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May 10, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Ms. E. G. Adensam, Chief
Licensing Branch No. 4

Re: Catawba Nuclear Station
Docket Nos. 50-413 and 50-414

Dear Mr. Denton:

Section 15.2.4.2 of the Catawba Safety Evaluation Report discusses Open Item 17, Alarm in the Control Room for Boron Dilution Modes in All Modes of Operation. In response to this open item, attached is revised Catawba FSAR Section 15.4.6 and responses to Questions 440.56, 88, and 92. These revised pages will be included in Revision 10 to the FSAR.

The attached analysis is based on Catawba Cycle 1 core parameters. If necessary, the analysis will be revised by the first refueling in order to bound subsequent cycles.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

ROS/php

Attachment

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15.4.5 A MALFUNCTION OR FAILURE OF THE FLOW CONTROLLER IN A BWR LOOP THAT RESULTS IN AN INCREASED REACTOR COOLANT FLOW RATE

(Not applicable to Catawba.)

15.4.6 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT RESULTS IN A DECREASE IN BORON CONCENTRATION IN THE REACTOR COOLANT

15.4.6.1 Identification of Causes and Accident Description

Reactivity can be added to the core by feeding primary grade water into the Reactor Coolant System via the reactor makeup portion of the Chemical and Volume Control System. Boron dilution is a manual operation under administrative control with procedures calling for a limit on the rate and duration of dilution. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water during normal charging to that in the Reactor Coolant System. The Chemical and Volume Control System is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

The opening of the reactor water makeup control valve provides makeup to the Reactor Coolant System which can dilute the reactor coolant. Inadvertent dilution from this source can be readily terminated by closing the control valve. In order for makeup water to be added to the Reactor Coolant System at pressure, at least one charging pump must be running in addition to a reactor makeup water pump.

The rate of addition of unborated makeup water to the Reactor Coolant System when it is not at pressure is limited by administratively limiting the output of the reactor makeup water pumps. Normally, only one reactor makeup water pump is operating while the other is on standby. With the RCS at pressure, the maximum delivery rate is limited by the control valve.

The boric acid from the boric acid tank is blended with primary grade water in the blender and the composition is determined by the preset flow rates of boric acid and primary grade water on the control board.

In order to dilute two separate operations are required:

1. The operator must switch from the automatic makeup mode to the dilute mode;

2. The start button must be depressed.

Omitting either step would prevent dilution.

Information on the status of the reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of the pumps in the Chemical and Volume Control System. Alarms are

actuated to warn the operator if boric acid or demineralized water flow rates deviate from preset values as a result of system malfunction.

This event is classified as an ANS Condition II incident (an incident of moderate frequency) as defined in Section 15.0.1.

15.4.6.2 Analysis of Effects and Consequences

Method of Analysis

To cover all phases of the plant operation, boron dilution during refueling, cold shutdown, hot standby, startup, and power operation are considered in this analysis.

Dilution During Refueling

An uncontrolled boron dilution accident cannot occur during refueling as a result of a reactor coolant makeup system malfunction. This accident is prevented by administrative controls which isolate the Reactor Coolant System from the potential source of unborated water.

Q440.89 Valves 1NV231, 1NV237, 1NV241, 1NV244, and 1NV240 in the CVCS will be locked closed during refueling operations. These valves will block the flow paths which could allow unborated makeup to reach the reactor coolant system. Any makeup which is required during refueling will be borated water supplied from the refueling water storage tank.

The most limiting alternate source of uncontrolled boron dilution would be the inadvertent opening of a valve in the Boron Thermal Regeneration System (BTRS). For this case highly borated RCS water is depleted of boron as it passes through the BTRS and is returned via the volume control tank. The following conditions are assumed for an uncontrolled boron dilution during refueling:

Technical Specifications require the reactor to be borated to a concentration of 2000 ppm at refueling. The critical boron concentration is conservatively estimated to be 731 ppm for Cycle 1.

Dilution flow is assumed to be 120gpm. This is assumed although normally neither the reactor makeup system nor the BTRS is operated at refueling conditions.

Mixing of the reactor coolant is accomplished by the operation of one residual heat removal pump.

A minimum water volume (3588 ft^3) in the RCS is used. This is the minimum volume of the RCS for residual heat removal system operation.

Dilution During Cold Shutdown

Conditions at cold shutdown require the reactor to be shut down by at least 1.0% Δk. The critical boron concentration is conservatively estimated to be 731 ppm for Cycle 1. The following conditions are assumed for an uncontrolled boron dilution during cold shutdown:

Dilution flow is assumed to be 120 gpm.

Mixing of the reactor coolant is accomplished by the operation of one residual heat removal pump.

A minimum water volume (3588 ft^3) in the RCS is used. This is the minimum volume of the RCS for residual heat removal system operation.

Dilution During Hot Standby

Conditions at hot standby require the reactor to have available at least 1.30% Δk shutdown margin. This mode of operation is analyzed both with and without the most reactive rod cluster control assembly (RCCA) stuck out of the core. The stuck rod case is assumed to occur immediately after a reactor trip and is therefore analyzed at no-load conditions. The case with no stuck rod is analyzed at 350°F which is conservative since this is the lowest permissible temperature in this mode. The critical boron concentrations are conservatively estimated to be 622 ppm (without stuck RCCA) and 448 ppm (with stuck RCCA) for Cycle 1. The following conditions are assumed in each case for a continuous boron dilution during hot standby:

1. Dilution flow is assumed to be output of two reactor makeup water pumps (240 gpm).

2. A minimum water volume (9029 ft^3) in the Reactor Coolant System is used. This corresponds to the active volume of the Reactor Coolant System while on natural circulation, i.e., the reactor vessel upper head and the pressurizer are not included.

Dilution During Startup

Conditions at startup require the reactor to have available at least 1.30% Δk shutdown margin. The critical boron concentration is conservatively estimated to be 847 ppm for Cycle 1. The following conditions are assumed for a continuous boron dilution during startup:

Dilution flow is assumed to be a conservatively high charging flow rate (300 gpm) consistent with Reactor Coolant System operation at 2250 psia and 557°F.

A minimum water volume (9800 ft^3) in the Reactor Coolant System is used. This volume corresponds to the active volume of the Reactor Coolant System minus the pressurizer volume.

The operator is alerted to an uncontrolled reactivity insertion by a reactor trip at the Power Range High Neutron Flux low setpoint (nominally 25% RTP).

Dilution During Full Power Operation

With the unit at power and the Reactor Coolant System at pressure, the dilution rate is limited by the capacity of the charging pumps (analysis is performed assuming all charging pumps are in operation although only one is normally in operation). The effective reactivity addition rate is a function of the reactor coolant temperature and boron concentration. The reactivity insertion rate calculated is based on a conservative value for the critical boron concentration for Cycle 1 (847 ppm) as well as a conservative charging flowrate capacity (125 gpm).

The Reactor Coolant System volume assumed (9800 ft^3) corresponds to the active volume of the RCS excluding the pressurizer.

The operator is alerted to an uncontrolled reactivity insertion by an overtemperature ΔT trip or by the rod insertion alarms depending on whether the plant is in manual or auto rod control.

15.4.6.3 Environmental Consequences

There would be minimal radiological consequences associated with a Chemical and Volume Control System malfunction that results in a decrease in boron concentration in the reactor coolant event. The reactor trip causes a turbine-trip, and heat is removed from the secondary system through the steam generator power relief valves or safety valves. Since no fuel damage occurs from this transient, the radiological consequences associated with this event are less severe than the steamline break event analyzed in Section 15.1.5.

15.4.6.4 Results

Dilution During Refueling

During refueling, an inadvertent dilution from the reactor makeup water system is prevented by administrative controls which isolate the RCS from the potential source of unborated makeup water.

The most limiting conditions for an inadvertent dilution from either the BTRS or the reactor makeup water system occurs with the RCS drained to 26" above the borrom ID of the reactor vessel inlet nozzles. The high flux at shutdown alarm, set at $\sqrt{10}$ times the background flux level measured by the source range nuclear instrumentation, is available at these conditions to alert the operator that a dilution event is in progress.

For this case, the operator has 96.3 minutes from the high flux at shutdown alarm to recognize and terminate the dilution before shutdown margin is lost and the reactor becomes critical.

Dilution During Cold Shutdown

While in cold shutdown, the high flux at shutdown alarm, set at $\sqrt{10}$ times the background flux level measured by the source range nuclear instrumentation, is available to alert the operator that a dilution event is in progress.

During the cold shutdown mode while operating on the residual heat removal system (RHRS) with the RCS drained to 26" above the bottom ID of the reactor vessel inlet nozzles, the operator has 17.9 minutes from the high flux at shutdown alarm to recognize and terminate the uncontrolled reactivity insertion before shutdown margin is lost and the reactor becomes critical.

Dilution During Hot Shutdown

Analysis for a dilution during hot shutdown is bounded by the analysis for a dilution during cold shutdown and hot standby.

Dilution During Hot Standby

While in hot standby, the high flux at shutdown alarm, set at $\sqrt{10}$ times the background flux level measured by the source range nuclear instrumentation, is available to alert the operator that a dilution event is in progress.

For the case with a stuck rod, the operator has 37.2 minutes from the high flux at shutdown alarm to recognize and terminate the uncontrolled reactivity insertion before shutdown margin is lost and the reactor becomes critical.

For the case without a stuck rod, the operator has 26.3 minutes from the high flux at shutdown alarm to recognize and terminate the uncontrolled reactivity insertion before shutdown margin is lost and the reactor becomes critical.

Dilution During Startup

In the event of an unplanned approach to criticality or dilution during power escalation while in the startup mode, the operator is alerted to an uncontrolled reactivity insertion by a reactor trip at the Power Range High Neutron Flux low setpoint (nominally 25% RTP). After reactor trip there is at least 27.0 minutes for operator action prior to return to criticality.

Dilution During Full Power Operation

1. With the reactor in automatic control, the power and temperature increase from boron dilution results in insertion of the rod cluster control assemblies and a decrease in the shutdown margin. The rod insertion limit alarms (low and low-low settings) provide the operator with adequate time (of other order of 65 minutes) to determine the cause of dilution, isolate the primary grade water source, and initiate reboration before the total shutdown margin is lost due to dilution.

2. With the reactor in manual control and if no operator action is taken, the power and temperature rise will cause the reactor to reach the overtemperature ΔT trip setpoint. The boron dilution accident in this case is essentially identical to rod cluster control assembly withdrawal accident. The maximum reactivity insertion rate for boron dilution is approximately .90 pcm/sec and is within the reange of insertion rates analyzed. Prior to the overtemperature ΔT trip, an overtemperature ΔT alarm and turbine runback would be actuated. There is adequate time available (of the order of 27.0 minutes) after a reactor trip for the operator to determine the cause of dilution, isolate the primary grade water sources and initiate reboration before the reactor can return to criticality.

15.4.6.5 Conclusions

The results presented above show that there is adequate time for the operator to manually terminate the source of dilution flow. Following termination of the dilution flow, the reactor will be in a stable condition. The operator can then initiate reboration to recover the shutdown margin. The calculated sequence of events is shown on Table 15.4.1-1. The radiological consequences of this event would be less limiting than the steamline break event analyzed in Section 15.1.5.

15.4.7 INADVERTENT LOADING AND OPERATION OF A FUEL ASSEMBLY IN AN IMPROPER POSITON

15.4.7.1 Identification of Causes and Accident Description

Fuel and core loading errors such as can arise from the inadvertent loading of one or more fuel assemblies into improper positions, loading a fuel rod during manufacture with one or more pellets of the wrong enrichment or the loading of a full fuel assembly during manufacture with pellets of the wrong enrichment will lead to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. Also included among possible core loading errors is the inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods.

Any error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes which are more peaked than those calculated with the correct enrichments. There is a 5 percent uncertainty margin included in the design value of power peaking factor assumed in the analysis of Condition I and Condition II transients. The incore system of moveable flux detectors which is used to verify power shapes at the start of life is capable of revealing any assembly enrichment error or loading error which causes power shapes to be peaked in excess of the design value.

To reduce the probability of core loading errors, each fuel assembly is marked with an identification number and loaded in accordance with a core loading diagram. During core loading, the identification number will be checked before each assembly is moved into the core. Serial numbers read during fuel movement are subsequently recorded on the loading diagram as a further check on proper placing after the loading is completed.

The power distortion due to any combination of misplaced fuel assemblies would significantly raise peaking factors and would be readily observable with incore flux monitors. In addition to the flux monitors, thermocouples are located at the outlet of about one third of the fuel assemblies in the core. There is a high probability that these thermocouples would also indicate any abnormally high coolant enthalpy rise. Incore flux measurements are taken during the startup subsequent to every refueling operation.

This event is classified as an ANS Condition III incident (an infrequent incident) as defined in Section 15.0.1.

TABLE 15.4.1-1 (Page 2)

Time Sequence of Events for Incidents which Cause Reactivity and Power Distribution Anomalies

Accident	Event	Time (sec.)
	Rods begin to fall into core	66.2
	Minimum DNBR occurs	67.1
Startup of an inactive reactor coolant loop at an incorrect temperature	Initiation of pump startup	1.0
	Power reaches P-8 trip setpoint	13.4
	Rods begin to drop	13.9
	Minimum DNBR occurs	15.0
CVCS Malfunction that results in a decrease in the boron concentration in the reactor coolant		
1. Dilution during refueling	Dilution begins	0
	High flux at shutdown alarm occurs	7634
	Criticality occurs	13506
2. Dilution during cold shutdown	Dilution begins	0
	High flux at shutdown alarm occurs	2064
	Criticality occurs	3137
3a. Dilution during hot standby (w/o stuck rod)	Dilution begins	0
	High flux at shutdown alarm occurs	2977
	Criticality occurs	4553
3b. Dilution during hot standby (w/stuck rod)	Dilution begins	0
	High flux at shutdown alarm occurs	4002
	Criticality occurs	6233
4. Dilution during startup	Power range low setpoint reactor trip due to dilution	0
	Criticality occurs (if dilution continues after trip)	1620

TABLE 15.4.1-1 (Page 3)

Time Sequence of Events for Incidents which Cause Reactivity and Power Distribution Anomalies

Accident	Event	Time (sec.)
5. Dilution during full power operation		
a. Automatic reactor control	Operator receives low-low rod insertion limit alarm due to dilution	0 3900
	Shutdown margin lost (if dilution continues after trip)	
b. Manual reactor control	Reactor trip setpoint reached for overtemperature ΔT	0
	Shutdown margin is lost (if dilution continues after trip)	1620
Rod Cluster Control Assembly Ejection		
1. Beginning-of-Life, Full Power	Initiation of rod ejection	0.0
	Power range high neutron flux setpoint reached	0.05
	Peak nuclear power occurs	0.14
	Rods begin to fall into core	0.55
	Peak fuel average temperature occurs	2.3
	Peak heat flux occurs	2.36
	Peak clad temperature occurs	2.37

to respond because of the large initial cooldown associated with a steam-line break transient.

Feedwater Line Break: See Table Q440.56-2

For a feedwater line break, auxiliary feedwater is initiated automatically as is Safety Injection. For the feedline break downstream of the main feedwater isolation valves the required operator actions are similar in nature to the required actions for the steamline break.

The first required operator action is to identify the faulted steam generator and isolate CA flow to that steam generator. The primary indication to the operator will be a comparison of individual steam line pressures after steam-line isolation has occurred. After identifying the faulted steam generator, the operator is instructed to isolate CA flow to that steam generator by shutting the steam generator CA isolation valve. The steam line pressure indicators and CA isolation valves are safety grade.

Where possible, the operator should also increase the auxiliary feedwater flow to the intact steam generators in order to shorten the time until primary temperatures begin to decrease. As a minimum, the operator must provide the decay heat removal through the intact steam generators by maintaining steam generator water level using auxiliary feedwater as a makeup supply. The operator can use the Steam Dump System or the steam generator PORV's to begin a controlled cooldown, or the unit may be maintained in hot standby by using the steam side safety valves for decay heat removal.

Finally, the operator must modulate the high head safety injection pumps to control primary pressure and pressurizer level. The operator must observe the primary steam pressure-temperature relationship to ensure that voiding does not occur in the Reactor Coolant System. The operator uses safety grade instrumen-tation and controls to manually control the primary system pressure and main-tain normal pressurizer level.

The analysis presented in Section 15.2.8 assumes a 30-minute delay until these actions occur.

Boron Dilution

Several indicators are available to the operator to determine that a boron dilution event is occurring depending on the mode of operation the plant is in. At startup or power the high neutron flux, overtemperature delta T, and rod position alarms are available. During other modes of operation, the operator relies on the high flux at shutdown alarm. Once the operator determines that a reactivity addition event is in progress, he should perform emergency boration utilizing the charging/SI pumps. By taking this action, the operator will prevent the core from returning to criticality. After verifying that the rate of reactivity addition is being reduced, the operator should determine the source of the boron dilution and terminate it.

Operator action is required for a boron dilution event in any mode of operation. Ample operator action time (from the receipt of alarm till the core returns to criticality) is available as discussed in Section 15.4.6.

Steam Generator Tube Rupture: See Table 440.3-3

The accident examined is the complete severance of a single steam generator tube. The operator is expected to determine that a steam generator tube rupture has occurred, and to identify and isolate the faulty steam generator on a restricted time scale in order to minimize contamination of the secondary system and ensure termination radioactive release to the atmosphere from the faulty unit. The recovery procedure can be carried out on a time scale which ensures that break flow to the secondary system is terminated before water level in the affected steam generator rises into the main steam line. Sufficient indications and controls are provided to enable the op-erator to carry out these functions satisfactorily. Consideration of the indications provided at the control board, together with the magnitude of the break flow, leads to the conclusion that the is-olation procedure can be completed within 30 minutes of accident initiation. Included in this 30 minute time period would be an al-lowance of approximately 6 minutes to trip the reactor (automatic action), 10 minutes to identify the accident as a steam generator tube rupture and 15 minutes to isolate the faulted steam generator. Preliminary diagnosis of a steam generator tube rupture can be ini-tiated prior to reactor trip. Consequently, although it may take slightly longer than 5 minutes for automatic reactor trip to occur, identification and isolation of the affected steam generator is expected to be completed within 30 minutes.

Immediately apparent symptoms of a tube rupture accident such as falling pressurizer pressure and level and increased charging pump flow are also symptoms of small steam line breaks and loss of cool-ant accidents. It is therefore important for the operator to deter-mine that the accident is a rupture of a steam generator tube in order that he may carry out the correct recovery procedure. The ac-cident under discussion can be identified by the following method. In the event of a complete tube rupture, it will be clear soon after the trip that the level in one steam generator is rising more rapidly than in the others.

Also this accident could be identified by either a condenser air ejector exhaust high radiation alarm or a steam generator blowdown radiation alarm.

The operator carries out the following major operator actions sub-sequent to reactor trip which lead to isolation of the faulted steam generator and minimizing primary to secondary leakage.

1. Identification of the faulted steam generator.

2. Isolation of the faulted steam generator.

Response:

See response to Question 440.128.

440.86 (15.3.3) Demonstrate that a rotor seizure and shaft break in a reactor coolant pump will not generate a more serious condition or result in a loss of function of the reactor coolant system or containment barriers.

Response:

The peak RCS pressure observed during a locked rotor or a shaft break in a reactor coolant pump was 2570 psia. This is well below 110% of design pressure (2750 psia). During the short time that RCS pressure is high enough to open pressurizer safety valves, the relief rate was within the maximum capacity of the safety valves. Thus, a loss of function of the reactor coolant system or containment barriers will not occur.

440.87 (15.3.3, 15.3.4) You classify the reactor coolant pump shaft break and locked rotor accidents as ANS Condition IV (limiting fault). The Standard Review Plan for Sections 15.3.3 and 15.3.4 classify these accidents as Condition III (infrequent incident).

Show that the transients meet the acceptance criteria for an infrequent incident.

Response:

The reactor coolant pump locked rotor and reactor coolant pump shaft break events are classified according to ANS as Condition IV events - limiting faults. Westinghouse follows this classification in Chapter 15 safety analysis. The results of the Catawba locked rotor shaft break events meet acceptance criteria for an infrequent event. The peak RCS pressure is maintained well below 110% of design pressure (2750 psia). The peak clad temperature is well below 2700°F, thus no clad failures are expected to occur.

440.88 (15.4.6) Reference or describe the analytical model used for obtaining the results in Section 15.4.6.2. Discuss the degree for conservatism incorporated in this model.

Response:

The boron dilution analysis is performed using a hand calculation to solve a differential equation of boron concentration as a function of time. Perfect mixing is assumed. The time at which criticality occurs is a function of the dilution flowrate, active volume in the Reactor Coolant System (RCS), and boron concentration. Minimum RCS

volume and maximum dilution flowrates are conservatively assumed for each case analyzed. Minimum shutdown margins and conservatively high boron concentrations were assumed to maximize the effect of the dilution event and therefore minimize action time.

in the event of a boron dilution assuming the highest WORTH RCCA stuck out. In addition the boron dilution analysis at power analysis for Catawba calculated operator action time from the time the overtemperature ΔT trip occurred, where the shutdown reactivity inserted at trip took credit for the highest worth RCCA stuck out.

440.91 (15.4.6) A PWR recently experienced a boron dilution incident due to inadvertent injection of NaOH into the reactor coolant system while the reactor was in cold shutdown condition. Discuss the potential for a boron dilution event caused by the chemical addition portion of the CVCS and by dilution sources other than the CVCS (for example, via the engineered safety systems).

Response:

Other than the CVCS, the sources of water of a boron concentration which could be less than the RCS boron concentration are the Recycle Holdup Tanks (RHT's) in the Boron Recycle System, and the Reactor Makeup Water (RMW) via the Reactor Makeup Water System.

There are also reactor makeup water connections to the Boron Recycle Systems; but as the only path between them and the RCS is via the RHT's, they do not constitute a separate dilution source.

No single failure can allow flow from the RHT's and the RMW to enter the RCS, so dilution of the RCS from these sources need not be considered.

The Containment Spray System does not have a spray additive tank thus there is no source for injecting NaOH into the Reactor Coolant System.

440.92 (15.4.6) The staff has specific time criteria for acceptable operator action during a boron dilution event, namely:

1. 30 minutes during refueling, and

2. 15 minutes at all other times.

The reference point for "starting the clock" is when there is an identifiable alarm in the control room alerting the operator to the situation.

For each of the cases evaluated in the SAR, identify the alarm that alerts the operator, provide the time interval from this alarm to when the core would go critical, and identify Limiting Conditions of operation for the Technical Specifications related to the sensors, alarms, and equipment necessary to mitigate all of these events.

Response:

The high flux at shutdown alarm set at $\sqrt{10}$ times background alerts the operator that a boron dilution is occurring in Modes 3, 4, 5 and 6. In Modes 1 and 2, either the overtemperature ΔT or high neutron flux alarm will alert the operator to the transient core condition.

FSAR Section 15.4.6.4 and Table 15.4.1-1 identify the time intervals from receipt of alarm to core criticality for all modes of operation.

440.93 (15.5.2) Describe the transient due to a CVCS malfunction that increases the reactor coolant inventory by injection of borated water. Identify the most limiting conditions and show that for these conditions the acceptance criteria are not exceeded and the consequences are bounded by the referenced accidents (Sections 15.5.1 and 15.4.6).

Response:

Transients due to CVCS malfunctions that increase the reactor coolant inventory can be divided into three categories;

Category 1: CVCS malfunctions that result in the injection of water with boron concentration greater than the RCS boron concentration.

Category 2: CVCS malfunctions that result in the injection of water with a boron concentration less than the RCS boron concentration.

Category 3: CVCS malfunctions that result in the injection of water with a boron concentration equal to RCS boron concentration.

There are two possible criteria for evaluating these transients, core integrity and overfilling of the pressurizer. Transients of the types listed in Category 1 are bounded by the "Inadvertent Operation of Emergency Core Cooling System Analysis" presented in Section 15.5.1. Transients of the type listed in Category 2 are bounded by the "CVCS Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant" presented in Section 15.4.6.

CVCS malfunctions of the type described under Category 3 will not result in any significant nuclear power or RCS temperature transient, and therefore are not presented in Chapter 15. This type of transient may result in filling the perssurizer water solid. Attached is an analysis of the CVCS malfunction that result in injection of water with a boron concentration equal to the RCS boron concentration.