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SUBJECT: Forwards plant-specific info re TMI Action Item II.K.3.5 per Section IV of SER issued w/Generic Ltr 85-12.Reactor coolant subcooling methodology will be used to determine reactor coolant pump setpoint.

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Mr. Hugh L. Thompson, Jr. Director
Division of Licensing
United States Nuclear Regulatory Commission
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23
RESPONSE TO TMI ACTION ITEM II.K.3.5
GENERIC LETTER 85-12

Dear Mr. Thompson:

Carolina Power & Light Company hereby submits information in response to TMI Action Item II.K.3.5. The enclosure provides the plant specific information requested in Section IV of the Safety Evaluation Report issued with Generic Letter 85-12.

Generic Letter 85-12 reiterates many of the requests and suggestions made in Generic Letter 83-10d. Carolina Power & Light Company has already provided much of the requested information regarding H. B. Robinson, Unit No. 2 in our responses to Generic Letter 83-10d. We have supplemented that information where requests were made in Generic Letter 85-12.

This response is submitted in accordance with the schedule agreed upon with your HBR2 project manager.

Yours very truly,

S. R. Zimmerman
Manager
Nuclear Licensing Section

SRZ/SDC/ccc (1843SDC)

Enclosure

cc: Dr. J. Nelson Grace (NRC-RII)
Mr. G. Requa (NRC)
Mr. H. Krug (NRC Resident Inspector - RNP)

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ATTACHMENT 1

Item 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 to A.1:

As indicated in our January 16, 1984 letter, Carolina Power & Light Company (CP&L) has chosen to use the reactor coolant subcooling methodology. Detailed information concerning the core cooling monitors at H. B. Robinson - Unit 2 (HBR2) was included in our letters dated December 31, 1979, March 31, 1981, and April 26, 1983.

A core subcooling monitor which provides a continuous on-line indication of the primary coolant saturation condition has been installed at HBR2. The heart of this system is a microprocessor which receives inputs from 4 primary system pressure transmitters, 6 loop RTDs, and 16 core exit thermocouples and outputs a margin to saturation in degrees fahrenheit subcooling or superheat.

The subcooling monitor possesses the required redundancy in that it is comprised of two channels which operate completely independently. Each channel is powered from a vital instrument bus which receives its power from off site or the emergency diesels. In addition, the warning lights, alarms, meter movements, and associated electronics are testable through use of front panel test switches.

The core subcooling monitor makes use of redundant control grade temperature inputs from each hot and cold leg RTD. Since the RTDs are control grade, they are backed up by multiple core exit thermocouples. In addition, each channel is provided with three pressure signals, one narrow range safety grade pressure and two wide-range control grade pressures. All safety grade sensors are isolated from the subcooling monitor by isolation amplifiers.

CORE SUBCOOLING MONITOR INPUTS

Temperature	RTD -	2 hot leg per channel (1 dedicated, 1 shared)
	-	2 cold leg per channel (1 dedicated, 1 shared)
Temperature	T/C -	8 per channel
Pressure Transmitters	-	2 wide-range loop (shared)
		2 narrow-range pressurizer

Item A (Continued):

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.

Response to A.2:

Calculation of instrument uncertainties for normal and adverse containment conditions is documented in the HBR Setpoint Study. The study shows that the instrument uncertainties associated with determining subcooling margin using RTDs and pressure indication are 25°F for normal containment conditions and 35°F for adverse containment conditions. The HBR Setpoint Study was performed to support the plant specific implementation of Revision 1 of the Westinghouse Owners' Group (WOG) Emergency Response Guidelines (ERGs) (for low pressure Safety Injection (SI) plants). The methodology and baseline conditions for performing each setpoint determination is contained in the executive volume and background volumes for the WOG ERG's.

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.

   If a licensee determines that the WOG alternative criteria are marginal for preventing unneeded RCP trip, it is recommended that a more discriminating plant-specific procedure be developed. For example, use of the NRC-required inadequate core-cooling instrumentation may be useful to indicate the need for RCP trip. Licensees should take credit for all equipment (instrumentation) available to the operators for which the licensee has sufficient confidence that it will be operable during the expected conditions.

Response to A.3:

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 programs results, assuming no changes in the initial plant conditions (i.e., full power, pressurizer level, all SI and Auxiliary Feedwater (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:

- Break Flow
- SI Flow

- Decay Heat
- Auxiliary Feedwater Flow

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

BREAK FLOW - To conservatively simulate a double ended tube rupture in safety analyses, the break flow model used in LOFTRAN includes substantial amounts 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.

SI FLOW - 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%.

DECAY HEAT - 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.

AUXILIARY FEEDWATER - 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 the 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 H. B. Robinson-2 unit is +1°F 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.

Item B:

Potential Reactor Coolant Pump Problems

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

Response to B.1:

At HBR2 there are two levels or degrees of containment isolation, Phase A and B. Phase A containment isolation has no impact on RCP operation since none of the essential services to the RCP's are lost. Phase B containment isolation does result in isolation of some of the essential services supporting RCP operation. Phase B isolations are initiated either by automatic or manual initiation of containment spray.

As stated in our April 22, 1983 response, the current procedures at HBR2 require tripping the RCPs if essential services are lost (regardless of the cause of isolation, unless RCP operation is required to prevent core damage). In order to restart the RCP's current procedures require the verification that essential services are available.

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 to B.2

Components required for RCP tripping:

- Reactor Turbine Generator Board (RTGB) Switch
- Control Power for Loop 1 and 3 RCPs (Station Battery "A")
- Control Power for Loop 2 RCP (Station Battery "B")
- Trip Coil and Auxiliary Contacts on 4160V Switch Gear Breaker

All components necessary to trip the RCPs are located outside the Reactor Containment Building and, thus, are not affected by adverse containment conditions.

Item C:

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 to C.1:

The HBR2 operators have been trained to trip the RCPs as soon as RCS subcooling reaches 25°F (35°F for adverse containment conditions) during a depressurization event unless it is a planned and controlled depressurization during the longer term recovery actions. This training was accomplished during the implementation training for the new EOP's conducted in 1984. This training is reinforced during regularly scheduled simulator retraining.

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

Response to C.2:

All of the above RCP trip related operations are addressed in the Robinson Plant Specific Emergency Operating Procedures (EOPs). These EOPs are based on the Westinghouse Owners' Group Emergency Recovery Guidelines (ERGs), Revision 1. The ERGs have been submitted to the NRC. In addition, normal RCP operation and less serious abnormal conditions are addressed in the following plant procedures.

OP-101, Reactor Coolant System and Reactor Coolant Pump Start-up and Operation

GP-001, Fill and Vent of the Reactor Coolant System

AOP-018, Reactor Coolant Pump Abnormal Conditions

The EOPs along with the procedures listed above are available for review on site.

(1843SDC/ccc)