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SUBJECT: Submits status of NUREG-0737,Items II.K.2.13,II.K.2.17, II,K.3.1,II.K.3.5,II.K.3.25 & II.K.3.30.

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December 29, 1981
AEP:NRC:0649

Donald C. Cook Nuclear Plant Unit Nos. 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
Status of NUREG-0737 Items II.K.2.13, II.K.2.17, II.K.3.1,
II.K.3.2, II.K.3.5, II.K.3.25, II.K.3.30

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

Dear Mr. Denton:

The purpose of this letter is to inform you of the status of the subject NUREG-0737 items. These items have either been completed since our initial response to NUREG-0737 was submitted on Janurary 8, 1981 (AEP:NRC:00398) or carry January 1, 1982 submittal dates.

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:

This item requires a detailed analysis of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater. The Westinghouse Owners Group is performing an analysis for generic Westinghouse plant groupings to address this issue. It is our understanding that this analysis will be submitted to the NRC by the end of 1981. This generic study will be applicable to the Donald C. Cook Nuclear Plant.

ITEM II.K.2.17 - POTENTIAL FOR VOIDING IN THE RCS DURING TRANSIENTS:

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The Westinghouse Owners Group has performed a study which addresses the potential for void formation in Westinghouse designed nuclear steam supply systems during natural circulation cooldown/ depressurization transients. This study has been sumbitted to the NRC by the Westinghouse Owners Group by letter OG-57, dated April 20, 1981 and is applicable to the Donald C. Cook Nuclear Plant.

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In addition, the Westinghouse Owners Group has developed a generic natural circulation cooldown guideline that takes the results of the study into account so as to preclude void formation in the upper head region during natural circulation cooldown/depressurization transients, and specifies those conditions under which upper head voiding may occur. These Westinghouse Owners Group generic guidelines have been submitted to the NRC by letter OG-64, dated November 30, 1981. The generic guidance developed by the Westinghouse owners Group will be utilized in updating, if necessary, our existing natural circulation cooldown procedure as part of the efforts for item I.C.1. As stated in our response to item I.C.1 in our January 8, 1981 letter (AEP:NRC:00398) our natural circulation cooldown procedure already incorporates adequate provisions to prevent void formation in the reactor head and is consistent with the concerns expressed in IE Circular No. 80-15.

ITEMS II.K.3.1 - INSTALLATION AND TESTING OF AUTOMATIC POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM; AND II.K.3.2 - REPORT ON OVERALL SAFETY EFFECT OF POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM:

The Westinghouse Owners Group has submitted WCAP-9804 entitled, "Probabilistic Analysis and Operational Data in Response to Item II.K.3.2 for Westinghouse NSSS Plants" by letter OG-52, dated March 13, 1981. This report is applicable to the Donald C. Cook Nuclear Plant and shows that the installation of an automatic PORV isolation system (II.K.3.1) is not necessary. No further action is necessary for these NUREG-0737 items.

ITEM II.K.3.5 - AUTOMATIC TRIP OF REACTOR COOLANT PUMPS DURING LOSS-OF-COOLANT ACCIDENT:

The Westinghouse Owners Group has performed an analysis of delayed reactor coolant pump trip during small-break LOCAs. This analysis is documented in WCAP-9584 entitled "Analysis of Delayed Reactor Coolant Pumps Trip During Small Loss of Coolant Accidents for Westinghouse NSSS" dated August 1979. In addition, the Westinghouse Owners Group has performed test predictions of LOFT Experiments L3-1 and L3-6. The results of these predictions are documented in letters OG-49 dated March 3, 1981, OG-50 dated March 23, 1981 and OG-60 dated June 15, 1981.

Based on: 1) the Westinghouse analysis, 2) the excellent prediction of the LOFT Experiment L3-6 results using the Westinghouse analytical model, and 3) Westinghouse simulator data related to operator response time, it is the position of both the Westinghouse Owners Group and ourselves is that automatic reactor coolant pump trip is not necessary since sufficient time is available for manual tripping of the pumps.

Our understanding of the schedule for final resolution of this issue is:

A) Once the NRC formally approves the Westinghouse model, a 3-month study period will ensue during which the Westinghouse Owners Group will attempt to demonstrate compliance with some NRC acceptance criteria for manual RCP trip. The NRC acceptance criteria will accompany their formal approval of the Westinghouse models.

B) If, at the end of the 3-month period, the Westinghouse Owners Group cannot show compliance with the acceptance criteria, the NRC will formally notify utilities that they must submit an automatic RCP trip design.

ITEM II.K.3.25 - EFFECT OF LOSS OF AC POWER ON REACTOR COOLANT PUMP (RCP) SEALS:

This supplements our response to item II.K.3.25 provided by our letter of Janurary 8, 1981 (AEP:NRC:00398). During operation, seal injection flow from the Centrifugal Charging Pumps (CCP's) is provided to cool the RCP seals and the Component Cooling Water (CCW) system provides flow to the thermal barrier heat exchanger to limit the heat transfer from the reactor coolant to the RCP internals. In the event of loss of offsite power the RCP's are tripped and both of these cooling supplies are terminated momentarily. The CCP's and CCW pumps are powered from the emergency diesel generators which automatically start restoring seal injection flow and component cooling water to the thermal barrier heat exchanger automatically within seconds. Either of these cooling supplies is adequate to provide seal cooling and prevent seal failure due to loss of seal cooling during a loss of offsite power. No further action is necessary for this item.

ITEM II.K.3.30 - REVISED SMALL-BREAK LOCA METHODS TO SHOW COMPLIANCE WITH 10 CFR 50, APPENDIX K:

This supplements our response to item II.K.3.30 provided in our letter of October 1, 1980 (AEP:NRC:00398A).

Westinghouse feels very strongly and we are in agreement that the small-break LOCA analysis model currently approved by the NRC for use on Cook Plant is conservative and in conformance with Appendix K to 10 CFR Part 50. However, Westinghouse believes that improvement in the realism of small-break calculations is a worthwhile effort and has

committed to revise its small-break LOCA analysis model to address NRC concerns (e.g., NUREG-0611, NUREG-0623, etc.). This revised Westinghouse model is currently scheduled for submittal to the NRC by April 1, 1982 as documented in letter NS-EPR-2524 dated November 25, 1981.

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

Very truly yours,

R. S. Hunter
Vice President

cc: John E. Dolan - Columbus
R. W. Jurgensen
D. V. Shaller - Bridgman
R. C. Callen
G. Charnoff
Joe Williams, Jr.
NRC Resident Inspector at Cook Plant - Bridgman

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