January 19, 1979

THREE MILE ISLAND COMMISSION
[illegible] EXHIBIT 2
DATE 8/22/79

MEMORANDUM FOR: N. C. Moseley, Director, Division of Reactor Operations Inspection, IE

H. D. Thornburg, Director, Division of Reactor Construction Inspection, IE

FROM: James G. Keppler, Director, RIII

SUBJECT: RECOMMENDATION FOR NOTIFICATION OF LICENSING BOARDS AND REQUEST FOR TECHNICAL ASSISTANCE (AITS F30468H2)

The enclosed inspector memorandum dated January 8, 1979, with enclosures, identifies several potential problems which are being or will be pursued at Davis-Besse 1 which appear to be generic to B&W plants. In addition to the items identified in the memorandum, an issue (described in enclosed Action Item AITS F30385H2) concerning GDC 17 which was recently resolved at Davis-Besse 1 could possibly be common to other plants under review by NRR (e.g., Davis-Besse 2 and 3). The GDC 17 item and some of the other items may only be generic to B&W plants having Bechtel as the architect-engineer. We are aware that some of the items have been previously identified and dispositioned at other plants.

In accordance with the inspector's recommendation, RIII supervision has reviewed the materiality and relevancy of these matters to all pending cases before Boards involving B&W as the NSSS supplier. Based on information we have on those cases (Davis-Besse 2 and 3, Midland 1 and 2, Greene County, Three Mile Island 2), guidance given in MC 1530, and a liberal interpretation of the MC 1530 words "...any new information that could reasonably be regar ed as putting a new or different light upon an issue before the Board or as raising a new issue", RIII believes NRC policy dictates that the information be forwarded to all sitting Boards for cases involving B&W as the NSSS supplier. To our knowledge, none of the information relates to specific issues under consideration in the pending hearings. RIII does not know the significance of the information as it may affect current staff positions.

8001200003

			Norelius 1/18/79	Roy 1/18/79	Keppler 1/18/79
RIII					
Streeter 1/18/79	Knop 1/18/79	Tambling 1/18/79	Vandel 1/18/79	Jones 1/18/79	Hayes 1/18/79

Although RIII believes that NRC policy as described in MC 1500 dictates the transmittal of this information to sitting Boards, RIII questions the appropriateness of doing so. It would seem that a more effective and less premature way of handling this information would be for NRR to review and disposition the information during the development of the SER and SER Supplement relating to OL issuance for the affected plants. In the case of Three Mile Island 2 and other operating plants where the SER and SER Supplement have already been issued, the information could be evaluated for application to those plants as an NRR generic review task.

For your information, listed below is the status of reviews at Davis-Besse 1 of the items in the inspector memorandum:

Item 1 - During a recent inspection the licensee was requested to provide information to reconcile the apparent inconsistency between the FSAR statement on fuel assembly net holddown force and the administrative requirement to place restrictions on starting the fourth reactor coolant pump. This information will be available February 1979.

Item 2 - We have been following the licensee's efforts to determine the magnitude of the power oscillations. To date the maximum oscillations have been approximately 1.5% and do not appear to present a safety problem.

Item 3 - The pressurizer level question is presently the subject of communications between NRR and the licensee. We have not addressed the possibility that Tcold and makeup instrumentation do not meet GDC 17.

Item 4 - To our knowledge, this problem has not developed at DB 1. We plan to inspect this item in February 1979.

Item 5 - In response to an item of noncompliance, the licensee is developing criteria for detector substitution when the reactor is operated with incore strings out of service.

Item 6 - To our knowledge, this problem has not developed at DB 1. We plan to inspect this item in February 1979.

RIII will use the results of any technical reviews conducted which relate to items in the inspector memorandum to disposition the items as they relate to Davis-Besse 1. By copy of this letter, the Assistant Directors for Technical Programs and Field Coordination are requested to provide RIII with answers to the following questions:

Assistant Director for Technical Programs

1. Has NRR generically determined that the B&W core lift problem is not an unreviewed safety question?

2. Has NRR generically determined that the B&W power oscillation problem is not an unreviewed safety question?

3. Does the failure of Tcold and makeup instrumentation to follow the transient constitute a GDC 13 problem?

Assistant Director for Field Coordination

1. Is there a need to develop standard B&W technical specifications for continued plant operations with failed incore detector strings?

2. Is there a need to develop standard B&W technical specifications for restrictions on starting a fourth reactor coolant pump below certain temperatures?

For your convenience, the items in the inspector memorandum have been retyped on separate pages. If you need additional information please contact J. S. Creswell (387-9311) or J. F. Streeter (387-9223) of my staff.

James G. Keppler
Director

Enclosures:
1. Memorandum from J. S. Creswell to J. F. Streeter, dtd, 1/8/79
2. Retyped excerpts (6) from the 1/8/79 memorandum
3. Memorandum from J. F. Streeter to R. W. Woodruff, dtd, 6/9/78

cc: w/enclosures
E. L. Jordan, IE
S. E. Bryan, IE
J. S. Creswell, RIII

POOR ORIGINAL

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

7120160

January 8, 1979

Docket No. 50-300/501
50-329/330

MEMORANDUM FOR: J. F. Streeter, Chief, Nuclear Support Section 1

FROM: J. S. Creswell, Reactor Inspector

SUBJECT: CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2

During the course of my inspections at Davis-Besse, certain issues have come to my attention which I am submitting for consideration for forwarding to the Atomic Safety and Licensing Board which has proceedings pending for the aforementioned facilities. This submittal is made pursuant to Regional Procedure 1330A (November 16, 1978), step 3 and information supplied to me per step 1. The issues for consideration are:

1. During a recent inspection at Davis-Besse Unit 1 information has been attained which indicates that at certain conditions of reactor coolant viscosity (as a function of temperature) core lifting may occur. The licensee informed the inspector that this issue involves other B&W facilities. The Davis-Besse FSAR states in Section 4.4.2.7:

   > The hydraulic force on the fuel assembly receiving the most flow is shown as a function of system flow in Figure 4-39. Additional forces acting on the fuel assembly are the assembly weight and a hold down spring force, which resulted in a net downward force at all times during normal station operation.

   The licensee states that there is a 500°F interlock for the starting of the fourth reactor coolant pump. However, no Technical Specification requires that the pump be started at or above this temperature. A concern regarding this matter would be if assemblies moved upward into a position such that control rod movement would be hindered.

2. Inspection Report 50-346/78-06, paragraph 4, reported reactivity - power oscillations in the Davis-Besse core. These oscillations have also occurred at Oconee and are attributed to steam generator level oscillations. B&W report BAW-10007 states in A5.2:

7904200627

POOR ORIGINAL

The OTSG laboratory model test results indicated that periodic oscillations in steam pressure, steam flow, and steam generator primary outlet temperatures could occur under certain conditions.

It was shown that the oscillations were of the type associated with the relationships between feedwater heating chamber pressure drop and tube nest pressure drop, which are eliminated or reduced to levels of no consequence (<u>no feedback to reactor system</u>) by adjustment of the tube nest inlet resistance. As a result of the tests, an adjustable orifice has been installed in the downcomer section of the steam generators to provide for adjustment of the tube nest inlet resistance and to provide the means for elimination of oscillations if they should develop during the operating lifetime of the generators. The initial orifice setting is chosen conservatively to minimize the need for further adjustment during the startup test program.

We also note that the effect on the incore detector system for monitoring core parameters during the oscillations is not clear.

3. Inspection and Enforcement Report 50-346/78-06 documented that pressurizer level had gone offscale for approximately five minutes during the November 29, 1977 loss of offsite power event. There are some indications that other B&W plants may have problems maintaining pressurizer level indications during transients. In addition, under certain conditions such as loss of feedwater at 100% power with the reactor coolant pumps running the pressurizer may void completely. A special analysis has been performed concerning this event. This analysis is attached as Enclosure 1. Because of pressurizer level maintenance problems the sizing of the pressurizer may require further review.

   Also noted during the event was the fact that Tcold went offscale (less than 520°F). In addition, it was noted that the makeup flow monitoring is limited to less than 160 gpm and that makeup flow may be substantially greater than this value. This information should be examined in light of the requirements of GDC 13.

4. A memo from B&W regarding control rod drive system trip breaker maintenance is attached as Enclosure 2. This memo should be evaluated in terms of shutdown margin maintenance and ATWS considerations particularly in light of large positive moderator coefficients allowable with B&W facilities.

5. Inspection and Enforcement Report 50-346/78-17, paragraph 6 refers to inspection findings regarding the capability of the incore detector system to determine worst case thermal conditions. The reactor can be operated per the Technical Specifications with the center incore string out of service. If the peak power locations is in the center of the core (this has been the case at Davis-Besse), factors are not applied to conservatively monitor values such as Fq and F delta H.

6. Enclosure 3 describes an event that occurred at a B&W facility which resulted in a severe thermal transient and extreme difficulty in controlling the plant. The aforementioned facilities should be reviewed in light of this information for possible safety implications.

J. S. Creswell
Reactor Inspector

Enclosures: As stated

cc w/o enclosures:
G. Fiorelli
R. C. Knop
T. N. Tambling

POOR ORIGINAL

POOR ORIGINAL

Attachment 5