Omaha Public Power District

1623 HARNEY ■ OMAHA, NEBRASKA 68102 ■ TELEPHONE 536-4000 AREA CODE 402

December 31, 1981

Mr. Robert A. Clark, Chief
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Operating Reactors Branch No. 3
Washington, D.C. 20555

Reference: Docket No. 50-285

Dear Mr. Clark:

The TMI Action Plan, Items II.K.2.13, II.K.2.17, II.K.3.25, and II.K.3.30, requires licensees to evaluate specific concerns regarding the plant's response to various transients. Omaha Public Power District's responses or schedule for responding to the four items are provided in Attachments 1, 2, 3, and 4, respectively.

Sincerely,

W C Jones

W. C. Jones
Division Manager
Production Operations

WCJ/KJM/TLP:jmm

Attachments

cc: LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N.W.
Washington, D.C. 20036

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TMI ACTION PLAN ITEM II.K.2.13: THERMAL MECHANICAL REPORT EFFECT OF HIGH-PRESSURE INJECTION ON VESSEL INTEGRITY FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENT WITH NO AUXILIARY FEEDWATER

Omaha Public Power District is participating in Combustion Engineering Owners Group (CEOG) activities that respond to the requirements of TMI Action Plan Item II.K.2.13, "Thermal Mechanical Report--Effect of High-Pressure Injection on Vessel Integrity for Small-Break Loss-of-Coolant Accident With No Auxiliary Feedwater." As a result of this effort, the CEOG submitted to the Commission for review report CEN-189, "Evaluation of Pressurized Thermal Shock Effects Due to Small Break LOCA's With Loss of Feedwater for Combustion Engineering NSSS," December 1981, by a letter dated December 31, 1981. This report describes the methods used in all analytical evaluations and presents plant-specific analysis results as separate appendices. In particular, Appendix A, "Evaluation of Pressurized Thermal Shock Effects Due to Small Break LOCA's With Loss of Feedwater for the Fort Calhoun Reactor Vessel," contains the results for the Fort Calhoun Station vessel. The District hereby references report CEN-189 and Appendix A to report CEN-189 in response to the documentation requirements of TMI Action Plan Item II.K.2.13.

TMI Action Plan Item II.K.2.13, as clarified in NUREG-0737, requires that "a detailed analysis shall be performed on the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater." The requirement "deals with the potential for thermal shock of reactor vessels resulting from cold safety injection flow." Report CEN-189 details the results of the required analysis.

The justification for continued safe operation of the Fort Calhoun Station, with the thermal shock effects of a small break LOCA, is based upon existing fluence to the vessel. The power distributions used to calculate the Fort Calhoun reactor's vessel fluence were provided to Combustion Engineering by the District. The power distributions for Cycles 1, 2, and 3 were calculated by CE using the PDQ code to support the surveillance capsule fluence calculation. The PDQ code has also been utilized by the District to calculate the power distributions for Cycles 4, 5, 6, and 7.

The PDQ code has been utilized to calculate the power distributions for the reload safety analysis for Cycles 2, 3, 4, and 5. The PDQ code is a part of the CE core analysis package which is being utilized by the District. The District participated with CE to assess the physics uncertainty associated with this package for operating CE reactors, including the Fort Calhoun Station. This assessment verified the capability of the code package to accurately predict power distributions and is contained in Reference 1. The NRC's acceptance of the associated physics uncertainty for this package is contained in Reference 2.

The results contained in report CEN-189 and in Appendix A demonstrate that the Fort Calhoun Station can safely withstand a small break loss of coolant accident with extended loss of feedwater for the full design life of the plant without crack initiation.

The District is a continuing participant in a CEOG program to evaluate the effects on reactor vessel integrity of accident scenarios other than the small break loss of coolant accident with no auxiliary feedwater. Preliminary results from this program have been previously discussed with the NRC staff in meetings on April 29, 1981, July 30, 1981, and October 7, 1981. Documentation of the results from this program will be provided to the NRC staff for review during 1982. In particular, the District will submit to the Commission in January 1982 results which demonstrate the safety of continued operation of the Fort Calhoun Station in consideration of the pressurized thermal shock issue. This submittal will be made in response to the NRC's letters dated August 21, 1981 and December 18, 1981. Also to be included in that response will be a plan that will define actions and schedules for resolution of this issue as required in the August 21, 1981 letter.

REFERENCES

(1) CENPD-153-P, Rev. 1-P-A, May 1980, "Evaluation of Uncertainty in the Nuclear Power Peaking Measured by the Self-Powered, Fixed In-Core Detector System."

(2) Letter from James R. Miller (DOL) to A. E. Scherer (CE) dated July 2, 1980.

Attachment 2

TMI ACTION PLAN ITEM II.K.2.17: POTENTIAL FOR VOIDING IN THE REACTOR COOLANT SYSTEM DURING TRANSIENTS

Omaha Public Power District is participating in a Combustion Engineering Owners Group (CEOG) program to respond to the requirement of TMI Action Plan Item II.K.2.17, "Potential for Voiding in the Reactor Coolant System During Transients." This item is a requirement to "analyze the potential for voiding in the reactor coolant system (RCS) during anticipated transients." The CEOG program analyses have been completed. These analyses demonstrate that, if voiding occurs in a CE RCS during anticipated transients, the NRC acceptance criteria for these transients are still met. The documentation of the analyses results is presently being prepared and is expected to be submitted by the CEOG to the NRC for review no later than March 31, 1982.

The analyses results from the CEOG program will be justified regarding applicability to the Fort Calhoun Station. When the documentation of these results is submitted to the Commission by the CEOG, the District will reference these results as applicable for the Fort Calhoun Station.

It should be noted that the concern of RCS voiding during natural circulation has been addressed separately in the District's letter to the Commission dated November 13, 1981. The conclusions and commitments made in the November 13, 1981 letter are still valid and will not be addressed as part of the CEOG program described above.

TMI ACTION PLAN ITEM II.K.3.25: EFFECT OF LOSS OF ALTERNATING CURRENT ON PUMP SEALS

The TMI Action Plan position is that licensees should determine, by analysis or experiment, the effect of loss of cooling water to the reactor circulation pump seal coolers. The pump seals should be designed to withstand a complete loss of Alternating Current (AC) power for at least two hours. The TMI Action Plan also states that, if seal failure is a consequence of loss of cooling water to the Reactor Coolant Pump (RCP) seal coolers due to loss of offsite power, an acceptable solution would be to supply emergency power to the component cooling water pump.

Based upon a recent event at the Fort Calhoun Station, the District believes the RCP seals can survive a two hour loss of cooling water. This recent event resulted in a loss of cooling water to the RCP seal coolers for a one hour and thirty-three minute period. The loss of cooling water was caused by loss of a DC bus (see Fort Calhoun Station LER 81-003) and resulted in no seal degradation. The District is confident that the seals would have survived a full two hours.

It should also be noted that the component cooling water (CCW) system which provides cooling water to the RCP seal coolers is powered from an emergency bus. The Fort Calhoun Station emergency procedures for loss of AC power provides the operators guidelines for restoring CCW flow to the seal coolers. This operator action is required because the RCP seals could be isolated from the component cooling water system if the transient were to initiate a containment isolation actuation signal (CIAS). However, the component cooling water is not isolated from a number of coolers following CIAS, including the fan coolers and other components which are currently recognized as necessary for emergency safeguards operation. The District is considering a modification to the reactor coolant pump seal coolers isolation logic such that the coolers will not be isolated upon CIAS, thus eliminating the need for operator action to restore flow.

TMI ACTION PLAN ITEM II.K.3.30:
REVISED SMALL BREAK LOSS-OF-COOLANT
ACCIDENT METHODS TO SHOW COMPLIANCE WITH
10 CFR PART 50, APPENDIX K

Omaha Public Power District has participated in a Combustion Engineering Owners Group (CEOG) program to develop a response to the requirements of TMI Action Plan Item II.K.3.30, "Revised Small Break Loss-of-Coolant Accident Methods to Show Compliance With 10 CFR Part 50, Appendix K." The details of this program have been discussed by representatives of the CEOG and the NRC staff in meetings on January 26, 1981 and October 30, 1981. By letter dated November 18, 1981 from the Chairman of the CEOG to Mr. D. G. Eisenhut, Director of the NRC Division of Licensing, a request was made for extension of the deadline for response to Item II.K.3.30 to March 31, 1982.

The results of the CEOG program which addresses Item II.K.3.30 are applicable to the Fort Calhoun Station. When the results of this program are submitted to the NRC for review, the District will reference these results and justify the report's findings for applicability to the Fort Calhoun Station. It is expected that this report and plant specific justification will be provided to the Commission by March 31, 1982.