Docket No. 50-346

License No. NPF-3

Serial No. 1-411

March 6, 1984

TOLEDO EDISON

RICHARD P. CROUSE
Vice President
Nuclear
(419) 259-5221

PRINCIPAL STAFF			
RA	has	DRP	
D/RA		DE	has
A/RA		DRSS	has
SC		DRMA	
PAO		SGS	
SGA		ML	
ENF		FILE	has

vorig+1

Mr. James G. Keppler, Regional Administrator
United States Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Dear Mr. Keppler:

On March 2, 1984, the Davis-Besse Nuclear Power Station, Unit 1, experienced a plant trip in which a Main Steam Safety Valve failed to fully close. The nature of the transient resulted in the activation of the Toledo Edison Emergency Plan with its classification as an Unusual Event. A limited Nuclear Regulatory Commission (NRC) Site Team was dispatched to observe Toledo Edison's actions.

In discussions with the NRC Site Team Leader, several items were agreed to be addressed in a report to the Regional offices prior to entering Operating Mode 2. Toledo Edison reviewed these items with members of your team for clarification on March 3, 1984. This report is in support of your March 3, 1984 Confirmatory Action Letter, Item #7. Attachment 1 to this letter addresses each of the six items identified.

With the evaluations performed and actions discussed in Attachment 1, we are confident in returning Davis-Besse to full operational service with no adverse effects to the health and safety of our employees or the public. Representatives of your staff have been involved in the review of the event. No items have been identified by Toledo Edison's staff that would indicate that this report would be found unacceptable prior to entering Operational Mode 2, therefore, we have initiated preparations for heatup with plans for Mode 2 operations on Wednesday, March 7, 1984.

Very truly yours,

R. P. Crouse / TJM

RPC:SGW:TJM:nlf
encl.
cc: DB-1 NRC Resident Inspector

MAR 8 1984

Item 1a: Review the transient to determine if all parameters remained within the prior analysis. If any conditions are determined to be outside the analysis, assess the significance of such deviations.

Response: At 12:20 hours on March 2, 1984, Davis-Besse Unit 1 was operating at approximately 99% of full power. The plant was in full automatic control and Instrument & Control (I&C) personnel were performing a surveillance test on the Steam and Feedwater Rupture Control System (SFRCS) Channel 2. When the I&C technician relieved the pressure on the switch being tested at 12:21 hours, the #2 Main Steam Isolation Valve (MSIV) closed due to an undetected failure in SFRCS Channel 4, simultaneous with the test of SFRCS Channel 2. The rapid increase in #2 Steam Generator pressure, due to the #2 MSIV closing, increased the Reactor Coolant System (RCS) cold leg water temperature of the #2 Steam Generator. The increase in #2 Steam Generator pressure caused the feedwater to increase into the #1 Steam Generator. Steam Generator #1 level reached a maximum level of 85%. This dropped the #1 Steam Generator cold leg temperature causing reactor power to increase. A reactor trip on high flux occurred at 12:22 hours.

After the reactor trip. the Rapid Feedwater Reduction (RFR) circuit closed the main feedwater control valves, terminating the excessive feedwater flow to the #1 Steam Generator. Main Feed Pump (MFP) #2 had lost its steam supply from the Steam Generator since MSIV #2 had closed. The MFP #1 began to control improperly, even with the control station transferred to manual. Personnel were dispatched locally to the MFP #1, and local control was established.

On Steam Generator #2, pressure was observed to be decreasing well below the expected post trip value. A local observation of the Main Steam Safety Valves determined a safety valve had failed to close on Steam Generator #2. The SFRCS was manually initiated at 12:38 hours on low Steam Generator level as required by procedures. Since the safety valve was still open on Main Steam Line #2, at 12:38 hours #2 Steam Generator pressure reached the 612 psig low pressure trip setpoint and the SFRCS realigned both auxiliary feedwater pumps to feed the #1 Steam Generator as designed.

By 12:42 hours, pressurizer level was restored to approximately 100 inches, and the RCS pressure had been stablized out at approximately 2100 psig. The #1 Steam Generator was being used to remove the pump heat and core decay heat, and the #2 Steam Generator was dry and depressurized.

Since the #2 Steam Generator was dry, cooldown was accomplished quite slowly minimizing shell to tube differential temperature considerations. The RCS cooldown was conducted at a rate to match the temperature of the #2 Steam Generator shell. By 14:13 hours, the RCS had been cooled down to 490°F. By 17:20 hours, the RCS had been cooled to 450°F. By 19:50 hours, the RCS was cooled to 409°F. By 07:20 hours on March 3, 1984, the RCS temperature had been cooled to 340°F. The stuck open safety valve was replaced with a spare, and at 07:30 hours, the refill of the #2 Steam Generator began. By 07:45 hours, the #2 Steam Generator was restored to operable status. The plant continued its cooldown, entering Mode 4 at 12:40 hours on March 3, 1984.

The primary and secondary plant responded as expected for the transient. Engineered Safety Features Systems actuated within the Technical Specifications limits, and performed their designed functions. An evaluation of the transient event by Babcock & Wilcox concludes that each of the primary pressure boundary components still meet all of the requirements of the ASME Boiler and Pressure Vessel Code, Section III.

Item 1b: Also revise normal or emergency procedures to assure that they take into account lessons learned from this event.

Response: Several procedures have been revised to incorporate lessons learned from this event. PP 1102.10, Plant Shutdown and Cooldown, was revised to improve the section on cooldown with only one steam generator. EP 1202.24, Steam Supply System Rupture, was revised to incorporate improvements to both immediate and supplementary operator actions. These procedural changes have been included in training as operator required reading that must be completed by each operator prior to assuming the shift watch in Mode 2 or above for this startup. These changes are considered adequate based on the recent transient.

Longer lead time operating experience review efforts could yield additional procedural improvements in the future, but are not part of our review under Item 1b prior to this restart.

Item 2: Assess the effect of the transient on Steam Generator No. 2 to assure that it has maintained its desired integrity.

Response: The transient has been evaluted with respect to potential affects on the Steam Generator. Toledo Edison and Babcock & Wilcox evaluated this event to see if it would be enveloped by existing design transients.

The results of that evaluation (B&W Engineering Information Record Document identified 51-1150398-00) identify that this transient can be depicted and enveloped by the combination of a normal cooldown cycle in combination with a "Stuck Open Turbine Bypass Valve" design transient cycle. Therefore, the steam generator and its components during this event were subjected to no effects beyond the design transient and no degradation was sustained.

Item 3: Determine the cause of the Main Steam Isolation Valve closure which initiated the reactor trip, and the cause of other valve performance problems during the event, and take corrective action to minimize recurrence of a similar event.

Response: The Main Steam Isolation Valves (MSIV's) at Davis-Besse receive a closure actuation signal both from the Steam and Feedwater Rupture Control System (SFRCS) and the Safety Features Actuation System (SFAS). The control and display circuitry for these valves utilize various relays and contacts controlled by either of these two systems.

The loop 2 MSIV (MS-100) inadvertently closed during the performance of ST 5031.14, Steam and Feedwater Rupture Control System Monthly Functional Test. Portions of this test require isolation of specific sensor strings which initiates a half channel trip. On March 2, 1984, as part of the above Surveillance Test, the pressure sensing string associated with SFRCS Channel 2 was isolated causing a trip of SFRCS Logic Channel 2.

It is noted that on March 1, 1984, the data light for MS-100 in SFRCS Channel 4 was discovered to be off. However, since the associated data light (DS43A) in the SFAS Channel 4 was still energized, it was concluded that the SFRCS Channel 4 Control Relay (K104A) was operable. The de-energized SFRCS Channel 4 light for MS-100 was attributed to a bad socket at that time. Further troubleshooting revealed that an optical isolator in the relay driver card for relay K104A was defective. This resulted in a pre-existent SFRCS Channel 4 half trip on MS-100. This pre-existent (half) trip, when combined with the (other half) trip of Logic Channel 2 during the Surveillance Test, resulted in the automatic closure of MS-100.

During the above troubleshooting, some wiring anomalies were noted in the control switch circuitry for MS-100. Contrary to the schematic drawings, there were cross-connections between the Channel 2 and Channel 4 control circuitry for this valve. This enabled the DS43A SFAS light to incorrectly remain energized with an inoperable and faulty K104A SFRCS relay driver card.

This wiring anomaly was due to apparent ambiguity in the cable nomenclature utilized in the installation of the control switch for MS-100. During the troubleshooting, a broken cable in the MS-100 reset circuitry was also discovered. This broken wire did not relate to the inadvertent closure of MS-100 and will be replaced prior to this plant startup. It is emphasized that the above wiring anomaly did not, in any way, effect the closure of MS-100 if a full SFRCS or SFAS actuation of the associated actuation channel(s) were to occur.

The wiring anomaly in the loop 2 MSIV (MS-100) control circuitry was corrected and acceptably tested on March 5, 1984. In addition, such anomaly was verified not to be present in the control circuitry for the loop 1 MSIV (MS-101). ST 5031.14, SFRCS Monthly Functional Test, is being modified to verify proper operation of the MS-100 data lights. Correction of the wiring anomaly and the broken cable prior to this plant startup provides adequate assurance that recurrence of a similar event is minimized.

During the reactor trip other valve problems occurred. The cause of the performance problems and the corrective actions taken to minimize recurrence of a similar event are as follows:

1. Main Steam Safety Valve (MSSV), SP17A1

   MSSV SP17A1, having a set pressure of 1050 psig, failed to open following the reactor trip. A visual inspection of the valve indicated it appears to be satisfactory condition. SP17A1 will be hydroset in accordance with ST 5070.01, "MSSV Setpoint Test", in Mode 2 during this startup to ensure proper setpoint. An evaluation as to the operability of SP17A1 will be made based on the as-found lift point setting of the valve.

2. Steam Generator No. 2 Auxiliary Feedwater Isolation Valve, AF599

   Prior to refilling #2 Steam Generator, AF599 failed to open using the Limitorque Operator. The valve was manually opened, after which it cycled normally using the motor operator. Troubleshooting of AF599 under MWO 1-84-0819-00 did not reveal any mechanical or electrical problems with the valve or motor operator.

Facility Change Request (FCR) 84-039 was implemented which changed the torque switch settings for both AF599 and its other train counterpart AF608 (Steam Generator No. 1 Auxiliary Feedwater Isolation Valve) from 1.50 for open and close to 1.50 for open and 1.00 for close. The reduced close setting will prevent excessive closing force on the valve into the seat. The new torque switch settings were based on a review and application of a Torrey Pines Technology Report for Davis-Besse: Limitorque Valves Troubleshooting Program.

3. Main Feedwater Valve FW601

After restoring level to the #2 Steam Generator with the Auxiliary Feedwater System, the normal feedwater flow path was aligned to the Steam Generator. During this evolution, FW 601 (#2 Main Feedwater Stop Valve) torqued out in the open direction and had to be manually opened. This was a re... of the unusually large differential pressure (approximately 800 PSIG) across the eighteen inch valve disc as the Startup Feed Pump (SUFP) was in operation and pressurizing the upstream side of the valve. During all normal operations this valve is open and its function is only required to shut on a Steam and Feedwater Rupture Control System (SFRCS) or a Safety Features Actuation System (SFAS) actuation. The valve performed its tasked function satisfactorily in the safety direction. The only proceduralized opening would exist under abnormal conditions of a complete loss of Main and Auxiliary Feedwater event. However, during this procedure, FW601 is opened before the SUFP is started. During the event, the SUFP was already in operation and feeding the #1 Steam Generator, so the large differential pressure could not be avoided.

Item 4a: Determine the cause of the Safety Relief Valve to reseat (sic). Examine similar valves to assure that they do not have similar problems.

Response: Main Steam Safety Valve (MSSV), SP17A4, is a Dresser type 3707 RA-RT 21 valve. SP 17A4, having a set pressure of 1070 psig, lifted and failed to fully close.

Inspection of SP17A4 revealed the release nut cotter pin (Item 16A of Enclosure 1), which secures the release nut (Item 16) to the valve spindle (Item 8) was missing. The release nut had moved down on the spindle approximately one inch and was jammed on the top lever (Item 17). The spindle was also bent. Apparently, the release nut cotter pin was left out of the release nut or had broken during plant operation and vibrated out,

allowing the release nut to travel down the threaded end of the spindle while the valve was open. The bottom of the release nut made contact with the top lever preventing complete spindle travel for valve closure.

A Dresser representative inspected valve SP17A4. He concurred with Toledo Edison's assessment of the failure. All other MSSV's were visually inspected for similar problems with specific attention for missing or damaged release nut cotter pins. Visual inspection revealed no abnormalities on the remaining MSSV's.

The existing release nut cotter pins will be replaced on all MSSV's before returning to power operation. Modifications were made to existing maintenance procedures to verify installation of new release nut cotter pins after maintenance or testing in the future.

Item 4b: Also compare the cause with prior events of safety valves not closing properly to determine if any generic problems are evident. Assure correction of identified problems.

Response: A review of maintenance history could not substantiate a similar occurrence at Davis-Besse. In discussion with maintenance personnel, there may have been one (1) similar, but far less significant, occurrence at Davis-Besse about 5 years ago in which the loose release nut vibrated down the valve stem during valve operation and interferred with the lever arm upon valve reclosure resulting in slight valve leakage (weeping).

St. Lucie Nuclear Station had an apparently similar occurrence on February 9, 1984, in which one of their Main Steam Safety Valves failed to close in the same manner.

As mentioned in Item 4a above, Toledo Edison is replacing, on all MSSV's, the existing release nut cotter pins.

Toledo Edison is inspecting 60% of the nuclear safety related safety valves for similar problems with specific attention for missing or damaged release nut cotter pins. These include valves manufactured by Dresser, Lanagran, and Crosby. Eighteen of these twenty-eight selected valves are the MSSV's.

Item 4c: Verify proper set point settings on the replaced valves (sic).

Response: MSSV SP17A4, the only valve replaced, will be hydroset to 1070 psig (± 1%) in accordance with the Technical Specifications. This will be done in accordance with ST 5070.01, MSSV Setpoint, to ensure proper set points in Mode 2 during this startup to satisfy Technical Specifications 4.7.1.1, Table 4.7-1 (which requires the lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure).

Item 5: Assure the operability of an adequate number of Steam Generator shell temperature detectors for post-incident monitoring or provide alternate means of obtaining equivalent information.

Response: The existing Steam Generator shell thermocouples were utilized during the event to aid the cooldown process. This effectively guaranteed that any unusual stresses put on the generator and its tubes would be minimized. This fact and the availability of the data was a significant contributor to the evaluation of loads needed prior to restart. Since temperature differentials are not a postulated failure mechanism of the Steam Generator tubes, per B&W Engineering Information Record Document identified 51-1150398-00, there is no safety concern for taking the plant to cold shutdown. Therefore, Steam Generator shell thermocouple data is not required information for post-incident monitoring.

Item 6: Develop a plan for increased monitoring of water chemistry and leak rate detection from Steam Generator No. 2. The plan should identify the frequency and duration of such monitoring.

Response: Toledo Edison will implement the following program for monitoring of water chemistry and leak detection from Steam Generator No. 2.

a. During this startup, prior to Mode 2, Steam Generator water samples will be analyzed for Tritium every four (4) hours when samples are available. From Mode 2 until 50% reactor power, Steam Generator water samples will be analyzed for Tritium once per shift. Above 50% reactor power, monitoring of the Condensate Pump Discharge will be conducted on a normal weekly basis for Gross Beta Activity.

b. During this startup, the Main Steam Line Radiation Monitors (RE 600 and RE 609) will be in the Gross Count Mode and monitored once per hour until Mode 2 is reached. Above Mode 2, the monitors will be shifted from the Gross Count Mode to the Analyze Mode for monitoring N-16 gamma radiation. Readings will be taken once per shift until reaching 50% reactor power. Above 50% reactor power, the normal alarm functions of the radiation monitors will be used to provide indication of any leakage.

c. During this startup, after the Main Steam Isolation Valves are opened, hourly readings will be taken on the Steam Jet Air Ejector Radiation Monitor (RE 1003 A or B) until Mode 2 is reached. Above Mode 2, readings will be taken once per shift until reaching 50% reactor power. Above 50% reactor power, the normal alarm functions of the radiation monitor will be used to provide indication of any leakage.

Docket No. 50-346
License No. NPF-3
Serial No. 1-411
Enclosure 1
Page 1 of 1