close and letdown flow loss will be terminated. Consequently, guidance in FR-C.2 and FR-C.1 to depressurize the steam generators (and RCS) to deliver SI accumulator and low head SI pump makeup water to the RCS also function to reduce inventory loss out of the RCS, including any letdown flow loss. The net effect of this guidance is to make letdown flow loss insignificant relative to other operator actions in FR-C.2 and FR-C.1 directed toward depressurizing the RCS to deliver SI accumulator and low head SI pump makeup water to the RCS.

SER Item 26. Specific instructions are not provided for loss of high pressure makeup (i.e., centrifugal charging and/or safety injection pumps) before the occurrence of inadequate core cooling. This was required by NUREG-0737, Page I.C.1-2, item (3). These procedures should be provided in the next guideline revision.

WOG Response 26. Revision 1 addresses this concern. FR-C.2, "Response to Degraded Core Cooling" provides instructions to restore core cooling during accident conditions prior to the onset of inadequate core cooling symptoms.

NRC Request 26. Provide clarification as to how this item has been addressed in the Revision 1 ERGs.

WOG Clarification 26. An inadequate core cooling scenario is characterized by an initiating loss of reactor coolant event combined with the loss of high pressure SI flow. The inability to deliver high pressure SI flow is initially addressed in Guideline E-0, the entry guideline for the ERGs. Steps 8, 15 and 18 verify, respectively, that SI pumps are operating, SI flow exists and SI valves are properly aligned. If the expected plant response is not obtained, the operator is instructed to try to establish the desired conditions. These steps precede the event-related diagnosis steps in E-0 and, hence, are performed for all emergency transients.

E-0 Steps 20 [LP 21], 22 [LP 23], 23 [LP 24], 24 [LP 25], and 25 [LP 26] provide event-related diagnosis of the plant condition. The loss of high pressure SI flow does not affect the LOCA diagnosis actions provided in E-0 Step 24 [LP 25]. Consequently, the operator will transition to Guideline E-1 Step 1 following diagnosis of the LOCA. For LOCA scenarios characterized by RCS pressures greater than the shutoff head of the low head SI pump, E-1 Step 13 [LP 18] transitions the operator to Guideline ES-1.2. This guideline provides guidance to cooldown (at rates up to 100°F/hr.) and depressurize the RCS to cold shutdown conditions. For the more probable case where the high pressure SI pumps are running, Guideline ES-1.2 provides guidance to reduce and terminate high pressure SI pump flow in combination with the plant cooldown and depressurization. For the case where high pressure SI pumps are not operating, Guideline ES-1.2 functions to cooldown and depressurize the RCS. Since RCS pressure will follow saturation pressure for RCS temperature, this RCS cooldown will result in delivery of the SI accumulator contents followed by delivery of the low-head SI flow.

If the operator diagnoses the plant condition and is transitioned out of guideline E-0 per the above discussion, the ERG rules of usage direct the operator to initiate monitoring of the Critical Safety Function (CSF) Status Trees. Alternatively, if event-related diagnosis is not accomplished, Guideline E-0, Step 27 [LP 28] directs the operator to start monitoring the CSF Status Trees.

During operator response to an emergency transient, the plant safety state is monitored by the CSF Status Trees. The Status Tree for the Core Cooling CSF is the one that monitors the severity of the core cooling challenge. If the RCS is saturated (as it would be for a LOCA without

high pressure SI flow) the core cooling CSF will indicate either a saturated (Yellow), degraded (Orange) or inadequate (Red) condition depending on inventory in the RCS. If the RCS is saturated but inventory is sufficient to cover the core, the core cooling CSF condition is saturated (Yellow) and the operator is directed to Guideline FR-C.3. This guideline directs the operator to verify SI system operation and RCS vent paths isolated. If RCS inventory is not sufficient to fully cover the core, the condition is degraded (Orange) and the operator is directed to Guideline FR-C.2. This guideline includes the guidance in FR-C.3 plus guidance to perform an RCS cooldown (at rates less than 100°F/hr.) to reduce RCS pressure and thereby initiate SI accumulator injection followed by low-head SI flow. If RCS inventory is insufficient to adequately cool the core, the condition is inadequate (Red) and the operator is directed to Guideline FR-C.1. This guideline includes the guidance in FR-C.2 (except that RCS cooldown is at maximum rate) plus guidance to locally dump steam and start reactor coolant pumps if necessary to reestablish adequate core cooling.

In summary, the Revision 1 ERGs have been developed to include actions to restore high pressure SI flow (i.e., Guidelines E-0 and FR-C.3) or to cool down and depressurize the RCS to deliver SI accumulator or low head SI flow (i.e., Guidelines ES-1.2 and FR-C.2) prior to the occurrence of inadequate core cooling. These actions have been developed with due consideration to other CSF concerns, such as limiting RCS cooldown to less than 100°F/hr. so that the Integrity CSF is not challenged by operator actions taken in response to a degraded core cooling condition. Only if these actions are not effective will an inadequate core cooling condition develop and require implementation of Guideline FR-C.1.

The usage of the ERGs in response to an inadequate core cooling scenario (2500 gpm LOCA without high pressure SI flow) was evaluated as part of the Revision 1 ERG Validation Program. The validation program demonstrated the adequacy of Guidelines ES-1.2 and FR-C.2 in recovering from the subject scenario without the onset of an inadequate core cooling condition (i.e., the core cooling CSF did not experience a red challenge). Refer the Section 4.1.4 of WCAP-10599 (Emergency Response Guidelines Validation Program Final Report) for a description of the subject validation scenario. This document was transmitted to the NRC by letter OG-129 (J. J. Sheppard to H. L. Thompson, Jr.) dated August 14, 1984.

Within the ERG development program, the loss of all high pressure SI makeup is addressed as a functional failure and not an initiating accident (i.e., loss of reactor coolant, loss of secondary coolant, steam generator tube rupture). The ERGs are developed to address the loss of all high pressure SI makeup in combination with one or more of the accident initiators. Should the loss of all high pressure SI makeup be an initiating failure during normal plant operation, the ERGs would function to maintain the plant safe during plant recovery. Recovery could vary depending on plant model and available equipment. The following outlines how the ERGs would address this functional failure which would initially appear as the loss of all charging pumps during normal operation.

The loss of all charging pumps due to a common mode mechanical failure or any other unforeseen reason is an initiator of a disturbance to the plant. That disturbance, due to loss of normal RCS makeup capability, will prohibit the maintenance of some nominal reactor coolant system parameters (e.g., pressurizer level) for power operation. The response that the plant would take depends in part on the actions of the operating crew and in part on any other faults that may exist at that time. These in turn would lead to a set of symptoms observed by the operator. Entry into the ERGs is based on the symptoms that a reactor trip or a safety injection has either occurred or is required. If neither a reactor trip nor safety injection occurs and neither are required, then the failure that the plant is experiencing would be considered an abnormal occurrence. The failure of all charging capability would in itself fall into this category. However, if a reactor trip or safety injection did occur or was required, then the ERGs should be entered. The path followed through the ERGs would again depend on the symptoms observed which are a function of what is happening to the plant in addition to the loss of charging capability. The following is an example of one of the possible scenario trajectories, and is based on only the loss of charging capability with a subsequent reactor trip. The high pressure (HP) version of the Revision 1 ERGs is used for the example case.

1. Following the reactor trip, the operator enters E-0 at Step 1 and starts to work through the ERGs. At Step 4, the operator makes a transition to ES-0.1 when safety injection is not actuated nor required.

2. While in ES-0.1, the operator is instructed to initiate safety injection when either pressurizer pressure drops below the SI actuation setpoint in Step 5 or pressurizer level cannot be maintained as required on the foldout page. The operator is then instructed to go to E-0 and this time proceeds beyond Step 4.

3. Since plant symptoms requiring transition out of E-0 will not exist, the operator would remain for some time in E-0. Eventually the reactor coolant system pressure will drop to the shutoff head pressure of the high head SI pumps and stabilize.

4. At some time later, pressurizer level will return on scale due to makeup from the high head SI pumps. At that time, plant symptoms will require transition out of E-0. The operator will make the transition from E-0, Step 26 to ES-1.1.

5. In ES-1.1 at Step 9 the operator will make a transition to ES-1.2 since RCS pressure is not greater than the shutoff head of the high head SI pumps and they are needed for inventory makeup.

6. In ES-1.2, the operator will perform a controlled cooldown to cold shutdown while using the high head SI pumps as necessary to maintain pressurizer level.

7. While performing the above actions in the Optimal Recovery Guidelines, the operator will be monitoring the Critical Safety Function Status Trees. No Critical Safety Function will be jeopardized although a yellow path on low pressurizer level will be expected to exist at various times.

The plant used for the example case is the HP reference plant which includes high head safety injection pumps that play a key role in refilling the pressurizer, thus permitting the transition out of E-0. The example case is also applicable to the low pressure (LP) reference plant which also includes high head safety injection pumps. Should the high head SI pumps also be inoperable, the operator would remain in E-0 unless some other fault is occurring which results in plant symptoms that permit transition out of E-0. Thus items 3, 4, 5 and 6 of the example would not be performed. However, plant safety will be maintained through the monitoring of the Critical Safety Function Status Trees and implementation of Function Restoration Guidelines as required.

For the case where loss of high pressure SI makeup is the initiating failure, the WOG concludes that if an ERG entry is required, that the ERGs adequately address the loss of high pressure SI makeup.

SER Item 27. Approval of the instructions for use of the Reactor Vessel Level Instrumentation System (RVLIS) is granted only insofar as it does not conflict with the resolution required by generic letter 82-28 of December 19, 1982 (Ref. 22). Additional guidance should be provided in the event that RVLIS fails or is otherwise unavailable.

WOG Response 27. For both BASIC and Revision 1, RVLIS is assumed to be a safety grade instrument and is handled accordingly.

NRC Request 27. Provide clarification as to what guidance is provided should RVLIS fail or is unavailable. It is not apparent how this item has been addressed in the Revision 1 ERGs.

WOG Clarification 27. The ERG development program has consisted of a reanalysis of transients and accidents to provide improved guidance for operator response to emergency transients. As part of this effort, the WOG evaluated the plant symptoms upon which to base operator response actions. Where appropriate the WOG guidelines utilize multiple parameters to diagnose the plant condition and provide feedback to the operator on plant response to operator control actions. In the Revision 1 ERGs, RVLIS is utilized for the following categories of operator actions:

a) To diagnose and respond to degraded and inadequate core cooling conditions.

b) To terminate and reinitiate safety injection under select conditions (e.g., potential pressurized thermal shock) where pressurizer level is too conservative.

c) To diagnose and respond to voids in the reactor vessel upper head.

The first two categories of operator actions are the most important for accident mitigation. When RVLIS is used to direct operator actions in these categories, at least one additional diverse plant parameter is used in combination with RVLIS. For example, in checking core cooling Guideline FR-C.1 and FR-C.2 use RVLIS in combination with either core exit temperature (i.e., FR-C.1 Step 16 and FR-C.2 Step 7) or reactor coolant system hot leg temperature (i.e., FR-C.1 Steps 16 and 23 and FR-C.2 Step 18). In checking for termination/reinitiation of safety injection under select conditions where pressurizer level may not be available, Guideline FR-P.1

uses RVLIS in combination with RCS subcooling (i.e., FR-P.1 Step 5 for SI termination and FR-P.1 Step 12 for SI reinitiation) and Guideline ECA-3.2 uses RVLIS in combination with core exit temperature (i.e., ECA-3.2 Step 20 [LP 16]) for SI reinitiation. In all cases, the additional plant parameter is sufficient by itself to guide operator actions with respect to SI termination/reinitiation and degraded/inadequate core cooling.

The third category of operator actions pertain to responding (e.g., collapsing or venting) to voids in the reactor vessel upper head. This category of operator actions is not nearly as significant from an overall accident mitigation standpoint. Guidance to address voids in the reactor vessel is provided in Guideline FR-I.3, Response to Voids in Reactor Vessel. This guideline provides guidance to first collapse voids through increasing RCS pressure. This guidance can be implemented without RVLIS based on operator training on pressurizer level behavior with voids in the reactor vessel. If void collapse is not achieved through increasing RCS pressure, Guideline FR-I.3 provides guidance for venting the reactor vessel head.

In summary, the additional diverse plant parameters used in the ERGs in combination with operator training on the use of diverse plant parameters are considered sufficient to guide operator actions should RVLIS fail or be unavailable.

SECTION 4.3

Section 4.3, Plant Specific Items That Must Be Addressed in Revision 1, identifies plant specific items that should be addressed in the Revision 1 version of the ERGs. The WOG interpreted Section 4.3 to not require specific responses and stated that these items have been factored into the Revision 1 development process where appropriate. The WOG stated that the final responsibility for addressing plant specific design details resides with the individual utilities. Items 7, 8, 9, 11 and 12 of the SER are provided below, followed by the NRC requested information and the WOG clarification to the items. For completeness, other Section 4.3 items are also included below and are followed by the WOG clarification as to how the items have been addressed in the Revision 1 ERGs.

SER Item 1. Step 3 of Guideline E-0 gives the operator a choice between "AC emergency bus voltage--NORMAL" and "not energized". The word "energized" should be changed to "NORMAL" or vice-versa if this is what is meant, or the operator should be told what to do if an energized bus has non-normal voltage. Voltage deviations constituting non-normal voltage may be plant specific.

WOG Clarification 1. Revision 1 of Guideline E-0 uses the "energized, not energized" terminology in the subject step.

SER Item 2. Step 2 of ES-0.2 directs the operator to verify boron concentration by sampling, but omits any explicit requirement that (1) the concentrations in the RCS hot leg and in the letdown line approach a common value, and that (2) there be a check of nuclear instrumentation [per the background, page 16, line 4]. The requirements for sampling points and instrumentation may be plant specific.

WOG Clarification 2. Revision 1 of Guideline ES-0.2, Natural Circulation Cooldown, Step 3 has been revised to include the words "Enter plant specific means" to verify cold shutdown boron concentration by sampling. The Background Document Step Description Table for the subject step provides necessary information to explain the basis for the step and to permit utilities to make the step plant specific.

SER Item 3. In Guideline ES-1.1, the operator should check for high coolant radiation upon opening letdown lines at step 9(H) or 8(L) if the plant monitors coolant activity at this location.

WOG Clarification 3. Revision 1 of Guideline ES-1.1, Step 11 [LP 9] has been modified to include the words "Enter plant specific means" to establish letdown. The Background Document Step Description Table for the subject step identifies an operator knowledge requirement that the evaluation of the consequences of establishing letdown should be made prior to the action if excessive activity levels in the RCS are expected.

SER Item 4.	In Guideline E-1(L), prior to Step 11, the operator is cautioned not to thermal stress the RCP seals when reinstating charging flow. More definitive guidance is needed for reinitiation of charging flow. Each plant design should justify the adequacy of this procedure to prevent thermal stress. Additionally, the background material should address the reasons for interrupting the charging flow to the RCP seals.
WOG Clarification 4.	As discussed in the clarification to Section 4.2 Item 2, the WOG has systematically addressed the issue of RCP seal cooling in the Revision 1 ERGs. Due to design differences between the HP and LP reference plants, its treatment is slightly different in the two ERG versions. The caution that was used in BASIC version of Guideline E-1 has been deleted. The Revsion 1 version of LP Guideline E-1 includes guidance as to the necessary conditions for starting a charging pump. The detailed instructions as to how to restore seal injection flow to an RCP that had lost seal cooling remains plant specific and must be based on guidance provided by the pump manufacturer which may vary slightly per pump model.
SER Item 5.	Step 22 of Guideline E-1(L) directs the operator to identify and isolate containment leakage if the auxiliary building radiation level is abnormal. The guidelines should identify the need for a plant specific checklist to accomplish this.
WOG Clarification 5.	Revision 1 of Guideline E-1 has consolidated the check on auxiliary building radiation within Step 17 which initiates the evaluation of plant status. The subject step includes the words "enter plant specific list" for auxiliary building radiation and "enter plant specific means" to identify and isolate leakage to the auxiliary building. The Background Document provides additional information on the step, however, does not identify the need for a checklist to accomplish this step. The provisions to accomplish this step (e.g., inclusion of information in the procedure step, attached checklist, attached procedure, etc.) should be determined at the plant level based on the plant specific EOP writers guide.
SER Item 6.	For ECA-2, "Loss of AC Power", the background material (page 13) states that, in the absence of ac power, "an orderly cooldown [at approximately 100°F/hr] is not likely because of (1) the slow action/feedback loop between the operator...and the people at the equipment stations..., and (2) the possible limitations on steaming rates...with only the turbine driven AFW pump running". There is no analysis of the effects of a non-orderly cooldown. The discussion should describe what the operator can expect to see under such conditions and behavior that may be plant specific.

WOG Clarification 6.

The referenced BASIC background material was intended to note that plant cooldown operations under a loss of all ac power condition may differ from normal plant cooldown due to limitations on equipment availability and operability from the control room. It was not intended to indicate that the plant response would not be orderly. The referenced BASIC background material has been deleted from the Revision 1 ERGs. In Revision 1 to ECA-0.0, the need for operator actions outside the control room has been highlighted elsewhere in the background material, e.g., potential operator actions outside the control room are summarized in Section 3.2 of the Background Document. The transient analysis information provided in the Background Document is representative of what the operator can expect to see under a loss of all ac power condition.

SER Item 7.

For ECA-2, "Loss of AC Power", the background material (page 52) indicates that the operator should begin secondary depressurization if he "does not have positive indication that ac power will be immediately restored". More guidance should be provided on this, specifically, on how long the WOG believes the operator should be allowed to operate in a blackout condition before having to justify not immediately beginning depressurization. A discussion of a plant-specific decision point should be included.

NRC Request 7.

The specific concern identified in this comment has been resolved by instructing the operator to initiate secondary depressurization. However, the term maximum rate appears to be ambiguous, and should be more specific.

WOG Clarification 7.

In the Revision 1 ERGs, Step 16 of ECA-0.0 (Loss of All ac Power) and the associated background document material have been modified to address this SER item. Ambiguous terms (e.g., use of "rapid cooldown" in the BASIC guideline) have been deleted from the guideline and the guideline and background material have been modified to require the operator to perform a secondary depressurization in Step 16 of ECA-0.0.

In the Revision 1 ERGs, the wording "dump steam at maximum rate" is used selectively and has precise meaning, i.e., all available steam dump (e.g., condenser or atmospheric as indicated in the ERG step) valves full open. This wording is used in Guidelines ECA-0.0 Step 16, E-3 Step 14 and FR-C.1 Step 11 and 14 where the operator is intended to depressurize the steam generators as quickly as possible. Consequently, this is not considered an ambiguous term.

SER Item 8. The Z-Function, "Maintenance of Containment Integrity", appears to have been translated into the need to avoid one event: "overpressurization of containment vessel". Avoidance of this event is not sufficient to prevent failure as caused, for example, by underpressurization of the containment vessel. Such underpressurization could be caused by a severe containment overcooling transient at a plant such as Kewaunee which has a dual containment structure. The WOG should describe the steps taken to ensure that the Function Restoration Guidelines are complete and have not become so event-specific they do not adequately protect the six critical safety functions or the three underlying barriers (namely the fuel rod, the RCS pressure boundary, and the containment). More specifically, the WOG should provide guidelines for negative pressure or other features that may be plant specific that need to be considered for maintaining containment integrity.

NRC Request 8. The last sentence of the item requested the WOG to provide guidelines on negative pressure that need to be considered for maintaining containment integrity. This is considered a generic item. How and where has this comment been addressed.

WOG Clarification 8. In evaluating the CONTAINMENT CSF, the WOG concluded that the negative pressure concern did not apply to the reference plant containment and could be generically addressed through proper location and structuring of steps in the Optimal Recovery Guidelines to stop containment spray when containment pressure is below the reset setpoint. Due to the variation in containment and containment system designs that exist on Westinghouse designed plants, the WOG concluded the underpressurization concern could be best addressed on a plant specific basis. Refer to item 4 of Section 4.2, above.

SER Item 9. An in-depth review of the three barriers should be performed to ensure "completeness", that is, to identify a need for additional guidelines to address other threats not yet considered (for example, "underpressurization"). Plant specific design that may influence such threats should be identified in the guidelines.

NRC Request 9. This item is a generalized extension of Item 8 (i.e., negative pressure inside containment was an example of a threat to a CSF that was not generically addressed). Provide assurance that the subject review was performed and other threats to plant safety do not exist.

WOG Clarification 9	The CSF Status Trees and associated Function Restoration Guidelines were developed to be compatible with the Optimal Recovery Guidelines and as a total guideline set, to address the spectrum of NUREG-0737 Item I.C.1 requirements. The CSF status trees were subjected to extensive review within the WOG Procedures Subcommittee as they evolved into their present Revision 1 configuration. The WOG has concluded that the Status Trees in their Revision 1 configuration adequately address generic threats to the plant safety state. The basis for the CSF Status Trees and the selected branches are provided in the F-0 Background Document, Critical Safety Function Status Trees, and the F-0.1 thru F-0.6 Background Documents for each of the individual Status Trees. The Plant Specific Application section of the F-0 Background Document acknowledges that individual plants may have unique configurations or limitations which would require modification to the generic tree structure.
SER Item 10.	In Functional Recovery Guideline FR-C.1, "Response to Inadequate Core Cooling", the operator should be instructed to first check the containment hydrogen meter (Step 10) rather than to take a containment sample, if a meter has been installed. Guidance should be provided on starting a recombiner or other plant-specific feature for reducing hydrogen concentration.
WOG Clarification 10.	Revision 1 of Guideline FR-C.1, Response to Inadequate Core Cooling, Step 8 has been revised to "obtain a hydrogen concentration measurement: Enter Plant Specific Means". This wording permits each utility to modify the step consistent with plant provisions for hydrogen monitoring and permits plants to utilize hydrogen monitors. The step has further been structured to instruct the operator to start the hydrogen recombiner system if hydrogen concentration is between 0.5 and 6.0% in dry air. If hydrogen concentration is greater than 6.0% in dry air, the operator is instructed to consult the plant engineering staff for additional recovery actions.
SER Item 11.	Instructions should be provided in Status Tree F-0.5, "Containment and Associated Guidelines", for monitoring additional containment safeguards equipment including secondary containment exhaust fans, penetration pressurization equipment, return air fans, hydrogen igniters, or other plant-specific equipment. Additional branches may need to be added to the Status Tree.
NRC Request 11.	This item requests additional instructions to be added to Status Tree F-0.5 and associated FR-Z Function Restoration Guidelines. How and where has this comment been addressed.

WOG Clarification 11.	The ERG reference plant utilized for guideline development included a containment spray system, containment fan coolers and hydrogen recombiners for control of containment environment, typical of large dry containment designs. This is reflected in the F-0.5 Status Tree and associated Function Restoration Guidelines. Plant specific equipment unique to other containment designs such as the ice condenser containment or subatmospheric containment must be addressed at the plant specific level. Due to the variation in containment and containment system designs that exist on Westinghouse plants, the WOG concluded that these unique features could be best addressed on a plant specific basis. The plant specific EOP writing process established by the NRC in Supplement 1 to NUREG-0737 emphasizes the need to identify plant specific design differences from the generic reference plant and address them in the EOP writing process.
SER Item 12.	Status Tree F-0.5 should provide for checks of high containment hydrogen concentration via the hydrogen monitor and should provide for early treatment of hydrogen accumulation. Plant-specific designs with small free volume, such as that for an ice condenser plant, should include an additional branch.
NRC Request 12.	Hydrogen inside containment is considered to be a generic concern that can best be addressed by including a check for high hydrogen concentration in the F-0.5 Status Tree. Please clarify the basis for not including a check on hydrogen concentration in the F-0.5 Status Tree.
WOG Clarification 12.	In development of the ERGs, containment hydrogen has been addressed in those areas in which it is potentially present and where containment integrity is challenged. For the reference plant, these areas include plant conditions associated with loss of reactor coolant transients, inadequate core cooling transients and transients with an associated containment pressure increase above the value (high-2 pressure) at which hydrogen ignition poses a containment integrity concern. The hydrogen generation associated with a loss of reactor coolant is addressed in Guideline E-1 Step 12 through sampling and operation of hydrogen recombiners, if appropriate. This treatment serves to initiate early actions to address hydrogen, if required. The Status Trees are used to diagnose inadequate core cooling conditions (the precursor of significant hydrogen generation and high containment hydrogen concentrations) and containment high pressure conditions (which in combination with high hydrogen concentrations pose a potential integrity concern). Associated FRGs FR-C.1 and FR-Z.1, respectively, then direct the operator to sample

containment hydrogen concentration and initiate appropriate actions based on hydrogen concentration. This treatment of the hydrogen concern is considered an acceptable alternative to including hydrogen concentration in the F-0.5 Status Tree.

Plants with unique containment designs and systems that differ from the ERG reference plant must address design differences at the plant specific level in the EOP development process.