

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 ± 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 ± 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.

EXHIBIT 1
page 3 of 3

2283 319



METROPOLITAN EDISON COMPANY

POST OFFICE BOX 542 READING, PENNSYLVANIA 19603

TELEPHONE 215 - 929-3551

July 24, 1978
GQL 1227

Mr. B. H. Grier, Director
Office of Inspection & Enforcement
Region 1
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Dear Sir:

Three Mile Island Nuclear Station Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320

In accordance with the requirements of Section 6.9.2.A of the TMI-2 Technical Specifications, enclosed please find a Special Report concerning the TMI-2 ECCS Actuation which occurred on April 23, 1978.

Sincerely,

J. G. Herbein
Vice President-Generation

JGH:RAL:tas

Enclosure: Special Report concerning the TMI-2 ECCS
Actuation of April 23, 1978

2283 320

DUPLICATE DOCUMENT

Entire document previously
entered into system under:

ANO

No. of pages:

7904100172
5

Babcock & Wilcox

Power Generation Group

M-500
M-517

RECEIVED
NOV 24 1976
POWER ENG.

RT 12/4/76

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

Telephone: (804) 384-5111

	November 22, 1976	
TO		
FROM		
SUBJECT		
RECEIVED		
DATE		
BY		
CHKD		
APPROVED		
CMG		

SOM #209 620-0014
T3.3.1
SIP #14/114
12B59 DP 1101.01 &
DP 1101.02

*Send Co
to R.P. Jones
11/24/76*

Mr. J. G. Evans, Station Super
Davis-Besse Nuclear Power Stat
5501 North State Route #2
Oak Harbor, Ohio 43449

Subject: Recommendations for Avoiding Pressurizer Off-Scale Indications

Dear Jack:

Experience has shown that the B&W 177 Fuel Assembly Plants with the pressurizer level indication range of only 320 inches are susceptible to below zero level indications on reactor/turbine trips and load rejection transients. Our Control Analysis Unit in Lynchburg has reviewed this problem and provided the following generic resolution:

1. For a plant with normal operating level of the pressurizer of 180 inches, raise the nominal level to 200 \pm 20 inches rather than 180 inches. Operating history of automatic pressurizer level control shows a deviation of approximately \pm 10 inches. Any additional increase in level will be in conflict with the assumptions employed in the Anticipated Transient Without Scram study for the NRC.
2. The amount of blowdown of the steam safety relief valves has been assumed to be 5% or approximately 50 psi for the safety valves with the lowest setting (1050 psig). Measured steam line pressures at operating plants of this type indicate that the actual blowdown is about 7% or 75 psi and even as large as 8.5%. The minimum reactor coolant system average temperature following a reactor trip should not decrease below 548°F and the minimum steam generator discharge pressure should exceed 975 psig at the same time. Should the measured steam safety valve blowdown exceed 7%, the valve blowdown should be readjusted to approximately 5% at your earliest convenience.

*DB-1
326*

POOR ORIGINAL

2283 521

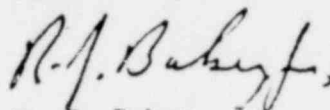
EXHIBIT 3
page 1 of 2

The pressurizer level alarms should remain the same with the exception of the low level alarm. The low level alarm should be raised to 180 inches from 160 inches. Pressurizer alarms are selected with an adequate margin for the operator to take action before the pressurizer level achieves a critical high or low value. This change will increase that margin.

FRP
GRP
Implementation of these recommendations will require changes in Plant Set-points, Plant Limits & Precautions, and all procedures with a reference to normal pressurizer level and pressurizer low level alarm setpoint.

If you have any questions in this matter, please do not hesitate to call.

Yours truly,



R. J. Baker, Jr.
Site Operations Manager

RJB:RES:mlf

cc: W. H. Spangler
J. A. Lauer
E. L. Logan
R. L. Pittman
E. R. Michaud
R. W. Winks

E. C. Novak, TECo
J. D. Lenardson, TECo

2283 322

POOR ORIGINAL

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. | File No.
TECO | or Ref.

Subj. | Date
SPR 396 | 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 2688 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.

POOR ORIGINAL

C.W. Tally
C.W. Tally

2283 23

EXHIBIT 4

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

ARKANSAS POWER & LIGHT COMPANY

INTRA COMPANY CORRESPONDENCE

April 15, 1975

RECEIVED

APR 16 1975

ARKANSAS POWER & LIGHT CO.
ARKANSAS NUCLEAR ONE

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.

WC:DAR:ls

Attachment

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

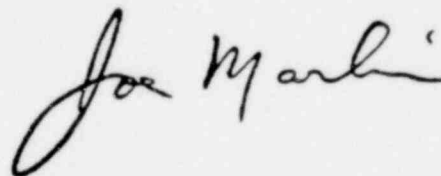


EXHIBIT 5
page 1 of 3

2283 524

Babcock & Wilcox

Power Generation Group

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

Telephone: (804) 384-5111

April 3, 1975

POOR ORIGINAL

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.

RECEIVED
APR 21 1975
ARKANSAS POWER & LIGHT CO.
ARKANSAS NUCLEAR ONE

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.

2283 525

EXHIBIT 5
page 2 of 3

bck & Wilcox

nney/Govers to Cavanaugh

-2-

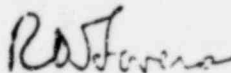
April 3, 1975

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.

2283 326

EXHIBIT 5
page 3 of 3