March 25, 1988

Docket No.: 50-285

Mr. R. L. Andrews
Division Manager - Nuclear Production
Omaha Public Power District
1623 Harney Street
Omaha, Nebraska 68102

Dear Mr. Andrews:

SUBJECT: SAFETY EVALUATION REPORT RELATING TO IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5 IN REGARD TO FORT CALHOUN STATION, UNIT 1 (GENERIC LETTER 86-06 AND TAC NO. 49696)

The staff and its consultants, EG&G Idaho Inc. have completed their review concerning the Fort Calhoun response to Generic Letter 86-06 (TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps During Loss-of-Coolant Accident").

Based upon our review of your submittals, we find them acceptable. Therefore, you have satisfied the requirements for the TMI Action Item II.K.3.5. Enclosed please find the staff's Safety Evaluation, Enclosure 1, and our consultant's Technical Evaluation Report, Enclosure 2.

If you have any further questions, please contact me at (301) 492-1345.

Sincerely,

/S/

Anthony Bournia, Project Manager
Project Directorate - IV
Division of Reactor Projects - III, IV, V and Special Projects

Enclosures:
As Stated

cc w/enclosure:
See next page

DISTRIBUTION

Docket File	NRC & LOCAL PDR's
PD IV R/F	DCrutchfield
ADR4/Collins	JCalvo
ABournia	PNoonan
OGC-WF	E. Jordan
J. Partlow	ACRS (10)
Plant File	

DOCUMENT NAME: FT. CALHOUN SER

LA:PDIV	PM:PDIV	D:PDIV
PNoonan:jc	ABournia	JCalvo
03/25/88	03/25/88	03/25/88

8804010084 880325
PDR ADOCK 05000285
PDR

March 25, 1988

Docket No.: 50-285

Mr. R. L. Andrews
Division Manager - Nuclear Production
Omaha Public Power District
1623 Harney Street
Omaha, Nebraska 68102

Dear Mr. Andrews:

SUBJECT: SAFETY EVALUATION REPORT RELATING TO IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5 IN REGARD TO FORT CALHOUN STATION, UNIT 1 (GENERIC LETTER 86-06 AND TAC NO. 49696)

The staff and its consultants, EG&G Idaho Inc. have completed their review concerning the Fort Calhoun response to Generic Letter 86-06 (TMI Action Item II.K.3.5, "Automatic Trip of Peactor Coolant Pumps During Loss-of-Coolant Accident").

Based upon our review of your submittals, we find them acceptable. Therefore, you have satisfied the requirements for the TMI Action Item II.K.3.5. Enclosed please find the staff's Safety Evaluation, Enclosure 1, and our consultant's Technical Evaluation Report, Enclosure 2.

If you have any further questions, please contact me at (301) 492-1345.

Sincerely,

/S/

Anthony Bournia, Project Manager
Project Directorate - IV
Division of Reactor Projects - III,
IV. V and Special Projects

Enclosures:
As Stated

cc w/enclosure:
See next page

DISTRIBUTION

Docket File	NRC & LOCAL PDR's
PD IV R/F	DCrutchfield
ADR4/Collins	JCalvo
ABournia	PNoonan
OGC-WF	E. Jordan
J. Partlow	ACRS (10)
Plant File	

DOCUMENT NAME: FT. CALHOUN SER

LA:PDIV PN	PM:PDIV for	D:PDIV JAC
PNoonan:jc	ABournia	JCalvo
03/25/88	03/25/88	03/25/88

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

March 25, 1988

Docket No.: 50-285

Mr. R. L. Andrews
Division Manager - Nuclear Production
Omaha Public Power District
1623 Harney Street
Omaha, Nebraska 68102

Dear Mr. Andrews:

SUBJECT: SAFETY EVALUATION REPORT RELATING TO IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5 IN REGARD TO FORT CALHOUN STATION, UNIT 1 (GENERIC LETTER 86-06 AND TAC NO. 49696)

The staff and its consultants, EG&G Idaho Inc. have completed their review concerning the Fort Calhoun response to Generic Letter 86-06 (TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps During Loss-of-Coolant Accident").

Based upon our review of your submittals, we find them acceptable. Therefore, you have satisfied the requirements for the TMI Action Item II.K.3.5. Enclosed please find the staff's Safety Evaluation, Enclosure 1, and our consultant's Technical Evaluation Report, Enclosure 2.

If you have any further questions, please contact me at (301) 492-1345.

Sincerely,

Anthony Bournia

Anthony Bournia, Project Manager
Project Directorate - IV
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
As Stated

cc w/enclosure:
See next page

Mr. R. L. Andrews
Omaha Public Power District

Fort Calhoun Station
Unit No. 1

cc:
Harry H. Voigt, Esq.
LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, NW
Washington, D.C. 20036

Mr. Jack Jensen, Chairman
Washington County Board
of Supervisors
Blair, Nebraska 68008

Mr. Phillip Harrell, Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 309
Fort Calhoun, Nebraska 68023

Mr. Charles B. Brinkman, Manager
Washington Nuclear Operations
C-E Power Systems
7910 Woodmont Avenue
Bethesda, Maryland 20814

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
Office of Executive Director
for Operations
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Harold Borchert, Director
Division of Radiological Health
Department of Health
301 Centennial Mall, South
P.O. Box 95007
Lincoln, Nebraska 68509

W. G. Gates, Manager
Fort Calhoun Station
P. O. Box 399
Fort Calhoun, Nebraska 68023

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. 86-06)
OMAHA PUBLIC POWER DISTRICT
FORT CALHOUN UNIT 1
DOCKET NOS. 50-285

1.0 SUMMARY

In Generic Letter 86-06 (Ref. 1) we reported that the information provided by the CE Owners Group (CEOG) in support of alternative Reactor Coolant Pump (RCP) trip criteria was acceptable on a generic basis. The 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 CEOG methodology. This Safety Evaluation Report (SER) contains the staff's findings concerning this issue for Omaha Public Power District's Fort Calhoun Unit 1.

Reference 1 required owners of CE Nuclear Steam Generating Systems to evaluate their plants with respect to 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 86-06 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 TMI Action Item II.K.3.5.

2.0 BACKGROUND

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 since tripping the reactor coolant pumps (RCPs) was not considered the ideal solution. Automatic trip of the RCPs in the case of a small-break LOCA was recommended until a better solution was found. A summary of both the industry programs and the NRC programs concerning RCP trip is provided in Generic Letters 83-10a through f, which are included in the NRC report, SECY-82-475 (Ref. 2). SECY-82-475 also provided the NRC guidelines and criteria for the resoultion of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps."

The CEOG proposes using a trip-two/leave-two (T2/L2) strategy. The T2/L2 trip strategy consists of tripping two RCPs, located in diametrically opposed coolant loops, very early in a transient on a low reactor coolant system (RCS) pressure signal independent of the nature of the event. The remaining two RCPs are tripped subsequently after trip setpoints indicating a LOCA are reached.

The licensee addressed this issue in References 3 and 4, 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 detail in the attached TER, the licensee has satisfied the requirements of GL 86-06. The staff finds that Omaha Public Power District has complied with the requirements of Generic Letter 86-06 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 first two RCPs are tripped if the pressurizer pressure falls below 1350 psia. The last two RCPs are tripped if the steam generator recording pressure does not fall below 800 psia and the containment pressure is increasing. This agrees with the approved CE Owners Group guidelines and hence is acceptable.

B. Instrumentation Uncertainties for Normal and Adverse Environments
The licensee has demonstrated that the instrument uncertainties are conservatively bounded in the plant specific analyses. We conclude these uncertainties are acceptable.

C. Generic and Plant-Specific Analyses Uncertainties
The licensee has demonstrated that the results of the CEOG generic analyses are conservative for Fort Calhoun. Therefore we consider these acceptable.

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

4.0 CONCLUSION

Each of the points identified in Reference 1 has 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. F. J. Miraglia, USNRC, letter to all applicants and licensees with CE designed Nuclear Steam Supply Systems (except Maine Yankee), "Implementation of TMI Action Item II.K.3.5, Automatic Trip of Reactor Coolant Pumps," Generic Letter 86-06, May 29, 1986.

2. W. 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.

3. R. L. Andrews, OPPD, letter to NRC, "Response to Generic Letter 86-06," LIC-87-649, September 30, 1987.

4. R. L. Andrews, OPPD, letter to NRC, "Additional Questions on OPPD Response to Generic Letter 86-06," LIC-88-055, February 4, 1988.

ENCLOSURE 2

TECHNICAL EVALUATION REPORT
CONFORMANCE TO GENERIC LETTER 86-06
IMPLEMENTATION OF NUREG-0737, TMI ACTION ITEM II.K.3.5
FORT CALHOUN, UNIT 1
DOCKET NO. 50-285

1. INTRODUCTION

TMI Action Plan Item II.K.3.5 of NUREG-0737 requires all licensees to consider other solutions to 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 and NRC programs concerning RCP trip. In Generic Letter 86-06[2] the staff accepted the Combustion Engineering Owner's Group (CEOG) trip-two/leave-two staggered RCP trip strategy.[3,4]

The CEOG developed a trip-two/leave-two (T2/L2) strategy as the basis for RCP trip. The T2/L2 strategy consists of tripping two RCPs, located in diametrically opposed coolant loops, early in a transient on a low reactor coolant system (RCS) pressure signal regardless of the nature of the event. The remaining two RCPs are later tripped if setpoints indicating a LOCA are reached. The goal of the T2/L2 trip strategy is to trip all four RCPs in the case of a small break LOCA but to have two or more RCPs operating for non-LOCA events. These would include steam line breaks, steam generator tube ruptures, or an anticipated operational occurrence.

The CEOG reports addressed the selection of trip parameters, evaluation of LOCA and non-LOCA events, evaluation of NRC criteria, justification of manual RCP trip, and instrumentation capabilities. The generic information presented by the CEOG, however, did not address plant specific concerns about instrumentation uncertainties, potential RCP problems, and operator training and procedures. This information, specifically identified in Generic Letter 86-06, was requested from each C-E licensee to enable the staff to assess implementation of the RCP trip criterion.

2. DISCUSSION

Omaha Public Power District's (OPPD's) response to Generic Letter 86-06, Section IV, for Fort Calhoun, Unit 1, was provided in a letter dated September 30, 1987.[5] OPPD's response to a request for information was submitted February 4, 1988.[6] This information was reviewed to verify OPPD provided the required information. This review found the licensee endorsed the CEOG reports and provided plant specific details, such as pressure setpoints, emergency operating procedures, and instrument uncertainties. A summary of OPPD's response to Generic Letter (GL) 86-06 and EG&G Idaho's basis for acceptance is provided below.

2.1 <u>GL 86-06, Item 1 - Reactor Coolant Pump Trip Criteria</u>

The NRC requested the licensee to identify the instrumentation used to determine the RCP trip setpoints, including the degree of redundancy for each measurement needed for the criteria chosen.

Response for Fort Calhoun:

The first two RCPs are tripped if the pressurizer pressure falls below 1350 psia. The last two RCPs are tripped if the steam generator secondary pressure does not fall below 800 psia and the containment pressure is increasing or the secondary system reactivity alarms do not actuate. Only the RCS and secondary pressures are compared to specific setpoints. The other parameters (containment pressure and secondary reactivity) are reviewed for trending information. In Reference 6, OPPD stated that the choice of these trip setpoints was based on the information provided in CEN-152, <u>Emergency Procedure Guidelines</u>.

RCS wide range pressure is [illegible] from two channels of instrumentation. [illegible] pressure is available from six channels per steam generator. [illegible] can be determined from four instrument channels. [illegible] reactivity can be determined from the condenser [illegible] radiation [illegible] and one radiation monitor per steam generator [illegible] blowdown line.

EG&G Idaho evaluation:

The licensee identified the pump trip critera and setpoints for Fort Calhoun. The setpoints discussed above (trip the first two pumps if RCS pressure is less than 1350 psia and trip the second two pumps if secondary pressure is above 800 psia and containment pressure increasing or no secondary radiation alarms) are different from those proposed in the CEOG report, CEN-268. However, this approach is based on the emergency procedure guidelines in CEN-152. These guidelines provided alternate means to determine a SBLOCA had occurred. CEN-152 was reviewed by the NRC and an SER accepting this report was issued by the NRC.[7] Use of alternate, approved methods for determining when a plant is experiencing a SBLOCA is considered acceptable. The licensee also identified the instrumentation needed to implement the chosen pump trip criteria. Adequate redundancy is available for this instrumentation. The response to Item 1 is acceptable.

2.2 GL 86-06, Item 2 - Instrumentation and Environment

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

Response for Fort Calhoun:

The licensee used adverse containment conditions to determine the instrument uncertainty to be applied to the plant specific setpoints recommended in CEN-268. For the RCS pressure setpoint to trip the first set of pumps, an uncertainty of ±115 psi was determined based on the containment thermal-hydraulic response and radiation dose from a small break LOCA up to the time where the emergency operating procedures direct operator actions. Uncertainty in the secondary pressure was determined to be ±62 psi.

These uncertainties were conservatively bounded in the plant specific RCS pressure setpoint for tripping the first set of pumps. The setpoint chosen, 1350 psia, exceeds the setpoint recommended in CEN-268, 1210 psia, by 140 psi.

The licensee noted there is sufficient instrument redundancy and that the instruments are physically separated so that local conditions such as pipe whip or fluid jets will not affect the ability of the operators to diagnose the need to trip the pumps.

EG&G Idaho evaluation:

The licensee based its estimate of instrument uncertainty on adverse containment conditions which envelope normal conditions. The pressure setpoint for tripping the first set of pumps, 1350 psia, accommodates an uncertainty of 140 psi based on the recommended setpoint of 1210 psia for Fort Calhoun in CEN-268. The actual uncertainty was determined to be 115 psia.

The licensee also demonstrated local conditions will not impact the measurements required to implement the T2/L2 strategy. Thus, OPPD's response to this item is considered acceptable.

2.3 GL 86-06, Item 3 - Generic and Plant Specific Analyses

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

Response for Fort Calhoun:

The licensee endorsed the CEOG report, CEN-268, as providing a thermally conservative value of RCS pressure to use in tripping the first set of pumps. The licensee noted the plant specific instrument uncertainties were conservatively bounded in determining the plant specific setpoint to trip the first set of pumps. In addition, the setpoint for tripping the first set of pumps, 1350 psia, is also below the SIAS setpoint of 1600 psia. Therefore, the SIAS can be used as a warning to alert the operators that the first set of pumps may need to be tripped.

The licensee also noted the 1210 psia setpoint recommended for Fort Calhoun in CEN-268 was not based on the computer analysis, but rather it was based on a energy balance between the primary and secondary as the RCS pressure plateau is established in a small break LOCA. The analysis was performed to determine the primary to secondary pressure difference necessary to remove decay heat. As such, except for plant specific instrument uncertainties, no further uncertainties need to be considered.

EG&G Idaho evaluation:

The licensee endorsed the CEOG report as providing a conservative value of RCS pressure at which to trip the first set of pumps. The analysis referred to by the licensee to justify the 1210 psia setpoint for the first two pumps was discussed in response to question 48 to 55 in CEN-268, Supplement 1. The licensee provided plant specific information in Reference 6 that could be used to verify the analysis in CEN-268 and CEN-268, Supplement 1, provided a pressure that was represenstative of the Ft. Calhoun pressure plateau that occurs in a SBLOCA. Using this information and assuming a primary pressure of 1200 psia, it was found that the break flow alone would be sufficient to remove the decay heat and pump energy. Therefore, the pressure plateau at Ft. Calhoun would be below 1200 psia while CEN-268 recommended a pressure setpoint of 1210 psia. This is not unexpected because the power at Ft. Calhoun is 1500 MW versus the 2700 MW power for the class of reactors Ft. Calhoun was grouped with in CEN-268. Review of the licensee's response to this item also found the plant specific instrument uncertainties identified in the response to Item 3 were adequately accounted for in determining the pump trip setpoints. This calculation indicates OPPD's response is adequately accounts for the uncertainties in the analysis and instrumentation.

2.4 GL 86-06, Item 4 - Operator Procedures and Training

The NRC requested the licensee to identify plant procedures that require RCP trip guidelines and describe the training and procedures that provide direction for use of individual steam generators with and without operating RCPs.

Response for Fort Calhoun:

The following EOPs were identified as requiring the use of the RCP pump trip strategy:

a. EOP-01, Reactor trip.

b. EOP-02, Electrical emergency.

c. EOP-03, Loss of coolant accident.

d. EOP-04, Steam gererator tube rupture.

e. EOP-05, Uncontrolled heat extraction.

f. EOP-06, Loss of all feedwater.

g. EOP-20, Functional recovery.

The licensee stated the pump trip implementation strategy complied with the CEOG generic Emergency Procedure Guidelines, CEN-152, Rev. 2.

The licensee provided information on operator training in Reference 6. Enlcosed were sample lesson plans for the RCP Trip Strategy and Natural Circulation Review. The licensee stated these were included in the standard training plan. The licensee also included a representative list showing dates and persons participating in the training. These listings showed training was provided for EOPs -02, -03, -04, -05, and -20 listed above.

EG&G Idaho evaluation:

The licensee identified the procedures requiring the use of RCP trip guideline and stated the Fort Calhoun pump trip strategy complied with CEOG emergency procedure guidelines, which is acceptable. Operator training was discussed in Reference 6 and indicated that operators were recieving training on the pump trip strategy and natural circulation as well as the EOPs. The response to this item is considered adequate.

3. CONCLUSION

OPPD's responses for Fort Calhoun, Unit 1, to Generic Letter 86-06 were reviewed. The information in these responses clarifies the plant specific implementation of the CEOG strategy for reactor coolant pump trip. The review found the submittal for Fort Calhoun, Unit 1, meets the NRC position established in the review of the CEOG report.

4. REFERENCES

1. W. J. Dircks, NRC, "Staff Resolution of the Reactor Coolant Pump Trip Issue," SECY-82-475, November 30, 1982.

2. F. J. Miraglia, NRC, ltr to All Applicants and Licensees with CE Designed Nuclear Steam Supply Systems (Except Maine Yankee), "Implementation of TMI Action Item II.K.3.5, Automatic Trip of Reactor Coolant Pumps," Generic Letter 86-06, May 29, 1986.

3. Justification of Trip-Two/Leave-Two Reactor Coolant Pump Trip Strategy During Transients, CEN-268, March 1984.

4. Response to NRC Request for Additional Information on CEN-268, CEN-268 Supplement 1-NP, November 1984.

5. R. L. Andrews, OPPD, ltr to NRC Document Control Desk, "Response to Generic Letter 86-06," LIC-87-649, September 30, 1987.

6. R. L. Andrews, OPPD, ltr to NRC Document Control Desk, "Additional Questions on OPPD Response to Generic Letter 86-06," LIC-88-055, February 4, 1988.

7. Combustion Engineering Emergency Procedures Guidelines, CEN-152, Rev. 2, May 1984.

8. J. A. Zwolinski, NRC, ltr to R. W. Wells, CEOG, "Supplement 1 to Safety Evaluation Report for CEN-152, Combustion Engineering Emergency Procedure Guidelines," April 16, 1985.