Metropolitan Edison Company
Post Office Box 480
Middletown, Pennsylvania 17057
717 944-4041

August 30, 1979

John G. Kemeny, Chairman
President's Commission on the
Accident at Three Mile Island
2100 M Street, N.W.
Washington, D.C. 20037

Dear Chairman Kemeny:

During the Commission hearing held on May 30, 1979, a number of questions were asked regarding problems encountered at the Davis-Besse nuclear power plant prior to the accident at Three Mile Island (TMI). Other questions in the same hearing related to documentation of certain events at Davis-Besse, and to our knowledge or awareness of the Davis-Besse problems through receipt of such documentation. I was asked specifically to determine from our corporate records whether we received NRC Inspection Report 50-346/78-06, covering the results of NRC Inspection and Enforcement Divisions inspections conducted at Davis-Besse (Tr. 57-58).

Inspection Report 50-346/78-06 (Enclosure 1) was issued in April, 1978, presenting results of inspections conducted at Davis-Besse in December, 1977 and March, 1978. The report covers a number of subjects, including a loss of pressurizer level indication which was experienced during a transient at Davis-Besse on November 29, 1977. The discussion of loss of pressurizer level indication, due to volumetric contraction of water with reduced temperature ("shrink") following the reactor trip in November, appears on pages 2 and 3 under the section "Further Review of Reactor-Turbine Trip with Loss of Offsite Power".

As we have indicated to the Commission staff in prior correspondence, we did not receive a copy of Inspection Report 50-346/78-06 prior to the May 30 hearing; after the hearing, we obtained a copy from the NRC's Public Document Room in Washington.

We have performed a further review of the official notices that we received or could have received regarding the November 29, 1977 Davis-Besse transient. We regard such official notices to include materials sent by the NRC or by the nuclear steam supply (NSSS) vendor (in this case Babcock & Wilcox) ("B&W"). This review provides considerable insight into both the substance and format of incident reporting available to us prior to the TMI-2 accident.

The existence of the April, 1978 inspection report is mentioned in the May, 1978 issue of the NRC's Monthly Operating Units Status Report (NUREG 0020) (the "Grey Book") on page 2-30 (Enclosure 2). However, the brief summary of

8001230374

of this inspection report contained in the Grey Book makes no reference to the November 29, 1977 transient at Davis-Besse. The Davis-Besse transient is indirectly referenced in a letter dated February 2, 1979, from B&W to Met-Ed (Enclosure 3) requesting information on incidents of loss of pressurizer level indication. The B&W letter transmitted to Met-Ed earlier correspondence between B&W and an NRC inspector in which the NRC sought information on events at all B&W plants which resulted in loss of pressurizer level indication due to shrink. In response to B&W's letter, we provided information on two occurrences where pressurizer level indication had been lost at TMI-2 (Enclosure 4), and we sent a representative to attend a B&W meeting on loss of pressurizer level indication held on February 13, 1979, in Lynchburg, Virginia. At the February meeting, B&W presented the results of a 1975 analysis of loss of pressurizer level indication at Arkansas Unit 1. Based on the comments at this meeting, no action was deemed necessary for the TMI plants. To our knowledge, no other utility present at the February B&W meeting took action as a result of that meeting.

Other possible sources of information from B&W have been reviewed, but none have been found that make reference to the November 29, 1977 transient. No mention of it is made in the minutes of B&W Owner's Group meetings. And, while Met-Ed receives weekly Operating Plant Service Bulletins from B&W, the description of the November 29, 1977 incident included in these bulletins (where it is incorrectly identified as taking place on November 30) (see Enclosure 5) makes no reference to loss of pressurizer level; in any event, the brief nature of the discussions included in these bulletins renders them of little value in promoting understanding of complex transients.

The last document I have enclosed with reference to the November 29, 1977 transient is a letter dated March 29, 1979, and distributed by the NRC to Atomic Safety and Licensing Boards and parties to pending NRC licensing proceedings, including Met-Ed (Enclosure 6). This document contains a number of attachments, all of which are internal NRC communications or excerpts thereof, and none of which were received by us in any form prior to their distribution with the March 29 letter to Licensing Boards. Included among those attachments is the January 8, 1979, internal NRC memorandum referred to at the May 30, 1979, Commission hearing. Needless to say, we had no access to the information contained in these documents prior to the TMI-2 accident.

Recognizing the Commission's general interest in the notice that we may have received of problems experienced at other plants and its particular interest in the event experienced at the Davis-Besse plant in September, 1977, we have tracked as well the various reports of that event through and including our receipt of the reports.

Enclosures 7 and 8 are Licensee Event Report NP-32-77-16 and a Supplement thereto ("the Supplement") submitted to the NRC by the Davis-Besse licensee (Toledo-Edison) following the incident that occurred at Davis-Besse on September 24, 1977. As the Licensee Event Report (LER) reflects, that incident involved a trip of the Steam and Feedwater Rupture Control System (SFRCS). This trip of the SFRCS increased primary system pressure causing the pressurizer power relief valve to open. This valve then failed to close resulting in reduction in primary pressure and subsequent safety features actuation. Initial operator response and action was similar to that at TMI-2. Met-Ed never received a copy of the LER or the Supplement. Generally, Met-Ed has not received LERs regarding any plants outside the GPU System.

A copy of this LER might have been received in the GPU Service Corporation's offices in New Jersey through an informal arrangement whereby the Edison Electric Institute collects LERs of member utilities and periodically distributes them to its members for their information. We have been unable to determine, however, from our files or those of EEI whether this particular LER was sent or received.

We have now determined that notice of the September 24, 1977 event at Davis-Besse and the ensuing LER was available to us through several sources. One source is the NRC's Monthly LER PWR Listings, dated December 8, 1977 (Enclosure 9), where information on the Davis-Besse September 24, 1977 event appears in a nine-line summary on page 39. This document was received by Met-Ed and routed through the Licensing Department at Reading and the Training Department at TMI. The Training Department reviewed the report as to the need for possible guidance to the operating staff. No action was deemed necessary.

A second source of information on the September 24 event at Davis-Besse was through the November, 1977 issue of the NRC's Grey Book (Enclosure 10), where a one paragraph description appears on page 2-28. A follow-up item appeared in the December, 1977 volume of the Grey Book on page 2-28 (Enclosure 11). These documents were available in the Met-Ed Generation Library in Reading and in the Training Department at TMI, but the Davis-Besse incident was not identified as requiring action.

A third possible source of information appeared on pp. 2-3 of the December, 1977 issue of a publication titled "Current Events - Power Reactors" from the NRC (Enclosure 12). This discussion, while far more detailed than that contained in the above two NRC monthly reports, did not mention the operator action at Davis-Besse — throttling the high pressure injection pumps — which was an important aspect in our accident. Additionally, the Davis-Besse transient is discussed under a section on valve malfunctions, not operator error. Thus, the organization of the document does not readily direct one's attention to operating procedures. Finally, while Met-Ed was to receive this publication, the issue that appears as Enclosure 12 cannot be located among company records, suggesting that this particular issue may not have been received. This same discussion was reprinted in a commercially circulated publication known as the Atomic Energy Clearing House in its January 9, 1978 issue on pp. 17-18 (Enclosure 13). The Atomic Energy Clearing House is received both in Reading and at TMI; copies at TMI are circulated to certain individuals with directions to take note of specific items. Though this document was circulated to Met-Ed and TMI personnel, the Davis-Besse event was not identified as requiring action at TMI. (We should note that the monthly LER PWR Listings are also typically reprinted in the Clearing House document.)

Finally, a commercially published newsletter from Nuclear Power Experience, Inc. (NPE) discussed the Davis-Besse transient. This document is received monthly in the offices of GPU Service Corporation in New Jersey, and is not circulated to Met-Ed personnel. The July, 1978 issue carried a summary of the Supplement (Enclosure 14), which provided considerable detail on the Davis-Besse event, including the operator's throttling of high pressure injection. Among the dozens of items reported on each monthly NPE issue, some reactor events are identified by the publisher as "Alert" items, meaning that they are believed significant from "outage causing, generic, safety, etc. standpoints" (Enclosure 15). The Davis-Besse transient, which appears as Item 95 on page 4 of the table of contents for the July, 1978 issue, was not identified as an "Alert" item.

With respect to the NSSS vendor, Met-Ed representatives at the B&W Owner's Group meetings have no record of the September, 1977 event being mentioned at those meetings. At one B&W User's Group Meeting, there was cursory mention of the event; the enclosed minutes of such meeting (Enclosure 16) confirm that it was given only limited attention (see page 11). Finally, the September 30, 1977 issue of the B&W Operating Plant Service Bulletin contains a two paragraph report on the event (Enclosure 17).

Other than the enclosed reports, there are no other publications regarding the September 1977 event through which we would have been informed. A table summarizing the information on the September, 1977 event contained in each of the sources available to Met-Ed or GPUSC is attached as Enclosure 18.

The documents enclosed with this letter are intended to provide the Commission with an illustration of our actual notice of information on certain operating problems encountered at other plants. We chose the 1977 Davis-Besse events because of their relationship to the accident at Three Mile Island and the attention these particular events have received.

I do not mean to suggest through this letter that information was not available in addition to the documentation we can now determine we received of these events. All LERs and inspection reports are at least available in the NRC's Public Document Room in Washington. However, it is not more information on every event which occurs at every nuclear power plant that is needed by utilities. What is needed is an interpretation and prioritization of the vast number of items to which every licensee is now exposed regarding both insignificant and significant problems experienced at other plants.

Our company's review of these materials did not indicate the real significance of the September, 1977 event nor its applicability to TMI operations. Nor did the reviews of the same materials by other utilities or industry consultants cause them to analyze the Davis-Besse events any more closely or imaginatively than we did. In fact, as the record of the hearings before this Commission vividly reveals, only two individuals may have grasped the full nature of the September, 1977 Davis-Besse transient, the indications on plant conditions that the transient provided to the operators, the operator action based on those indications that took place at Davis-Besse, and the potential effect that such operator action could have on the actual, but unrecognized plant conditions. Moreover, these two individuals who may have gained an understanding of the significance of the Davis-Besse event (Mr. Dunn of B&W and Mr. Creswell of NRC) apparently did so after focusing on that transient for an extended period of time; they did not gain their appreciation from the review of reports on that event from the myriad of other event reports available in the literature.

While we plan to intensify our efforts to review all documentation we receive of transients and events that occur at other plants, the potential for real benefits in our review would be greatly enhanced by effective presentation of the information we receive. Met-Ed would welcome the Commission's recommendations in this area of nuclear industry communications.

Sincerely,

RCArnold

R. C. Arnold

RCA:bas
Enclosures

cc: S. Gorinson, Esq.
Ms. B. Jorgenson

ENCLOSURE 1

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION III

Report No. 50-346/78-06

Docket No. 50-346 License No. NPF-3

Licensee: Toledo Edison Company
Edison Plaza
300 Madison Avenue
Toledo, OH 43652

Facility Name: Davis-Besse Nuclear Power Station, Unit 1

Inspection At: Davis-Besse Site, Oak Harbor, OH

Inspection Conducted: December 6-8, 1977 and March 6-8, 1978

Inspector: [illegible] 4/19/78

Accompanying Personnel: E. A. Connaughton

Approved By: W. S. Little, Chief 4/19/78
Nuclear Support Section

Inspection Summary

Inspection on December 6-8, 1977 and March 6-8, 1978 (Report No. 50-346/78-06)

Areas Inspected: Routine, unannounced inspection of startup testing, natural circulation performance, and followup associated with rod drop incident which occurred on December 4, 1977. The inspection involved 35 onsite inspector-hours by one NRC inspector.

Results: Of the three areas inspected, no items of noncompliance were identified in two areas. One item of noncompliance was identified in regard to startup testing. (Infraction - Failure to follow procedures - Paragraph 6)

POOR ORIGINAL

Dup

7905110094

DETAILS

1. Persons Contacted

*T. Murray, Station Superintendent
*L. Stalter, Technical Engineer
*D. Green, Administrative Coordinator
J. Lingerfelter, Nuclear and Performance Engineer

The inspector also talked with and interviewed other licensee employees, including members of the technical and operations staffs.

*Denotes those attending the exit interview.

2. Further Review of Reactor-Turbine Trip with Loss of Offsite Power

On November 29, 1977, a reactor trip occurred due to a short in a patch panel used for startup testing.[1] Since the reactor was cooled with the reactor coolant pumps tripped, the licensee desired to use data accumulated during the event to support the conclusion that sufficient natural circulation capability exists. The inspector reviewed charts, data and logs associated with the event in order to ascertain whether the data and conditions under which the event occurred allowed an accurate determination of natural circulation capability.

In the exit interview the inspector stated that based upon his review, the data did not adequately support the requirements of the natural circulation test. Subsequent to this the licensee presented the data to NRR in a meeting held February 7, 1978. By letter from R. S. Boyd to L. E. Roe dated February 16, 1978, NRR informed the licensee that the natural circulation would have to be performed per their commitment in the FSAR. This letter did allow a deferment of 120 days before the test had to be performed.

During the review, the inspector noted that the pressurizer level indication had gone offscale for approximately 5 minutes, and the minimum pressurizer level was not known. The licensee later furnished the inspector with a calculation that the level fell approximately 9 inches below the lower level sensing tap. This

1/ IE Inspection Rpt No. 50-346/77-34.

calculation assumed Loop 1 hot leg temperatures were the same as Loop 2 hot leg temperatures. Actual data for Loop 1 hot leg temperatures were not available. Review of RCS pressure data associated with the reactimeter showed several pressure variations on the order of 50-100 psi during the event. The licensee maintains the indications are erroneous due to faulty instrumentation. Discussions with a reactor operator and review of control room strip charts tend to confirm the licensee's position that the readings were faulty.

No items of noncompliance or deviations were identified.

3. Review of Dropped Rod Bank Event

On December 4, 1977, at 10:33 p.m., safety rod groups 1 and 3 dropped into the core without explanation.[2] Generator output prior to the event was approximately 140 MWe. Power after the rod drop was approximately 50 MWe.

At 9:51 p.m., the licensee had successfully completed auto transfer on "A" and "B" safeguards busses simultaneously. This transfer was accompanied by a computer alarm, "Any Trip Device B/D Tripped."

In addition, the CRD programmer indicated a fault. Electronic technicians were immediately dispatched to determine the cause of the alarms, but they could not determine the source.

The operator recalled seeing safety groups 1 and 3 dropping. When the safety groups dropped, rod groups 6 and 7 withdrew from 25% and 41% withdrawn. Using a rod speed of 30 in/min, it can be inferred that it took the operator about 40 seconds to take manual control of the control rods after the safety groups dropped.

It was determined after the rod drop that Technical Specifications require that the reactor must be placed in Mode 3 in 6 hours. Generator power was decreased to 10 MWe and the turbine output breakers tripped at approximately 11:15 p.m. Rod groups 5, 6, and 7 were then inserted and the reactor taken into the hot shutdown mode and a shutdown margin calculation performed.

After the event, the rod control system vendor was contacted. Personnel from the company reviewed the event and examined the

2/ LER NP 33-77-102, dated 12/27/77.

circuitry, but could not determine the cause for the dropped rods.

No items of noncompliance or deviations were identified.

4. Review of Reactivity Coefficient Determination at Power

The inspector reviewed data and logs associated with Test No. TP 800.05.1 which was performed November 29, 1977. Review of the reactimeter traces revealed reactivity oscillations of a roughly sinusoidal form. The period of the variations was approximately 4 seconds and the amplitude was approximately 1.5 pcm. Reactor power during the test was 40%.*

The licensee attributes the oscillations to fluctuations in steam generator level and to the resulting variations in water temperature and density occurring in the downcomer region of the core. A vendor representative estimated the effect to be less than 1°F variation in T cold temperature. This representative states that similar oscillations have been noted at another of the vendor's sites and that the oscillations diminish as power increases.

Because of the oscillations, the reactivity data must be corrected to remove the oscillating component. Figure 1 shows the behavior of the reactivity trace during the movement of control rod groups 6 and 7. Groups 6 and 7 movement is measured incrementally by reed switch outputs.

During the conduct of the test, the reactor coolant system Tave was lowered 5°F. This resulted in BTU limits being received on the steam generators. Since this was unexpected, the licensee did a setpoint calculation for the BTU limit. This analysis showed that the value programmed into the calculation for reactor coolant flow ($67 \times 10^6$ lb/hr/loop) was probably in error. The licensee stated that when full power operation is achieved, the setpoint program will be properly calibrated. In order to avoid the BTU limits problem, the licensee issued a temporary procedure change to lower the Tave setpoint 2°F instead of 5°F. Further review of the test will await data reduction and take place in a future inspection. This matter is considered unresolved.

5. Incore Detector System Review

A review of TP 800.24, Incore Detector Testing, and ST 5033.03, Incore Instrument Channel Calibration, was performed on December 7 and 8, 1977. From this review, several questions have arisen.

*Test performed at 75% power with Tave increased 5°F show an almost complete damping of the oscillations.

Section 7.3 of TP 800.24 under "Acceptance Criteria" states, Normalized power corrected SPND value plots for each string are reasonable and consistent with respect to similar symmetrical strings and/or general flux shapes as determined by NPI per 6.8 (5033.03)." There is some uncertainty as to what constitutes "reasonably and consistent", and therefore, whether or not 7.3 (TP 800.24) is definitive enough to be considered an acceptance criteria.

ST 5033.03 was written and is performed to satisfy Technical Specification 4.3.3.2, calling for a channel calibration which does not include the neutron detectors, to be performed at least once per 18 months. ST 5033.03 is essentially a check of the software used to analyze detector signals and a comparison between detectors located in similar symmetrical strings. The "Acceptance Criteria" for this comparison is as described in the previous item. ST 5033.03 is not clearly a channel calibration by definition.

Information concerning the treatment of background detector signals, now deleted from the incore data analysis, was unavailable at the time of this inspection.

Technical information on the aluminum oxide insulated detectors currently in use as part of the incore detector system was unavailable at the time of this inspection.

These items are considered unresolved at this time, pending further investigation.

6. Rod Drop Testing at 40% Power

The inspector interviewed personnel and reviewed records associated with rod drop testing at 40% power. The following sequence of events along with comments was developed.

10/25/77 - Pretest meeting notification

11/26/77 - Pretest checklist completed
Checklist items include:
- Procedure review complete and approved
- Procedure available
- Pretest deficiency list prepared

- Pretest deficiencies resolved per Administrative Directive 1801.04, Resolution of Test Deficiencies

11/27/77 - TP 0800.29, Dropped Control Rod Test, run first time.

Procedure Phase I Prerequisite 6.2.7 states that the axial power shaping rods will be positioned to establish a near axial imbalance of near zero. Actual imbalance was on the order of - 10% at the start of the test.

(Phase II of test not performed on November 27, 1977)

Procedure Step 7.1.1 states to obtain data specified by the Core Power Distribution Procedure, TP 0800.11, Section 7.0. No record was found of Enclosure 1, Prerequisite and Procedure Signoff, for this step. This step was verified as completed.

Procedure Step 7.1.8 states to repeat Step 7.1.1. Again the Enclosure 1 data was not found for this step. This step was verified as completed.

Procedure Step 7.1.9 states to use Enclosure 1 to compare quadrant power tilt manual calculations with those generated by the computer. Enclosure 1 for this test was not found. This step was verified as completed.

Procedure Step 7.1.19 states to calculate minimum DNBR and the maximum LHR using a copy of Enclosure 2. A copy of Enclosure 2 was not found. This step was verified as completed.

The following is an entry in the chronological test log at 1400 hours: "Reviewing the data, we have found that the KW/ft criteria was not met. The problem appears to be in the initial rod position which resulted in approximately 71% WD position of 6/7 when 5-7 was at 0% WD. Ideally, the 6/7 position would have 78% (100% FP insertion limit), but due to small rod worth errors, it was not. The additional 6% on 6/7 will probably be sufficient to modify the flux distribution to meet the KW/ft criteria. The test will be rerun (only the 5-7 rod at 0% WD portion) with an initial rod position of approximately 72% on 6/7 which should put 6/7 at 78% WD when 5-7 is at 0% WD."



POOR ORIGINAL

Administrative Procedure AD 1801.04, Resolution of Test Deficiencies, defines deficiencies at "A deviation from an authorized requirement document in a component or system identified prior to a test, during a test, or during the review of test results that, were it to remain uncorrected, could adversely affect design and safe operation of the nuclear power plant at any time throughout the expected lifetime of the plant".

The extrapolated value of MLHR to 100% power yielded a value of 20.5 KW/ft. This was in excess of the procedure acceptance criteria of 20.19 KW/ft and the Technical Specifications safety limit basis of 20.4 KW/ft.

Section 7.1 of Administrative Procedure 1801.04 states that if during the analysis of the test data and the review of the test results it is determined that the acceptance criteria were not met, the deficiency shall be documented on Enclosure 1. Enclosure 1 is entitled "Deficiency Report." Not filing a Deficiency Report per AD 1801.04 is considered to be an item of noncompliance with Technical Specification 6.8.1.a.

Not notifying the NRC per the requirements of Technical Specification 6.9.1.8.i after the extrapolated value of MLHR exceeded 20.4 KW/ft is considered to be an unresolved item.

12/5/77 - Phase I of TP 0500.29 was rerun. When the test was performed, the test leader obtained the shift foreman's permission to run the test. The test leader checked off selected steps of the procedure to be performed as follows:

Steps 7.1.1, 7.1.2, 7.1.6, 7.1.7, 7.1.8, 7.1.10, 7.1.11, 7.1.16, 7.1.17, 7.1.18, and 6.1.20.

The steps performed were not verified and dated. Section 6.2, Phase I, Prerequisites, were not verified and dated. Not even a temporary procedure change accompanied the performance of the test to address the procedure modification. Performing the test without an adequately reviewed and approved procedure is an apparent item of noncompliance with Technical Specification 6.8.2.

Again, the test results failed the acceptance criteria in that the maximum linear heat rate extrapolated to 100% power was 21.1 KW/ft.

Again, there was no notification of the NRC per Technical Specification 6.9.1.8.i.

12/17/77- Test Deficiency Report filed by test leader.

Test Deficiency Report states "Extrapolated values of MLHR exceeded fuel melt limit when extrapolated to 100% FP. Values acceptable if extrapolated to trip setpoint of 75% plateau (85% FP)."

The recommended action was to request the reactor vendor to review extrapolation techniques to eliminate unnecessary conservatisms.

The responsible section head did not sign the Deficiency Report until February 17, 1978.

The Deficiency Report was not signed off as noted by the plant superintendent as of the date of the inspection.

On December 15, 1977, the test program manager requested vendor review and comments by issuing a Request for Review and Comments (No. 186). The request states that the enclosed test documentation includes no test deficiencies. A note at the bottom of the request states, however, "This is a preliminary review only. The complete package for review will be forwarded following resolution of test deficiency." On December 16, 1977, the vendor noted in their comments that the extrapolation of linear heat rate to 100% full power does not meet the acceptance criteria. However, an extrapolation to the next overpower trip setpoint (85% of full power) will meet the acceptance criterion and does not represent a safety concern for testing at the 75% plateau.

The vendor stated that they were presently evaluating conservatism in the extrapolation and would provide the resolution to this test deficiency prior to finishing the 75% testing. Until such resolution was provided, the test deficiency was a restraint to increasing the overpower trip setpoint above 85%.

Section II of AD 1EO91.04 states that the test program manager will maintain a log of the status of all test deficiencies. As of the date of the inspection, the deficiency associated with MLHR had not been logged by the test program manager.

On January 20, 1978, the vendor sent memo SOM No. 335 to TECo personnel which addressed revised hand calculations for Fq, $F_{\Delta H}$ and LHR. Salient points in the memo are as follows:

1. A radial local peak of 1.066 can be used based on our improved core models. (SOM 253 dated May 23, 1977, had described a radial local peak of 1.10)

2. The calculation of LHR should use the densification spike factor for the axial level where the peak occurs and not the factor for the 5 foot level.

Licensee personnel reviewed SOM 335 and requested additional information which was furnished in SOM 336 dated January 31, 1978. This memo explained "The hand calculation of LHR uses conservative factors which add a total of 25% correction to the calculation. Two of these factors (peak to average segment power = 1.04 and the radial local peak - 1.066) total 10.9% and are picked to conservatively to cover any power distribution.

The online computer curve/surface fit routines calculate these factors for the specific power distribution present, and hence, generally calculate numbers less than or equal to the hand calculation numbers.

On the first dropped rod test, the online computer calculated 3.8% as the combined factor of these two items; on the second test the computer calculated 2.4%."
The vendor concluded that the 10.9% factor could be replaced with the aforementioned values to calculate MLHR. The resulting calculation yielded a MLHR of 18.84 KW/ft for the test performed on November 27, 1977, and 19.10 KW/ft for the test performed on December 9, 1977.

SOM 336 also states future calculations of LHR should continued to use the conservative factors listed SOM 335, and if any further limits are exceeded, a detailed analysis of that power distribution will also be required.

7. <u>Review of Quadrant Power Tilting During Rod Drop Testing</u>

Review of online computer calculated tilts at the dropped rod C2 withdrawn condition on November 27, 1977, revealed the following:

| Quadrant | WX | XY | YZ | ZW |
|---|---|---|---|---|
| Incore | 13.504 | 12.474 | -9.6515 | -14.297 |
| Out-of-Core | 1.839 | 10.233 | 5.7556 | -15.912 |

Tilts observed at the 50% withdrawn condition on November 27, 1977, were:

| Quadrant | WX | XY | YZ | ZW |
|---|---|---|---|---|
| Incore | 8.6494 | 7.589 | -6.2598 | -9.9785 |
| Out-of-Core | .5981 | 6.2744 | 4.5303 | -11.404 |

A review of the incore versus out-of-core values for tilt reveal significant differences in quadrants WX and YZ. The reason for these differences will be examined in a future inspection. It was also noted that out-of-core channel 8 was reading approximately 10% lower in power than the actual power level. These matters are considered unresolved.

8. <u>Unresolved Items</u>

Unresolved items are matters about which more information is required in order to ascertian whether they are acceptable items, items of noncompliance or deviations. Unresolved items are identified in Paragraphs 4, 5, 6, and 7.

9. <u>Exit Interviews</u>

The inspector met with licensee representatives (denoted in Paragraph 1) on December 8, 1977, to summarized the findings of the inspection. The following items were discussed:

a. Review of the November 29, 1977, event with regards to natural circulation capability. The inspector requested that the test be conducted as per the procedures. (Paragraph 2)

b. Review of the December 4, 1977, safety groups rod drop. (Paragraph 3)

c. Review of reactivity coefficient at power determination. (Paragraph 4)

d. Review of incore system data. (Paragraph 5)

The following commitments were made by the licensee:

a. Faulty instrumentation associated with reactor coolant system pressure monitoring will be investigated and repaired if necessary.

b. An analysis will be provided to determine the minimum pressurizer level attained during the November 29, 1977, event.

c. That an analysis of the November 29, 1977, event with regards to natural circulations capabilities will be forwarded to NRR for review. This analysis will include a detailed description of the conditions surrounding the event.

d. That further monitoring of the rod control system would be explained in the licensee event report concerning the December 4, 1977, event.

On March 8, 1978, an additional exit interview was held with the following item discussed:

The performance of rod drop testing at the 40% power plateau.

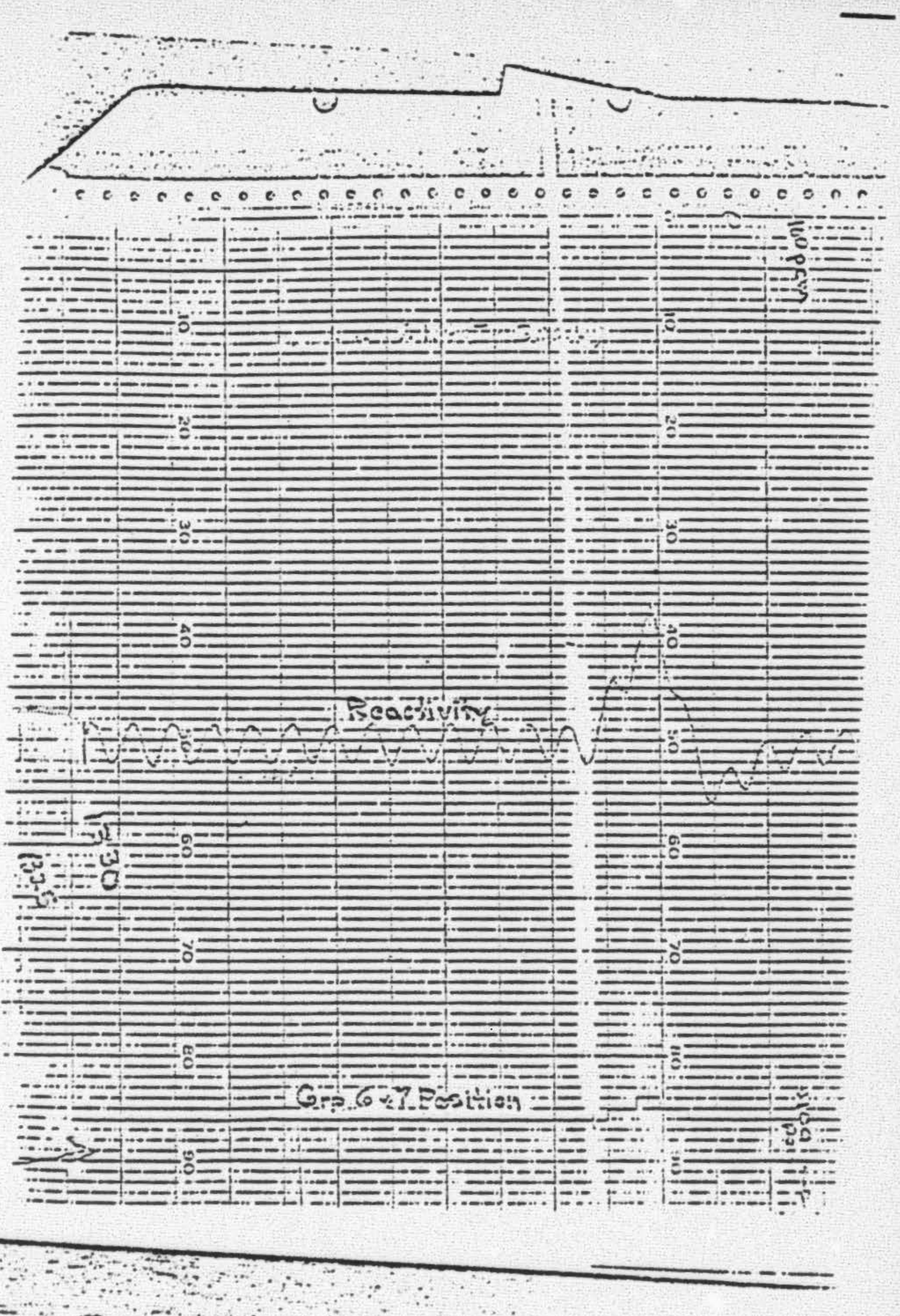
a. Lack of notification of NRC regarding MLHR limits. (Paragraph 6)

b. Lack of filing a Deficiency Report after acceptance criteria was exceeded. (Paragraph 6)

c. Lack of proper review and approval of procedure used during testing which took place on December 9, 1977. (Paragraph 6)

In reply to the item concerning the notification of the NRC, the licensee stated they felt that the axial imbalance experienced during the test represented an overly conservative condition when compared to conditions that would be experienced at 100% power. With regard to filing a Deficiency Report after the first test, the licensee agreed that a mistake was made. The licensee also concurred that management controls over the test run on December 9, 1977, were not as normally exercised and that more emphasis on management controls would be more for exercised for further testing.

Attachment:
Reactivity Chart


Reactivity
Grp 6 & 7 Position

ENCLOSURE 2

NUREG 0020
VOL. 2 NO. 5
MAY 1978

# OPERATING UNITS STATUS REPORT

DATA AS OF 4-30-78



## LICENSED OPERATING REACT

### DATA FOR DECISIONS

- Department Of Energy
- Nuclear Regulatory Commission



ERRATA SHEETS FOR CORRECTIONS TO HIGHLIGHTS SECTION – MARCH, 1978 DATA (NUREG-0020, VOL. 2, NO. 4, APRIL, 1978)

# DAVIS-BESSE 1

NRC POWER RESTRICTIONS: [illegible]

## FACILITY DATA

[illegible]

POOR ORIGINAL

## INSPECTION STATUS

### ENFORCEMENT STATUS

SUMMARY OF TECHNICAL SPECIFICATION 6.8, PROCEDURES WERE NOT FOLLOWED. [illegible]

7. OTHER SIGNIFICANT ITEMS

A- SYSTEMS AND COMPONENT PROBLEMS:

* MAIN CONDENSER TUBES WILL BE STAKED DURING APRIL 7, 1978, OUTAGE TO ELIMINATE A VIBRATION PROBLEM. (78-0[illegible])

B- FACILITY ITEMS/PLANS AND PROCEDURES:

NONE.

C- MANAGERIAL ITEMS:

NONE.

LAST IE SITE INSPECTION DATE: DECEMBER 6-8, 1977; MARCH 6-10 AND 21-23, 1978

* INSPECTION REPORT NO: 78-06 AND 78-07

8. SUMMARY

INSPECTION SUMMARY

* INSPECTION ON DECEMBER 6-8, 1977 AND MARCH 6-8, (78-06): ROUTINE, UNANNOUNCED INSPECTION OF STARTUP TESTING, NATURAL CIRCULATION PERFORMANCE, AND FOLLOWUP ASSOCIATED WITH [illegible] INCIDENT WHICH OCCURRED ON DECEMBER 4, 1977. THE INSPECTION INVOLVED [illegible] INSPECTOR HOURS ONSITE BY ONE NRC INSPECTOR. OF THE THREE AREAS INSPECTED, NO ITEMS OF NONCOMPLIANCE WERE IDENTIFIED IN TWO AREAS. ONE ITEM OF NONCOMPLIANCE WAS IDENTIFIED IN REGARD TO STARTUP TESTING (FAILURE TO FOLLOW PROCEDURES).

* INSPECTION ON MARCH 8-10 AND 21-23, (78-07): ROUTINE, UNANNOUNCED INSPECTION OF MAINTENANCE ACTIVITIES, FOLLOWUP ON CIRCULARS, PART 21 REPORT AND OTHER OPEN ITEMS. THE INSPECTION INVOLVED [illegible] INSPECTOR HOURS ONSITE BY ONE NRC INSPECTOR. NO ITEMS OF NONCOMPLIANCE OR DEVIATIONS WERE IDENTIFIED.

PLANT STATUS

* LICENSEE INITIATED SHELF PUMP OPERATION EARLY APRIL FOR [illegible] OF [illegible] PENDING RESOLUTION OF BURNABLE POISON ROD PROBLEM DISCOVERED AT CRYSTAL RIVER. PLANNED APRIL OUTAGE POSTPONED UNTIL APRIL 28, 1978, TO INCORPORATE ANY NECESSARY CORRECTIVE ACTION ON THE BURNABLE POISON RODS.

## REPORTS RECEIVED FROM LICENSEE

| NUMBER | DATE OF EVENT | DATE OF REPORT | SUBJECT OF REPORT | SUBJECT | PUBLIC ANNOUNCEMENT |
|---|---|---|---|---|---|
| 78-025 | 03/[illegible]/78 | 03/[illegible]/78 | 30 DAY REPORT | UNIDENTIFIED RCS LEAKAGE GREATER THAN 1 GPM. | [illegible] |
| | 04/[illegible]/78 | 04/[illegible]/78 | PART 21 REPORT | [illegible] | [illegible] |
| 78-026 | 03/[illegible]/78 | 04/[illegible]/78 | 30 DAY REPORT | [illegible] | [illegible] |
| 78-027 | 03/[illegible]/78 | 04/[illegible]/78 | 30 DAY REPORT | FAILURE OF [illegible] | [illegible] |
| 78-028 | 03/[illegible]/78 | 04/[illegible]/78 | 30 DAY REPORT | [illegible] INDICATOR INOPERABLE. | [illegible] |
| 78-029 | 04/05/78 | 04/[illegible]/78 | [illegible] DAY REPORT | BURNABLE POISON RODS. | [illegible] |
| 78-030 | 03/22/78 | 04/[illegible]/78 | 30 DAY REPORT | CONTAINMENT [illegible] INOPERABLE. | [illegible] |
| 78-031 | 03/[illegible]/78 | 04/[illegible]/78 | 30 DAY REPORT | CONTROL ROOM VENTILATION [illegible] CHANNEL [illegible] | [illegible] |
| 78-032 | 03/[illegible]/78 | 04/[illegible]/78 | 30 DAY REPORT | [illegible] CHANNEL [illegible] PLACED IN TEST TRIP BYPASS. | [illegible] |
| 78-033 | 03/[illegible]/78 | 04/20/78 | 30 DAY REPORT | STEAM GENERATOR 1-2 LEVEL LIMIT EXCEEDED. | [illegible] |
| 78-0[illegible] | 03/[illegible]/78 | 04/21/78 | 30 DAY REPORT | [illegible] INOPERABLE. | [illegible] |

REVIEWED BY OLSPA LOEB: M.R. Beebe DATE: 05/10/78

REVIEWED BY NRR DOR: ABEngle DATE: 05/1?/78

ENCLOSURE 3

Babcock & Wilcox

Power Generation Group
P.O. Box 1260, Lynchburg, Va.
Telephone (804) 384-5111

February 2, 1979
TMI-79-16

Mr. J. F. Hilbish
Manager, Licensing
Metropolitan Edison Company
Post Office Box 542
Reading, PA 19603

Subject: Three Mile Island Nuclear Station
Loss of Pressurizer Level Indication

Reference: NRC Letter, J. E. Foster to J. H. Taylor, dated 1/31/79

Dear Mr. Hilbish:

B&W has received the attached request for information concerning the loss of pressurizer level indication at B&W plants from the NRC Region III office. The Region III office is requesting this information based on their feeling that a loss of pressurizer level indication violates Criterion 13 - Instrumentation and Control, which states, "Instrumentation shall be provided to monitor variables and systems over their anticipated range for normal operations, for anticipated operational occurrences, and for accident conditions as appropriate to ensure adequate safety....."

Because of the tight schedule of their request, we would appreciate your prompt response to this request. If you choose to supply the requested information for your plant(s), it would be desirable for the information to be in our hands by February 8, 1979, so we can compile the information and develop the proper strategy.

Present plans call for a strategy meeting on Tuesday, February 13, 1979, in Lynchburg to discuss the concern for utilities that plan to attend the February 14, 1979, meeting.

Please contact us by February 8 if you plan to attend and indicate whether you will investigate past occurrences at your plant.

Very truly yours,

Joel T. Canin
Service Manager

JTC/mb
Attachment
cc: [illegible] M. Klingaman L. L. Lowe
D. F. Miller J. L. Snellinger
J. D. Lagan C. H. Montgomery
L. C. Rogers
J. [illegible] Smith



POOR ORIGINAL