THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

cc: R Berchin, FJ Levandosk, ER Kane, FR Faist, WH Spangler, RC Luken, AHL/DBT, JA Lauer, GA Meyer, JD Dempsey, DL Rice, AH McBride, CW Bruney - Mt. Vernon, H Behnke - Mt. Vernon, RN Bottorf - Mt. Vernon

(all w/att)

To	DISTRIBUTION
From	J. A. LAUER, PROJECT MANAGER (X-2156)
Cust.	TOLEDO EDISON COMPANY DB-1
File No. or Ref.	NSS-14
Subj.	REPORT ON DEPRESSURIZATION EVENT
Date	OCTOBER 11, 1977

This letter to cover one customer and one subject only.

DISTRIBUTION

R. B. Davis J. S. Tulenko
R. M. Douglass (Mt. Vernon) R. K. Kennedy

The attached NRC letter received September 30 outlined corrective actions required as a result of the DB-1 depressurization event. Based upon the attached B&W letters BWT-1578 and BWT-1579 dated October 5 and October 7, NRC permitted TECO to startup DB-1 again. However, TECO is obligated to provide a formal report on the incident which includes a detailed analysis of the long-term effects (i.e., fuel life, fatigue cycles, etc.). This requirement is summarized in the exit interview notes dated October 7 (also attached). Note that the report is intended to provide information for the NRC Accident Analysis group to evaluate other accidents.

The B&W input to TECO's report should include four separate sections. SPR 372 may be used as the source document for information regarding the transient. The responsibility and suggested content of the sections is as follows:

DAVIS 1. Description of the event, including
- a. sequence of events
- b. system interactions
- c. plots of available data

DOUGLASS 2. Evaluation of reactor coolant system, including
- a. stresses in the pressure boundary
- b. evaluation of depressurization transient
- c. evaluation of boiling SG dry
- d. effect upon fatigue analysis
- e. jet impingement on SG shell

KENNEDY 3. Evaluation of RC pumps including
- a. cavitation damage to impellers
- b. damage to bearings
- c. damage to seals
- d. start-up testing

Ptf's Ex. 6PU 431 For ID
Deft's 1-27-82 J.R. Danyo

TULENKO 4. Evaluation of fuel including
- a. heat generation
- b. pressure differential on cladding
- c. core lift-off
- d. effect upon fuel life

All charges are to be accumulated on L-order 620-0014-91-29 so that they may be recovered from TECO. Please forward WA's for the manhours and any computer hours necessary to prepare this report. We would like to have all sections of the report ready to mail to TECO by October 28. Please advise immediately if this date cannot be met.

J. A. Lauer

J. A. LAUER

JAL/hj
Attachment

TELECOPY TO: R. P. WILLIAMS, NUCLEAR SERVICE (2 pages)

(from F. E. Falct, Davis-Besse)

NRC OIE
EXIT INTERVIEW
OCTOBER 7, 1977

OCT 10 1977

ATTENDEES:

NRC OIE
T. Tambling

TECO QA
J. Buck

TECO D-B
T. Murray
M. Green

SF

1. From previous discussions regarding September 24 depressurization incident, TECo has committed to the following:

   a. Required prior to Mode 2 - Testing of AFP governors using a modification to the monthly ST on AFPs. This to be done after main steam pressure of greater than 800 psia is available.

   b. Required prior to Mode 2 - Testing of the Electromatic Relief (on pressurizer). This is to be approximately five (5) open/close cycles with the isolation valve open. In addition the proper operation of the circuit and solenoid to be checked using simulated pressure signal (with block valve closed).

   c. Installation of recorders to monitor the SFRCS System during startup.

   d. Refresher Training on SFRCS for Operators to be completed on October 22, 1977.

2. As part of making SFRCS System more reliable, we are looking into making a change so that SFRCS alarms have a time delay dropout to insure they get picked up by the computer when it scans once per second.

3. Regarding the missing relay in the Electromatic Relief valve control circuit, the NRC is concerned that this might have been industrial sabotage. If we don't feel that it was sabotage, what sort of controls should we have to prevent a similar problem. We told Tom about the inspection that we made of all relay cabinets to insure that all fuses and relays are installed as required.

4. Movement of the local reset pushbuttons for the SFRCS components is being considered.

BLW 5. The initial monitoring of RCPs is complete but the program will continue until the detailed check to be made at 1300 psi is completed.

6. In order to get approval to startup, we had to commit to providing a formal report on the incident, which must include a detailed analysis of the long term effects (fuel life, fatigue cycles, etc.).

   BAW

   Must clarify why in first B&W letter say "We are also evaluating to determine whether this transient has an effect upon the calculated fatigue life of the components, ... " and in the second letter it is stated that "There is no change in the calculated fatigue life of the components".

   The primary purpose of the report is to provide information for the NRCs Accident Analysis Group who will use the information in evaluating other accident analyses.

We must call next week to give them the time frame for submittal of this report.

WES/dmw

bcc: A. H. McBride
R. Berchin
W. H. Spangler
G. A. Meyer
R. K. Kennedy
C. W. Bruny, Mt.V.
A. H. Lazar
D. B. Tulodieski
J. A. Lauer
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Records Center
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J. P. Jones

Babc

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

ctober 7, 1977

BWT-1579
File: T1.2
12B

cc: J. D. Lenardson
J. C. Lewis
D. J. DeLaCroix
E. C. Novak/2c

Mr. C.
Nuclea
Toledo
Power on
300 Madison Avenue
Toledo, Ohio 43652

Subject: Toledo Edison Company
DEPRESSURIZATION EVENT OF SEPTEMBER 24, 1977
Davis-Besse Unit 1
NSS-14

Dear Mr. Domeck:

Our letter BWT-1578 dated October 5, 1977 advised that B&W is reviewing the available data regarding the depressurization event of September 24 and we consider that there has been no degradation of safety in the plant. By telecon of October 6, you have advised that NRC would like more information regarding the basis for our conclusions.

The components are designed for forty cycles of a depressurization transient in which the pressure drops 1400 psi and the temperature drops 62°F in fifteen minutes. In this actual transient the pressure dropped 1250 psi and the temperature dropped 45°F in 7.5 minutes. The stresses due to pressure are not sensitive to time in these loading ranges. Since the pressure change of the actual transient is less than the pressure change of the generalized transient, the stress effect due to pressure would be less than calculated for the generalized transient. Although the rate of temperature change is higher in the actual transient, the overall temperature change is 17° less. These two differences tend to offset each other such that the resulting stresses for the actual transient were no worse than the calculated stresses for the design transient.

One steam generator apparently boiled dry during the depressurization event because the auxiliary feedwater turbine failed to come up to speed. The design transients include twenty cycles in which feedwater flow is lost to one generator and the generator is evaporated to a dry pressurized condition. The introduction of feedwater into a dry steam generator is a design condition and will have no harmful effects. The major concern while the steam generator is dry is variation in the tube-to-shell temperature differences. In the actual transient the steam generator was dry for a short period of time and the generator remained pressurized. During this time of approximately 13 minutes, the reactor coolant temperature dropped a total of 50°. This temperature drop was not sufficient to cause excessive stress or permit deformation of the tubes, and is within the established design limits.

The actual stresses were no worse than the calculated stresses from the design transients and consequently the fatigue usage resulting from the actual transient is no worse than that for the design transients. The predicted fatigue usage for this transient is the same as that of one design cycle of rapid depressurization and one design cycle of startup of a dry steam generator. There is no change in the calculated fatigue life of the components. Since the stresses and deformations resulting from design transient analysis are acceptable, there is no reason to expect overstressing or material deformatic in the RC system due to the actual transient. We do not consider it necessary to

conduct a detailed inspection of hangers and restraints for evidence of damage or deformation.

The reactor coolant pumps were all operated at or near saturation pressure (A2 and B1 for about one minute and A1 and B2 for about 45 minutes). There is some risk of cavitation damage to the impellers and also a risk that saturated steam would cause dry bearings and resulting damage. In addition, radial offsets due to cavitation may damage the seals. All four pumps suffered either loss of or erratic seal injection flow for about 1-3/4 minutes after containment isolation valves were closed. We have reviewed these conditions with the pump manufacturer and concluded that the risk of damage is small. Disassembly and inspection of the seals, bearings, and impellers would not provide 100% assurance that they will operate properly. Therefore, we have recommended that the pumps be instrumented to measure shaft vibration, seal cavity pressures, RC pressure, standpipe leakage, and seal injection flow and temperature. Each pump has been run for two minutes with this instrumentation in Mode 5 and the observed parameters show no indication of damage. We expect to have similar test runs in Mode 3 when the RC pressure is above 1300 psi. If these runs also show no indication of damage, B&W would then recommend that the pumps may be safely operated as designed.

B&W has evaluated the 9/24 incident with regard to its effects upon fuel performance and has concluded that there are no safety concerns with respect to the reactor fuel. This conclusion is based upon the following considerations:

- Prior to the subject transient the reactor had been operating for approximately one week at a maximum of 15% of rated power; immediately prior to trip the power level was approximately 10% of rated power, therefore, the heat generation in the core (decay heat) during the depressurization transient was extremely low and significantly less than that produced by the reactor coolant pumps.

- The core burnup on 9/24 was approximately 1 EFPD.

- During the transient the maximum fuel rod internal pressure has been conservatively estimated to have been no more than 300 psi greater than the minimum RC system pressure; the maximum fuel rod cladding temperature was 550F. The tensile stresses imposed on the cladding as a result of the 300 psi pressure differential existed for less than one hour. For cladding with low irradiation exposure exposed to this temperature/pressure combination no deformation or failure would be predicted.

- Reactor coolant temperature, pressure, and flow rate data obtained during the course of this transient indicated that there was no significant heat generation in the reactor core; this data further indicates that no significant boiling occurred in the core.

Very truly yours,

A. H. Lazar, Senior Project Manager

J. A. Lauer

J. A. Lauer, Project Manager

JAL/hj

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B&W TELECOPY

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P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

October 5, 1977

BWT-1578
File: T1.2/12B

cc: J. D. Lenardson
J. C. Lewis
D. J. DeLaCroix
E. C. Novak/2c

Mr. C
Nucle
Toled
Power Engineering and Construction
300 Madison Avenue
Toledo, Ohio 43652

Subject: Toledo Edison Company
DEPRESSURIZATION EVENT OF SEPTEMBER 24, 1977
Davis-Besse Unit 1
B&W Reference NSS-14

Dear Mr. Domeck:

B&W is reviewing the available data regarding the transient on the evening of September 24 in which the reactor coolant pressure dropped from almost 2300 psig to about 950 psig in 7.5 minutes. During this depressurization or the recovery period immediately thereafter, there was apparently some steam formation in the reactor coolant system, one steam generator secondary side went dry, and the reactor coolant pumps were operated a few minutes without seal injection water.

This transient does not appear to have caused any excessive stresses on the pressure boundary of the reactor coolant or secondary system because it is reasonably close to the design transients. We are reviewing those areas which exceed design transient conditions to confirm that stress levels are within ASME Code allowable values. We are also evaluating to determine whether this transient has any effect upon the calculated fatigue life of the components, but this would be of no immediate concern.

The reactor coolant pumps were exposed to conditions which could have caused cavitation damage to the impellers, damage to the bearings, and seal leakage or failure. None of these things appear to have happened and we feel that the pumps may be operated with only minor risk of additional damage to the pumps. We suggest a series of progressive start-up tests under close observation in order to minimize the risk of furthering any latent damages. This is not considered to be a safety concern.

Based upon our evaluation of the transient conditions, we do not feel there has been any violation of fuel integrity which would preclude safe startup. However, we have made no evaluation regarding effect upon fuel life.

In summary, B&W considers that there has been no degradation of safety in the plant and it may be operated as designed. As an economic concern, we urge progressive startup of the RC pumps under close observation.

Please advise if you need further advice.

Very truly yours

A. H. Lazar, Senior Project Manager

J. A. Lauer

J. A. Lauer, Project Manager

JAL/hj