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AUTH.NAME AUTHOR AFFILIATION
ALEXICH,M.P. Indiana Michigan Power Co. (formerly Indiana & Michigan Ele
RECIP.NAME RECIPIENT AFFILIATION
MURLEY,T.E. Office of Nuclear Reactor Regulation, Director (Post 870411

SUBJECT: Forwards WCAP-11908, "Containment Integrity Analysis for Donald C. Cook Nuclear Plant Units 1 & 2." SGG SUBJ WCAP 11908

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Power Company
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Columbus, OH 43216

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AEP:NRC:1024D
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Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
CONTAINMENT LONG-TERM PRESSURE ANALYSIS TO SUPPORT RHR CROSS-TIE VALVE CLOSURE

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Attn: T. E Murley

August 22, 1988

Dear Dr. Murley:

The purpose of this letter is to transmit an analysis of containment long term pressure effects due to a postulated LOCA. The analysis was performed for us by Westinghouse Electric Corp. (Westinghouse). The analysis, printed as WCAP 11908, is contained in Attachment 2.

The analysis supports operation of both units of the Donald C. Cook Nuclear Plant with the residual heat removal (RHR) or safety injection (SI) system cross-tie valves closed. The analysis is also part of a program we are undertaking for Unit 1 which will allow operation of the unit at reduced temperatures and pressures. Background information and a summary of the analysis is provided in Attachment 1.

Approval of the containment analysis, as well as several other analyses related to cross-tie closure, is necessary by January 31, 1989, so that ASME code-required stroke testing of certain RHR valves can be performed in Unit 1 without bringing the unit to a shutdown condition. (Details of this are included in Attachment 1.) Therefore, we request that your review be completed by January 2, 1989, in order to avoid an unnecessary unit shutdown and to allow for outage planning for other units.

A check in the amount of $150 is enclosed for NRC processing of the analysis.

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This document has been prepared following Corporate procedures which incorporate a reasonable set of controls to ensure its accuracy and completeness prior to signature by the undersigned.

Sincerely,

M. P. Alexich
Vice President

MPA/eh

Attachments

cc: D. H. Williams, Jr
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Bruchmann
G. Charnoff
NRC Resident Inspector - Bridgman
A. B. Davis - Region III

ATTACHMENT 1 TO AEP:NRC:1024D

BACKGROUND INFORMATION AND

SUMMARY OF WESTINGHOUSE ELECTRIC CORPORATION

CONTAINMENT LONG-TERM PRESSURE ANALYSIS

Background

T/S 3.5.2 states that two emergency core cooling system (ECCS) subsystems must be operable; it defines an operable ECCS subsystem as including one operable charging pump, safety injection (SI) pump, residual heat removal (RHR) pump, RHR heat exchanger, and associated flow paths. This T/S allows the operator to remove one ECCS subsystem for up to 72 hours while in Modes 1, 2 or 3 while maintaining an operable flow path for the redundant subsystem. The RHR and safety injection pump configuration at the Cook Nuclear Plant is such that any one pump can deliver flow to all four reactor coolant loops. This is accomplished by means of cross-tie valves. With the cross-tie valves closed, each pump can only supply flow to two reactor coolant loops. The current small-break and large-break LOCA analyses for Cook Nuclear Plant Unit 1 assume that the cross-tie valves in the SI and RHR lines are open. This requires that the cross-tie valves be open to satisfy the operable flow path requirements of T/S 3.5.2.e for Modes 1, 2 and 3.

In the past, there were instances in which the Cook Nuclear Plant Units 1 and 2 were operated in Modes 1, 2 and 3 with the cross-tie valves closed. The valves were closed to allow maintenance and testing of various system components. Because this operation was not in agreement with the existing safety analyses, it was the subject of an Enforcement Conference held at Region III headquarters on January 21, 1987.

Since some maintenance and testing work can only be performed on the RHR or SI systems in Modes, 1, 2 and 3 with the cross-tie valves closed, we decided to pursue new analyses which would support two-loop injection. Analyses have been transmitted on several occasions, as described in the table below.

Submittal No.	Date	Description
AEP:NRC:1024	March 23, 1987	Unit 2 small-break LOCA evaluation.
AEP:NRC:1024A	May 13, 1987	Unit 1 small-break LOCA evaluation; Units 1 and 2 large-break LOCA evaluations.
AEP:NRC:1024C	October 13, 1987	Evaluation of limiting break size for Unit 1.
AEP:NRC:1024E	February 29, 1988	Evaluation of limiting break size for Unit 2.

ATTACHMENT 2 TO AEP:NRC:1024D

WESTINGHOUSE CONTAINMENT ANALYSIS, WCAP 11908

The analysis of containment long-term pressure is the last portion of the analytical work necessary to support operation of the Cook Nuclear Plant with the cross-tie valves closed. NRC acceptance of the analyses is requested by January 2, 1989, because of an inservice testing program requirement to stroke 2 valves in the RHR system that can only be stroked in Modes 1, 2 or 3 with the RHR cross-tie valves closed. (The units are currently operating under temporary relief granted by the NRC in safety evaluation reports dated December 19, 1986, and June 8, 1988. The next required test will be for the Unit 1 valves, which must be tested by January 31, 1989.)

In addition to supporting the closure of the cross-tie valves, the containment analysis transmitted with this letter also forms part of the analytical effort which will support operation of Unit 1 at reduced temperatures and pressures during its next fuel cycle (Cycle 11). The reduced temperature and pressure program was undertaken in order to prevent steam generator degradation of the type experienced in Unit 2. The balance of the analyses which support reduced temperature and pressure operation will be transmitted by October 15, 1989. The containment long-term pressure analysis portion was performed early in order to support the inservice testing requirements.

Summary of Westinghouse Containment Long-Term Pressure Analysis

As discussed above, the Westinghouse analysis is intended to address cross-tie closure as well as reduced temperature and pressure operation. The analysis was performed such that it is bounding for both Units 1 and 2.

The analysis is performed in two parts. First, the mass and energy released to containment as a result of a LOCA is determined. The results of this calculation are then used to determine the resulting containment pressure. The mass and energy release is determined using the NRC-approved methodology outlined in WCAP 10325-P-A, entitled "Westinghouse LOCA Mass and Energy Release for Containment Design - March 1979 Version." Two cases are analyzed, one assuming maximum ECCS flow, the other assuming minimum ECCS flow (i.e., failure of one train of ECCS water and closure of the RHR cross-tie valves).

The break is taken as a double ended hot leg guillotine break in the pump suction portion of the RCS piping, located between the steam generator and the reactor coolant pump. This location has been shown in previous Westinghouse studies to result in the highest mass and energy release rates to the containment building. The mass and energy release calculation used a different

steam/water mixing model than that discussed in WCAP 10325-P-A. The enhanced model is based on empirical test data from experiments which closely simulated the flow regions and gravitational effects that would occur in PWRs.

The mass and energy release from the minimum ECCS flow case was determined to be bounding. This is because less ECCS water is available to condense steam in the RCS. The mass and energy release calculational results for the minimum ECCS flow case were then input into the LOTIC code, in accordance with the NRC-approved methodology outlined in WCAP 8345-P-A, entitled "Long Term Ice Condenser Containment Code - LOTIC Code." The LOTIC results determined a peak containment pressure of 11.89 psig, which is below the containment design pressure of 12 psig.