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SUBJECT: Forwards response to 850628 Generic Ltr 85-12 re NRC conclusions concerning Westinghouse Owners Group submittals on automatic trip of reactor coolant pumps.

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L-85-471

Office of Nuclear Reactor Regulation
Attention: Mr. Hugh L. Thompson, Jr., Director
Division of PWR Licensing - A
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Thompson:

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Automatic Trip of Reactor Coolant Pumps
Generic Letter No. 85-12

Generic Letter No. 85-12 dated June 28, 1985 provided the NRC Staff conclusions regarding the Westinghouse Owners Group submittals on reactor coolant pump trip in response to Generic Letters 83-10c and d, and provided guidance concerning implementation of the reactor coolant pump trip criteria.

Section IV.C of the NRC Safety Evaluation required the submittal of certain information regarding determination of reactor coolant pump trip criteria, potential reactor coolant pump problems and operator training and procedures. The information for Turkey Point Units 3 and 4 is attached.

If you have any questions concerning this information, please call us.

Very truly yours,

C. O. Woody
Group Vice President
Nuclear Energy

COW/TCG/gp

Attachment

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# RESPONSE TO NRC
# GENERIC LETTER 85-12
# FOR TURKEY POINT UNITS 3 and 4

As a result of the accident at Three Mile Island, NRC developed the "TMI Action Plan" (NUREG-0660, Reference 1) and the "Clarification of the TMI Action Plan" (NUREG-0737, Reference 2). One issue identified as requiring study and resolution as a result of TMI was Item II.K.3.5 "Automatic Trip of Reactor Coolant Pumps During Loss-of-Coolant Accident." Subsequent to issuance of NUREG-0737, extensive studies were performed to better understand the dynamic response of PWRs to small break LOCAs. As a result of greater knowledge of these events, NRC issued Generic Letter 83-10 "Resolution of TMI Action Item II.K.3.5 'Automatic Trip of Reactor Coolant Pumps'." This letter provided specific guidance for developing RCP trip criteria.

The Westinghouse Owners Group (WOG) submitted two reports to the NRC in response to the Westinghouse specific Generic Letters, 83-10c and d. The first report provided an "Evaluation of Alternate RCP Trip Criteria" (Reference 3). The second report provided the "Justification of Manual RCP Trip for Small Break LOCA Events" (Reference 4). The WOG also provided additional information (Reference 5) in response to an NRC request based on the review of the WOG submittals.

From their review, NRC has issued Generic Letter 85-12 "Implementation of TMI Item II.K.3.5 'Automatic Trip of Reactor Coolant Pumps'" (Reference 6). NRC has established that the methods employed by the WOG to justify manual reactor coolant pump (RCP) trip are consistent with the guidelines in Generic Letters 83-10c and d. The NRC, however, requires plant specific information regarding implementation of the WOG methodology prior to final resolution of Item II.K.3.5. The following information is provided in response to Section IV of Generic Letter 85-12:

A. Determination of RCP Trip Criteria

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

<u>FPL Response</u>

Calculations in accordance with the methodology provided in Reference 7 were performed to determine which of the three alternative RCP trip criteria is most appropriate in providing pump trip discrimination between Loss of Coolant Accidents (LOCAs) and Steam Generator Tube Rupture (SGTR) or non-LOCA events.

Of the three RCP trip criteria used by Westinghouse; RCS pressure, RCS subcooling and Secondary pressure dependent RCS pressure $P_{RCS} - P_{steamline}$); only RCS pressure did not meet the WOG's acceptance criteria for discrimination between LOCA and SGTR or non-LOCA events. Because the secondary pressure dependent RCS pressure trip parameter requires the reactor operator to look at two instruments (RCS pressure and Steam Generator pressure) and RCS subcooling only requires the operator to look at one instrument (subcooled margin monitor), RCS subcooling was chosen as the desired parameter for pump trip.

The subcooled margin monitor (SMM) is a fully redundant, qualified system as required by TMI Action Item II.F.2 "Instrumentation for Detection of Inadequate Core Cooling." The SMM takes input from redundant hot leg pressure transmitters and temperature elements. These inputs are then used in a calculational program that determines RCS subcooling which is then displayed in the control room.

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 conditions such as fluid jets or pipe whip which might influence the instrumentation reliability.

<u>FPL Response</u>

The instrument error for use of RCS subcooling was calculated for Turkey Point for both normal and adverse containment conditions. The minimum RCS pressure calculated for SGTRs and non-LOCAs is approximately 1135 psig. The subcooling uncertainty for normal containment conditions at this pressure is 22.3°F. At Residual Heat Removal (RHR) system pressure, 450 psig, the subcooling uncertainty is 25.5°F. Therefore, the trip setpoint for the RCP's under normal containment conditions is less than 25.5°F subcooling.

Under adverse containment conditions, the RCP trip setpoint at the non-LOCA SGTR lower pressure limit of 1135 psig was determined to be subcooling 65°F.

The instrument uncertainties used by Westinghouse consider uncertainties from the transmitter or temperature sensor, through the electronics to the display itself. Separately, Westinghouse has determined at what point in an event the adverse containment conditions setpoint must be used. While the temperature sensors associated with the SMM are not sensitive to containment conditions, the RCS pressure transmitters exhibit higher uncertainty under adverse containment conditions. The permissive for using the adverse containment setpoint is either 180°F containment temperature or $1.3 \times 10^5$ R/hr.

The design of the subcooled margin monitor (SMM) software is such that invalid or failed instrument inputs, such as might be caused by pipe whip, are not used. The arrangement of instrumentation at Turkey Point precludes failure of the SMM due to pipe whip or single failure.

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.

<u>FPL Response</u>

The following response to this question was provided by the Westinghouse Owner's Group:

"The LOFTRAN computer code was used to perform the alternate

RCP trip criteria analyses. Both Steam Generator Tube Rupture (SGTR) and non-LOCA event were simulated in these analyses. Results for 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 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 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 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 shows 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 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 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 Turkey Point units is +1 to +5 °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.

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

<u>FPL Response</u>

a. The RCP seals at Turkey Point are provided with redundant means of cooling, seal injection via the charging system and thermal barrier cooling via the Component Cooling Water (CCW) system. While both systems are normally operating, either is sufficient to provide adequate seal cooling for up to 24 hours.

   For an event which occurs causing a safety injection signal with off site power available (i.e., RCPs remain running), thermal barrier cooling will continue so long as the High-High containment pressure signal setpoint of 20 PSIG is not reached. Seal injection will be lost on an S.I. signal (charging pumps tripped on S.I.). On High-High containment pressure, Phase B containment isolation is initiated and thermal barrier cooling will be automatically isolated. If seal injection is re-established, operating procedures permit continued RCP operation until upper or lower motor bearing temperatures reach 200°F. Upon reaching 200°F, the RCPs are manually stopped. If neither CCW or Seal Injection are available, the RCPs will be tripped.

b. The RCPs at Turkey Point are capable of operating for several minutes without cooling water with no significant seal degradation. The RCPs must, however, be shutdown when the RCP motor bearing temperature reaches 200°F. The RCP seals will not leak excessively even if cooling water is lost for an extended period of time after the RCPs are tripped (Reference 8).

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.

FPL Response

Manual trip of an RCP motor requires the availability of 125V DC power, the motor control switch, and the motor breaker. This provides a reliable means of tripping the RCP. With the exception of the motor and cabling, all the components associated with the RCP motor are outside containment. Therefore, adverse environmental conditions will not prevent RCP trip when required.

References


1) NUREG-0660 "TMI Action Plan"

2) NUREG-0737 "Clarification of TMI Action Plan requirements," dated November 1980.

3) Westinghouse Owners Group Letter OG-110, "Evaluation of Alternate RCP Trip Criteria," December 1, 1983.

4) Westinghouse Owners Group Letter OG-117 "Justification of Manual RCP Trip for SBLOCA Events," March 9, 1984.

5) Westinghouse Owners Group Letter OG-137 "Response to NRC Question on RCP Trip," October 25, 1984.

6) NRC Generic Letter 85-12 "Implementation of TMI Item II.K.3.5 'Automatic Trip of Reactor Coolant Pumps,'" dated June 28, 1985.

7) Westinghouse Emergency Response Guidelines, Rev. 1, Executive Summary "Generic Issue - RCP Trip/Restart," dated September 1, 1983.

8) WCAP 10541 "Reactor Coolant Pump Seal Performance Following a Loss of All AC Power," dated April 1984.

C. Operator Training and Procedures (RCP Trip)

1. Describe the operatory training program for RCP trip. Include the general philosophy regarding the need to trip pumpts verusus the desire to keep pumps running.

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

(a) RCP trip using WOG alternate criteria
(b) RCP restart
(c) Decay heat removal by natural circulation
(d) Primary system void removal
(e) Use of steam generators with and without RCPs operating
(f) RCP trip for other reasons

FPL Response

Plant operators are trained to recognize the normal range of the parameters monitored and to trip the RCPs if appropriate. For these events the operators are trained that the RCP trip may be initiated manually by the operator or by any of a number of RCP protective trips.

RCP trip criteria have been developed and incorporated into the Updated Emergency Operations Procedures that were trained on by all licensed operators during the requalification simulator training in 1985 conducted by Westinghouse at the Westinghouse Simulator in Pittsburgh, PA.

RCP trip criteria provides for RCP trip when required for SBLOCAs and to minimize the probability of RCP trip when not required.

It is desirable to keep the RCPs running during a steam generator tube rupture (SGTR) and other non LOCAs to 1) maintain normal pressure control using pressurizer spray and thereby avoiding opening of the pressurizer PORVs, 2) prevent the formation of a stagnant water volume in the upper head region which may flash and form a steam bubble during subsequent cooldown and depressurization, 3) minimize potential pressurized thermal shock challenges and 4) minimize operator action such as tripping the RCPs and then restarting them later.

Those procedures that include RCP trip related operations are as follows:

a. RCP trip using WOG alternate criteria

1) E-0 - Reactor Trip or Safety Injection
2) ES-0.4 - Natural Circulation Cooldown with Steam Void in Vessel Without RVLMS (QSPDS)
3) E-3 - Steam Generator Tube Rupture
4) ECA-2.1 - Uncontrolled Depressurization of all Steam Generators
5) E-1 - Loss of Reactor or Secondary Coolant.

b. RCP restart

1) ES-0.2 - Natural Circulation Cooldown
2) ES-0.3 - Natural Circulation Cooldown with Steam Void in Vessel with RVLMS (QSPDS)
3) ES-0.4 - Natural Circulation Cooldown with Steam Void in Vessel (Without RVLMS)
4) E-3 - Steam Generator Tube Rupture
5) ES-1.1 - SI Termination
6) ES-1.2 - Post LOCA Cooldown and Depressurization
7) ECA-2.1 - Uncontrolled Depressurization of all Steam Generators
8) ECA-3.1 - SGTR With Loss of Reactor Coolant - Sub-Cooled Recovery Desired
9) ECA-3.2 - SGTR With Loss of Reactor Coolant - Saturated Recovery Desired
10) ECA-3.3 - SGTR Without Pressurizer Pressure Control
11) FR-P.1 - Response to Imminent Pressurized Thermal Shock Condition
12) FR-I.3 - Response to Voids in Reactor Vessel

c. Decay heat removal by natural circulation

1) ES-0.2 - Natural Circulation Cooldown
2) ES-0.3 - Natural Circulation Cooldown with Steam Void in Vessel (With RVLMS)
3) ES-0.4 - Natural Circulation Cooldown with Steam Void in Vessel (Without RVLMS)

d. Primary system void removal
1) FR-I.3 - Response to Voids in Reactor Vessel

e. Use of steam generators with and without RCPs operating
1) See above listed procedures

f. RCP trip for other reasons

1) ONOP-1108.1 - Reactor Coolant Pump - Off-Normal Conditions
2) ECA-1.1 - Loss of Emergency Coolant Recirculation