50-329

MAR 08 1982

MEMORANDUM FOR: Brian Sheron, Chief
Reactor Systems Branch
Division of Systems Integration

FROM: Carl H. Berlinger, Chief
Core Performance Branch
Division of Systems Integration

SUBJECT: RETURN TO POWER IN MIDLAND STEAM LINE BREAK

The Core Performance Branch has prepared the enclosed evaluation of the applicant's response to the request for further information on departure from nucleate boiling during the subcritical return to power following the steam line break. We conclude that departure form nucleate boiling does not occur.

Carl H. Berlinger, Chief
Core Performance Branch
Division of Systems Integration

Enclosure:
As stated

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ENCLOSURE

Return to Power in the Midland Steam Line Break Analysis

In response to the staff's concerns regarding possible Departure form Nucleate Boiling (DNB) during the subcritical return to power following the large steam line break event in Midland the applicant submitted a generic analysis which was performed for 205 assembly Babcock and Wilcox plants. The Reactor Physics Section of the Core Performance Branch has reviewed the submittal. Our evaluation follows.

The procedure used in the analysis consisted of the use of a systems code using a point kinetics core model to obtain the average core response during the transient. The core power, inlet enthalpy, pressure, moderater temperature and total coolant flow were then used as input to a three-dimensional FLAME 3 model calculation which included a stuck rod. The radial and axial power distribution calculated by FLAME 3 was then used with the CHATA and TEMP computer codes to obtain the hot channel DNBR value. The minimum hot channel DNBR was 1.7 at return to power.

We conclude that departure form nucleate boiling does not occur during subcritical return to power in the Midland reactor. This conclusion is based on the following:

1. The systems code overpredicts the maximum power reached during the transient. The reactivity feedback due to cool down is overpredicted because no account is taken of voids formed in the vicinity of the peak assembly.

2. Conservative values are used for the moderator temperature coefficient and steam generator inventory.

3. The FLAME 3 calculation assumes equilibrium between prompt and delayed neutron power distributions which tends to increase power peaking.

4. The generic analysis applies to Midland since the important core parameters for 177FA and 205FA are nearly the same for the two plants.