CRESWELL

94

MEMO TO PERSONAL FILE

On Friday, March 16, 1979 (from 9:00 to 11:30 a.m.) C. Norelius, L. Spessard, J. Streeter, J. Foster, T. Tambling, J. Kohler and J. Cresswell met to discuss the findings of a recent investigation at Davis-Bessie, including possible items of noncompliance.

C. Norelius opened with a general comment on the investigation, and comments on the purpose of the meeting: to resolve differences of interpretation on the findings of the investigation and clarify remaining issues.

A package representing major points found was followed throughout the presentation by Kohler and Foster. This package is attached.

Discussions were held on several points, including information gathered, issues not covered, technical significance of the two areas investigated. J. Cresswell indicated that he had passed his concerns regarding Loss of Pressurizer Level Indication (LOPLI) on to the ASLB.

Mr. Cresswell posed several questions during the discussions, including:

1. Were the investigators certain that the previous instances of LOPLI had been reviewed by the NRC?
Answer: Yes, although no report had been made of the Arkansas events, each of the transients had been reviewed, and LOPLI was present during the events. In some cases, it may be that no mention of LOPLI was specifically noted.

2. Had the investigators reviewed the performance of the Once Through Steam Generators during the November transient? This was from a remark in paragraph 3 of a memo demonstrating timliness of TECO LOPLI evaluation.
Answer. No. Not within the investigation's scope.

3. Had anyone questioned the "sanctity" of the thirty-second ECCS criteria during the investigation?
Answer: No. Not within scope of the investigation or expertise of the investigators.

4. Had any part of the TECO procedure for initiating an FCR been violated during issue and implementation of FCR's related to the undervoltage relay setpoint?
Answer: Not investigated. Not within scope of the investigation.

5. Had the investigators considered memos from Mr. Buck, which advised of outstanding FCR's?
Answer: Yes, and the memos will be documented in the investigation report.

6. Would it be useful for J. Smith to review the July 18, 1978 sumbittal to NRR which resulted in the approval of Ammendment #7?
Answer: No. NRR has reviewed and approved the submittal.

Inclusion of an item of noncompliance related to a lift wire procedure which violated a Technical Specification was discussed. This issue was not included in the investigation as no new information was required to establish non-compliance.

Following long discussion, it was agreed that no items of noncompliance be cited with the investigation report, but concerns as to TECO's management would be contained in the report transmittal letter.

PAQ

Mr. Creswell attempted to excuse himself approximately 3/4 of the way through the meeting. He stated that "this has become a management discussion now....I have some memos to write." Mr. Streeter indicated that he should stay until the conclusion of the meeting.

Throughout the meeting, Mr. Cresswell's attitude suggested that he thought that the issues he had raised had not been properly investigated in that the investigation had not been widened to other issues. Questions he raised on OTSGs, FCR procedures, ect., appeared to be grasping at straws to find some significance to the issues he had raised.

On a more personal level, and on the basis of statements Mr. Cresswell has made to me over several weeks, the following opinions are made:

1. He will not give up his"concerns" even in the face of overwhelming evidence that they are not significant.

2. He will pursue any aspect of the issues raised which he can, within the regular inspection program. I refer specifically to performance of OTSG's during the November transient AND FCR PROCEDURES.

3. He has several other concerns related to Davis-Bessie which were not given to the investigators. He will continue with these, and object to any resolution of these issues.

4. I feel that his attitude borders on paranoia, and that he will become increasingly distrustful of regional management. I believe that he now feels that I was "soft" on TECO, and so was Joel Kohler.

5. If the progressive distrust of regional personnel proceeds unabated, he will complain to higher levels of NRC and finally "go public" perhaps with the support of some anti-nuclear group. I hope that this does not occur, as I feel that he is technically competent, and sincere. However, his outlook on Davis-Bessie clouds his technical judgement.

6. Some concerns as to TECO management are well—founded. However, there is no evidence that they knowingly violated NRC regulations.

James E. Foster

March 19, 1979

5/2/78 (CORR.
(CORRECTED)

SCOPE

I. LOSS OF PRESSURIZER LEVEL INDICATION (LOPLI)

A. TIMELINESS OF EVALUATION OF LOPLI BY TECO FOLLOWING 11/29/77 TRANSIENT

B. POSSIBLE GENERIC OCCURRENCES OF LOPLI AT OTHER B&W PLANTS

INFORMATION TO BE GATHERED: 1. OTHER OCCURRENCES AT B&W PLANTS
2. CHRONOLOGY OF TECO EVALUATION

II. UNDERVOLTAGE RELAY SETPOINT

A. TIMELINESS OF IMPLEMENTATION OF TECH. SPEC. CHANGE

B. REASONS FOR ABOVE.

INFORMATION TO BE GATHERED: CHRONOLOGY OF EVENTS AND REASONS FOR SAME.

LOSS OF PRESSURIZER LEVEL INDICATION (LOPLI)

1. LOPLI HAS OCCURED AT OTHER B&W PLANTS AS NOTED:

ARKANSAS	10/74, 12/74, 5/75
THREE MILE ISLAND	4/78, 11/78
RANCHO SECO	74, 75, 78 (several)

2. THE ARKANSAS EVENT AND OTHERS HAD BEEN REVIEWED BY B&W (AND THE NRC).

3. THE B&W SIMULATOR IS PROGRAMMED TO LOOSE PRESSURIZER LEVEL UNDER CERTAIN CONDITIONS.

4. B&W STATED THAT LOPLI WAS AN OPERATIONAL "HEADACHE" BUT NOT A SAFETY PROBLEM, AS PRESSURIZER LEVEL CANNOT FALL BELOW A CERTAIN LEVEL WITHOUT A DROP IN THE RCS PRESSURE WHICH WILL INITIATE HPSI.

5. THE B&W ANALYSIS OF THE PHENOMENOM INDICATED THAT CORE COVERAGE WILL BE MAINTAINED AND ECCS WILL OCCUR AS ANALYZED.

6. B&W ADVISED TECO OF THE POSSIBILITY OF LOPLI DUE TO MAIN STEAM SAFETY RELIEF BLOWDOWN SETTINGS PRIOR TO POWER OPERATION. SETTINGS NOT IMPLEMENTED.

7. ON NOVEMBER 29, 1977, A REACTOR TRANSIENT RESULTED IN LOPLI AT DAVIS- BESSIE.

8. IN ADDITION TO STEAM SAFETY RELIEF VALVE SETTINGS BEING OUTSIDE OF RECOMMENDED SETTINGS, SIZE AND DESIGN OF THE AUXILIARY FEEDWATER PUMPS AT DAVIS-BESSIE CONTRIBUTED SIGNIFICANTLY TO LOPLI. THIS IS A DESIGN DIFFERENCE BETWEEN DAVIS-BESSIE AND OLDER B&W PLANTS. THE RESULTING LOPLI IS COMPOUNDED BY THIS DESIGN BUT THE CONCLUSION OF THE VENDOR IS THAT SYSTEM RESPONSE IS IDENTICAL.

9. DOCUMENTATION EXISTS THAT TECO DISCUSSED LOPLI WITH B&W SOON AFTER THE TRANSIENT AND CONCLUDED THAT NO SAFETY PROBLEM EXISTED. TECO WAS EXPLORING CORRECTIVE ACTION REGARDING AUXILIARY FEEDWATER PUMP FLOW CONTROL AS EARLY AS FEBUARY 10, 197[illegible]. NO NEW LOPLI ANALYSIS WAS PLANNED UNTIL INFORMATION WAS REQUESTED DURING AN RIII INSPECTION.

10. THE B&W ANALYSIS OF THE DAVIS-BESSIE TRANSIENT SHOWS THAT THE PRESSURIZER DID NOT VOID DURING THE EVENT. THE ANALYSIS IDENTIFIED ONE SPECIALIZED CASE WHERE THERE WOULD BE VOIDING OF THE PRESSURIZER IF SECONDARY STEAM PRESSURE FALLS BELOW <u>ANTICIPATED</u> VALUES.

11. NO ATTEMPT WAS MADE TO JUDGE THE TECHNICAL ADEQUACY OF THE ANALYSIS DESCRIBED BY B&W. QUESTIONS REGARDING THIS ANALYSIS SHOULD BE DIRECTED TO NRR.

12. NRR HAS REVIEWED THIS ISSUE AND DETERMINED THAT THERE IS NO SAFETY QUESTION. DISCUSSION WITH NRR ON MARCH 13, 1979, INDICATED THAT NRR HAS NO OPEN ITEMS ON THIS ISSUE.

13. THE INVESTIGATORS JUDGEMENT IS THAT TECO PERFORMED A PROPER REVIEW OF LOPLI. NO ITEMS OF NONCOMPLIANCE WERE INDICATED.

-DISCUSSION-

UNDERVOLTAGE RELAY SETPOINT (UVRS)

CHRONOLOGY:

7/27/76 MILLSTONE BULLETIN

10/1/76 UNDERVOLTAGE RELAYS INSTALLED IN PLANT (NOT ORIGIONAL DESIGN), SET TO 10 SECOND DELAY ON 90% UNDERVOLTAGE.

12/8/76 SER ISSUED, UNDERVOLTAGE STILL UNDER REVIEW

4/22/77 PLANT LICENSED. FOUR MONTH ALLOWANCE FOR FINAL REVIEW OF UNDERVOLTAGE IS LICENSE CONDITION. ADMINISTRATIVE PROCEDURES IN EFFECT FOR VOLTAGE. 10 SECOND DELAY CALIBRATED 4/11/77.

7/18/77 TECO SUBMITTAL TO NRR, ANALYSIS OF UNDERVOLTAGE. RELAY TO BE SET TO 9 SECOND DELAY TO COPE WITH CASE WHERE THERE IS A SA AND VOLTAGE IS LESS THAN 90%, MORE THAN 59% FOR MORE THAN NINE SECONDS. THIS WAS TO ELIMINATE THE POSSIBILITY OF REDUCED MARGINS FOR ECCS.

10/77 FCR 217 IMPLEMENTED RELAY SETPOINT AT NINE (9) SECONDS + or - 1.5. (ALLOWABLE RANGE OF 7.5 to 10.5 SECONDS).

10/5/77 FCR 217 IMPLEMENTED. RELAYS SET TO 9 SECONDS.

10/27/77 SUBMITTAL OF AMMENDMENT # 7. SETPOINTS CHANGED TO 7 + or- 1.5 SECONDS. THIS CHANGE WAS TO ASSURE THAT THE RELAYS WOULD ACTUATE WITHIN NINE SECONDS, INCLUDING A MAXIMUM .5 SECOND DRIFT (IN ACCORDANCE WITH REG GUIDE 1.105).

10/28/77 FCR 430 WRITTEN TO SET RELAYS AT 7 + or - 1.5 SECONDS, INSTALL PUSHBUTTON TO DEFEAT RELAY IF OPERATION PROBLEMS RESULT FROM RELAY SETPOINT CHANGE.

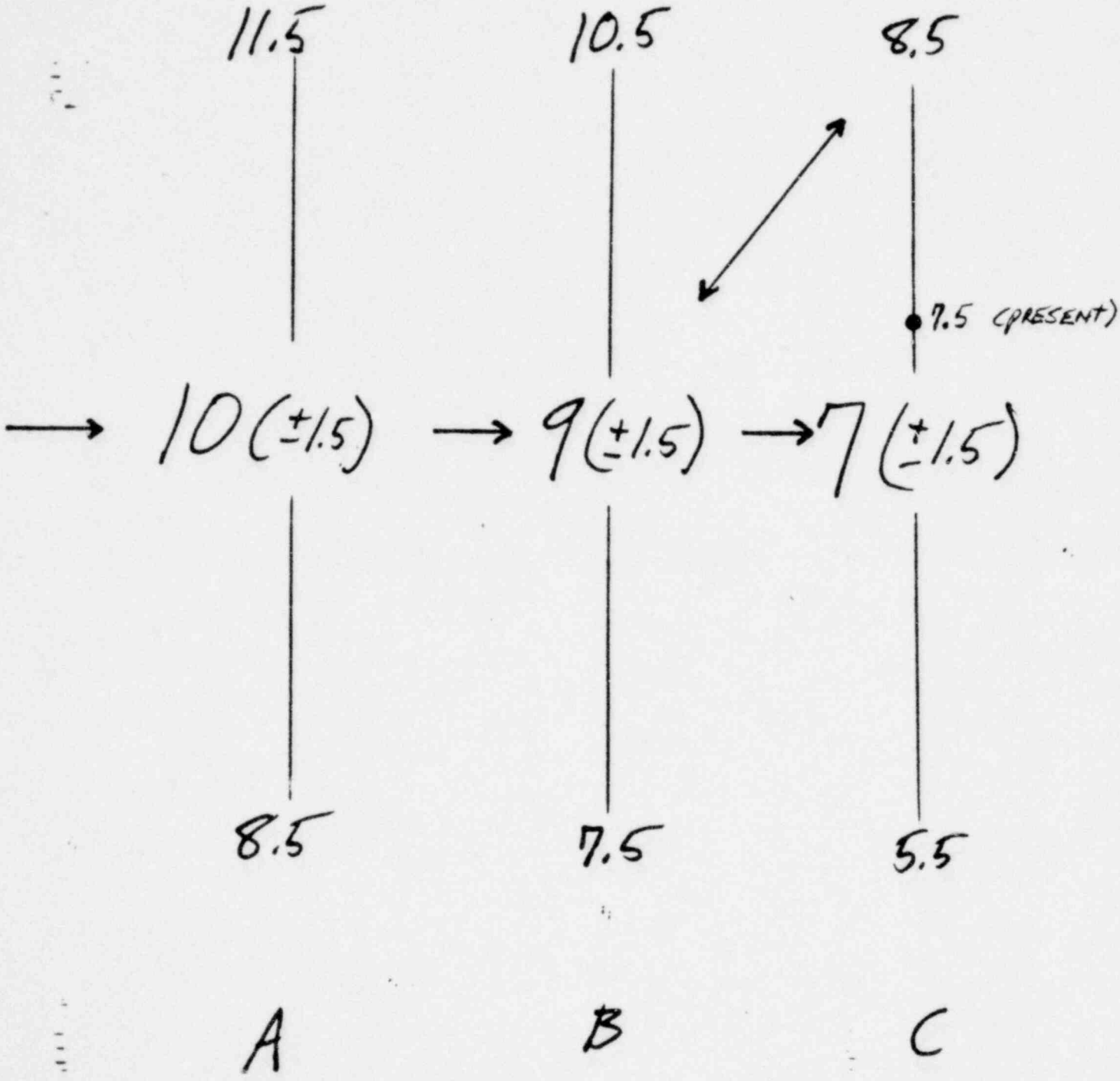
10/29/77 AMMENDMENT # 7 APPROVED. MODIFICATION: UNDERVOLTAGE RELAYS
REVISION: CHANGE RELAY SETPOINT
TECH. SPEC. CHANGE: 7 + or - 1.5 SECONDS

4/78 (approx) PLANT GOES TO COLD SHUTDOWN TO DEAL WITH FUEL PROBLEM.

6/12/78 SRB REVIEW OF FAILED SFAS TEST FINDS THAT RELAY SETPOINT CHANGE HAS NOT BEEN ACCOMPLISHED.

6/15/78 FCR 430 IS IMPLEMENTED, RELAYS SET TO 7 SECONDS.

6/23/78 LER -78-061 REPORTS DISCOVERY OF LACK OF RELAY SETPOINT CHANGE.



THE LICENSEE DID NOT IMPLEMENT AMMENDMENT # 7 FOR A PERIOD OF SEVEN MONTHS.

THE RESULT OF THIS FAILURE WAS AS FOLLOWS:

1. REGULATORY GUIDE 1.105, CONCERNING SETPOINT DRIFT WAS NOT FOLLOWED, WHICH WOULD ADD AN ADDITIONAL .5 SECOND MARGIN TO RESPONSE TIMES.

2. THE NINE SECOND TIME DELAY WAS IN EFFECT, BUT THE .5 SECOND DRIFT WAS NOT INCLUDED IN THE SETTING, AND COULD THEORETICALLY HAVE DRIFTED THE SETPOINT TO 9.5 SECONDS, THEREBY DELAYING ECCS BY .5 SECONDS UNDER CONDITIONS OF SA AND LOW VOLTAGE UNDER 90%, MORE THAN 59% FOR OVER 9 SECONDS. HOWEVER, THE LICENSEE REPORTED THAT THE AVERAGE RELAY SETPOINT WAS FOUND TO BE 8.99 SECONDS (LER-78-061).

3. THE TECHNICAL SPECIFICATIONS ALLOW + or - 1.5 SECOND RANGE FOR RELAY SETPOINT. BY NOT IMPLEMENTING AMMENDMENT # 7, UTILITY PERSONNEL COULD THEORETICALLY HAVE SELECTED A RELAY SETPOINT ABOVE NINE SECONDS. AS A WORST CASE, THIS WOULD HAVE PLACED THE RELAY SETPOINT 2 SECONDS FROM THAT REQUIRED. THIS 2 SECOND CASE INCLUDES A THEORETICAL .5 SECOND DRIFT IN A NONCONSERVATIVE DIRECTION. HOWEVER, THIS WAS NOT THE CASE, AS SETPOINT DOCUMENTATION INDICATED THAT THE RELAYS WERE SET FOR NINE SECONDS.

IN ORDER FOR THE LICENSEE TO FAIL TO MEET THE 30-SECOND INJECTION INTO THE VESSEL ASSUMING THE FSAR LOSS OF OFFSITE POWER TRANSIENT, THE FOLLOWING THINGS WOULD HAVE TO OCCUR SIMULTANEOUSLY:

1. LOCA
2. DEGRADED GRID VOLTAGE GREATER THAN 59% BUT LESS THAN OR EQUAL TO 90% ON THE INCOMING 4.16 KV BUSES FOR A TIME PERIOD OF GREATER THAN 9 SECONDS.
3. THE 90% RELAY SETPOINT WHICH WAS VERIFIED TO HAVE BEEN SET TO 9 SECONDS WOULD HAVE TO DRIFT NONCONSERVATIVELY TO 9.5 SECONDS.

THE NET RESULT OF THIS EVENT WOULD BE THAT ECCS WOULD BE DELAYED A MAXIMUM OF .5 SECONDS.

THE INVESTIGATORS NOTED THAT FOR ITEM 3, THERE IS AN EQUAL PROBABILITY THAT THE RELAYS WOULD DRIFT IN A CONSERVATIVE DIRECTION.

IT IS OUR OPINION THAT THE PROBABILITY OF ITEMS 1, 2, and 3 OCURRING SIMULTANEOUSLY IS OF A SUFFICIENTLY SMALL MAGNITUDE NOT TO REPRESENT A SAFETY CONCERN (ALL TIMES ARE FOR ANALYZED EVENTS WHOSE ACCURACY WE CANNOT VERIFY).

AS OF JUNE 15, 1978, THE SETPOINT ASSOCIATED WITH THE 90% UNDERVOLTAGE RELAY IS SET AT 7.0 + or - 1.5 SECONDS IN COMPLIANCE WITH THE TECH. SPECS. (RELAY SET TO 7.5 SECONDS).

TECO MANAGEMENT DEFICIENCIES

1. COMMUNICATION PROBLEM. FAILURE TO PROPERLY COMMUNICATE WITH LICENSING WHEN A TECHNICAL SPECIFICATION CHANGE WILL BE ACCOMPLISHED. ALSO, FAILURE TO CONTACT LICENSING TO ADVISE THEM THAT A TECHNICAL SPECIFICATION CHANGE WAS ISSUED WHICH THE FACILITY WAS NOT READY TO COMPLY WITH (RELAY SETPOINT).

2. FAILURE TO PROPERLY REVIEW FCR'S TO DETERMINE IF A CHANGE TO THE STATION TECHNICAL SPECIFICATIONS IS NEEDED DUE TO A FCR.

3. ENGINEERING REQUIRED TO IMPLEMENT FCR'S RELATED TO TECHNICAL SPECIFICATION CHANGES NOT PERFORMED PRIOR TO THE APPROVAL OF THE TECHNICAL SPECIFICATION CHANGE. THIS RESULTED IN A DELAY IN IMPLEMENTING TECHNICAL SPECIFICATION RELATED CHANGES.

4. LACK OF TRACKING OF FCR'S TO INSURE PROPER AND TIMELY IMPLEMENTATION, LARGE BACKLOG OF FCR'S.

5. INACCURATE COMMENTS INCLUDED ON FCR FORMS (SUCH AS REASON FOR UNDERVOLTAGE SETPOINT).

ENFORCEMENT

1. TECHNICAL SPECIFICATION VIOLATION IN THAT THE PLANT OPERATED FOR SEVERAL MONTHS WITHOUT IMPLEMENTING THE 7 + or - 1.5 SECOND SETPOINT REQUIRED BY TECH. SPEC.(AMMENDMENT #7). THIS WAS REPORTED BY THE LICENSEE ON JUNE 23, 1978, AND IS RELATED TO FCR DEFICIENCIES WHICH HAVE BEEN DISCUSSED WITH TECO MANAGEMENT AND COVERED IN OTHER INSPECTIONS.

2. NO ITEMS OF NONCOMPLIANCE WERE OBSERVED IN RELATION TO LOSS OF PRESSURIZER LEVEL INDICATION.

3. AS NOTED, FCR DEFICIENCIES ARE COVERED IN OTHER REPORTS.

4. THE "LIFT WIRE PROCEDURE" WRITTEN TO COPE WITH DIFFICULTIES EXPERIENCED FOLLOWING ADJUSTMENT OF THE UNDERVOLTAGE RELAYS IS NOT TREATED IN THE INVESTIGATION REPORT, BUT APPEARS TO BE CLEAR-CUT NONCOMPLIANCE.

5. DEFICIENCIES NOTED UNDER"TECO MANAGEMENT DEFICIENCIES" APPEAR TO BE MAJOR FINDINGS OF THE INVESTIGATION AND WERE COVERED IN THE EXIT INTERVIEW WITH TECO PERSONNEL. THESE POINTS SHOULD BE FURTHER HIGHLIGHTED IN THE REPORT TRANSMITTAL LETTER.

-DISCUSSION-

FORM 291
REV 11-70

ARKANSAS POWER & LIGHT COMPANY

INTRA COMPANY CORRESPONDENCE

April 15, 1975

NDC 2719

MEMORANDUM

TO: J. W. Anderson

FROM: William Cavanaugh

SUBJECT: Arkansas Nuclear One-Unit 1
Pressurizer Level Setpoint
(File: 3740)

Reference: 1. JWA-848
2. NDC-2360
3. Letter, Govers to Cavanaugh 3/3/75

Attached is reference 3 from B&W which provides their answers to PSC comments on loss of level indication in the pressurizer following a reactor trip. From that letter, it can be seen that as long as water remains in the pressurizer the core will remain covered and the HPSI setpoint will not be reached. If the pressurizer empties, HPSI will be automatically initiated due to the rapid pressure drop mentioned in their letter.

If you have further questions, please contact us.

Joe Marlin

WC:DAR:ls

Attachment

cc: Mr. D. A. Rueter
Mr. M. L. Pendergrass

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505

Telephone: (804) 384-5111

April 3, 1975

Mr. W. Cavanaugh, III
Manager, Nuclear Services
Arkansas Power & Light Company
P.O. Box 551
Little Rock, Arkansas 72203

Subject: Arkansas Nuclear One - Unit One
Pressurizer Level Setpoint
B&W Reference NSS-8

Reference: NDC 2360, 3/3/75

Dear Mr. Cavanaugh:

NDC 2360 expressed concern over the momentary loss of pressurizer level indication following a reactor trip and requested additional information to clarify that maintaining RC pressure above 1500 psig (HPSI automatic actuation setpoint) would ensure that the reactor core remains covered with water.

This protection can be demonstrated by using a very simple principle: reactor coolant system pressure is determined by the saturation pressure for the hottest water in the reactor coolant system. In all operating situations except extreme accident conditions, this water is, of course, pressurizer water at about 650°F, corresponding to a saturation pressure of 2155 psig while the average water temperature in the reactor core of 579°F has a saturation pressure of about 1300 psig. Within about 20-30 seconds after a reactor trip, all water in the reactor coolant system (except pressurizer water) will be below 579°F as the reactor power-sustained differential temperature across the core collapses and as the reactor coolant system is cooled to about 550°F (due to turbine bypass valves being set to control OTSG pressure at 1010 psig). Even though the pressurizer water out-surge during system cooldown will allow system pressure to fall below 2155 psig, data from reactor trips at B&W's operating plants shows that RC pressure remains well above 1500 psig. With the RC cooldown established by means of the turbine bypass valves' pressure setpoint, RC pressure will not drop to 1500 psig unless the pressurizer is completely drained. If the pressurizer were to drain completely, RC pressure would drop rapidly to the saturation pressure for the hottest water remaining in the RC system. The temperature of this water would be between 550°F and 579°F with a resulting RC pressure of 1010 psig to 1300 psig. This resulting RC pressure band if the pressurizer were to empty following a reactor trip is well below the 1500 psig HPSI automatic initiation setpoint. Thus 1500 psig is an adequate low pressure setpoint for ensuring that the reactor core remains covered with water.

you have any further questions in this matter, please advise.

Very truly yours,

J. D. Phinney, Manager
Operating Plant Services & Maint.

By:

R. A. Govers
Service Project Engineer

JDP/RAG/cs

cc: J. W. Anderson
J. A. Bailey
R. P. Lockett, Jr.

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

To

R.P. WILLIAMSON - NUCLEAR SERVICE

From

C.W. TALLY - CONTROL ANALYSIS (EXT. 2883)

BDS 663.5

Cust.

TECO

File No. or Ref.

Subj.

SPR 396

Date

FEBRUARY 10, 1978

This letter to cover one customer and one subject only.

Reference: 1. Letter BWT-1609, J.A. Lauer to C.R. Domeck, T1.2/12B, dated December 5, 1977.

Engineering has evaluated the transient described in SPR 396 resulting in the following comments:

1. The classification of the transient in Reference 1 was correct and no further comment on this aspect is required.

2. The decrease in pressurizer level (off-scale low) is indicative of rapid steam generator level increases following the initiation of AFW. This undesirable effect is symptomatic of high level setpoints. Conversations with Fred Miller of TECO Engineering have confirmed TECO's awareness of this problem and their desire to have it rectified. In view of the fact that Davis-Besse I has elevated loops, there should be little difficulty in decreasing the level setpoint with appropriate analysis. The funding for this work will be pursued through Project Management.

3. Engineering has been unable to satisfactorily resolve the dissimilar behavior of the two OTSG's during the transient. During the 5 to 15 minute period of the transient, the two steam pressures moved in opposite directions and were considerably apart. The plant computer printout says a main steam line warm up isolation valve was open during this time ("22:55:56 Z688 MN STM Line 2 WU ISO VLV CLOS"), but TECO Engineering says the valve indicator is wired backwards, indicating that it actually was closed until 22:55:56, when an operator opened it. If indeed it was closed until this time, there appears to be no logical explanation for the steam pressure differences. This should be passed on to TECO Engineering, since Plant Design has no further information with which to investigate this anomaly.

C.W. Tally

cc: J.R. Burris
R.B. Davis
J.A. Lauer
R.W. Winks

TOLEDO EDISON COMPANY
DAVIS-BESSE UNIT ONE NUCLEAR POWER STATION
SUPPLEMENTAL INFORMATION FOR LER NP-32-78-07

DATE OF EVENT: June 12, 1978

FACILITY: Davis-Besse Unit 1

IDENTIFICATION OF OCCURRENCE: Incorrect setpoints on essential bus undervoltage relays

Conditions Prior to Occurrence: The unit was in Mode 6 with Power (MWT) = 0, and Load (MWE) = 0.

Description of Occurrence: On June 12, 1978, during the Station Review Board review of the "Safety Features Actuation System (SFAS) 18 Month Test", ST 5031.07, it was found that the time delay setpoints of the essential bus undervoltage relays were incorrect and that the monthly channel functional test was not being performed.

The initial investigation showed the Facility Change Request (FCR) 77-217 which was implemented on October 4, 1977, called for the time delay to be set at 9 seconds. FCR 77-430 was prepared on October 28, 1977, to correct the setpoints to 7 $\pm$ 1.5 seconds, but had not yet been issued for implementation on June 12, 1978.

This occurrence is being reported in accordance with the provisions of Technical Specification 6.9.1.8f.

Designation of Apparent Cause of Occurrence: The cause of this occurrence is procedure inadequacy.

Analysis of Occurrence: There was no danger to the health and safety of the public or to unit personnel. The intent of the 7 $\pm$ 1.5 second time delay setpoint is to ensure that a bus trip will occur in 9 seconds after the bus voltage degrades to less than 90% of the normal voltage. The average time delay setting of the relays was found to be 8.99 seconds.

Corrective Action: FCR 77-430 was immediately implemented and at that time it was also found that the voltage setpoints were incorrectly set to a maximum of 2.5% less than the technical specification minimum. One relay was found to be defective and was replaced. The time delay and voltage setpoints were adjusted to values in compliance with Table 3.3-4 of Technical Specification 3.3.2.1. A modification (T-2870) was prepared for a test to be performed in conjunction with ST 5031.07 to satisfy the monthly functional check. A new surveillance test procedure will be written to assure the monthly functional test is completed when the unit is in the applicable modes. This work was completed on June 15, 1978 under Maintenance Work Order 78-1397.

Failure Data: This is not a repetitive occurrence.