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

MAR 28 1979

MEMORANDUM FOR: Dudley Thompson, Executive Officer for Operations Support, IE

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, IE

SUBJECT: NOTIFICATION OF LICENSING BOARDS

On February 23, 1979, six items concerning Babcock and Wilcox designed nuclear plants were sent to you for forwarding to the appropriate licensing boards. At that time only a preliminary evaluation had been done. We have completed our evaluation of each of the items and that information is enclosed. This additional information should be forwarded to the licensing boards.

Norman C. Moseley
Director
Division of Reactor
Operations Inspection, IE

Enclosure:
Evaluations of Concerns

cc: S. E. Bryan
E. L. Jordan
R. F. Heishman, RIII
J. C. Stone
D. Kirkpatrick
~~G. C.~~ Gower
V. D. Thomas

CONTACT: J. C. Stone
(x28019)

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

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.

DISCUSSION AND EVALUATION

The potential for core lifting in B&W plants is a concern which has been previously reviewed by NRR. The concern was first raised in connection with the Oconee 2 and 3 reactors, where the primary coolant flow rates were found to be in excess of the design flow rates. For example, the Unit 2 flow rate was found to be 111.5% of the design flow rate. Since this was very near the predicted core lift flow rate of 111.9%, an analysis was done by B&W to determine what effect core lifting would have on the previous safety analysis for these plants. This analysis (dated May 2, 1975) indicated that the potential for core lifting did not result in an unreviewed safety question. A subsequent review of this B&W analysis by NRR also concluded that an unsafe condition did not exist (letter from R. A. Purple to Duke Power, dated 9/24/75). It should be noted that the potential vertical displacement of the core is limited to a very small distance by the upper core support structure. Core lifting at power would result in a slight reduction in reactivity since the rising fuel would tend to engage the withdrawn control rods to a slightly greater extent than it would in the bottomed condition. The amount of this change in reactivity is, of course, available for reinsertion should the fuel settle back to its original position. The potential reactivity increase caused by the settling of the 16 centrally located control rod assembly elements (assumed to have been subject to lifting in the Oconee 2 reactor) was calculated to be 0.1% Δ K/K. This value is insufficient to have much effect on the accident and transient safety analyses.

An additional concern was the potential for damage to the fuel assembly and fittings which might be caused by fretting due to repetitive fuel movement. Consequently, Duke Power was requested by NRR to make certain examinations of the Oconee 2 fuel during the first refueling to confirm that fuel element motion was not occurring. The results of this examination (letter from W. O. Parker to R. C. Rusche dated 7/21/75) showed that no fuel lifting or other type of motion had occurred during the first cycle of operation.

After the core lift concern was identified, B&W developed newer types of fuel holddown springs which provide more margin against core lifting than the previous springs did. It is our understanding that the newer types of springs have been installed in all B&W reactors.

For these reasons, we believe that there is presently little likelihood that core lifting will occur during normal power operation. At lower temperatures, there is an increased flow induced lifting force on the fuel due to the higher viscosity of the reactor coolant. Consequently, we view the restriction against 4 pump operation below 500°F as a prudent precaution against fuel fretting. However, since the potential for core lifting has little safety significance and because critical operation below 500°F is not permitted, we have no basis to recommend including this restriction in the Technical Specifications.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

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-10027 states in A9.2:

> 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 (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 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.

DISCUSSION AND EVALUATION

Power Oscillations of the order of 1.5% of full power have been observed at all of the Oconee plants and are considered normal. In 1977 the power oscillations experienced by the Oconee 3 reactor increased to a maximum of 7.5% of full power. At that time the problem was reviewed by NRR with the conclusion that there was no significant safety consideration at that value (Note to B. C. Buckley from S. D. MacKay, dated January 27, 1978). It should be noted that the 7.5% power oscillations cause about a 1°F oscillation in core average temperature due to the short period of the oscillations. The important core safety parameters, which are, the departure from nucleate boiling ratio and the average maximum linear heat generation rate are affected very little by oscillations of this amplitude. The primary cause of the power oscillations is believed to be a fluctuation of the secondary water level in the steam generators. This can be minimized by increasing the flow resistance in the downcomer region of the steam generators. The corrective effort at Oconee 3 was complicated by the fact that the orifice plate provided for this purpose could not be fully closed.

However, the oscillations at other B&W plants have been kept to about 1.5% of full power by appropriate adjustment of the downcomer flow resistance. For these reasons, the power oscillations at B&W plants are not considered to be a significant safety concern.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1978, FROM J.S. CRESWELL TO J.F. STREETER

3. Inspection and Enforcement Report 50-346/77-06 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.

DISCUSSION AND EVALUATION

The event at Davis Besse which resulted in loss of pressurizer level indication has been reviewed by NRR and the conclusion was reached that no unreviewed safety question existed.

The pressurizer, together with the reactor coolant makeup system, is designed to maintain the primary system pressure and water level within their operational limits only during normal operating conditions. Cooldown transients, such as loss of offsite power and loss of feedwater, sometimes result in primary pressure and volume changes that are beyond the ability of this system to control. The analyses of and experience with such transients show, however, that they can be sustained without compromising the safety of the reactor. The principal concern caused by such transients is that they might cause voiding in the primary coolant system that would lead to loss of ability to adequately cool the reactor core. The safety evaluation of the loss of offsite power transient shows that, though level indication is lost, some water remains in the pressurizer and the pressure does not decrease below about 1600 psi. In order for voiding to occur, the pressure must decrease below the saturation pressure corresponding to the system temperature. 1600 psi is the saturation pressure corresponding to 605°F, which is also the maximum allowable core outlet temperature. Voiding in the primary system (excepting the pressurizer) is precluded in this case, since pressure does not decrease to saturation.

The safety analysis for more severe cooldown transients, such as the loss of feedwater event, indicates that the water volume could decrease to less than the system volume exclusive of the pressurizer. During such an event, the emptying of the pressurizer would be followed by a pressure reduction below the saturation point and the formation of small voids throughout much of the primary system. This would not result in the loss of core cooling because the voids would be dispersed over a large volume and forced flow would prevent them from coalescing sufficiently to prevent core cooling. The high pressure coolant injection pumps are started automatically when the primary pressure decreases below 1600 psi. Therefore, any pressure reduction which is sufficient to allow voiding will also result in water injection which will rapidly restore the primary water to normal levels.

For these reasons, we believe that the inability of the pressurizer and normal coolant makeup system to control some transients does not provide a basis for requiring more capacity in these systems.

General Design Criterion 13 of Appendix A to 10 CFR 50 requires instrumentation to monitor variables over their anticipated ranges for "anticipated operational occurrences". Such occurrences are specifically defined to include loss of all offsite power. The fact that T cold goes off scale at 520°F is not considered to be a deviation from this requirement because this indicator is backed up by wide range temperature indication that extends to a low limit of 50°F. Neither do we consider the makeup flow monitoring to deviate since the amount of makeup flow in excess of 160 gpm does not appear to be a significant factor in the course of these occurrences.

The loss of pressurizer water level indication could be considered to deviate from GDC 13, because this level indication provides the principal means of determining the primary coolant inventory. However, provision of a level indication that would cover all anticipated occurrences may not be practical. As discussed above, the loss of feedwater event can lead to a momentary condition wherein no meaningful level exists, because the entire primary system contains a steam water mixture.

It should be noted that the introduction to Appendix A (last paragraph) recognizes that fulfillment of some of the criteria may not always be appropriate. This introduction also states that departures from the Criteria must be identified and justified. The discussion of GDC 13 in the Davis Besse FSAR lists the water level instrumentation, but does not mention the possibility of loss of water level indication during transients. This apparent omission in the safety analysis will be subjected to further review.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

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.

DISCUSSION AND EVALUATION

Our investigation of the above circuit breaker problems has revealed that eight failures of reactor scram circuit breakers to trip during test have been reported from Babcock & Wilcox (B&W) type operating facilities since 1975. In each case, the faulty circuit breaker was identified as a GE type AK-2 series (i.e., AK-2A-15, 24, or 50). The causes for failure were attributed to either binding within the linkage mechanism of the undervoltage trip device (UV) and trip shaft assembly or an out-of-adjustment condition in the same linkage mechanism. B&W and GE determined that the binding and the out-of-adjustment conditions resulted from inadequate preventive maintenance programs at the affected operating facilities.

In addition to the breaker problems experienced at the B&W facilities, three circuit breakers of the aforementioned GE type failed in similar fashion at the Oyster Creek operating facility on November 26, 30, and December 2, 1978. As in each case above, cleaning and relubricating of the UV/trip shaft assembly within the circuit breaker was required to correct the problem. It is significant to note that during the November 30, 1978 event, both redundant service water pump circuit breakers failed to trip as required during the loss of off-site power test. These failures in turn created a potential overload condition on the emergency busses during the sequential bus loading by each diesel generator.

However, both diesel generators successfully picked up their required bus loads without experiencing a unit shutdown from an overload condition. With respect to the generic implications and safety significance of this issue, both B&W and GE are in the process of issuing alert letters to their customers. These letters are scheduled for issuance by late March and will describe the causes for failure and provide recommendations to resolve the problem.

Based on our study findings and on information obtained in discussions about the breaker problem with the knowledgeable people from B&W, GE and Region II, we plan to issue an IE Circular covering the matter. The thrust of the Circular will be directed toward the need for adequate preventive maintenance programs at all operating facilities. Specific recommendations from GE to resolve the above breaker problem will also be mentioned in the Circular.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

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 F_Q and F delta H.

DISCUSSION AND EVALUATION

We do not believe that there is a valid basis for requiring the center string of incore detectors to be always operable in B&W reactors.

The power distributions for various plant conditions, throughout the fuel cycle, are calculated prior to the operation of the reactor. The power distribution is verified at the beginning of operation, and periodically thereafter, by comparison with the available incore detectors. The power in fuel assemblies that lack detectors (including those with failed detectors) is derived by using the known power distribution to determine the power ratios between such an assembly and nearby assemblies that have detectors. These ratios can then be multiplied by the power in the measured assemblies to derive the power level in any specific unmeasured assembly. The central assembly is not fundamentally different than any other assembly in this respect. Although this assembly is the highest powered assembly in the Davis Besse reactor at the beginning of the fuel cycle, this is not the case at all reactors. Nor does the central assembly have the highest power, in the Davis Besse reactor, at the end of the first fuel cycle. Since there is some variation between the calculated power distributions and the actual ones, an appropriate margin is assumed for this variation in establishing the allowable power peaking factors.

Fixed incore detectors must function in an extremely harsh environment and are subject to high failure rates. In order to ensure that an adequate number will survive the fuel cycle, many more detectors are installed than are necessary for the power distributions determinations. To require the central string to be always operable would likely result in unnecessary power restrictions. Neither the standard Technical Specifications (STS) for B&W plants nor the STS for CE plants (which also have fixed incore detectors) require the central detectors to be operable.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

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.

DISCUSSION AND EVALUATION

Following the cooldown transient at Rancho Seco, NRR evaluated the event and concluded that no structural damage had occurred to the primary coolant system which would preclude future operation of Rancho Seco. However, in their safety evaluations they concluded that positive steps should be taken to preclude similar transients and that the generic implications of this event should be reviewed. In addition, IE initiated a Transfer of Lead Responsibility, Serial No. IE-ROI 78-04, dated April 25, 1978, recommending that:

1. NRR perform a generic review of the non-nuclear instrumentation power supplies for other B&W units, if design changes to the non-nuclear instrumentation (NNI) power supplies are required at Rancho Seco.

2. NRR evaluate the susceptibility of B&W plants to other initiating events or failures which could cause similar significant cooldown transients.

This event is currently being evaluated by NRR.

8/18/9 12N

Barbara:

Ms. Goldfrank accepted the 7 boxes (from Harrisburg) which accompanied this letter of which one also addressed to Stan Gorinson.

Simone

SHAW, PITTMAN, POTTS & TROWBRIDGE
1800 M STREET, N. W.
WASHINGTON, D. C. 20036

AUG 17 1979
3

(202) 331-4100

TELECOPIER
(202) 296-0694 & 296-1760

TELEX
89-2693 (SHAWLAW WSH)
CABLE "SHAWLAW"

JOHN H. SHARON
EDWARD B. CROSLAND
COUNSEL

August 17, 1979

Ms. Barbara Jorgenson
President's Commission on the
Accident at Three Mile Island
2100 M Street, N.W.
Washington, D.C. 20037

Dear Barbara:

By letter to me of August 7, 1979, Mr. Gorinson requested that we provide to you a response to Commissioner Pigford's question of Mr. Arnold, during the May 30 hearing, regarding receipt and distribution of NRC Inspection and Enforcement Division's Report 50-346/78-06. I am informed that this report was not received by any of the GPU System companies prior to the May 30 hearing. After the hearing we obtained a copy from NRC's Public Document Room in Washington.

Mr. Arnold is preparing a response to the Commission which will confirm the substance of this letter and address as well communications the Company received of Davis-Besse events in September and November, 1977, which have received considerable attention since the TMI-2 accident. Mr. Arnold's letter should be completed next week.

Sincerely,

Ernie Blake

Ernest L. Blake, Jr.

cc: Stanley M. Gorinson, Esq.

on the Accident at Three Mile Island
2100 M Street, NW Washington, DC 20037

August 7, 1979

Mr. Ernest Blake, Esq.
Shaw, Pittman, Potts & Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

Dear Mr. Blake:

During Mr. Robert Arnold's testimony before the President's Commission on the Accident at Three Mile Island on May 30, 1979, Commissioner Thomas Pigford asked him to supply the Commission with information concerning the receipt and distribution of a certain 1978 report from the Nuclear Regulatory Commission's Office of Inspection and Enforcement.

Commissioner Pigford was interested in learning whether General Public Utilities and Metropolitan Edison were aware of I&E report #50-3467806 which conveyed "new information to licensing boards" concerning Davis-Besse Unit 1 and 2 and Midlands Units 1 and 2. He asked that Mr. Arnold detail who at GPU and Met Ed were aware of the report, when they became aware of it, and how the information was communicated to others in the organizations. For your convenience, I have attached the transcript pages at which the request appears.

Please forward your response by August 17 to the attention of Barbara Jorgenson at the letterhead address. If you have any questions, please contact me at (202) 653-7660.

Sincerely,

Stanley M. Gorinson

Stanley M. Gorinson
Chief Counsel

cc: Barbara Jorgenson
Commissioner Paul Marks

/ew

bcc: Commissioner Pigford
V. Johnson
W. Rockwell