CATAWBA NUCLEAR STATION

EMERGENCY PROCEDURE GUIDELINES

DESCRIPTION OF DEVIATIONS FROM GENERIC

EMERGENCY RESPONSE GUIDELINES

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CONTENTS

I. 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 generic Westinghouse Owner's Group (WOG) Emergency Response Guidelines (ERGs).

The EPGs are based on Revision 1 of the ERGs. 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:

- Plant specific design differences
- Preference for the ERG Revision 0 approach over Revision 1 in some 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.

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 straight forward 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.

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.

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 EPGs 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 following discussion will demonstrate the similarity of the EPGs and ERGs and identify and justify deviations.

- Table 1 illustrates a comparison of the Optimal Recovery Guidelines (ORGs).
- Table 2 illustrates a comparison of the Emergency Contingency Action Guidelines (ECAs) for the EPGs and Revision 1 of the generic ERGs.

- 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.

Deviation 1: The generic ES-0.0 Rediagnosis guideline has been deleted in the EPGs. Based on the verification and validation activities it was determined that this guideline was not useful. It is apparent that some use for ES-0.0 may have been warranted during the ERG development process due to the relative unfamiliarity of the operating crew involved in the generic ERG validation program. Increasing familiarity with the EPGs will enable the operators to understand the overall intent. In addition, the Critical Safety Function Status Trees and the Function Restoration Guidelines will ensure that the plant is maintained in a safe condition should the operating crew become temporarily disoriented in the Optimal Recovery Guidelines due to the more unlikely multiple failure scenarios.

Deviation 2: The generic ES-0.2 and ES-0.3 guidelines have been merged to reduce unnecessary guideline duplication. The technical content has been preserved.

Deviation 3: The generic ES-0.4 guideline has been deleted in the EPGs since it is only required for those plants not equipped with a RVLIS system.

Deviation 4: The EPGs include an additional guideline ES-0.3, SI Termination Following Spurious SI, to respond to indications of a spurious and unnecessary actuation of safety injection equipment. The recovery actions are designed to optimize the recovery from this specific event rather than any safety injection termination sequence.

Deviation 5: The generic E-1, Loss of Reactor or Secondary Coolant, has been some what modified in the EPGs. The EPG E-1, High Energy Line Break Inside Containment, differs from the generic in that secondary breaks outside contain-

TABLE 1

Comparison of EPGs and ERG Revision 1

Optimal Recovery Guidelines

ERG Revision 1		EPGs
E-0	Reactor Trip or Safety Injection	Same
ES-0.0	Rediagnosis	(Deviation 1)
ES-0.1	Reactor Trip Response	Same
ES-0.2	Natural Circulation Cooldown	(Deviation 2)
ES-0.3	Natural Circulation Cooldown with Steam Void In Vessel (With RVLIS)	(Deviation 2)
ES-0.4	Natural Circulation Cooldown with Steam Void In Vessel (Without RVLIS)	(Deviation 3)
		(Deviation 4)
E-1	Loss of Reactor or Secondary Coolant	(Deviation 5)
ES-1.1	SI Termination	Same
ES-1.2	Post-LOCA Cooldown And Depressurization	Same
ES-1.3	Transfer To Cold Leg Recirculation	Same
ES-1.4	Transfer to Hot Leg Recirculation	Same
E-2	Faulted Steam Generator Isolation	(Deviation 6)
		(Deviation 7)
E-3	Steam Generator Tube Rupture	Same
ES-3.1	Post-SGTR Cooldown Using Backfill	Same
ES-3.2	Post-SGTR Cooldown Using Blowdown	(Deviation 8)
ES-3.3	Post-SGTR Cooldown Using Steam Dump	Same

TABLE 2

Comparison of EPGs and ERG Revision 1

Emergency Contingency Action Guidelines

ERG Revision 1		EPGs
ECA-0.0	Loss of All AC Power	Same
ECA-0.1	Loss of All AC Power Recovery Without SI Required	Same
ECA-0.2	Loss of All AC Power Recovery With SI Required	Same
ECA-1.1	Loss of Emergency Coolant Recirculation	Same
ECA-1.2	LOCA Outside Containment	Same
ECA-2.1	Uncontrolled Depressurization of All Steam Generators	(Deviation 9)
ECA-3.1	SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired	Same
ECA-3.2	SGTR With Loss of Reactor Coolant - Saturated Recovery Desired	Same
ECA-3.3	SGTR Without Pressurizer Pressure Control	Same

TABLE 3

Comparison of EPGs and ERG Revision 1

Critical Safety Function Status Trees

ERG Revision 1		EPGs
F-0.1	Subcriticality	(Deviation 10)
F-0.2	Core Cooling	Same
F-0.3	Heat Sink	Same
F-0.4	Integrity	(Deviation 11)
F-0.5	Containment	(Deviation 12)
F-0.6	Inventory	Same

TABLE 4

Comparison of EPGs and ERG Revision 1

Function Restoration Guidelines

ERG Revision 1		EPGs
FR-S.1	Response To Nuclear Power Generation/ATWS	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 Saturated Core Cooling	Same
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 Loss Of Normal Steam Release Capabilities	Same
FR-H.5	Response To Steam Generator Low Level	Same
FR-P.1	Response To Imminent Pressurized Thermal Shock Condition	Same
FR-P.2	Response To Anticipated Pressurized Thermal Shock Condition	
		(Deviation 13)
FR-Z.1	Response To High Containment Pressure	Same
FR-Z.2	Response To Containment Flooding	Same
FR-Z.3	Response To High Containment Radiation Level	Same
FR-I.1	Response To High Pressurizer Level	Same
FR-I.2	Response To Low Pressurizer Level	Same
FR-I.3	Response To Voids In Reactor Vessel	Same

ment are covered in E-2, Steam Line Break outside Containment (refer to Deviation 6). This modification was implemented since the symptoms of a high energy line break inside containment are very explicit, and the verification of containment systems actuation and performance is only applicable for that type of transient. The generic approach to LOCA and steam line break mitigation has been preserved. The deviation serves to enhance operator response by addressing only applicable symptoms.

Deviation 6: The generic E-2, Faulted Steam Generator Isolation, has been modified in the EPGs to cover only steam line breaks outside containment. The deviation is complementary with Deviation 5 discussed above. The EPG E-2, Steam Line Break Outside Containment, preserves the generic approach to steam line break mitigation.

Deviation 7: The EPGs include an additional guideline, ES-2.1, SI Termination Following Excessive Cooldown, which has been included to address the safety concern associated with overcooling transients, (i.e., pressurized thermal shock, reactor vessel head voiding, etc). The generic guidance for these concerns is included in the Function Restoration Guidelines. The basis for this additional guideline is to address these concerns during transient mitigation and recovery along the optimal recovery path, rather than relying solely on the Function Restoration Guidelines. This allows a preventive rather than a reactive approach to transient management. The technical bases for ES-2.1 are consistent with the generic bases.

Deviation 8: The generic ES-3.2, Post-SGTR Cooldown Using Blowdown, has been deleted in the EPGs since the Catawba blowdown system cannot perform the functions required by the guideline.

Deviation 9: The generic ECA-2.1, Uncontrolled Depressurization Of All Steam Generators, has been merged into EPG E-2, Steam Line Break Outside Containment, and other interfacing guidelines. The generic technical guidance has been preserved.

Deviation 10: 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 11: The generic CSF Status Tree F-0.4, 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 and cold-overpressurization considerations. 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. A new EPG, FR-P.3 Response to High Reactor Coolant System Pressure, provides the appropriate function recovery actions (Refer to Deviation 13).

Deviation 12: 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.

Deviation 13: 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 11. 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 the generic ERGs 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 14: In the generic E-0, Reactor Trip Or Safety Injection, the sequence of transfers to other ERGs are based on diagnosis of 1) faulted steam generators (SLB) 2) SGTR, and 3) LOCA. The EPG E-0, the sequence of transfers to other EPGs are based on 1) high energy line break in containment 2) SLB, and 3) SGTR. The EPG sequence essentially accomplishes the same mitigation actions as the generic ERGs. However, the guideline wherein the specific steps are included may, in some cases, be different.

Deviation 15: On the foldout for the generic E-0, Reactor Trip or Safety Injection, is Item 3, Red Path Summary. The usefulness of this item has not been demonstrated and it has been deleted throughout the EPGs. A member of the control room operating crew will be continuously monitoring the CSF status trees under any abnormal condition.

Deviation 16: In the generic ES-0.1, Reactor Trip Response, Step 7 verifies that offsite power is available. This step is deleted throughout the EPGs since it is standard operating procedure that whenever offsite power has been interrupted, the operators will verify loading of the diesel generators and initiate action to restore offsite power.

Deviation 17: The generic ERGs and the EPGs include reactor coolant pump (RCP) trip criteria on loss of subcooling for SBLOCA mitigation. 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 and inventory depletion.

Deviation 18: 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.

Deviation 19: 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 (RWST). This will establish a redundant ECCS suction inventory should sump recirculation be interrupted.

Devi‹tion 20: In the generic ERG E-3, Steam Generator Tube Rupture, the status of the RCPs is not checked until Step 35. 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.

Deviation 21: In the generic ERG ECA-0.0, Loss of All AC Power, Step 6 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.

Deviation 22: In the EPG ECA-1.2, LOCA Outside Containment, an option to initiate feed and bleed cooling is included. This approach essentially reduces the loss of ECCS suction inventory and preserves some capability for sump recirculation. This approach initiates melting of the ice condenser inventory which contributes to the accumulation of water in the containment sump for recirculation. Initiating feed and bleed cooling is also included as an option for a SGTR which has resulted in a significant depletion of RWST inventory due to a non-isolable tube leak. This scenario is essentially a LOCA outside containment.

Deviation 23: 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.

Deviation 24: 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.

Deviation 25: In the EPG, FR-H.1, Response To Loss Of Secondary Heat, 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.

Deviation 26: 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.