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October 4, 1985

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Washington, D. C. 20555

Attention: Mr. Hugh L. Thompson, Jr., Director
Division of Licensing

Gentlemen:

IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5,
"AUTOMATIC TRIP OF REACTOR COOLANT PUMPS"
(GENERIC LETTER 85-12)
SALEM UNITS 1 AND 2
DOCKET NOS. 50-272 AND 50-311

Pursuant to your request for additional information dated June 28, 1985, PSE&G herewith submits its response in the attachment to this letter.

Should you have any questions, we will be pleased to discuss them with you.

Sincerely,

Attachment

C Mr. Donald C. Fischer
Licensing Project Manager

Mr. Thomas J. Kenny
Senior Resident Inspector

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PUBLIC SERVICE ELECTRIC AND GAS COMPANY
SALEM GENERATING 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: Salem has selected low RCS Pressure less than 1450 psig as the basic criteria for tripping the Reactor Coolant Pumps (RCP's) from the three options provided in Westinghouse Owners Group (WOG) developed Emergency Response Guidelines (ERG's), High Pressure Version. For a Steam Generator Tube Rupture (SGTR) or non-LOCA event, reactor coolant subcooling less than 10°F (in addition to instrument uncertainties) is used to trip the RCP's.

RCS pressure is sensed and transmitted by two redundant pressure transmitters PT403 and PT405. These two transmitters with different power sources are located on the same loop, but physically separate. For RCS subcooling determination RCS wide range hot leg RTD's and Core Exit Thermocouples (CET's) are used in addition to the above RCS pressure transmitters.

The highest of four (4) Wide Range (WR) hot leg RTD's and five (5) designated thermocouple indications are used to determine the subcooling from the subcooling tables for normal and adverse containment conditions. Instrument uncertainties for RCS pressure transmitters, WR RTD's and CET's are built into these tables so that the determination of RCS subcooling is straight forward.

2. Identify the instrumentation uncertainties for both normal and adverse containment conditions. Describe the basis for the selection of the adverse containment parameters. Address, as appropriate, local condition such as fluid jets or pipe whip which might influence the instrumentation reliability.

Response: a. The instrumentation uncertainties for both normal and adverse containment conditions as calculated are provided below:

Instrument	No. of Instrument	Normal Conditions	Adverse Conditions
RCS WR Pressure	2	±51 PSI	±279 PSI
Hot Leg WR RTD's	4	±15.4 °F	±15.4 °F
Core Exit Thermocouples	5/64	±5.26 °F	±5.26 °F

The channel accuracy was calculated using the square root of the sum of the squares method. The accuracies of each individual component in the loop were identified and used in the calculation.

b. If either containment pressure exceeds 4 PSIG or the containment radiation exceeds 10^5 R/hr., the operators would implement the established procedures using the adverse containment process parameter values within those procedures. This is consistent with WOG developed criteria in ERG's.

c. The pressure transmitters PT403/405 are mounted in protective panels in the containment above flood level. The protective panels are designed to withstand the increased pressure due to High Energy Line Break (HELB) and LOCA without damage to the internal components. Panel 797-1B (PT-403) is protected from HELB in SG blowdown line by a sleeve and restraints. Panel 797-1A (PT-405) has no high energy line nearby.

3. In addressing the selection of the criterion, consideration to 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 Salem parameters in Table 1 of Reference 1. 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 critical time period when minimum pressure and subcooling is determined.

The major causes of uncertainties and conservatism in the computer program results, assuming no change in the initial plant conditions (i.e. full power, pressurizer level, all SI and AFW pumps run) are due to either models 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 lower than expected predicted minimum pressure.

The SI flow input used was 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 5.12 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 input used in the WOG analyses are best estimate values, assuming that all auxiliary feed pumps are 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 a 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. Although specific aux. feedwater flow rate sensitivities were not done for four loop plants, it would be expected that four loop results would be similar to the two and three loop results and the conclusion would be the same.

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 Salem Units 1 & 2 are +30 to +200 psi for the RCS/pressure and -2 °F to +20 °F for 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: Component Cooling Water System provides cooling water to the reactor coolant pump thermal barrier heat exchanger, as well as to the upper and lower motor bearing oil coolers. Seal injection flow, at a slightly higher pressure and at a lower temperature than the Reactor Coolant System, enters the pump through a pipe penetration between the pump radial bearing and the thermal barrier heat exchanger. Most of the seal injection flow (directed to the seals) is discharged through the No. 1 seal leakoff, which is piped to the Volume Control Tank.

Reactor coolant pump seal damage is prevented by an assured supply of cooling water from the Chemical and Volume Control System (seal injection) and from the component Cooling Water 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 the isolation status. Thus, seal damage or failure will be avoided for LOCA, non-LOCA or inadvertent 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 water to the thermal barrier heat exchanger and motor bearing oil coolers 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. On Phase B isolation, Containment Spray is actuated. The motors for the RCP's can not run under spray conditions. The operators are

trained to trip the RCP's promptly upon the Phase B containment isolation. Also The RCP's could have been tripped off manually under RCS pressure or subcooling criteria as discussed above. Thus, there will be no continued operation which will lead to pump damage. The No. 1 seal leakoff line isolates on Phase A containment isolation which occurs concurrently with safety injection actuation. Although Phase A isolation could occur during SGTR and other non-LOCA transients, seal leakoff flow would continue via the relief valve on the seal leakoff line. The relief valve is set at 150 psig and it discharges to the Pressurizer Relief Tank which is located inside the containment.

2. Identify the components required to trip the RCP's, 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 RCP's are manually tripped by depressing the trip push buttons (one for each pump) on the control room console. This action will energize an auxiliary relay which interfaces with the 125 VDC breaker control circuit to energize the breaker trip coil in each of the 4160 volt breakers causing the breakers to open and hence the RCP's to trip. There are four RCP switchgear assemblies per unit, one for each RCP motor, located in the turbine building.

All RCP switchgear protective relays, including the trip coils, rely on 125 VDC power to trip the breakers. The breaker control power is furnished by the 125 VDC Power System. The 125 VDC feed to each switchgear train is manually interlocked to a backup 125 VDC source. The distribution panels supplying the feeds are located in the room below the control room. The manually interlocked switches are located at the switchgear cubicles in the Turbine Building. Additionally the loss of 125 VDC control power is alarmed in the control room.

The auxiliary relay operated from the control pushbutton is a 28 VDC relay which is fed from the 28 VDC Power System. The station is equipped with two 28 VDC power systems which feed separate 28 VDC distribution panels. The distribution panels are equipped with manually interlocked switches so that the distribution panel may be fed from the opposite 28 VDC battery system if its normal feed were to fail. Additionally, the 28 VDC control circuit is equipped with a loss of power alarm which is alarmed in the control room. In the event control power is lost, the RCP switchgear breakers may be manually tripped by the operation of a manual trip pushbutton located on the breaker.

Since none of the critical components are located inside the containment building, adverse containment conditions have no impact on the RCP trip reliability.

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: Reactor Coolant Pump Training is conducted formally during various types of class room training sessions. These sessions include Initial Reactor Operator License Training, Initial Senior Reactor Operator License Training and Licensed Operators 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: Failure of Thermal Barrier Heat Exchanger, High Vibrations, Improper Oil Level, High Bearing Temperature, RCP Seal Failure, Low RCS Pressure and Loss of RCS Subcooling.

Further training on Reactor Coolant Pump Trips include formal classroom training on abnormal and Emergency Operating Procedures. Emergency Instruction "Failure of a Reactor Coolant Pump" describes the various Symptoms, Immediate Actions and Subsequent Actions the operator is required to take in the event of a pump malfunction. Several Emergency Operating Procedures (EOP's) describe the required pump trip following RCS pressure reduction and loss of RCS subcooling.

Various industry incident reports pertaining to the Reactor Coolant Pumps are discussed with all licensed personnel in the classroom. The RCP trip criteria for the simultaneous tripping of all four RCP's are discussed each time the lesson is taught. All procedures discussed in the paragraphs above are taught during Simulator Requalification sessions conducted during the annual Licensed Operator Requalification Program. Any NRC and/or INPO recommendations concerning RCP's are discussed, analyzed and implemented whenever applicable.

At Salem, for all transients and accidents other than Small Break LOCA (SB LOCA), continued operation of RCP's is desirable. However, for SB LOCA, RCP's are tripped if RCP trip setpoint is reached.

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

   a. RCP trip using WOG alternate criteria.

Response:

EOP-TRIP-1	- Reactor Trip or Safety Injection
EOP-TRIP-2	- Reactor Trip Response
EOP-LOCA-1	- Loss of Reactor Coolant
EOP-LOCA-2	- Post LOCA Cooldown and Depressurization
EOP-LOCA-3	- Transfer to Cold Leg Recirculation
EOP-LOCA-5	- Loss of Emergency Recirculation
EOP-LOCA-6	- LOCA Outside Containment
EOP-LOSC-1	- Loss of Secondary Coolant
EOP-LOSC-2	- Multiple Steam Generator Depressurization

EOP-SGTR-1	- Steam Generator Tube Rupture
EOP-SGTR-2	- Post SGTR Cooldown
EOP-SGTR-3	- SGTR with LOCA-Subcooled Recovery
EOP-SGTR-4	- SGTR with LOCA-Saturated Recovery
EOP-SGTR-5	- SGTR Without Pressurizer Press Control

b. RCP restart

Response:

EOP-TRIP-1	- Reactor Trip or Safety Injection
EOP-TRIP-4	- Natural Circulation Cooldown
EOP-TRIP-5	- Natural Circulation, Rapid Cooldown Without RVLIS
EOP-TRIP-6	- Natural Circulation, Rapid Cooldown with RVLIS
EOP-LOCA-2	- Post LOCA Cooldown & Depressurization
EOP-LOSC-1	- Loss of Secondary Coolant
EOP-LOSC-2	- Multiple S.G. Depressurization
EOP-SGTR-1	- Steam Generator Tube Rupture
EOP-SGTR-3	- SGTR with LOCA-Subcooled Recovery
EOP-SGTR-4	- SGTR with LOCA-Saturated Recovery
EOP-LOPA-2	- Loss of All AC Power/Safety Injection not Required
EOP-FRCC-1	- Response to Inadequate Core Cooling
EOP-FRTS-1	- Response to Imminent Pressurized Thermal Shock Conditions
EOP-FRCI-3	- Response to Void in Reactor Vessel

c. Decay heat removal by natural circulation.

Response: All Salem 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 EOP-FRCI-3, Response to Void in Reactor Vessel.

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

Response: In the Salem EOP's, steam generators are used for decay heat removal and for RCS cooldown. Decay heat removal is addressed in C.2c above. The emergency procedures which use the steam generators for cooling down are:

EOP-TRIP-4	- Natural Circulation Cooldown
EOP-LOCA-2	- Post LOCA Cooldown and Depressurization
EOP-SGTR-1	- Steam Generator Tube Rupture
EOP-SGTR-2	- SGTR Cooldown Using Steam Dump
EOP-SGTR-2	- SGTR Cooldown Using Backfill
EOP-SGTR-2	- SGTR Cooldown Using Blowdown
EOP-SGTR-3	- SGTR with LOCA-Subcooled Recovery
EOP-SGTR-4	- SGTR with LOCA-Saturated Recovery
EOP-SGTR-5	- SGTR without Pressurizer Press Control
EOP-LOCA-5	- Loss of Emergency Recirculation
EOP-LOPA-1	- Loss of All AC Power
EOP-FRCC-1	- Response to Inadequate Core Cooling
EOP-FRCC-2	- Response to Degraded Core Cooling
EOP-FRHS-2	- Response to S.G. Overpressurization

f. RCP trip for other reasons.

Response: RCP's are tripped in the Salem EOP's for two additional reasons:

1) Prevention of pump damage from operation under abnormal conditions.

2) Prevention of heat addition to the reactor coolant system under degraded heat sink conditions.

The emergency procedures which contain these additional trip instructions are:

EOP-TRIP-3	- SI Termination Following HELB Inside Containment
EOP-LOCA-2	- Post LOCA Cooldown and Depressurization
EOP-TRIP-3	- SI Termination Following Steam line Break
EOP-SGTR-2	- SGTR Cooldown Using Steam Dump
EOP-SGTR-2	- SGTR Cooldown Using Backfill
EOP-SGTR-2	- SGTR Cooldown Using Blowdown
EOP-SGTR-3	- SGTR with LOCA; Subcooled Recovery
EOP-SGTR-4	- SGTR with LOCVA; Saturated Recovery
EOP-SGTR-5	- SGTR without Pressurizer Pressure Control
EOP-LOCA-5	- Loss of Emergency Recirculation
EOP-FRHS-1	- Response to Loss of Secondary Heat Sink

REFERENCES:

1. Background information for Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERG's) Generic Issue - RCP Trip/Restart in Executive Volume.