CATAWBA NUCLEAR STATION

EMERGENCY PROCEDURE GUIDELINES

DESCRIPTION OF DEVIATIONS FROM GENERIC

EMERGENCY RESPONSE GUIDELINES

(WESTINGHOUSE OWNER'S GROUP ERG-BASIC EDITION)

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CONTENTS

1. INTRODUCTION

The objective of this document is to describe and justify any safety-significant deviations between the Catawba Nuclear Station plant-specific Emergency Procedure Guidelines (EPGs) and the NRC approved generic Westinghouse Owner's Group (WOG) Emergency Response Guidelines (ERGs) - BASIC Edition. The EPGs are based on Revision 1 of the ERGs which are derived from ERG-BASIC. The NRC SER dated June 1, 1983 concludes that the BASIC ERGs are acceptable for implementation. NRC review of the Catawba plant-specific deviations described in this document will complete the review process.

Development of the EPGs was necessary due to the plant specific design differences between Catawba and the generic Westinghouse plant which was the basis for the ERGs. In addition, development of the EPGs includes additions to, deletions from, and restructuring of the generic ERGs. These enhancements or "deviations" were implemented in order to upgrade the comprehensiveness and usefulness of the station emergency procedures. The bases and justification for these deviations originated from the following:

- NRC comments on the BASIC ERGs as documented in the SER.
- Plant specific design differences
- Preference for the ERG Revision 1 approach over the BASIC ERGs in many guidelines
- Engineering evaluations
- Operating philosophy
- Operating experience
- Experience with other vendor guidelines
- Verification and validation activities

The description and justification of the deviations that follow will demonstrate that the overall intent, structure, and format of the generic guidelines have been preserved. The associated verification and validation activities have confirmed the correctness and appropriateness of the EPGs. A maintenance program has also been developed and implemented to address and incorporate potential EPG revisions as new information relevant to transient and accident mitigation and recovery is generated by the NSSS vendor, NRC, INPO, industry

operating experience, and from within Duke Power Company. The EPG development program has resulted in a comprehensive and well-integrated set of guideline documents which serve as the basis for station emergency procedures.

The station emergency procedures which are based on the plant-specific EPGs have been reviewed in accordance with 10 CFR 50.59, and have been determined not to involve an unreviewed safety question.

II. PLANT SPECIFIC DESIGN DEVIATIONS

The design of the Catawba Nuclear Station includes some features important to transient and accident mitigation and recovery which differ from the generic Westinghouse 4-loop plant which was utilized as the reference plant in the development of the generic ERGs. The major design deviations are listed in this section and the utilization of these systems and equipment are discussed. Since the utilization in all cases is relatively straightforward and consistent with the plant licensing basis, or good operating practice, the specific deviations associated with these design differences will only be discussed in this section.

On page 6-3 of the Duke Power Company response to Supplement 1 to NUREG-0737, it is stated that "Major differences in the Catawba design and the Westinghouse reference plant are being considered in additional analyses performed by Westinghouse." During the development of the Catawba EPGs, the need for and usefulness of any additional analyses were reviewed with Westinghouse. Based on this review it was concluded that additional analyses were not required in order to implement the Catawba specific design differences into the generic guidelines.

Upper Head Injection

The Catawba Emergency Core Cooling System includes an upper head injection (UHI) accumulator which is a passive pressurized injection tank and is necessary for acceptable plant response to large break LOCA events. The UHI accumulator is pressurized to approximately 1200 psig with a nitrogen cover gas, and

is isolated following injection on low level in order to preclude injection of the nitrogen cover gas.

Utilization of the UHI system in the EPGs is specified to ensure injection and isolation on low level, as designed, whenever injection should have occurred, or when injection would be beneficial to the transient mitigation effort. Similarly, the UHI system is isolated whenever its function is not required (similar to a normal shutdown sequence).

Ice Condenser and Containment Systems

Catawba utilizes the ice condenser containment design rather than the standard dry containment used as the reference design in the ERGs. All generic guideline steps related to dry containment systems have been replaced with steps for the corresponding ice condenser systems.

The ice condenser counterpart of the dry containment spray system is the combined capability of the upper containment spray system and the residual heat removal auxiliary containment spray system. Instructions concerning these systems in the EPGs are basically to verify automatic actuation on high containment pressure and automatic termination on low containment pressure for the upper containment sprays, and manual actuation and control of the auxiliary containment spray system.

The ice condenser design includes the Containment Air Return and Hydrogen Skimmer Fan System which circulates the post-accident containment atmosphere through the ice condenser in order to remove energy. These fans also serve to mix the containment atmosphere in order to prevent stagnant pockets of hydrogen. Proper actuation and performance of this system is ensured in the EPGs.

The Annulus Ventilation System filters leakage from the containment into the annulus between the containment vessel and the Reactor Building, and discharges it to the unit vent. Proper actuation and performance of this system is verified in the EPGs.

The hydrogen recombiners, the Emergency Hydrogen Mitigation System (igniters), and the aforementioned hydrogen skimmer fans are utilized and guidance is

provided to the operator concerning optimum methods of post-inadequate core cooling hydrogen mitigation. These methods are consistent with the licensing basis and good operating practice.

Standby Shutdown Facility

The Standby Shutdown Facility (SSF) is utilized in the ERGs as a source of reactor coolant pump seal injection following a loss of all AC power. This capability provides an extra level of redundancy with respect to maintaining reactor coolant pump seal integrity which is not available in the generic reference plant. This capability is utilized to bypass the cooldown and depressurization sequence in the loss of all AC power EPG, provided that the Reactor Coolant System inventory confirms that pump seal integrity has been maintained.

Reactor Vessel Level Indication System

The Reactor Vessel Level Indication System (RVLIS) for a UHI plant differs slightly from the generic RVLIS of the reference plant. The capabilities of the UHI design are however, essentially identical to the reference design.

One deviation from the generic utilization of the RVLIS exists in the EPGs. In the EPGs, the upper range indication is not utilized if any reactor coolant pumps are running. The utilization of the generic RVLIS in this mode has been determined to be of extremely low probability, and adds additional complexity to RVLIS utilization with essentially no significant benefit.

Setpoints

The EPGs utilize many setpoints as criteria for performing subsequent mitigation and recovery actions. Generic bases for specifying the setpoints have been included in the ERG documentation. The plant specific setpoints in the EPGs have, in some cases, been modified based on safety or operational concerns with the generic setpoint bases. In all cases, the EPG setpoints have either been revised in the conservative direction (more margin to challenging critical safety functions and equipment operability limits) or were revised in order to

improve the capability to achieve the overall intent of a mitigation or recovery sequence.

III. GUIDELINE SET CONFIGURATION DEVIATIONS

The EPG set configuration is based on Revision 1 to the generic ERGs and includes additions, deletions, and restructuring intended to enhance the quality of the guidelines. The Revision 1 guideline set is similar to the BASIC ERGs in many aspects. The following discussion will demonstrate the similarity of the EPGs and ERGs and identify and justify deviations between the EPGs and the BASIC ERGs..

- Table 1 illustrates a comparison of the Optimal Recovery Guidelines (ORGs).
- Table 2 illustrates a comparison of the Emergency Contingency Action Guidelines (ECAs).
- Table 3 illustrates a comparison of the Critical Safety Function Status Trees.
- Table 4 illustrates a comparison of the Function Restoration Guidelines (FRGs).

A discussion of each deviation follows. Deviations 5, 6, 8, 10, 11, 13, 15, 17, 19, 23 and 32 are consistent with Revision 1 of the generic ERGs which were submitted to the NRC by Westinghouse Owner's Group letter OG-111 dated November 30, 1983.

Deviation 1: The generic guidelines ES-0.2A, Natural Circulation Cooldown With No Accident In Progress, and ES-0.2B, Natural Circulation Cooldown With Potential for Steam Void in Vessel Upper Head (With RVLIS) and No Accident In Progress, have been combined into one EPG, ES-0.2, Natural Circulation Cooldown. The combination of the two guidelines serves to delete unnecessary duplication of steps. The technical content has been preserved, and therefore no safety-significance is associated with the change.

Deviation 2: The generic guideline ES-0.2C, Natural Circulation Cooldown With Potential for Steam Void in Vessel Upper Head (Without RVLIS) and No Accident In Progress, has been deleted from the EPG set. This guideline is only

applicable for those plants not equipped with a RVLIS indication, and therefore it is not required for Catawba.

Deviation 3: The generic guidelines E-1, Loss of Reactor Coolant, and E-2, Loss of Secondary Coolant, have been revised in the Catawba EPGs. The corresponding EPGs are E-1, High Energy Line Break Inside Containment, and E-2, Steam Line Break Outside Containment. As indicated by a comparison of the ERG and EPG titles, the main difference is that in the EPGs all high energy line breaks inside containment are mitigated in the EPG E-1, and only steam line breaks outside containment are mitigated in E-2. The basis for this change is that the symptoms of all high-energy line breaks in containment are very explicit, and the verification of containment systems actuation and performance is only applicable for that type of transient. Similarly, all steam line breaks outside containment have common symptoms and operator mitigation strategy. In addition, since a new EPG ECA-1.2, LOCA Outside Containment, has been added, the distinction between the EPG that is applicable for a LOCA inside or outside containment is clear (refer to Deviation 10).

The operator actions required to mitigate a high energy line break as detailed in the generic ERGs have been preserved in the EPGs. This deviation is better described as an enhancement to the generic guidelines that provides a more symptomatic diagnosis and enhances operator response by addressing only applicable symptoms.

Deviation 4: The generic guidelines ES-2.1, SI Termination Following Loss of Secondary Coolant, and ES-2.3, SI Termination Following Excessive RCS Cooldown, have been combined into the EPG ES-2.1, SI Termination Following Excessive Cooldown. The combination of the two guidelines serves to delete unnecessary duplication of steps. The technical content has been preserved, and therefore no safety-significance is associated with the change.

Deviation 5: The generic guideline ES-2.2, Transfer to Cold Leg Recirculation Following Loss of Secondary Coolant, has been deleted from the EPG set since it is identical to ES-1.3, Transfer to Cold Leg Recirculation Following Loss of Reactor Coolant. Any ECCS realignment for recirculation mode is performed

using ES-1.3, Transfer to Cold Leg Recirculation. No guidance has been deleted or revised and therefore no safety-significance is associated with the change.

Deviation 6: The generic guideline ES-3.1, SI Termination Following Steam Generator Tube Rupture, has been merged into the EPG E-3, Steam Generator Tube Rupture. This change is editorial only and does not affect the technical content.

Deviation 7: The generic guideline ES-3.2B, SGTR Alternate Cooldown Using Steam Generator Blowdown, has been deleted from the EPGs since the Catawba blowdown system cannot accommodate the fluid volume generated using this cooldown approach. The generic guidelines include three options for cooldown of an isolated and ruptured steam generator. All three options are not required to be available at each plant, and the selection of which options are incorporated into emergency procedures is left to the discretion of the utility based on plant specific design and operating preference. The remaining two options are incorporated into the guidelines ES-3.1, Post-SGTR Cooldown and Depressurization, and ES-3.2, SGTR Alternate Cooldown Using Backfill.

Deviation 8: Four generic guidelines which are related to steam generator tube rupture scenarios with multiple failures have been combined into two EPGs which provide the equivalent technical guidance. The generic guidelines ES-3.3, SGTR With Secondary Depressurization, ECA-3, SGTR Contingencies, ECA-7, Combined SGTR and LOCA, and ECA-8, Unisolatable SGTR, have been combined into the EPGs ECA-3.1, SGTR With Continuous Reactor Coolant System Leakage: Subcooled Recovery, and ECA-3.2, SGTR With Continuous Reactor Coolant System Leakage: Saturated Recovery. These guidelines provide guidance for multiple tube ruptures and tube ruptures concurrent with SBLOCAs and steam line breaks. The EPGs provide mitigation guidance that addresses the plant response characteristic of these multiple scenarios and due mainly to the excessive and possibly unisolatable primary to secondary leakage. Combining the four generic guidelines into two is desirable in order to reduce the number of guidelines, since this can be achieved without affecting the technical content.

Deviation 9: The generic guideline ECA-4, Response to Multiple Steam Generator Depressurization, has been merged into EPG E-2, Steam Line Break Outside

Containment, and other applicable guidelines. This change was desired since the technical content specifically addressing the scenario of interest was very limited, and the balance of the guideline duplicated the guidance included in the generic E-2, Loss of Secondary Coolant. This change allows for the deletion of a guideline while maintaining the technical content, and is therefore desirable and justified.

Deviation 10: The EPGs include an additional guideline ECA-1.2, LOCA Outside Containment, in order to address an NRC requirement for development of such a guideline as stated in the ERG-BASIC SER on p. 4-4 (Item 3). ECA-1.2 directs the operator to attempt to isolate potential causes of a LOCA outside containment, and to initiate makeup to the refueling water storage tank since a loss of ECCS suction inventory is occurring. Also, the option of initiating feed and bleed cooling is available and the decision to do so is left to the discretion of station management. The basis for initiating feed and bleed is the desirability of accumulating water in the containment sump to enable recirculation mode upon depletion of the refueling water storage tank inventory. In addition, bleeding of Reactor Coolant System inventory will melt ice in the ice condenser which will also accumulate in the containment sump. Initiating feed and bleed cooling is also included as an option for a SGTR which has resulted in a significant depletion of ECCS suction inventory due to a non-isolable tube leak. This scenario is also essentially a LOCA outside containment.

Deviation 11: The sequence of the six Critical Safety Functions in ERG-BASIC is Subcriticality, Core Cooling, Integrity, Heat Sink, Containment, and Inventory. This sequence has been maintained in the EPGs with the exception that Integrity and Heat Sink have been switched in the sequence. The significance of this change is that for a Critical Safety Function alarm of the same level of severity, the Heat Sink alarm condition would now be responded to before an Integrity alarm condition. This change was implemented based on engineering evaluations following release of ERG-BASIC, which concluded that the loss of heat sink represented a more severe challenge to overall plant safety.

Deviation 12: The generic Critical Safety Function (CSF) Status Tree F-0.1, Subcriticality, has been slightly modified in order to be compatible with the design basis of the Safety Parameter Display System (SPDS). The SPDS design

requires that the Subcriticality CSF alarm indicate a GREEN condition (i.e., CSF satisfied) during normal operation. Since the generic F-0.1 is only intended to be applicable following reactor trip, the first question "REACTOR POWER >5%" will generate a RED condition during normal operation. In order to correct this incompatibility, an additional question "REACTOR TRIP REQUIRED" has been added, prior to checking the above question, in the EPG F-0.1. This modification preserves the intent of the generic F-0.1 and satisfies the SPDS design requirement.

Deviation 13: The generic Critical Safety Function Status Tree F-0.3, Integrity, has been expanded to alarm on Reactor Coolant System pressure and temperature values that indicate the potential for a cold-overpressurization transient. This change broadens the monitoring of challenges to Reactor Coolant System integrity and is an enhancement to ERG-BASIC.

Deviation 14: The generic Critical Safety Function Status Tree F-0.3, Integrity, has been modified to alarm on challenges to Reactor Coolant System (RCS) integrity due to overpressurization considerations. This enhancement complements the existing alarms based on pressurized thermal shock. An ORANGE condition (i.e., a severe challenge to the CSF) is alarmed if RCS pressure exceeds 2400 psig (2250 psig if degraded containment environment instrument errors are in effect). The setpoint has been selected to indicate that a pressurization transient may challenge the pressurizer code safety valves, and thereby challenge the Integrity CSF. It is undesirable to challenge the pressurizer code safety valves since a stuck open safety valve cannot be blocked closed like a PORV. A new EPG, FR-P.3, Response to High Reactor Coolant System Pressure, provides the appropriate function recovery actions (Refer to Deviation 18).

Deviation 15: The generic Critical Safety Function Status Tree F-0.4, Heat Sink, ranks the occurrence of low steam generator level in one steam generator as more severe than a loss of steam generator steaming capability through the condenser dump valves and PORVs. The severity of these two occurrences has been reversed in the EPGs. This change was implemented since steaming capability is required during recovery from most transients, and low level in one steam generator does not affect heat sink capabilities since three other steam

generators may be available and only one is required for decay heat removal and cooldown.

Deviation 16: The generic CSF Status Tree F-0.5, Containment, has been modified to explicitly monitor the containment hydrogen concentration. If containment hydrogen concentration exceeds 0.5%, an ORANGE condition is alarmed, and the operator is referred to the appropriate hydrogen mitigation guidance in FR-Z.1, Response to High Containment Pressure. This change was recommended by the NRC in the ERG-BASIC SER on p. 4-9 (Item 12).

Deviation 17: The generic guideline FR-C.3, Response to Potential Loss of Core Cooling, has been merged into the generic guideline FR-C.2, Response to Degraded Core Cooling. Following release of the ERG-BASIC guideline set additional engineering evaluations concluded that the technical content was essentially the same and therefore two separate guidelines were not warranted or required. This change is desirable since it deletes an unnecessary guideline and can be achieved without affecting the technical content.

Deviation 18: The EPGs include an additional guideline, FR-P.3, Response to High Reactor Coolant System Pressure, to enhance response to potential integrity challenges due to overpressurization. Entry conditions to FR-P.3 from the Integrity CSF Status Tree are discussed in Deviation 14. The objectives of FR-P.3 are to depressurize the RCS to terminate the overpressure condition. Available systems are utilized in a manner consistent with design bases and good operating practice.

IV. GUIDELINE TECHNICAL DEVIATIONS SUMMARY

This section identifies and justifies safety-significant deviations between ERG-BASIC and the Catawba EPGs. Deviations due to plant specific design differences and guideline set configuration differences are detailed in Sections II and III, respectively, and are not included in the following summary. The items are listed in a sequence consistent with the guideline set as given in Tables 1-4.

Deviation 19: The generic ERG-BASIC criteria for termination of safety injection include setpoints based on Reactor Coolant System pressure. For example, Step 25 of E-0, Reactor Trip or Safety Injection, specifies that RCS pressure be greater than 2000 psig and increasing. Also, in Step 16 of E-3, Steam Generator Tube Rupture, RCS pressure must have increased by 200 psi prior to safety injection termination. In the ERG-BASIC SER on p. 4-5 (Items 12 and 14) the NRC requested the WOG to reevaluate the use of RCS pressure as an SI termination criteria. The staff concern was a resulting increase in primary-to-secondary leakage with a SGTR, if the RCS pressure was allowed to increase unnecessarily. An engineering evaluation of this NRC concern resulted in the deletion of quantitative RCS pressure setpoints as criteria for safety injection termination throughout the EPGs. Stable or increasing RCS pressure is utilized as a criterion without setpoints. The importance of maintaining adequate RCS pressure is also inherent in the requirement to maintain subcooling. The subcooling criterion is used throughout the EPGs.

Deviation 20: In the generic E-0, Reactor Trip or Safety Injection, the sequence of transfers to other guidelines to address specific events is given in Steps 29-31. The order of priority is indicated by transfers to E-2, Loss of Secondary Coolant, E-1, Loss of Reactor Coolant, and E-3, Steam Generator Tube Rupture. For a scenario with only one failure, for example a SBLOCA, the sequence of transfers from E-0 is not important since only one of the three major events has occurred. However, for the very unlikely multiple failures such as a SBLOCA with a simultaneous SGTR, the sequence is justified provided that the guidance for mitigation of a multiple failure is included, and that the sequence does not delay key mitigating actions.

The sequence of transfers in the EPGs are to E-1, High Energy Line Break Inside Containment, E-2, Steam Line Break Outside Containment, and E-3, Steam Generator Tube Rupture. The EPG guidance for the unlikely multiple failure scenarios of concern has been verified to be correct and technically equivalent or better than the generic guidelines. In all cases the key mitigation actions are performed in a timely manner.

Deviation 21: On the foldout for the generic E-0, Reactor Trip or Safety Injection, and on other foldouts, there exist two items entitled "Symptoms for FR-C.1, Response to Inadequate Core Cooling," and "Symptoms for FR-H.1, Re-

sponse to Loss of Secondary Heat Sink." These items are intended to monitor two key sets of symptoms which are part of the Critical Safety Function Status Trees. As part of the Status Trees these parameters will also be monitored by the Catawba Safety Parameter Display System (SPDS). During accident conditions the Shift Technical Advisor (STA) will be continuously monitoring the SPDS and will be alerted to a change in the status of any tree by both audible and visual alarms. Inclusion of these items on the foldout is therefore unnecessary and deletion will reduce the number of foldout items, which is desirable.

Deviation 22: In the generic ES-0.1, Reactor Trip Recovery, Step 13 directs the operator to restore offsite power if it was lost. This step is included in many ERGs once the mitigating actions have stabilized the plant transient response. This step is deleted throughout the EPGs since the operator actions in response to a loss of offsite power and restoration of offsite power are standard actions. Immediately following a loss of offsite power the operator verifies auto-start of the emergency diesel generators and sequencer loading of the emergency bus. The restoration of offsite power would be initiated once the higher priority mitigation actions have been performed and after the transient response had stabilized. Loss of offsite power is covered by the Catawba abnormal operating procedure AP/1/A/5500/07, Loss of Normal Power. The fact that loss and restoration of offsite power are time dependent events and can occur anytime suggests that specifying this action at a particular step in a guideline is not necessary. The appropriate restoration actions will be undertaken as required. Higher priority mitigation actions remain of higher priority.

Deviation 23: The generic ERGs include a reactor coolant pump trip criterion based on verification of SI flow along with Reactor Coolant System pressure below a setpoint, typically 1500 psig. In response to NRC Generic Letter 83-10c, the WOG submitted reports entitled "Evaluation of Alternate RCP Trip Criteria," by letter OG-110 dated December 1, 1983, and "Justification of Manual RCP Trip for Small Break LOCA Events," by letter OG-117 dated March 12, 1984. Based on the analyses and justification included in these reports, a reactor coolant pump trip criterion based on verification of SI flow along with loss of subcooling is used in the EPGs. This criterion provides the pump trip required for SBLOCA mitigation and in addition allows the pumps to remain in operation for moderate overcooling events and SGTR events up to and including a

single ruptured tube. As such, the pump trip criterion in the EPGs is more desirable than the generic ERG-BASIC criterion.

The EPGs also trip the RCPs during cooldown and depressurization sequences if subcooling is lost and cannot be promptly restored. This action is taken to prevent pump damage which might occur due to operating with inadequate NPSH, as well as the potential for inventory depletion due to a degradation in the subcooling margin. This pump trip guidance is superseded by the guidance included for mitigation of inadequate core cooling.

Deviation 24: In EPG E-1, High Energy Line Break Inside Containment, if a large LOCA has occurred, as indicated by an RCS depressurization sufficient to allow direct injection via the RHR pumps (<200 psig), the guidelines skip mitigation steps that are not required for that event and direct operator attention to the next critical action, transfer to sump recirculation. This change is desirable since operator response time can be more effectively utilized when only applicable guideline steps are required to be read and performed. Step 1 of the generic guideline E-1, Loss of Reactor Coolant, used the symptom of a rapidly decreasing refueling water storage tank level for the same purpose. However, in the ERG-BASIC SER on p. 4-3 (Item 1), the staff questioned the use of "rapidly decreasing" as a criterion, since it is not easily quantified. The use of a positive indication of flow via the RHR pumps in the EPG provides that quantitative criterion.

Deviation 25: In the sequence of mitigation actions following a SGTR, the generic guidelines do not check the status of the reactor coolant pumps until the 33rd step (Step 16 of ES-3.1, SI Termination Following Steam Generator Tube Rupture). The EPGs verify that at least one RCP is running in Step 7. The EPG approach is based on the desirability of establishing forced circulation, the normal and familiar operating mode, in parallel with initiating the RCS cooldown. Forced circulation also enables normal pressurizer spray, which is the optimum method of RCS depressurization. The EPGs also use two RCPs rather than one for the post-SGTR cooldown. Two RCPs operating is the normal cooldown configuration and is preferable to one RCP operating based on operating experience at the McGuire station which is of the same design as Catawba.

Deviation 26: In the EPGs, following transfer from safety injection mode to cold leg recirculation mode, the operators are instructed to make up to the refueling water storage tank. This will establish a redundant ECCS suction inventory should sump recirculation be interrupted. This change is consistent with an NRC recommendation in the ERG-BASIC SER on p. 4-4 (Item 8).

Deviation 27: In the generic ERG ECA-2, Loss of All AC Power, Step 3.c.2 disables automatic sequencing of large loads prior to restoration of power to the emergency bus. Subsequent loading of large loads is performed manually. In the EPGs, automatic sequencer loading is restored for the case where a safety injection signal is present. This deviation has been implemented in order to lessen the burden of manual operator actions and to expedite the loading process. Although unlikely, the potential for an operator error is also precluded. No reduction in reliability is associated with the change.

Deviation 28: In the generic ERG FR-C.1, Response To Inadequate Core Cooling, depressurization of the RCS by opening pressurizer PORVs is not undertaken unless core exit thermocouple temperatures are greater than 1200°F. The EPGs implement this action if core exit thermocouple temperatures are greater than 700°F and increasing. This modification avoids waiting for the core to heat up if the mitigation actions already performed have not been successful. This deviation can be considered as an enhancement of the generic guidelines. The loss of inventory in the form of steam relief through the pressurizer PORVs is offset by the increase in the safety injection flowrate resulting from the RCS depressurization.

Deviation 29: In the EPG FR-H.1, Response To Loss Of Secondary Heat Sink, an attempt to restore a partial heat sink by restarting a RCP in the loop with the highest steam generator level is included. This action is beneficial for the case where natural circulation is degraded due to uncovery of the U-tubes in any steam generator. RCP restart is only attempted if the steam generator wide range level indicates greater than 10%. The generic requirement that all RCPs be tripped prior to initiating feed and bleed cooling has been maintained, as required by the ERG-BASIC SER on p. 4-7 (Item 23).

Deviation 30: In the EPG FR-H.1, Response To Loss Of Secondary Heat Sink, following initiation of feed and bleed cooling, RCS temperature is monitored

to assess the success of core cooling in this mode. The need for additional or less feed and/or bleed capacity is determined. This deviation is a refinement of the generic guidance and enables better control of the plant during feed and bleed.

Deviation 31: In the EPG, FR-I.3, Response To Voids In Reactor Vessel, venting of the reactor vessel is permitted without the detailed precautions required for hydrogen venting if no symptoms of inadequate core cooling have been observed. If the void may contain a significant volume of hydrogen, then the generic venting precautions are followed. The dissolved hydrogen in the RCS would only increase the volumetric concentration in the containment by less than 0.1%, if it were continuously vented. It is therefore necessary for the core to undergo metal-water reaction of the cladding for a significant volume of hydrogen to be generated. In addition, the SPDS will alert the operator to containment hydrogen concentrations greater than 0.5%. For these reasons, continuously venting through the reactor vessel head vent is justified. Continuous venting is desirable and achievable without concern for excessive containment hydrogen concentrations.

Deviation 32: In the generic FR-I.3, Response To Void In Reactor Vessel, the option to condense a steam void by restarting one reactor coolant pump is not included. This option is, however, included in Step 1 of the generic ES-0.2B, Natural Circulation Cooldown With Potential for Steam Void In Vessel Upper Head (With RVLIS) and No Accident In Progress. This guidance has been incorporated in the EPG FR-I.3.

TABLE 1

Comparison of EPGs and BASIC ERGs

Optimal Recovery Guidelines

BASIC ERG		Catawba EPGs
E-0	Reactor Trip or Safety Injection	Same
ES-0.1	Reactor Trip Response	Same
ES-0.2A	Natural Circulation Cooldown With No Accident In Progress	(Deviation 1)
ES-0.2B	Natural circulation Cooldown With Potential for Steam Void in Vessel Upper Head (With RVLIS) and No Accident In Progress	(Deviation 1)
ES-0.2C	Natural Circulation Cooldown With Potential for Steam Void in Vessel Upper Head (Without RVLIS) and No Accident in Progress	(Deviation 2)
ES-0.3	SI Termination Following Spurious SI	Same
E-1	Loss of Reactor Coolant	(Deviation 3)
ES-1.1	SI Termination Following Loss of Reactor Coolant	Same
ES-1.2	Post-LOCA Cooldown And Depressurization	Same
ES-1.3	Transfer To Cold Leg Recirculation Following Loss of Reactor Coolant	Same
ES-1.4	Transfer to Hot Leg Recirculation	Same
E-2	Loss of Secondary Coolant	(Deviation 3)
ES-2.1	SI Termination Following Loss of Secondary Coolant	(Deviation 4)
ES-2.2	Transfer to Cold Leg Recirculation Following Loss of Secondary Coolant	(Deviation 5)
ES-2.3	SI Termination Following Excessive RCS Cooldown	ES-2.1
E-3	Steam Generator Tube Rupture	Same
ES-3.1	SI Termination Following Steam Generator Tube Rupture	(Deviation 6)
ES-3.2A	SGTR Alternate Cooldown by Backfilling RCS	Same
ES-3.2B	SGTR Alternate Cooldown Using Steam Generator Blowdown	(Deviation 7)
ES-3.3	SGTR With Secondary Depressurization	(Deviation 8)

TABLE 2

Comparison of EPGs and BASIC ERGs

Emergency Contingency Action Guidelines

BASIC ERGs		Catawba EPGs
ECA-1	Anticipated Transients Without Scram	FR-S.1
ECA-2	Loss of All AC Power	ECA-0
ECA-2.1	Loss of All AC Power Recovery Without SI Required	ECA-0.1
ECA-2.2	Loss of All AC Power Recovery With SI Required	ECA-0.2
ECA-3	SGTR Contingencies	(Deviation 8)
ECA-4	Response to Multiple Steam Generator Depressurization	(Deviation 9)
ECA-5	Loss of Emergency Coolant Recirculation	ECA-1.1
ECA-7	Combined SGTR and LOCA	(Deviation 8)
ECA-8	Unisolatable SGTR	(Deviation 8)
ECA-9	SGTR Without Pressurizer Pressure Control	ECA-3.3 (Deviation 10)

TABLE 3

Comparison of EPGs and BASIC ERGs

Critical Safety Function Status Trees

BASIC ERGs		Catawba EPGs
F-0	Critical Safety Function Status Trees	(Deviation 11)
F-0.1	Subcriticality	(Deviation 12)
F-0.2	Core Cooling	Same
F-0.3	Integrity	(Deviations 13 & 14)
F-0.4	Heat Sink	(Deviation 15)
F-0.5	Containment	(Deviation 16)
F-0.6	Inventory	Same

TABLE 4

Comparison of EPGs and BASIC FRGs

Function Restoration Guidelines

BASIC ERGs		Catawba EPGs
FR-S.1	Response To Nuclear Power Ceneration	Same
FR-S.2	Response To Loss of Core Shutdown	Same
FR-C.1	Response To Inadequate Core Cooling	Same
FR-C.2	Response To Degraded Core Cooling	Same
FR-C.3	Response To Potential Loss of Core Cooling	(Deviation 17)
FR-C.4	Response to Saturated Core Cooling Conditions	Same
FR-P.1	Response to Imminent Pressurized Thermal Shock Condition	Same
FR-P.2	Response to Anticipated Pressurized Thermal Shock Condition	Same (Deviation 18)
FR-H.1	Response To Loss of Secondary Heat Sink	Same
FR-H.2	Response To Steam Generator Overpressure	Same
FR-H.3	Response To Steam Generator High Level	Same
FR-H.4	Response to Steam Generator Low Level	FR-H.5
FR-H.5	Response To Loss Of Steam Generator PORVs and Condenser Dump Valves	FR-H.4
FR-Z.1	Response To High Containment Pressure	Same
FR-Z.2	Response To Containment Sump Level	Same
FR-Z.3	Response To High Containment Radiation Level	Same
FR-I.1	Response To Pressurizer Flooding	Same
FR-I.2	Response To Low System Inventory	Same
FR-I.3	Response To Void In Reactor Vessel	Same