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SUBJECT: Forwards info requested in Section IV of NRC safety evaluation re B&W Ownwers Group submittals on TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," per Generic Ltr 86-05.

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October 20, 1986

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

ATTENTION: Mr. J.F. Stolz, Project Director
PWR Project Directorate No. 6

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

Dear Mr. Denton:

By letter dated May 29, 1986, the NRC transmitted Generic Letter 86-05 to all licensees with Babcock & Wilcox (B&W) designed Nuclear Steam Supply Systems. Enclosed with the Generic Letter was the NRC's Safety Evaluation (SE) regarding the B&W Owners Group (B&WOG) Submittals on TMI Action Item II.K.3.5, "Automatic trip of reactor coolant pumps". By my letter dated July 21, 1986, I had advised you of Duke Power's intention to endorse the B&WOG methodology and that the information requested in Section IV of the NRC Staff's SE would be submitted by October 20, 1986. Accordingly, please find attached Duke's response.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

PFG/11/jgm

Attachment

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DUKE POWER COMPANY

OCONEE NUCLEAR STATION

RESPONSE TO GENERIC LETTER 86-05

A. Determination of RCP Trip Criteria

Item 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: The instrumentation used at Oconee to determine the RCS subcooled margin (SCM) is the Operator Aid Computer (OAC). Two Motorola 56 PH pressure transmitters (soon to be replaced by two Rosemount 1153GD9 pressure transmitters) located inside containment provide this system with the pressure signals. Temperature inputs come from two wide range RTDs (one per loop) located on the hot legs and twenty-four environmentally qualified thermocouples located at the core exit. A program within the OAC determines and uses the average temperature of the five highest thermocouples out of all twenty-four. Three SCMs are displayed to the operators - core exit, loop A, and loop B subcooled margins. Since the reactor coolant pumps will be on prior to RCP trip, the operators will primarily use the loop SCMs to determine the thermodynamic state of the reactor coolant system.

To further add reliability and precision in determining the subcooled margin, a safety related Inadequate Core Cooling (ICC) Monitoring System will be added to each Oconee unit. This system will be composed of two trains which are physically separate and are electrically isolated from one another. They will perform the same function using similar and redundant inputs. The pressure transmitters associated with this system will be located outside of containment so that adverse containment conditions will not effect them. The RTDs and core exit thermocouples used for the OAC will also be used as temperature inputs to the ICC. The only difference is that the ICC will determine and use the average of the five highest thermocouples out of the twelve designated for each train.

Item 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: The following table lists the instrumentation uncertainties associated with the OAC SCM for both normal and adverse containment conditions.

Instrumentation	Range	Containment Conditions Normal	Containment Conditions Adverse
Motorola 56 PH Transmitters	0-2500 psi	120 psi	225 psi
Rosemount 1153GD9 Transmitters	0-2500 psi	65 psi	250 psi
Hot Leg RTDs	50-650°F	15°F	15°F
Core Exit Thermocouples	50-2300°F	15°F	15°F

The adverse containment parameters are selected when containment pressure reaches 3 psig. This is a good indication that containment temperature is greater than 175°F, including the error associated with an uneven temperature distribution existing in containment. Based on this, the OAC automatically selects the proper polynomial coefficients used in the subcooled margin curves. Figure 1 shows the resulting subcooled margin curves. The difference between the saturation curve and the SCM curves represents the overall SCM uncertainty.

Fluid jets or pipe whip will not effect the ability of an operator to determine the subcooled margin in the RCS. The redundancy associated with the instrumentation used to calculate the subcooled margin ensures that the true state of the reactor coolant can be determined at any time. Operators will be able to detect and disregard any anomalous indications caused by fluid jets or pipe whip.

Item 3: In addressing the selection of the criterion, consideration of uncertainties associated with the BWOG or plant specific 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 BWOG generic data group.

Response: As stated in a Duke letter dated March 30, 1984, both the BWOG and Duke Power analyses are conservative with respect to Oconee Nuclear Station. The BWOG generic analyses assumes an initial core power level of 2772 MW, which is greater than the Oconee full power rating of 2568 MW. The lower steady-state power at Oconee results in lower decay heat power levels and less severe consequences from a loss of primary coolant. The Oconee HPI capacity is also significantly greater than the flow assumed in the generic analysis. The additional injection flow would provide quicker recovering of the core following a delayed RCP trip. Furthermore, the radial and axial power peaking assumed in the generic analysis is much worse than the peaking seen in normal operation at Oconee. Less severe peaking would result in lower cladding temperatures following delayed reactor coolant pump trip.

The choice of assumptions used in the Duke Power analysis of a steam generator tube rupture (using the RETRAN02-MOD002 transient analysis computer code) provide significant conservatisms. The 435 gpm initial leak rate used in the analysis is the FSAR value, but the actual initial leak rate from a double-ended rupture is calculated to be less than 325 gpm. No credit was taken for makeup flow even though the automatic makeup system would provide at least 160 gpm to mitigate the loss of primary coolant as the pressurizer level dropped. The Engineered Safeguards (ES) actuation signal on low RCS pressure used i the analysis was 1500 psig. The actual plant ES setpoint is 1600 psig, which would result in a quicker system refill. Additionally, flow from three HPI pumps rather than the one assumed in this analysis would be available to compensate for the tube leakage and refill the RCS. Furthermore, the actual delay time for HPI flow following the receipt of the low pressure ES signal would be less than ten seconds rather than the assumed 35 seconds. Most importantly, the assumption of no operator action is highly pessimistic. The symptoms of a tube rupture are clear and unambiguous; the operators would certainly take prompt action to increase injection and shutdown the reactor, thus avoiding any possibility of a reactor coolant pump trip.

Uncertainties associated with the computer program calculations do not have a significant impact on the conclusions of the analyses. Both computer programs (CRAFT2 for the small break LOCA analysis and RETRAN02-MOD002 for the steam generator tube rupture analysis) have been reviewed by the NRC and approved for these applications. All transient thermal-hydraulic simulation programs have some uncertainty associated with them due to approximations and assumptions inherent in the equations used to model physical processes. In addition there are small inaccuracies introduced by the solution of those equations. However, experience indicates that these uncertainties are insignificant relative to the conservatism of the assumptions of the analyses, especially inlight of the margin demonstrated by the analysis results.

B. Potential Reactor Coolant Pump Problems

Section 5.4 of the BWOG generic report discusses the various aspects of the essential service water systems for the B&W plants in a generic fashion. Each licensee needs to identify and describe the plant specific features to:

Item 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, then they can be restored fast enough to prevent seal damage or failure once a non-LOCA situation is confirmed.

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

Response: a) Seal damage is prevented by maintaining or restoring either seal injection (HPI) or component cooling (CC) to the pump thermal barrier. As discussed in the next response, RCP seal cooling capability is always maintained.

b) For moderate to severe overcooling transients, non-essential Reactor Building isolation may occur on low RCS pressure (less than 1600 psig). This signal isolates only seal leakoff so that adequate water services are still provided to the RCPs. A high energy line break inside containment will actuate essential and non-essential Reactor Building isolation on high containment pressure (greater than 3 psig). This will result in the isolation of CC and low pressure service water (LPSW) flow, but seal injection will still be available to prevent any seal damage from occurring. It is expected that five to ten minutes are available to restore LPSW to the pump motor before a temperature limit is reached. This is the time available for operator action and is considered adequate.

Item 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 manually trip the RCPs, the following devices must be operable:

1) Control switch (located in control room)
2) 6900 volt switchgear breakers
3) Breaker trip coils

A RCP trip will occur when necessary since all of the components required to trip the RCPs are located in a mild environment.

C. <u>Operator Training and Procedures (RCP Trip)</u>

Item: In response to NRC questions concerning the identification and management of primary system voids, the BWOG response identified potential changes to the ATOG procedures to incorporate proposed detection and management schemes.

Each licensee should endorse this program as described, and provide an implementation schedule for the revised ATOG.

If a licensee does not endorse the provided proposal, then a suitable alternate proposal must be provided including an implementation schedule.

Response: In September of 1985, revised emergency procedures were implemented at Oconee which incorporate void identification and management directions that are consistent with the BWOG program. Void identification is currently based on indirect indications such as sudden and/or unexplained fluctuations in pressurizer level and difficulties in controlling and depressurizing the RCS. Void mitigation strategies include utilization of the reactor vessel and/or hot leg high point vents and RC Pump bumps. This guidance will be enhanced by use of the Reactor Coolant Inventory Monitoring System (RCIMS) which is presently being implemented at Oconee. Operator training will continue to emphasize the detection and mitigation of voids including the utilization of RCIMS for this purpose. It is stated in Generic Letter 86-05 that the reactor vessel head level measurement, when installed, is not to be used while venting. At Oconee the upper level tap does not come off of the head vent, so the indication can be used while venting.

The implementation schedule for the RCIMS is outlined in section 2.7 of Duke Power's Final Design Description - Instrumentation to Detect Inadequate Core Cooling, submitted on July 1, 1985.

Figure 1 Subcooled Margin Curves