March 15, 1988

Docket Nos. 50-269
50-270
50-287

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

SUBJECT: AUTOMATIC TRIP OF REACTOR COOLANT PUMPS (NUREG-0737, II.K.3.5)

Re: Oconee Nuclear Station, Units 1, 2 and 3

We have reviewed your responses to Generic Letter 86-05 and to NUREG-0737, Item II.K.3.5 on automatic tripping of the reactor coolant pumps. We were assisted in our review by our consultants, EG&G, who prepared a Technical Evaluation Report (Enclosure 2). In our safety evaluation (Enclosure 1), we have concluded that you have satisfied the requirements of TMI Action Plan Item II.K.3.5. Therefore, we consider this issue complete.

Sincerely,

Helen N. Pastis, Project Manager
Project Directorate II-3
Division of Reactor Projects I/II

Enclosures:
As stated

cc:
See next page

DISTRIBUTION:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATING TO IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5 "AUTOMATIC TRIP OF REACTOR COOLANT PUMPS" (RESPONSE TO GENERIC LETTER NO. 86-05) DUKE POWER COMPANY OCONEE NUCLEAR STATION UNITS 1, 2, AND 3 DOCKET NOS. 50-269, 50-270, 50-287

1.0 SUMMARY

In Generic Letter 86-05 (Ref. 1) we reported that the information provided by the Babcock and Wilcox Owners Group (BWOG) in support of alternative Reactor Coolant Pump (RCP) trip criteria was acceptable on a generic basis. This review noted that a number of considerations were assigned plant specific status. Accordingly, we requested that operating reactor licensees select and implement an appropriate RCP trip criterion based upon the BWOG methodology. This Safety Evaluation Report (SER) contains the staff's findings concerning this issue for Duke Power Company's Oconee Nuclear Station Units 1, 2 and 3.

Reference 1 required owners of B&W Nuclear Steam Generating Systems to evaluate their plants with respect to RCP trip. The objective was to demonstrate that their proposed RCP trip. The objective was to demonstrate that their proposed RCP trip setpoints assure pump trip for small break LOCAs, and in addition to provide reasonable assurance that RCPs are not tripped unnecessarily during non-LOCA events. A number of plant specific items were identified which were to be considered by applicants and licensees, including the selected RCP trip parameter, instrumentation quality and redundancy, instrumentation uncertainty, possible adverse environments, calculational uncertainty, potential RCP and RCP associated problems, operator training, and operating procedures.

The licensee has addressed the Generic Letter 86-05 criteria and we have reviewed this information with assistance from consultants at EG&G. We find the material submitted by the licensee to be acceptable and find that the licensee has satisfied the requirements in regard to RMI Action Item II.K.3.5.

2.0 BACKGROUND

TMI Action Plant Item II.K.3.5 of NUREG-0737 (Ref. 2) required all licensees to consider solutions pertinent to tripping RCPs under transient and Loss of Coolant Accident (LOCA) conditions. A summary of the industry and NRC programs concerning RCP trip was provided in SECY-82-475 (Ref. 3). Reference 3 also provided NRC guidance and criteria for resolution of II.K.3.5, and enclosed Generic Letters 83-10 (Ref. 4), which outlined requirements pertinent to RCP trip.

The B&W Owners Group responded to Reference 4 by developing RCP trip criteria based upon a loss of subcooling margin and provided information which individual utilities could use for plant-specific implementation (ref. 5). The staff then issued Generic Letter 86-05, and directed each applicant and licensee to provide RCP trip criteria and substantiating information.

The licensee addressed this issue in Reference 6, which we have reviewed with the assistance of EG&G consultants. Enclosure 3 is the technical evaluation report (TER) prepared by EG&G. We have reviewed their recommendations and concur that the licensee's submittal meets the requirement of Item II.K.3.5.

3.0 EVALUATION

As discussed in detailed in the attached TER, the licensee has satisfied the requirements of GL 86-05.

The staff finds that Duke Power Company has complied with the requirements of Generic Letter 86-05 and that they have therefore met the requirements in regard to implementation of TMI Action Item II.K.3.5.

These requirements include:

A. Determination of RCP Trip Criteria
The reactor coolant pump will be tripped if the indicated subcooling margin falls to 0° F considering instrument uncertainties. This agrees with B&W Owners Group guidelines and hence is acceptable.

A1. Instrumentation Identification including Redundancy and Quality Level
The licensee has identified all of the essential instrumentation and degree of redundancy used to trip the RCPs. We conclude these are acceptable.

A2. Instrumentation Uncertainties for Normal and Adverse Environments
The licensee has demonstrated that the instrument uncertainties do not exceed the allowable limits specified by NRC. We conclude these uncertainties are acceptable.

A3. Analysis Uncertainties
The licensee has demonstrated that the results of the BWOG generic analyses are conservative for Oconee Units 1, 2, and 3. Therefore we consider these acceptable.

B. Potential Reactor Coolant Pump Problems

B1. Containment Isolation Impact Upon RCP Operation
The licensee has demonstrated that the RCP water services can be restored following containment isolation. We consider this acceptable.

B2. Components Required for RCP Trip

Manual trip of the RCPs is accomplished from the main control room console. No critical components required to initiate manual RCP trip are located in harsh environment. We conclude that the licensee has provided an acceptable response to this item.

C. Operator Training and Procedures

The licensee has provided adequate operator training and procedures, which are consistent with the NRC staff guidelines. We thus conclude these are acceptable.

4.0 CONCLUSION

Each of the point identified in Reference 1 had been satisfactorily addressed by the licensee. Further, the licensee has considered items pertinent to RCP trip and operation which are in addition to the Reference 1 requirements. The staff finds the licensee treatment of RCP trip to be acceptable and the licensee has satisfied the requirements of TMI Action Item II.K.3.5.

5.0 REFERENCES

1. Miraglia, Frank J., "Implementation of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps' (Generic Letter No. 86-05)," letter from Director, Division of PWR Licensing-B, NRC, to all applicants and licensees with Babcock and Wilcox (B&W) designed nuclear steam supply systems (NSSSs), May 29, 1986.

2. "Clarification of TMI Action Plan Requirements," NUREG-0737, US NRC, November 1980.

3. Dircks, William J., "Staff Resolution of the Reactor Coolant Pump Trip Issue," Policy Issue for the Commissioners from the Executive Director for Operations, NRC, SECY-82-475, November 30, 1982.

4. Eisenhut, Darrell G., "Resolution of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps,' (Generic Letter No. 83-10)," letter to all applicants with (PWR vendor) designed nuclear steam supply systems from Director, Division of Licensing, NRC.

5. R. H. Bryan, "Reactor Coolant Pump Trip Philosophy," letter from R. H. Bryan, Chairman, B&W Owners Group Analysis Committee to J. R. Miller, Chief, Operating Reactor Branch No. 3, USNRC, June 18, 1984.

6. H. B. Tucker, Duke Power Company, to J. F. Stolz, Project Director, PWR Project Directorate 6, USNRC, "Implementation of TMI Action Item II.K.3.5," October 20, 1986.

Principal Contributor: S. L. Wu

EG&G IDAHO EVALUATION OF RESPONSE TO GENERIC LETTER 86-05, "IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5" DUKE POWER COMPANY OCONEE NUCLEAR STATION UNITS 1, 2, AND 3 DOCKET NOS. 50-269, -270, -287

1. INTRODUCTION

TMI Action Plan Item II.K.3.5 of NUREG-0737 required all licensees to consider other solutions to the small break loss of coolant accident (LOCA) problems, because tripping the reactor coolant pumps (RCPs) was not considered to be the ideal solution. NRC Report SECY-82-475[1] summarized the industry programs and the NRC programs concerning RCP trip. In Generic Letter 86-05[2] the staff accepted the Babcock and Wilcox Owner's Group (BWOG) criterion[3] for tripping the reactor coolant pumps during small break LOCAs which minimizes RCP trip for steam generator tube rupture (SGTR) and non-LOCA events.

The BWOG uses a loss of subcooling margin as the basis for RCP trip. The Abnormal Transient Operating Guidelines (ATOG) for the identification of an SGTR and for the mitigation of the SGTR event are considered to be an integral part of the RCP trip criterion. When the plant operator follows the ATOG procedures for an SGTR event, subcooling is maintained and the RCP trip will not be required. The BWOG analyses concluded that the trip signal and run-all/trip-all RCP philosophy were as adequate as partial trip schemes for maintaining acceptable fuel temperatures and were superior for ensuring pump integrity, pump availability, and minimizing operator error.

The generic information presented by the BWOG, however, does not address plant specific concerns about instrumentation uncertainties, potential reactor coolant pump problems and operator training and procedures as requested in Generic Letter 83-10. This information, specifically identified in Section IV of Generic Letter 86-05, was requested from each B&W licensee to enable the staff to assess implementation of the RCP trip criterion.

2. DISCUSSION

The Oconee response to Generic Letter 86-05 Section IV was provided in a letter dated October 20, 1986[4] and was supplemented by additional referenced material.[5,6,7,8] EG&G Idaho personnel reviewed the licensee material in detail to verify that the required information was provided. In those cases where the licensee's response was not clear, the referenced information was studied to enhance EG&G's understanding. Briefly, the licensee does endorse the BWOG methodology to use subcooling margin as the basis for RCP trip. Following is a summary of the licensee's responses to Generic Letter 86-05 and EG&G's basis for acceptance.

2.1 GL 86-05, Item A.1 - Reactor Coolant Pump Trip Criteria

The NRC requested the licensee to identify the instrumentation used to determine the RCP trip setpoint, including the degree of redundancy.

Oconee Response:

The licensee uses subcooling margin (SCM) as the criteria for assessing the need to trip the reactor coolant pumps. A polynomial expression for the RCS subcooling has been programmed into the Operator Aid Computer (OAC). The coefficients for the polynomial are based on the true saturation curve plus margin for instrument uncertainties. The OAC monitors the RCS and containment conditions and compares the data with the appropriate SCM curve. The operator is instructed to trip all RCPs if the indicated subcooling margin falls to 0°F.

Pressure inputs to the OAC are provided by two Rosemount 1153 GD9 pressure transmitters located inside containment. Temperature inputs come from two wide range RTDs (one per loop) located on the hot legs and 24 environmentally qualified thermocouples located at the core exit. Three SCMs are displayed to the operator--core exit, loop A, and loop B subcooled margins. The operator is instructed to use the most conservative indication of subcooling margin.

To increase reliability and precision in determining the subcooling margin, each Oconee unit will add a safety-related Inadequate Core Cooling (ICC) Monitoring System. This system will be composed of two trains, which are physically separate and are electrically isolated from each other. Each train will perform the same function using similar and redundant inputs.

EG&G Idaho Evaluation:

In their response[8] the licensee indicated that replacement of the Motorola 56 pH pressure transmitters with Rosemount units was complete. The Motorola units were installed as original plant

equipment and were due for replacement. The licensee verified that the Rosemount instrument uncertainties are less than or equal to the Motorola transmitters. Consequently, the existing polynomial coefficients for the subcooling curves are conservative and need not be adjusted. This is acceptable.

We have compared the licensee material with the information contained in the FSAR and in previous letters of correspondence with NRC. We find that the licensee has identified all of the essential instrumentation and degree of redundancy used to trip the RCPs. In addition, the licensee has cited the applicability of the generic Abnormal Transient Operating Guidelines (ATOG),[5] which has been reviewed and accepted by the NRC. We consider that the existing methods of measuring subcooled margin will provide an acceptable degree of redundancy for the purpose of assessing the RCP trip criteria. The licensee's response to Item A.1 is acceptable.

2.2 GL 86-05, Item A.2 - Instrumentation and Environment

The NRC requested the licensee to identify instrumentation uncertainties, adverse containment conditions, and the effects of localized factors (e.g., fluid jets or pipe whip) on instrument reliability.

Oconee Response:

The licensee submitted a table[4] listing the instrumentation uncertainties associated with the OAC SCM for normal and adverse containment conditions. The pressure transmitters have an operating range of 0 to 2500 psi with uncertainties of up to 120 psi (normal) and 225 psi (adverse). The RTDs and thermocouples have ranges 50-650°F and 50-2300°F respectively with a 15°F uncertainty under normal and adverse containment conditions. The adverse containment polynomial coefficients are selected by the OAC when the containment pressure reaches 3 psig (indicating containment temperature is greater than 175°F). The difference between the saturation curve and the subcooled margin curves represents the overall SCM uncertainty.

The licensee has considered the effects of localized factors and has concluded that the ability of the operator to determine the RCS subcooled margin can be maintained. 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. The operators will be able to detect and disregard any anomalous indications caused by fluid jets or pipe whip.

EG&G Idaho Evaluation:

The licensee stated [4] that the difference between the 0°F subcooled margin curves and the saturation curve represents the overall uncertainty. However, in Figure 1 the difference between the curves appeared to be smaller than the instrument uncertainties listed in Table 1.

In their response[8] the licensee explained that the pressure and temperature uncertainties were converted into degrees °F and were combined in a "square root of the sum of the squares (SRSS)" calculation. This SRSS term was then added to the saturation curve to develop the subcooled margin curves. Although, an algebraic sum of the individual uncertainties would have yielded a larger (and more conservative) margin, it is very unlikely that the maximum pressure and temperature uncertainties occur at the same time. We find the licensee's explanation to be acceptable since the SRSS method is a recognized industry practice.

We find that the licensee has demonstrated that the instrument uncertainties (normal and adverse containment conditions) do not exceed the specified subcooling margin. The licensee has identified all of the instrumentation and degree of redundancy used to trip the RCPs. The ability of the operator to determine the RCS subcooled margin will not be impaired by the effects of localized factors such as fluid jets or pipe whip. In addition, the licensee has committed to act in accordance with the Oconee and BWOG procedures and to use the most conservative indication of loss of subcooled margin. This is acceptable.

2.3 GL 86-05, Item A.3 - Generic and Plant Specific Analyses

The NRC requested the licensee to identify uncertainties associated with the BWOG analyses and atypical plant specific features.

Oconee Response:

The BWOG generic methodology has been endorsed by Oconee in letters dated July 21, 1986[5] and October 20, 1986.[4] The licensee pointed out that the BWOG generic analyses are conservative for several reasons:

1. The BWOG generic analyses assumed an initial core power level of 2772 MWT, which is greater than the Oconee full power rating of 2568 MWT. The lower steady state power at Oconee results in a lower decay heat power levels and less severe consequences from a loss of primary coolant.

2. The Oconee high pressure injection (HPI) capacity is much greater than the flow assumed in the BWOG analyses. The additional injection flow provides faster recovery of the core following a delayed RCP trip.

3. 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.

In addition, the licensee pointed out several areas of conservatism that were used in the Duke Power analyses of a steam generator tube rupture:

1. The 435 gpm initial leak rate used in the analyses is the FSAR value, but the actual initial leak rate from a double-ended rupture is calculated to be less than 325 gpm.

2. No credit was taken for makeup flow even though the automatic makeup system would provide at least 150 gpm to mitigate the loss of primary coolant as the pressurizer level dropped.

3. The analysis assumed that actuation of the Engineered Safeguards (ES) signal occurred at 1500 psig. The actual ES setpoint is 1600 psig which would result in a quicker system refill.

4. The flow from three HPI pumps instead of only one would be available to refill the RCS.

5. The actual time delay for HPI flow is less than 10 seconds compared to an assumed 35 seconds.

6. The assumption of no operator action is highly pessimistic. The symptoms of a tube rupture are clear and unambiguous; prompt operator action could be expected to increase injection and shut down the reactor.

EG&G Idaho Evaluation:

We find that the licensee has demonstrated that the results of the BWOG generic analyses and Duke Power analyses are conservative for Oconee. The uncertainties associated with the computer program calculations are insignificant relative to the degree of conservatism of the assumptions noted above. Also, the licensee has demonstrated that the instrument uncertainties and reliability are acceptable (see item A.2). The licensee's response to this item is acceptable.

2.4 GL 86-05, Item B.1 - Containment Isolation

The NRC requested the licensee to demonstrate that water services for RCP operation are capable of being restored following containment isolation and that the RCP will not be adversely affected if operated during containment isolation.

Oconee Response:

Damage to the pump seal is prevented by maintaining or restoring either seal injection (HPI) or component cooling (CC) to the pump thermal barrier. RCP seal cooling capability is always maintained.[4] For the combined loss of HPI and CCW the operator is instructed to trip the pumps. Pump restart requires the availability of either HPI or CCW.

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 component cooling 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.

EG&G Idaho Evaluation:

We find that the licensee has demonstrated that the RCP water services can be restored following containment isolation. The RC pump can be operated if either makeup seal injection or component cooling is functioning normally. Emergency operating procedures prevent RCP operation and restart for the combined loss of HPI or CCW. We agree that the operator has sufficient time to restore the loss of cooling services to the

RC pump and motor before exceeding the prescribed temperature limits. The BWOG analyses determined that the operator has at least 10 minutes to take action before the temperature limit of the core is approached. This is acceptable.

2.5 GL 86-05, Item B.2 - Components Required to Trip RCPs

The NRC requested the licensee to identify the components required to trip the RCPs and to justify their operability under the worst containment conditions.

Oconee Response:

To manually trip the RCPs, the control switch (located in the control room), 6900 volt switchgear breakers, and breaker trip coils must be operable. All of the components required to trip RCPs are located in a mild environment and will be capable of tripping the RCPs when necessary.[4]

EG&G Idaho Evaluation:

In the response to Items A.1 and B.2 the licensee has identified all of the essential instrumentation, switches, power sources, and degree of redundancy used to trip the RCPs. We find that the licensee has provided an acceptable response to this item.

2.6 GL 86-05, Item C - RCS Void Detection and Management

The NRC requested the licensee to address primary system void detection and management and to provide an implementation schedule for the revised ATOG procedures.

Oconee 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 is further 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.

EG&G Idaho Evaluation:

The licensee has indicated that the plant procedures have been revised and implemented to incorporate void identification and management guidelines that are consistent with the BWOG program. The licensee's commitment to utilize the RCIMS increases our confidence in the licensee's ability to address voids. We find the licensee's response to be acceptable.

3. CONCLUSION

EG&G Idaho has reviewed the Oconee responses to Generic Letter 86-05. The information clarifies the plant specific implementation of the Babcock and Wilcox Owner's Group (BWOG) philosophy for reactor coolant pump trip. EG&G Idaho finds that the Oconee submittal does meet the NRC position established in the review of the BWOG generic position.

4. REFERENCES

1. Wm. J. Dircks, Executive Director for Operations, USNRC, "Staff Resolution of the Reactor Coolant Pump Trip Issue," SECY-82-475, NRC Accession Number 8306030370, November 30, 1982.

2. F. J. Miraglia, Director, Division of PWR Licensing-B, USNRC, "Implementation of TMI Action Item II.K.3.5, Automatic Trip of Reactor Coolant Pumps," Generic Letter 86-05, May 29, 1986.

3. R. H. Bryan, "Reactor Coolant Pump Trip Philosophy," Letter from R. H. Bryan, Chairman, B&W Owners Group Analysis Committee to J. R. Miller, Chief, Operating Reactors Branch No. 3 USNRC, NRC Accession Number 8406210084, June 18, 1984.

4. H. B. Tucker, Vice President, Nuclear Production, Duke Power Company to J. F. Stolz, Project Director, PWR Project Directorate 6, U.S. Nuclear Regulatory Commission, "Implementation of TMI Action Item II.K.3.5", NRC Accession Number 8610310245, October 20, 1986.

5. H. B. Tucker, Duke Power Company to J. F. Stolz, USNRC, "Endorsement of B&W Methodology on TMI Action Item II.K.3.5", NRC Accession Number 8607280100, July 21, 1986.

6. H. B. Tucker, Duke Power Company to J. F. Stolz, USNRC, "Response to NRC Request for Additional Information on Inadequate Core Cooling (ICC) Instrumentation", NRC Accession Number 8507150631, July 1, 1985.

7. Babcock and Wilcox Owner's Group (BWOG), "Analytical Justification for the Treatment of Reactor Coolant Pumps During Accident Conditions," Report No. 77-1149091-00, NRC Accession Number 8404050168, March 30, 1984.

8. Duke Power Company Record of Conversation, "Clarification of Oconee response to TMI Action Item II.K.3.5", Teleconference held October 16, 1987 between H. Pastis (NRC), C. Kido and H. L. Magelby (EG&G Idaho), and Duke Power Company.