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August 22, 1985

Mr. Hugh L. Thompson, Jr., Director
Division of Licensing
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Subject: Catawba Nuclear Station
Docket Nos. 50-413 and 50-414
Generic Letter 85-12

Dear Mr. Thompson:

Please find attached our response to Generic Letter 85-12 for Catawba Nuclear Station concerning implementation of TMI Action Item II.K.3.5 "Automatic Trip of Reactor Coolant Pumps."

Very truly yours,

Hal B. Tucker/sm

Hal B. Tucker

LTP:smh

Attachments

A046
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cc: Dr. J. Nelson Grace, Regional Administrator
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Duke Power Company
Catawba Nuclear Station
Response to Generic Letter 85-12

A. Determination of RCP Trip Criteria

1. Identify the instrumentation to be used to determine the RCP trip setpoint, including the degree of redundancy of each parameter signal needed for the criterion chosen.

Response: Since the RCS subcooling criterion has been selected, both pressure and temperature must be determined. The calculation of subcooling will normally be made by a program on the plant computer using the following inputs:

Pressure

RCS Wide Range Pressure
RCS Low Range Pressure

Temperature

Loop A Wide Range Hot Leg Temperature
Loop B Wide Range Hot Leg Temperature
Loop C Wide Range Hot Leg Temperature
Loop D Wide Range Hot Leg Temperature
Core Exit Thermocouple Temperatures

The pressure input is selected from the RCS Low Range Pressure, if valid, since this instrument has the smaller uncertainty. Otherwise, an average of RCS Wide Range Pressures is used. If one of these indications is unavailable, the other is used alone. For the temperature input the five (5) highest valid thermocouple indications are averaged. This average is compared with the four (4) Loop Wide Range Hot Leg Temperatures. The highest valid indication from this comparison is used as the temperature input.

If the plant computer calculation is not available, a redundant means of determining subcooling is available to the operator via graphs maintained in the control room. All of the above instrumentation is available to the operator on control room indications. Another channel of RCS Wide Range Pressure is also available as an additional pressure indication.

2. a) Identify the instrumentation uncertainties for both normal and adverse containment conditions.

b) Describe the basis for the selection of the adverse containment parameters.

c) Address, as appropriate, local conditions such as fluid jets or pipe whip which might influence the instrumentation reliability.

Response:

a) The uncertainties use in the subcooling determination described above are listed below for each instrument.

Instrument	Normal Conditions	Adverse Conditions
RCS Wide Range Pressure	63 psi	213 psi
RCS Low Range Pressure	15 psi	125 psi
Loop Wide Range Hot Leg Temperature	8.1 °F	8.5 °F
Core Exit Thermocouple Temperature	7.2 °F*	8.5 °F*

*The core exit thermocouple error increases with increasing temperature. The error shown is for 530°F and is conservatively used for temperatures below 530°F.

b) A value of 3 psig, which corresponds approximately to the high-high containment pressure setpoint, is used as the basis for the selection of adverse containment parameters. This value is more appropriate, for an ice condenser plant such as Catawba, than the Westinghouse Emergency Response Guidelines (ERG) value, "approximately 5 psig", which is for a dry containment.

c) Local conditions which might influence instrument reliability have been addressed in the Catawba Nuclear Station Response to NUREG-0588, as currently revised in a letter from H. B. Tucker, Duke Power Co., to H. R. Denton, NRC, April 1, 1985.

3. In addressing the selection of the criterion, consideration of uncertainties associated with the WOG supplied analyses values must be provided. These uncertainties include both uncertainties in the computer program results and uncertainties resulting from plant specific features not representative of the generic data group.

Response:

The LOFTRAN computer code was used to perform the alternate RCP trip criteria analyses. Both Steam Generator Tube Rupture (SGTR) and non-LOCA events were simulated in these analyses. Results from the SGTR analyses were used to obtain all but three of the trip parameters. LOFTRAN is a Westinghouse licensed code used for FSAR SGTR and non-LOCA analyses. The code has been validated against the January 1982 SGTR event at the Ginna plant. The results of this validation show that LOFTRAN can accurately predict RCS pressure, RCS temperatures and secondary pressures especially in the first ten minutes of the transient. This is the critical time period when minimum pressure and subcooling is determined.

The major causes of uncertainties and conservatism in the computer program results, assuming no changes in the initial plant conditions (i.e., full power, pressurizer lever, all SI and AFW pumps running) are due to either models in or inputs to LOFTRAN. The following are

considered to have the most impact on the determination of the RCP trip criteria:

1. Break flow
2. SI flow
3. Decay heat
4. Auxiliary feedwater flow

The following sections provide an evaluation of the uncertainties associated with each of these items.

To conservatively simulate a double ended tube rupture in safety analyses, the break flow model used in LOFTRAN includes substantial amount of conservatism (i.e., predicts higher break flow than actually expected). Westinghouse has performed analyses and developed a more realistic break flow model that has been validated against the Ginna SGTR tube rupture data. The break flow model used in the WOG analyses has been shown to be approximately 30% conservative when the effect of the higher predicted break flow is compared to the more realistic model. The consequence of the higher predicted break flow is a lower than expected predicted minimum pressure.

The SI flow inputs used were derived from best estimate calculations, assuming all SI trains operating. An evaluation of the calculational methodology show that these inputs have a maximum uncertainty of ±10%.

The decay heat model used in the WOG analyses was based on the 1971 ANS 5.1 standard. When compared with the more recent 1979 ANS 5.1 decay heat inputs, the values used in the WOG analyses is higher by about 5%. To determine the effect of the uncertainty due to the decay heat model, a sensitivity study was conducted for SGTR. The results of this study show that a 20% decrease in decay heat resulted in only a 1% decrease in RCS pressure for the first 10 minutes of the transient. Since RCS temperature is controlled by the steam dump, it is not affected by the decay heat model uncertainty.

The AFW flow rate input used in the WOG analyses are best estimate values, assuming that all auxiliary feed pumps running, minimum pump start delay, and no throttling. To evaluate the uncertainties with AFW flow rate, a sensitivity study was performed. Results from the two loop plant study show that, a 64% increase in AFW flow resulted in only an 8% decrease in minimum RCS pressure, a 3% decrease in minimum RCS subcooling, and an 8% decrease in minimum pressure differential. Results from the 3 loop plant study show that, a 27% increase in AFW flow resulted in only a 3% decrease in minimum RCS pressure, a 2% decrease in minimum RCS subcooling, and a 2% decrease in pressure differential.

The effects of all these uncertainties with the models and input parameters were evaluated and it was concluded that the contributions from the break flow conservatism and the SI

uncertainty dominate. The calculated overall uncertainty in the WOG analyses as a result of these considerations for the Catawba units is -10°F to +10°F for the RCS subcooling RCP trip setpoint. Due to the minimal effects from the decay heat model and AFW input, these results include only the effects of the uncertainties due to the break flow model and SI flow inputs.

B. Potential Reactor Coolant Pump Problems

1. Assure that containment isolation, including inadvertent isolation, will not cause problems if it occurs for non-LOCA transients and accidents.

   a. Demonstrate that, if water services needed for RCP operations are terminated, they can be restored fast enough once a non-LOCA situation is confirmed to prevent seal damage or failure.

      Response: Reactor coolant pump seal damage is prevented by a continued supply of cooling water from the Chemical and Volume Control System (seal injection) and/or from the Component Cooling System (thermal barrier cooling). Only one of these supplies is necessary to prevent seal damage or failure. Although thermal barrier cooling is isolated on Phase B containment isolation, seal injection is maintained regardless of isolation status. Thus seal damage or failure will be avoided for LOCA, non-LOCA, or inadvertent containment isolation.

   b. Confirm that containment isolation with continued pump operation will not lead to seal or pump damage or failure.

      Response: The effects of containment isolation on reactor coolant pump seals are addressed in the previous item. Component cooling and service water to the reactor coolant pump motors and thermal barriers is isolated on Phase B containment isolation (high-high containment pressure). This is done to isolate any piping in these systems which might be damaged by pipe whip or jet impingement caused by a LOCA. The operators are trained to trip the reactor coolant pumps promptly upon the loss of water services due to Phase B containment isolation. Thus, there will be no continued operation which will lead to pump damage. There are no water services, needed for continued pump operation, which are lost due to Phase A containment isolation (high containment pressure).

2. Identify the components required to trip the RCPs, including relays, power supplies and breakers. Assure that RCP trip, when determined to be necessary, will occur. If necessary, as a result of the location of any critical component, include the effects of adverse containment conditions on RCP trip reliability. Describe the basis for the adverse containment parameters selected.

Response: The reactor coolant pumps are manually tripped by depressing the trip pushbuttons (one for each pump) on a panel located in the control room. This action will energize one of the two independent shunt trip coils located in each of the 6.9KV RCP switchgear assemblies, causing the RCP switchgear breakers to open (and hence the RCPs to trip). There are four RCP switchgear assemblies per unit, one for each RCP motor, located in the penetration rooms of the auxiliary building.

All RCP switchgear protective relays, including the trip coils, rely on 125VDC power to trip the breakers. The breaker control power is furnished by 125VDC Auxiliary Control Power System distribution centers located in the auxiliary building. In the event control power is lost, the RCP switchgear breakers may be manually tripped by means of the trip/close levers located in the breaker cubicles.

Since none of the critical component are located inside the reactor building, adverse containment conditions need not be considered.

C. Operator Training and Procedures (RCP Trip)

1. Describe the operator training program for RCP trip. Include the general philosophy regarding the need to trip pumps versus the desire to keep pumps running.

Response: The following is the Reactor Coolant Pump Trip Criteria taught to the Catawba Operators:

Response: Reactor Coolant Pump Training is conducted formally during various types of classroom training sessions. These sessions include License Prep Reactor Operator Training, License Prep Senior Reactor Operator Training and Requalification Training. During each of these training sessions a variety of topics are discussed. Among these topics are: The Purpose, Component Description, Component Operation, Instrumentation and Control, Setpoints, Limits and Precautions. During the Component Operation and Limits and Precautions discussions, a variety of Reactor Coolant Pump Trips and their associated limits are discussed. Among these trips are: High Upper and/or Lower Bearing Temperature Trips, High Motor Stator Winding Temperature Trip, Excessive Vibration on the Rotor and Casing, High Number One Seal Outlet Temperature Trips, High Pump Radial Bearing Temperature Trips and Loss of RCS Subcooling Trip.

Further training on Reactor Coolant Pump Trips include formal classroom training on Abnormal and Emergency Procedures. Abnormal Procedure "Malfunction of Reactor Coolant Pumps" describes the various Symptoms, Immediate Actions and Subsequent Actions the operator is required to take in the event of a pump malfunction. Several Emergency Procedures describe the required pump trip following loss of RCS subcooling. The Abnormal Procedure "Loss of Reactor Coolant Flow" will refer the operator to the "Malfunction of Reactor Coolant Pump" Abnormal Procedure in the event the loss of flow was due to a pump malfunction.

Various industry incident reports pertaining to NC Pumps are discussed with all Licensed personnel in the classroom. The WOG criteria for the simultaneous tripping of all four NC Pumps are discussed each time the lesson is taught. All procedures discussed in paragraphs above are taught during Simulator Requalification in both annual and biannual sessions. Any NRC and/or INPO recommendations concerning RCPs are discussed, analyzed and implemented wherever applicable.

2. Identify those procedures which include RCP trip related operations:

(a) RCP trip using WOG alternate criteria

Response: EP/01, Reactor Trip Or Safety Injection
EP/1C, High Energy Line Break Inside Containment
EP/1D, Steam Line Break Outside Containment
EP/1E, Steam Generator Tube Rupture

(b) RCP restart

Response: EP/1A1, Natural Circulation Cooldown
EP/1C1, SI Termination Following High Energy Line Break Inside Containment
EP/1C2, Post-LOCA Cooldown And Depressurization
EP/1D1, SI Termination Following Steam Line Break
EP/1E, Steam Generator Tube Rupture
EP/1E3, SGTR With Continuous NC System Leakage: Subcooled Recovery
EP/1E4, SGTR With Continuous NC System Leakage: Saturated Recovery
EP/1E5, SGTR Without Pressurizer Pressure Control
EP/2B1, Inadequate Core Cooling
EP/2C1, Loss Of Secondary Heat Sink
EP/2D1, Imminent Pressurized Thermal Shock Condition
EP/2D2, Anticipated Pressurized Thermal Shock Condition
EP/2F3, Void In Reactor Vessel

(c) Decay heat removal by natural circulation

Response: All Catawba emergency procedures, as appropriate, are written to address either forced or natural circulation decay heat removal.

(d) Primary system void removal

Response: Although several different procedures refer to void removal, the only one which restarts reactor coolant pumps for this purpose is EP/2F3, Void In Reactor Vessel.

(e) Use of steam generators with and without RCPs operating

Response: In the Catawba emergency procedures, steam generators are used for decay heat removal and for reactor coolant system cooldown. Decay heat removal is addressed in item C.2.(c) above.

The emergency procedures which use the steam generators for cooling down are

EP/1A1, Natural Circulation Cooldown
EP/1C2, Post-LOCA Cooldown And Depressurization
EP/1C5, Loss Of Emergency Coolant Recirculation
EP/1E, Steam Generator Tube Rupture
EP/1E1, Post-SGTR Cooldown And Depressurization
EP/1E2, SGTR Alternate Cooldown Using Backfilling
EP/1E3, SGTR With Continuous NC System Leakage: Subcooled Recovery
EP/1E4, SGTR With Continuous NC System Leakage: Saturated Recovery
EP/1E5, SGTR Without Pressurizer Pressure Control
EP/2B1, Inadequate Core Cooling
EP/2B2, Degraded Core Cooling
EP/2C3, S/G Overpressure
EP/03, Loss Of All AC Power

(f) RCP trip for other reasons

Response: Reactor coolant pumps are tripped in the Catawba emergency procedures for two additional reasons:

i) prevention of pump damage from operation under abnormal conditions

ii) prevention of heat addition to the reactor coolant system under degraded heat sink conditions

The emergency procedures which contain these additional trip instructions are

EP/1C1, SI Termination Following High Energy Line Break Inside Containment
EP/1C2, Post-LOCA Cooldown And Depressurization
EP/1C5, Loss Of Emergency Coolant Recirculation
EP/1D1, SI Termination Following Steam Line Break
EP/1E1, Post-SGTR Cooldown And Depressurization
EP/1E2, SGTR Alternate Cooldown Using Backfilling
EP/1E3, SGTR With Continuous NC System Leakage: Subcooled Recovery
EP/1E4, SGTR With Continuous NC System Leakage: Saturated Recovery
EP/1E5, SGTR Without Pressurizer Pressure Control
EP/2C1, Loss Of Secondary Heat Sink

The Catawba emergency procedures have already been reviewed and approved by the NRC as documented in Catawba SSER No. 4, December, 1984. The procedures are based on the Westinghouse Owners Group Emergency Response Guidelines (ERGs) which are currently under review by the NRC.