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November 6, 1985

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

Subject: Virgil C. Summer Nuclear Station
Docket No. 50/395
Operating License No. NPF-12
Reactor Coolant Pump Trip Criteria

Dear Mr. Denton:

On July 8, 1985, South Carolina Electric and Gas Company (SCE&G) received Generic Letter 85-12, "Implementation of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps'". This Generic Letter informed utilities of the NRC Staff's conclusions regarding the Westinghouse Owners Group (WOG) submittals on reactor coolant pump (RCP) trip provided in response to Generic Letters 83-10c and d, and gave guidance concerning implementation of the RCP trip criteria. The NRC Staff's safety evaluation was enclosed with the Generic Letter and identified information which each licensee needed to supply to complete their response to Generic Letters 83-10c and d. SCE&G is therefore providing this letter to supply the NRC Staff with the requested information. For your convenience, the information is provided below, identified to correspond to the items outlined in Section IV of the NRC's safety evaluation.

A. Determination of RCP Trip Criteria

The instruments used for identifying the RCP trip criteria at the Virgil C. Summer Nuclear Station are the two Post Accident Monitoring System (PAMS) grade Reactor Coolant System (RCS) wide range instruments. RCP trip is required when both channels indicate a value below the setpoint of 1380 psig. These instruments are located outside of containment and therefore their associated transmitters are not subjected to the containment harsh environment, pipe whips, or jet forces. The indications from these transmitters have an uncertainty of 3% which is taken into account in the determination of the setpoint.

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 Final Safety Analysis Report (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.

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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 level, all Safety Injection (SI) and Auxiliary Feedwater (AFW) pumps running) are due to either models or inputs to LOFTRAN. The following models have been used in the determination of the RCP trip criteria:

1. Break flow
2. SI flow
3. Decay heat
4. AFW flow

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

To simulate a double ended tube rupture in safety analyses, the break flow model used in LOFTRAN includes a 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 data. The break flow model used in the WOG analyses has been shown to be approximately 30% more 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 minimum pressure.

The SI flow inputs used were derived from best estimate calculations assuming all SI trains operating. An evaluation of the calculational methodology shows that these inputs have a maximum uncertainty of $\pm$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 are 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 inputs used in the WOG analyses are best estimate values assuming all auxiliary feed pumps are running, minimum pump start delay, and no throttling. To evaluate the uncertainties associated with AFW flow rate, a sensitivity study was performed. Results from the 3 loop plant study show that a 27% increase in

AFW 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 setpoint stated in the Virgil C. Summer Nuclear Station Emergency Operating Procedure (EOP) is 1380 psig and was calculated based on the RCS pressure methodology. This setpoint yields a margin of 148 psi for SGTR, 41 psi for steamline break and 528 psi for feedline breaks.

B. Potential Reactor Coolant Pump Problems

RCP seal cooling is provided by RCP seal injection flow which comes from the charging/safety injection pumps and from component cooling water (CCW) to the thermal barrier. No automatic closure signals are provided for seal injection; however, CCW is isolated via the containment spray/Phase B isolation signal. Either of the two seal cooling methods are suitable to preclude seal failure for extended periods of time. Furthermore, EOPs require the operator to trip the RCP's if a containment spray/Phase B isolation signal is generated.

RCP motor bearing cooling is provided by CCW. As stated above, this cooling system is isolated by the spray/Phase B signal which inturn requires that the RCPs be tripped. Inadvertent isolation of CCW to the motor bearing requires (via EOP) the RCP to be tripped after 10 minutes or when motor bearing temperatures reach 195°F.

The RCPs are controlled by 7200V breakers. These breakers utilize 125V DC power provided from the substation primary DC distribution panel to energize the trip coil and trip the RCPs. The trip coil is energized directly from a control switch located in the main control room and no auxiliary relays or devices are required to trip a RCP. The DC distribution panel used to energize the trip coil is non-safety related, but is battery backed and independent of the normal balance of plant (BOP) DC distribution system. If the RCP breaker failed to open when necessary, it could be opened by local manual operator action or the associated RCP BOP bus could be de-energized by its feeder breakers to initiate the trip.

C. Operator Training and Procedures (RCP Trip)

In EOP operator training, specific step-by-step instructions and their bases are discussed in class and during the simulator exercises. For example, the EOP for reactor trip/SI actuation covers trip of RCPs where Phase B isolation isolates CCW to the motor bearing coolers in one step, and where RCS pressure is monitored for RCP trip criteria

(steam break/LOCA event) response in another step. The RCP trip criteria and basis are consistent throughout the initial phase of the Optimal Recovery Guidelines (ORGs). These bases emphasize protection of the RCPs from damage on Phase B isolation and minimization of RCS inventory loss during steam break/LOCA events. Subsequent EOP actions to trip RCPs, for example, to minimize RCS heat input, are fully explained during the simulator and classroom sessions. Further recovery actions in the ORGs call for starting RCPs if none are running with the alternative of establishing natural circulation cooling only if a RCP cannot be started. The ORG organization of steps and general training make it clear to the operator that forced RCS cooling is the desired method.

RCP trip training is included in the EOP training sessions. In the past year this training has included 40 hours of general classroom and simulator training, 2.5 hours of EOP review, and entry into at least one EOP during each simulator session.

Attachment A to this letter identifies those EOPs at the Virgil C. Summer Nuclear Station which include RCP trip related operations. Copies of these procedures can be provided to the NRC upon request.

If you should have additional questions, please advise.

Very truly yours,

D. A. Nauman

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Attachment A

EOPs Including RCP Trip Related Operations

A. RCP Trip using WOG alternate criteria

EOPs 1.0, 2.0, 3.1, 4.0

B. RCP Restart

EOPs 1.1, 1.2, 1.3, 2.1, 4.0, 4.2, 4.3, 4.4, 14.0, 16.0, 18.2

C. Decay heat removal by natural circulation

EOPs 1.1, 1.2., 1.3, 2.1, 4.0, 4.2, 4.3, 6.0, 6.1, 6.2, 8.0, 15.0

D. Primary system void removal

EOP 18.2 for method

EOPs 1.3, 2.0, 2.1, 4.0, 4.1, 4.3, 6.0, 15.0 for monitoring

E. Use of steam generators with and without RCP's operating

EOPs 1.0, 1.1, 1.2, 1.3, 2.0, 2.1, 2.3, 2.4, 3.0, 3.1, 4.0, 4.1, 4.2, 4.3, 4.4, 6.0, 8.0, 13.0, 14.0, 14.1, 15.0, 15.2, 15.3, 15.4, 16.0, 16.1, 17.0, 17.1, 18.2

F. RCP Trip for other reasons.*

EOPs 1.0, 2.0, 2.1, 4.0, 4.1, 4.2, 4.4, 8.0, 14.0, 14.1, 15.0, 18.0
*Reasons include either loss of component cooling water or loss of seal leakoff.