DCS MS-016

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Docket Nos. 50-266
and 50-301

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Mr. C. W. Fay
Vice President Nuclear Power
Wisconsin Electric Power Company
231 West Michigan Street
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Dear Mr. Fay:

SUBJECT: NUREG-0737 ITEM II.K.2.17, POTENTIAL FOR VOIDING IN THE REACTOR COOLANT SYSTEM DURING TRANSIENTS

We have reviewed your submittals of December 23, 1980 and April 26, 1982 wherein you referenced work being performed by the Westinghouse Owner's Group to resolve the subject item for Westinghouse Plants. We have also reviewed the Westinghouse Owner's Group Report WOG-57 dated April 20, 1981 related to the subject item. The results of our review are contained in the attached Safety Evaluation. Based upon our review we find that the subject item is satisfactorily resolved for the Point Beach Nuclear Plant Units 1 and 2.

Sincerely,

Original Signed by J. R. Miller

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosure:
Safety Evaluation

cc: See next page

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

VOIDING IN THE REACTOR COOLANT SYSTEM DURING

ANTICIPATED TRANSIENTS IN WESTINGHOUSE PLANTS

I. INTRODUCTION

On April 14, 1979, just after the TMI-2 incident, the NRC issued IE Bulletin No. 79-06A (ref. 1) which, among other things, required all Westinghouse plant licensees to review the actions required by operating procedures for coping with transients and accidents with particular attention to:

a. Recognition of the possiblility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability,

b. Operator action required to prevent the formation of such voids, and

c. Operator action required to enhance core cooling in the event such voids are formed (e.g., remote venting).

On June 11, 1980, a steam bubble formed in the upper head region of a Combustion Engineering plant during a natural circulation cooldown (ref. 2). The issue of steam formation in the reactor coolant system (RCS) of Westinghouse plants was thereafter made part of the TMI Action Plan Item II.K.2.17 (ref. 3).

The June 11, 1980 event also resulted in the issuance of an NRC Generic Letter (ref. 4) which asked all PWR licensees to review their capabilities for performing natural circulation cooldown and to assess the potential for upper vessel voiding during the process. The natural circulation issue was evaluated separately as part of Multi Plant Action Item B-66.

II. DISCUSSION

Subsequent to Reference 4 the Westinghouse Owners Group undertook a study (ref. 5) to ascertain the potential for void formation in Westinghouse reactors during anticipated transients. For this study Westinghouse used the WFLASH computer program, which models the RCS with nodalized volumes connected by flow paths. This has two-phase flow capability, and tracks voids when they occur.

The potential for void formation during transients depends on, among other things, the initial temperature of fluid in the upper head region and the degree with which it mixes with colder fluid in other parts of the primary system. In Westinghouse plants the initial upper head temperature depends on how much cold leg fluid is diverted to this region. For the newer Westinghouse plants there is enough cold leg fluid diverted to make the temperature in the upper head region essentially equal to the temperature of the cold leg fluid. However, most currently operating Westinghouse plants have an amount of flow into the upper head region which results in an upper head fluid temperature that is between the cold leg temperature and the core outlet temperature. Since there will be more voiding in the plants with the hotter upper head regions, these are considered to be the limiting case. For these plants, Westinghouse conservatively assumed that the initial temperature of the fluid in the upper head region of the reactor vessel was equal to the core outlet temperature. Thus, in their analyses of loss of coolant transients with a loss of offsite power, void formation is assumed to occur in the upper head region whenever the RCS pressure drops to the saturation pressure corresponding to the intial core outlet temperature.

For Westinghouse plants with the reactor coolant pumps running, the flow into upper head region is from the upper downcomer through the spray holes. The flow out of the upper head region is downward through the guide tubes into the upper plenum region. If the reactor coolant pumps are stopped, this flow into the upper head slows, stops, and then reverses direction. This is because the water in the core is heated by the decay heat, so it has a lower density than the cold leg water in the downcomer. Thus without the reactor coolant pumps operating, the hot, low-density water in the core is buoyed up through the guide tubes into the upper head region. This hotter water increases the potential for creating voids. Thus a loss of offsite power with the consequential loss of the reactor coolant pumps will increase the amount of void formation in the upper head region.

To make the results of these analyses valid for all Westinghouse-designed 2, 3, and 4 loop plants, Westinghouse evaluated the variations in (1) thermal inertia of the upper head region (2) the power level to upper plenum volume ratio, and (3) the guide tube/spray nozzle flow path resistance. The analyses showed that the thermal inertia of the upper head region is largest for the highest power (3411MWth) 4 loop plant with an inverted top hat upper support plate, so this was modeled in the WFLASH program. It was also determined that the power level to upper plenum volume ratio was essentially the same for all 2, 3, and 4 loop plants and that the guide tube/spray nozzle flow path resistance is less in the 2 and 3 loop plants. From these evaluations Westinghouse concluded that the results of the transients analyses for steam voiding on a 4 loop 3411 MWth plant with an inverted top hat upper support plate bound those for all Westinghouse plants.

Steam voids can be created in the upper reactor vessel by either decreasing the pressure below the saturation pressure at the prevailing fluid temperature (i.e., a depressurization event) or increasing the temperature of the water above the saturation temperature. For all of the anticipated transients, including those where the temperature of the water is increased, Reference 5 states:

> "Previous analyses performed for preparation of --- safety analyses reported in plant licensing documentation explicitly account for void formation in the upper head region if it is calculated to occur. The results of the previous analyses indicate no safety concerns are associated with this possibility since voids generated in the upper head would be callapsed when they are brought in contact with the subcooled region of the system."

III. EVALUATION

Westinghouse has had the capability for calculating the effects of steam voids in reactor coolant systems since the FLASH program (Reference 6) was first developed in 1966. However, this program was too time consuming for large scale problems such as the calculation of voids in upper reactor vessels during transients. By 1969 Westinghouse had developed FLASH-4 (Reference 7) which, with the more rapid calculating ability provided by an implicit formation, did allow the calculation of voids in reactor vessels.

The ability to calculate voids was carried into LOFTRAN programs by greatly reducing the velocity of a fixed fraction of the flow, i.e., by creating a "dead volume".

Based on this knowledge and the availability of these computer programs we agree that the analyses performed for the anticipated transients reported in the licensing documentation of the Westinghouse plants listed in Table 1 account for the effects of void formation in the reactor coolant systems.

IV. CONCLUSION

The staff concludes that the voids generated in the reactor coolant systems of these Westinghouse plants during anticipated transients are accounted for in present analysis models. Furthermore, based on transient analyses performed by Westinghouse using these models, the staff further concludes that this steam void will not result in unacceptable consequences during anticipated transients in any of these Westinghouse plants listed in Table 1.

TABLE 1

Plant Name	
Beaver Valley 1	Prairie Island 1
Cook 1	Prairie Island 2
Cook 2	Robinson 2
Farley 1	Salem 1
Farley 2	Salem 2
Ginna	San Onofre 1
Haddam Neck	Sequoyah 1
Indian Point 2	Surry 1
Indian Point 3	Surry 2
Kewaunee	Trojan
McGuire 1	Turkey Point 3
North Anna 1	Turkey Point 4
North Anna 2	Yankee Rowe
Point Beach 1	Zion 1
Point Beach 2	Zion 2

REFERENCES

1. U.S. NRC, IE Bulletin No. 79-06A, "Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident", April 14, 1979.

2. Check, P. S. "Void Formation in Vessel Head During St. Lucie Natural Circulation Cooldown Event of June 11, 1980, dated August 12, 1980.

3. U.S. NRC, "Clarification of TMI Action Plan Requirements"; NUREG-0737; page II.K.2.17-1, dated November, 1980.

4. U.S. NRC, "Natural Circulation Cooldown (Generic Letter No. 81-21)", dated May 5, 1981.

5. Jurgensen, R. W.; "St. Lucie Cooldown Event Report"; WOG-57; April 20, 1981.

6. Margolis, S. G. and Redfield, J. A.; "FLASH: A Program for Digital Simulation of the Loss-of-Coolant Accident"; WAPD-TM-534; May 1966.

7. Porsching, T. A. et.al.; "FLASH-4: A Fully Implicit Fortran IV Program for the Digital Simulation of Transients in a Reactor Plant"; WAPD-TM-840; March 1969.