



NUCLEAR REGULATORY COMMISSION
 REGION III
 797 ROOSEVELT ROAD
 GLEN ELLYN, ILLINOIS 60127

January 8, 1979

Docket No. 50-500/301
 50-500/330

MEMORANDUM FOR: J. F. Sreeter, 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 submission is made pursuant to Regional Procedure 1530A (November 16, 1976), 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 obtained which indicates that at certain conditions of reactor coolant viscosity (as a function of temperature) core blocking may occur. The licensee informed the inspector that this issue involves other B&W facilities. The Davis-Besse B&W status is Section 4.4.2.7:

The hydraulic force on the fuel assembly receiving the flow is shown as a function of system flow in Figure 4-10. 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 reactor 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 50-345/78-06, 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 B&W-10027 states in 4.9.1:

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The OTSC 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 relationship between feedback bearing chamber steam flow and tube nest pressure drop, which are eliminated or reduced to levels of no consequence (no feedback to reactor system) 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 generator 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 eliminate 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 oscillation is not clear.

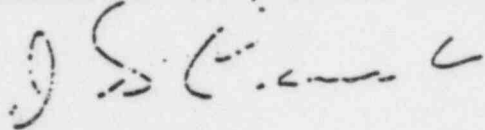
3. Inspection and Enforcement Report 30-316/78-35 documented that pressurizer level had gone offscale for approximately five minutes during the November 29, 1977 loss of offsite power event. There are some indications that other B&W plants may have problems maintaining pressurizer level fluctuations 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 Inclosure 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 320°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 QOC 13.

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

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5. Inspection and Enforcement Report SO-220/74-17, paragraph 6 refers to a separate finding regarding the capability of the flame detector system to determine some case thermal conditions. The detector can be operated per the technical specifications with the burner burner going out of service. If the peak power location is in the center of the core (this has been the case at Davis-Besse), factors are not applied to conservatively adjust values such as T_Q and T_{deliv} .
6. Enclosure 3 describes an event that occurred at a BWR 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. Crestall
Reactor Inspector

Enclosures: As stated

cc w/o enclosures:

G. M. ...

R. C. ...

T. W. ...