Exhibit 1

TITLE: AFPT Overspeed Trip Throttle Valve Problem

REPORT BY: R. J. Gradomski   Plan No. 1D

DATE PREPARED: 6/20/85

I. INTRODUCTION

The following report documents the investigation, analysis, and evaluation to support the action plan for determining the root cause of the 1) inability to immediately relatch both of the Auxiliary Feed Pump Turbine (AFPT) Overspeed Trip Throttle (T&T) Valve Linkages, and 2) difficulty in opening the T&T valves during the June 9, 1985 reactor trip. This report has been prepared in accordance with the "Guidelines to Follow When Troubleshooting or Performing Investigative Actions into the Root Cause Surrounding the June 9, 1985, Reactor Trip", Rev. 2.

II. SUMMARY OF DATA

A. Known information and operational data for conditions prior to, during, and after the transient.

1. Prior to the transient - Both pumps overspeed trip linkages were latched holding the T&T valves in the full open position. This is confirmed by the absence of computer alarm points S007 (AFPT 1-1) and S017 (AFPT 1-2). The alarm points are activated by limit/position switches that note the linkage position. Further, the T&T valves must be open for AFPT start. Both AFPT's did start on demand as

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discussed in Section A2. Surveillance testing was also performed on both pumps as discussed in Section B.

2. During the transient - At 0141.08 hrs. on June 9, 1985, an Steam and Feedwater Rupture Control System steam generator low pressure trip was initiated. At 23 and 36 seconds after that initiation, AFPT 1-1 and 1-2, respectively, tripped on overspeed. Both pumps correctly tripped by means of the turbine overspeed trip device set at 4500 ± 50 RPM as indicated by the computer printout of AFPT speed. (The overspeed event is documented in plan 1A & B.) The turbine overspeed trip device is connected to the overspeed trip linkage and ultimately to the T&T valve as indicated in Figure 1. The Equipment Operators (EO's) that were dispatched to investigate and restart the AFPT's reported that they could not relatch the trip linkage to the T&T valve in the normal manner. Normally, the T&T valve handwheel is rotated clockwise until the sliding nut/latch--up lever rises and engages with the trip hook. Then the T&T valve handle is turned counterclockwise until the valve is fully open. The EO's interviewed report that they followed the above mentioned sequence. They reported that the trip hook would not stay engaged with the latch-up lever and that they had great difficulty in opening the valve. It was necessary to manually pull (hold) the linkage/trip hook in the engaged position while using an extension on the handwheel to open the T&T valve. After

several minutes, they were successful in their attempts to hold/relatch the mechanism, open the T&T valve, and start both AFPT's.

3. After the transient - At approximately 1200 hrs. on June 9, 1985, tests to verify operability of both AFPT's were conducted. These tests consisted of a quick-start of the pump. Both overspeed trip mechanisms were in their correct positions, engaged and holding the T&T valves open. No overspeed trip occurred and the operability tests were successfully completed at approximately 1400 hrs. on June 9, 1985.

An informal visual inspection of the overspeed mechanisms was conducted on June 15, 1985. It was noted that both mechanism's linkages and valves were in the tripped condition. The duty Shift Supervisor was aware that they were tripped but did not know why. The inspection further revealed that the trip linkage was noticeably "bowed" on AFPT 1-2. Discussions with operations and maintenance personnel had confirmed that this bowed condition has existed for some time and was not recent. There was no obvious wear or damage on any visible components taking specific note of the trip hook to latch-up lever interfacing surfaces. Lastly, it was noted that the area of the trip mechanism on both turbines had built up dirt and loose particles of what appeared to be lagging.

B. Maintenance and Surveillance Testing History

10/15/83 - #2 AFPT overspeed trip during a feedwater transient. The pump did not accelerate to the trip setpoint. The overspeed linkage was reset, the T&T valve opened and the pump brought online 5 minutes after the trip. The apparent cause was listed as improper latching of the trip linkage. The corrective action was to exercise the linkage during performance of ST 5071.01, Auxiliary Feedwater System Monthly Test, and to verify proper latching during performance of ST 5071.04, Auxiliary Feedwater System Channel Functional Test.

*05/09/84 - Lubricated overspeed linkage at grease fittings in accordance with preventative maintenance #1702 for AFPT 1-2.

*12/21/84 - Completed PT 5150.01, Auxiliary Feed Pump Turbine Overspeed Test for AFPT's 1-1 and 1-2. Both turbines tripped within specified tolerance. No reports of linkage relatch or valve opening difficulties. This test utilizes auxiliary boiler steam at 235 psi to the uncoupled turbine.

*01/03/85 - Lubricated overspeed linkage at grease fittings in accordance with preventative maintenance #1700 for AFPT 1-1.

*05/21/85 - Completed ST 5071.04, Auxiliary Feedwater System Channel Functional Test, for AFPT 1-2. This test was conducted in Mode 1 and requires, in sequence, the T&T valve to be closed, MS107 open and then closed and finally to reopen the T&T valve. There is no requirement for tripping the overspeed mechanism. No problems were reported with either the linkage or valve opening.

*05/23/85 - Completed ST 5071.01, Auxiliary Feedwater System Monthly Test, for AFPT 1-1. This test requires the exercising of the overspeed trip mechanism, linkage and T&T valve without steam pressure on the inlet to the valve. No problems were reported with linkage relatch or valve opening.

*06/06/85 - Completed ST 5071.01, Auxiliary Feedwater System Monthly Test, for AFPT 1-2. The overspeed trip mechanism, linkage and T&T valve were exercised. No steam pressure on the inlet to the valve. No problems were reported with linkage relatch or valve opening.

*06/07/85 - Completed ST 5071.04, Auxiliary Feedwater System Channel Functional Test, for AFPT 1-1. The T&T and steam root valves were cycled manually in the correct sequence. No problems were reported.

* Most recently completed test or maintenance.

III. CHANGE ANALYSIS

- Overspeed Trip Linkage

There have not been any Facility Change Requests (FCR's) or Maintenance Work Orders (MWO's) prepared to either make design changes or adjustments to the overspeed trip mechanism or linkage. Regularly scheduled testing and preventative maintenance has been conducted with no reported problems. Inadvertent tripping of the overspeed trip device has not occurred since the 10/15/83 event.

- Trip Throttle Valve Opening Difficulty

There have not been any FCR's or MWO's prepared to either make design changes or adjustment to the T&T valves. A marked changed in the system has been noticed between all testing conditions and the system conditions that existed on June 9, 1985. All testing is done with either: 1) low pressure steam, 2) no steam pressure or 3) residual steam pressure trapped between the root steam valve and the T&T valve. The conditions that existed on June 9, 1985 were that: 1)

the steam root valves were continuously open and 2) steam pressure was $\approx$ 1000 psi.

IV. HYPOTHESES

- Overspeed Trip Linkage

Although there have been no reported problems associated with the relatching of the overspeed trip device during regularly scheduled testing, personnel involved in the evolutions have noticed some difficulty. After reviewing how the device should work, input from the Equipment Operators and the Terry Turbine Company, and the general condition of the equipment involved, it is possible that: 1) The tappet* of the turbine trip mechanism did not return to its normal position while attempting the relatch evolution, 2) The spring that provides the relatching force for the trip hook is defective or inadequate, 3) The pivot point of the trip hook is not sufficiently free to assure proper engagement of trip hook to latch-up lever, and/or 4) The linkage mechanism may not be adjusted correctly.

*Calvert Cliffs 1, May 1976, LER 76-16 reported an incorrect size tappet nut was the cause of difficulty in resetting the trip throttle latch mechanism for their Terry Turbine supplied AFPT. This condition was discovered during maintenance activities.

- Trip Throttle Valve Opening Difficulty

Although there have been no reported problems associated with opening the T&T valve during regularly scheduled testing, it is noted that significant differences exist between test and the system conditions of June 9, 1985. It is known that the T&T valves have rarely been opened, from full closed, against full steam generator pressure. The reactor trip of 10/15/83 provided the only known similar conditions to those that existed on June 9, 1985. It is unknown whether difficulties were encountered in the opening of the T&T valve for AFPT 1-2 on 10/15/83. It is therefore hypothesized that the valve may not correctly balanced/adjusted for opening against full steam generator pressure.

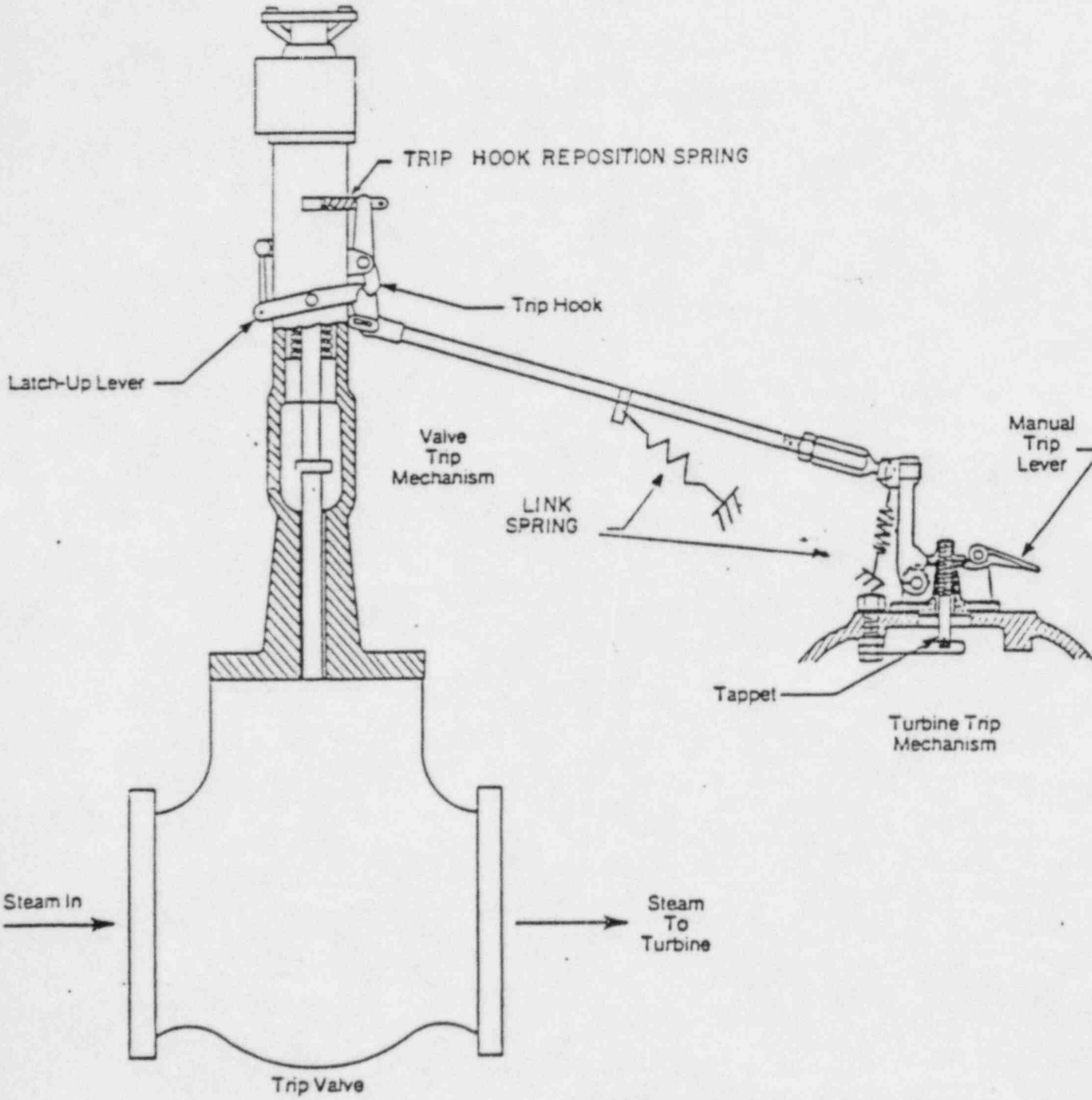
V. ACTION PLAN

The attached action plan contains checks, verifications, inspections, and exercising in order to identify the root cause of the inability to immediately relatch the overspeed trip device. The action plan contains recommended checks and inspections from the Terry Turbine Company. The action plan further identifies confirmatory testing of the hypothesis for difficulty in opening the T&T valve to be conducted during the testing proposed by action plans 1A and 1B for AFPT overspeed.

RJG:lrh

Attachment

Latching AFW Turbine Trip Valve (Typical Sketch) Figure 1



ACTION PLAN

TD 4408

TITLE

AUXILIARY FEEDWATER PUMP TURBINE (AFPT) TRIP THROTTLE VALVE PROBLEM

Rev. 0

PLAN NUMBER: 1D

PAGE: 1 of 3

DATE PREPARED: 6/20/85

PREPARED BY: R. Gradomski

SPECIFIC OBJECTIVE

To determine the root cause of the problem in relatching the Overspeed Trip Mechanism (OTM) and of Opening the Trip Throttle (T&T) Valves of AFPT 1-1 and 1-2 on June 9, 1985.

STEP NUMBER	ACTION STEPS	PRIME RESPONSIBILITY	ASSIGNED TO	START DATE	TARGET DATE	DATE COMPLETED
	ALL STEPS OF THIS ACTION PLAN ARE TO BE PERFORMED IN ACCORDANCE WITH THE LATEST REVISION OF "GUIDELINES TO FOLLOW WHEN TROUBLE-SHOOTING OR PERFORMING INVESTIGATIVE ACTIONS INTO THE ROOT CAUSES SURROUNDING THE JUNE 9, 1985, REACTOR TRIP".					
1	Document the present condition of the AFPT 1-1 and 1-2 OTM's. Provide detailed 35 mm photographs, that include reference scale, of each component (e.g., exposed turbine trip device, linkage and valve). Include detail sketches with dimensions, if necessary.	Gradomski	Thompson			
2	Inspect and document the condition of the internal portion of the overspeed trip device. Reference Terry Turbine Co. Instruction Manual, Section IX, Drawing C8680. This inspection can be performed through the overspeed trip adjustment port. Verify the integrity of the emergency tappet spring and roll pin, Part No's. 76 and 88 (e.g., is not cracked or bent). Additionally, verify that the emergency governor weight,	Gradomski	Thompson			

ACTION PLAN

TITLE

AUXILIARY FEEDWATER PUMP TURBINE (AFPT) TRIP THROTTLE VALVE PROBLEM

Rev. 0

PLAN NUMBER: 1D

PAGE: 2 of 3

DATE PREPARED: 6/20/85

PREPARED BY: R. Gradomski

SPECIFIC OBJECTIVE

To determine the root cause of the problem in relatching the Overspeed Trip Mechanism (OTM) and of Opening the Trip Throttle (T&T) Valves of AFPT 1-1 and 1-2 on June 9, 1985.

STEP NUMBER	ACTION STEPS	PRIME RESPONSIBILITY	ASSIGNED TO	START DATE	TARGET DATE	DATE COMPLETED
	Part No. 91, is in the correct position.					
3	Perform exercise of the overspeed trip linkage for AFPT 1-1 and 1-2 in accordance with ST 5071.01, Section 6.11. Ensure that the Equipment Operators involved on June 9, 1985 restart attempts are present. Ensure that a knowledgeable Terry Turbine Co. service representative is present. Perform the following checks and document the results:	Gradomski	Thompson			
	A. Freedom of movement of the tappet.					
	B. Condition of trip hook return spring.					
	C. Freedom of movement at all pivot points.					
	D. Proper adjustment of linkage and tightness of locknuts.					
	E. Condition of link springs.					
	F. Trip hook and latchup lever interfacing surface conditions.					
	G. Check for debris under tappet nut.					
	H. Check to ensure the tappet returns to its correct position when relatching.					
	I. Check movement of trip hook when relatching.					

ACTION PLAN

ED 4408

TITLE	Rev. 0	PLAN NUMBER: 1D	PAGE: 3 of 3
AUXILIARY FEEDWATER PUMP TURBINE (AFPT) TRIP THROTTLE VALVE PROBLEM		DATE PREPARED: 6/20/85	PREPARED BY: R. Gradomski

SPECIFIC OBJECTIVE

To determine the root cause of the problem in relatching the Overspeed Trip Mechanism (OTM) and of Opening the Trip Throttle (T&T) Valves of AFPT 1-1 and 1-2 on June 9, 1985.

STEP NUMBER	ACTION STEPS	PRIME RESPONSIBILITY	ASSIGNED TO	START DATE	TARGET DATE	DATE COMPLETED
4	In conjunction with step 10 of action plans 1A & 1B for AFPT 1-1 and 1-2:	Gradomski	Thompson			
	A. Trip the T&T valve to the closed position.					
	B. Against full steam generator pressure, attempt to open.					
	C. If valve opens with normal force, stop the test.					
	D. If valve does not open with normal force applied, adjust per manufacturer's instructions.					
	E. Repeat substeps A, B and C until T&T valve open with normal force.					
5	Submit all photographs and documentation to R. Gradomski	Assigned Individuals				
6	Submit results of investigations with recommended corrective actions to Project Manager.	Gradomski	Gradomski			

ap 1d

Questions/Issues On Sequence of Events

1) Any new information from analyses/evaluations which would affect the accuracy of the NRC produced Sequence of Events

2) Specifically, is anything more known about MSIV actuation from the data evaluations

3) What is the latest information on the operation of startup up feed control valve SP-7A

4) What plant data exists to determine if the startup feed pump flow was the first flow to go to steam generator #1

6/21/85

Q's for SOE meeting scheduled for 4:00PM

✓1. ? actual conditions regarding feed flow to S/G #2
via S/U FP.

✓2. Operator's indications re SP-7A, item #1.

X ~~3. Start of 2nd AFW pump; Cotthouse iso~~

✓4. Best avail info re: actuation of SFRCS at 01:35:31
what equip responded
what equip did not respond.
seal-in functions.

X ~~5. Any design info or test info re: capability of S/G feed iso valves (AF 599, 608) to re-open under press. conditions of June 9, 1985.~~

X ~~6. Is use of SFRCS "Initial Reset & Bypass" directed, suggested, or allowed by procedures during plant incident or during transient.~~

7. Best avail. info re. PORV performance
on 3rd lift.
Anything else happen at precisely same time.

8. Cause(s) of RCS press fall between
~01:51:18 → 01:51:42

9. Best avail explanation of SG#1 press. trace between approx. 01:47:30 → 01:50 (gentle rounding.)

10. Best avail explanation for S/G#1 press between ~ 01:50 – 01:51:17 (sharp fall)

11: Best avail sequence of operation of Steam Stm. Dumps Valves. auto & manual.

Exhibit #2

TOLEDO EDISON

Richard P. Crouse
Vice President
Nuclear
(419) 259-5221

Docket No. 50-346

License No. NPF-3

Serial No. 1100

November 12, 1984

File in Tech Spec Submittal Notebook

Director of Nuclear Reactor Regulation
Attention: Mr. John F. Stolz
Operating Reactor Branch No. 4
Division of Operating Reactors
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stolz:

Under separate cover, we are transmitting three (3) original and forty (40) conformed copies of an application for Amendment to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station Unit No. 1.

This application requests that the Davis-Besse Nuclear Power Station Unit No. 1 Operating License be amended to include the attached License Condition 2.C.(3)(t) regarding the Startup Feedwater Pump (SUFP).

The attachments identify the change and provide the Safety Evaluation and Significant Hazards Consideration for the License Condition concerning the SUFP. The NRC has requested Toledo Edison (TED) submit a License Condition for the operation of the SUFP and completion of the new SUFP system. This request results from a TED submittal dated October 18, 1984 (Serial No. 1070) concerning NRC review of the SUFP operation and approval for interim operation until the new SUFP is operable. The new SUFP project will be designed, procured and installed under the procedures of the Integrated Living Schedule Program and will be designated a Category A item. Any changes to the committed schedule will, therefore, require the prior approval of the NRC. This submittal is a revision to our letter dated October 18, 1984 and, therefore, no fee is incurred.

Toledo Edison requests expeditious review of this application to support the December, 1984 plant startup from the current refueling outage.

Very truly yours,

RP Crouse

RPC:GAB:lah

cc: DB-1 NRC Resident Inspector ✓
State of Ohio

~~8411160104~~

Page	Line	Correction or change and reason therefor
87	20	ALL REFERENCES TO "MAINTENANCE WORK ORDERS"
|	|	SHOULD BE CHANGED TO "SURVEILLANCE OR PERFORMANCE
		TESTS"

Page No. 1 DATE 6/25/85 SIGNATURE [signature]

Page	Line	Correction or change and reason therefor
105	20	The alarm when it comes on, it stays---
106	6	'fluctuation' should be 'actuation'
106	11	'reseat' should be "receipt"
107	5	Delete "but"
107	15	Change to 'spikes in the output of level string
↓	16	change 'strength' to 'string'
↓	21	'strength' to string
108	8	Change 'remain' to 'went' or go'
108	9	Change 'actual' to 'actuated'

Page No. ____ DATE 6/25/85 SIGNATURE [signature]

Page	Line	Correction or change and reason therefor	e = STENO error
96	6-8	ERROR IN MR ROSSI'S STATEMENT - DOESN'T	?
		MAKE SENSE AS WRITTEN	
96	15	ADD "ON" TO BEGINNING OF SENTENCE	e
96	20	"THAT" TO IT	e
99	14	BEFORE "EACH" ADD "FOR"	e
99	14	BEFORE "HAVE" ADD "WE"	e
100	7	DELETE "AUX"	e
100	10	Delete "IN"	e
100	23	CHANGE "RIGHT, THAT" TO "THE LIGHT"	e
101	16	PUT ("THE LIGHT THAT DIDN'T COME ON") IN ( )	CLARIFICATION
101	19	CHANGE "CHROME" TO "CONTROL ROOM"	e
102	15	AFTER "SHOW" ADD "FLOW"	e
103	3	AFTER "BOTH" ADD "HAD FLOW"	e
104	14	"EXTRA" SHOULD BE "NEXT"	e
104	20	CHANGE TO "QUESTION #2 IS A LITTLE BIT CONFUSING; YOU"	e
113	13	Delete "ON here" (CLARIFICATION)	
113	21	PLACE (THEY ARE ESSENTIALLY DRY AT THIS POINT IN TIME) IN ( )	(TO CLARIFY)
113	24	PERIOD AFTER HEATING UP.	(CLARIFICATION)
113	24	AFTER PERIOD "THAT DROP IN RCS PRESSURE IN	e
114	1	Delete "there"	e
115	6	NOT MR BEARD — MR BATCH	e
115	18	CHANGE "INTO" TO "UP TO"	e
116	18	Delete "AND" PUT PERIOD & CAPITALIZE He (MAKE 2 SENTENCES)	
116	20	Delete — ADD "to"	

Page No. 1 DATE 6/26/85 SIGNATURE [signature]

[signature] STAS

Page	Line	Correction or change and reason therefor	
117	6	CHANGE "IN" TO "ALONG WITH"	e
120	8	Delete 2ND "THAT"	e
121	15	CHANGE "WEAKENING" TO "WEEPING"	e
122	16	I BELIEVE THIS WAS NOT MR MURRAY	?
		BUT MR ROGERS	
123	11	Delete ? put .	e
124	3	AFTER MOVE ADD "THE VALVES"	CLARIFICATION
124	5	REMOVE 1ST THAT ADD "AN ALARM"	e
126	12	Delete "DOOR"	e
127	2	Delete "ZONES"	e
127	3	CHANGE TO "TWO LIGHTS ARE LIT ON THIS SIDE FOR THAT DOOR",	e
127	5	CHANGE "TWO" TO "BOTH" Delete "ON THAT"	e
127	6	CHANGE "MAKE" TO "MAKING"	e
127	6	AFTER COUPLE ADD "OF"	e
127	7	AFTER LIGHTS ADD "ARE", Delete "THERE"	e
127	8	Delete "THOUGH"	e
127	12	IT IS ST 5099.01 NOT --	e
130	9	-- SHOULD BE CORRELATE	e
130	22	"EVENT" SHOULD BE "EVENTS"	e
133	19	Delete First "IS"	e
133	23	"LAKE" SHOULD BE "LEG"	e
134	5	"PRESSURIZES" SHOULD BE "PRESSURIZER"	e

Page No. 2 DATE 6/26/85 SIGNATURE [signature]

APPLICATION FOR AMENDMENT

TO

FACILITY OPERATING LICENSE NO. NPF-3

FOR

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NO. 1

Enclosed are forty-three (43) copies of the requested change to the Davis-Besse Nuclear Power Station Unit No. 1 Facility Operating License No. NPF-3, with the Safety Evaluation for the requested change.

This amendment request concerns License Condition 2.C.(3)(t).

By /s/ C. T. Daft
C. T. Daft, Director
Quality Assurance

For R. P. Crouse
Vice President, Nuclear

Sworn and subscribed before me this 12th day of November, 1984.

/s/ Nora Lynn Flood
Notary Public, State of Ohio
My Commission Expires 9/1/87

S E A L

Docket No. 50-346
License No. NPF-3
Serial No. 1100
November 12, 1984

Attachments

I. Changes to the Davis-Besse Nuclear Power Station Unit No. 1, Operating License No. NPF-3 License Condition.

   A. Time required to Implement: This change is to be effective upon NRC approval.

   B. Reason for Change (Facility Change Request 84-192).

      NRC requested this License Condition to allow startup and operation utilizing the current SUFP configuration.

   C. Safety Evaluation
      (See Attached)

   D. Significant Hazards Consideration
      (See Attached)

SAFETY EVALUATION

This License Condition (2.C.(3)(t)) is being submitted to allow startup and operation, utilizing the current Startup Feedwater Pump (SUFP) configuration, of the Davis-Besse Nuclear Power Station Unit No. 1 until a new SUFP system can be installed.

The SUFP system itself performs no safety function. It is, however, used as a backup to the main and auxiliary feedwater systems for supplying water to the steam generators in case of the total loss of these two systems.

The SUFP is located in Room 238 which is the same room as one of the auxiliary feedwater pumps (AFP), pump 1-2. During the review of the high/moderate energy line break criteria as it relates to the AFW rooms, it was determined that the SUFP system can jeopardize AFP 1-2 in the event of a pipe leak or rupture.

The SUFP system is non-seismic downstream of valve FW91. The suction of the SUFP system utilizes the Deaerator or the Condensate Storage Tank (CST) as its water source. The normal source is the Deaerator. It also uses the non-seismic Turbine Plant Cooling Water (TPCW) system for pump cooling. The line from the Deaerator to the SUFP also runs through Room 237. Room 237 contains AFP 1-1. The discharge line from the SUFP is aligned to the inlet of the high pressure feedwater heaters. In the past, these systems were not valved off so as to provide immediate backup for the Main Feedwater System (MFWS) if needed.

Since this problem was identified, the SUFP system and the TPCW system in Rooms 237 & 238 have been isolated. This isolation has been accomplished by closing valve FW91 from the CST, valve FW32 from the Deaerator, valves CW196 and CW197 used for pump cooling, and valve FW106 in the discharge line.

The main concern with the location of the SUFP is the potential for pipe whip and jet impingement in Room 238 and flooding and high temperature in Room 237 or 238. The main concern with the TPCW system is the potential for flooding these rooms. These concerns will only be realized when the SUFP system is in operation, i.e., during startup and shutdown of the reactor.

During the period that the startup pump is running, the suction piping to the SUFP is a moderate energy line based on the criteria in USAR Section 3.6 which states that a line outside containment operating above 275 psig or 200°F is a moderate energy line. The discharge piping from the SUFP is a high energy line based on the USAR Section 3.6 which states that a line outside containment operating both above 275 psig and 200°F is a high energy line. If a high energy line is in service more than six hours, Section 3.6 requires that it must be analyzed for pipe rupture. The SUFP system could be in operation for as long as 72 hours for one reactor startup and shutdown cycle but can be in operation for as long as 168 hours during zero power physics testing. The total number of reactor trips in a year is conservatively assumed to be 8 times. The TPCW system

supply and return piping are neither moderate nor high energy lines but, since the lines are non-seismic, a flooding concern remains.

It has been postulated that during a seismic event the pipes would rupture in such a manner as to damage AFP 1-2 or possibly AFP 1-1 due to flooding. It has also been postulated that with a high energy pipe break the sheared pipe could damage AFP 1-2 due to jet impingement or pipe whip and would cause high temperatures and pressure in Room 238. A moderate energy pipe break could damage either AFP 1-1 or 1-2 due to flooding and high temperature in either Room 237 or 238. A rupture/break in the TPCW piping could damage either AFP 1-1 or 1-2 due to flooding in either Room 237 or 238.

A Probability Risk Assessment (PRA) study has been performed since this situation was discovered. The attached PRA justification documents that the worst case probability for a rupture/break in the SUFP and the TPCW piping causing the failure of AFP 1-2 is of the order of 3.2E-6/yr. The probability for failure of AFP 1-1 in Room 237 is smaller due to less SUFP piping in the room. The probability for failure of the AFWS due to pipe rupture/break in the SUFP and the TPCW system is documented in calculation number C-NSA-45.02-2. This probability is insignificant in light of the AFWS unavailability on the order of 1E-2/yr. for each train which was developed by EDS (now Impell) and submitted to the NRC in December, 1981.

Although these risks to the AFWS from the SUFP system are considered insignificant, the SUFP suction and discharge piping were hydrotested in the Fall, 1984, to the original acceptance criteria (ANSI B31.1) to ensure the integrity of the SUFP suction and discharge piping. In addition, certain precautionary measures will be observed during SUFP operation, when the SUFP is not being used as a source of auxiliary feedwater. An operator shall be positioned at the AFW room area when the SUFP is operating in Modes 1 through 3. Upon indication of a pipe leak the operator will either trip the SUFP locally or contact the control room to trip the SUFP. This may not reduce the probability for a pipe break in the SUFP system, however, it will reduce significantly the impact of a SUFP system failure resulting in a AFWS failure. He would then close all SUFP isolation valves which are external to the AFW rooms. This operator action is being taken since piping leaks are expected to occur prior to any complete piping rupture. If the SUFP is being used as a source of auxiliary feedwater, specific direction appropriate to the s uation will be provided to the operator by the shift supervisor.

Pursuant to the above it has been determined that the use of the SUFP system on an interim basis, until the new SUFP system is installed, does not significantly increase the probability of the loss of the AFWS.

SIGNIFICANT HAZARDS CONSIDERATION

The proposed License Condition (2.C.(3)(t)) would allow startup and operation of the Davis-Besse Nuclear Power Station Unit No. 1 utilizing the current Startup Feedwater Pump (SUFP) and also require installation of a new SUFP. This License Condition does not represent a Significant Hazard.

Toledo Edison (TED) was requested by the NRC Staff to submit a License Condition (2.C.(3)(t)) to allow startup and operation utilizing the current SUFP. This request results from TED submittal dated October 18, 1984 (Serial No. 1070) concerning NRC review of the SUFP operation and approval for interim operation until the new proposed SUFP is operable. The October 18, 1984 letter identified that SUFP system itself performs no safety function. It is, however, used as a backup to the main and auxiliary feedwater systems for supplying water to the steam generators in case of the total loss of these two systems.

The SUFP is located in Room 238 which is the same room as one of the auxiliary feedwater pumps (AFP), pump 1-2. During the review of the high/moderate energy line break criteria as it relates to the AFW rooms, it was determined that the SUFP system can jeopardize AFP pump 1-2 in the event of a pipe leak or rupture.

The SUFP system is non-seismic downstream of valve FW91. The suction of the SUFP system utilizes the Deaerator or the Condensate Storage Tank (CST) as its water source. The normal source is the Deaerator. It also uses the non-seismic Turbine Plant Cooling Water (TPCW) system for pump cooling. The line from the Deaerator to the SUFP also runs through Room 237. Room 237 contains AFP 1-1. The discharge line from the SUFP is aligned to the inlet of the high pressure feedwater heaters. In the past, these systems were not valved off so as to provide immediate backup for the Main Feedwater System (MFWS) if needed.

Since this problem was identified, the SUFP system and the TPCW system in Rooms 237 & 238 have been isolated. This isolation has been accomplished by closing valve FW91 from the CST, valve FW32 from the Deaerator, valves CW196 and CW197 used for pump cooling, and valve FW106 in the discharge line.

The main concern with the location of the SUFP is the potential for pipe whip and jet impingement in Room 238 and flooding and high temperature in Room 237 or 238. The main concern with the TPCW system is the potential for flooding these rooms. These concerns will only be realized when the SUFP system is in operation, i.e., during startup and shutdown of the reactor.

During the period that the startup pump is running, the suction piping to the SUFP is a moderate energy line based on the criteria in USAR Section 3.6 which states that a line outside containment operating above 275 psig or 200°F is a moderate energy line. The discharge piping from the SUFP is a high energy line based on the USAR Section 3.6 which states that a line outside containment operating both above 275 psig and 200°F is a high

energy line. If a high energy line is in service more than six hours, Section 3.6 requires that it must be analyzed for pipe rupture. The SUFP system could be in operation for as long as 72 hours for one reactor startup and shutdown cycle but can be in operation for as long as 168 hours during zero power physics testing. The total number of reactor trips in a year is conservatively assumed to be 8 times. The TPCW system supply and return piping are neither moderate nor high energy lines but, since the lines are non-seismic, a flooding concern remains.

It has been postulated that during a seismic event the pipes would rupture in such a manner as to damage AFP 1-2 or possibly AFP 1-1 due to flooding. It has also been postulated that with a high energy pipe break the sheared pipe could damage AFP 1-2 due to jet impingement or pipe whip and would cause high temperatures and pressure in Room 238. A moderate energy pipe break could damage either AFP 1-1 or 1-2 due to flooding and high temperature in either Room 237 or 238. A rupture/break in the TPCW piping could damage either AFP 1-1 or 1-2 due to flooding in either Room 237 or 238.

A Probability Risk Assessment (PRA) study has been performed since this situation was discovered. The attached PRA justification documents that the worst case probability for a rupture/break in the SUFP and the TPCW piping causing the failure of AFP 1-2 is of the order of 3.2E-6/yr. The probability for failure of AFP 1-1 in Room 237 is smaller due to less SUFP piping in the room. The probability for failure of the AFWS due to pipe rupture/break in the SUFP and the TPCW system is documented in calculation number C-NSA-45.02-2. This probability is insignificant in light of the AFWS unavailability on the order of 1E-2/yr. for each train which was developed by EDS (now Impell) and submitted to the NRC in December, 1981.

Although these risks to the AFWS from the SUFP system are considered insignificant, the SUFP suction and discharge piping were hydrotested in the Fall, 1984, to the original acceptance criteria (ANSI B31.1) to ensure the integrity of the SUFP suction and discharge piping. In addition, certain precautionary measures will be observed during SUFP operation, when the SUFP is not being used as a source of auxiliary feedwater. An operator shall be positioned at AFP room area when the SUFP is operating in Modes 1 through 3. Upon indication of a pipe leak the operator will either trip the SUFP locally or contact the control room to trip the SUFP. This may not reduce the probability for a pipe break in the SUFP system, however, it will reduce significantly the impact of a SUFP system failure causing a AFWS failure. He would then close all SUFP isolation valves which are external to the AFP rooms. This operator action is being taken since piping leaks are expected to occur prior to any complete piping rupture. If the SUFP is being used as a source of auxiliary feedwater, specific direction appropriate to the situation will be provided to the operator by the shift supervisor.

The Commission has provided examples of amendments which are not likely to involve a significant hazards consideration (48 FR 14870), such as a change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications: for example, a more stringent surveillance requirement (example ii). The License Condition will require TED to isolate the piping on the SUFP and station an operator in the SUFP/AFW pump area when the SUFP is operating. Also, the License

Condition will require TED to install a SUFP and associated piping and valves external to Rooms 237 and 238 before commencing startup of Cycle 6. These conditions are not presently required of TED and constitutes an additional restriction on operation and plant modification.

Based on the above information, this amendment request would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; or 2) create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) involve a significant reduction in a margin of safety.

Therefore, based on the above, the requested license amendment does not present a Significant Hazard.

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NO. 1

LICENSE CONDITION

2.C.(3)(t)

Toledo Edison shall operate the Startup Feedwater Pump (SUFP) System with the following operational restrictions:

1. Toledo Edison will station an operator in the Startup Feedwater Pump/Auxiliary Feedwater Pump (SUFP/AFW) area during operation of the SUFP to monitor SUFP/Turbine Plant Cooling Water (TPCW) piping status in the AFW Pump Rooms. In the event of SUFP/TPCW pipe leakage, the operator will trip the SUFP locally or notify the Control Room to trip the SUFP, and isolate the SUFP/TPCW piping.

2. Toledo Edison will isolate and maintain isolation outside the SUFP/AFW area of the SUFP suction, discharge, and turbine plant cooling water piping, when the SUFP is not in operation (Modes 1, 2, and 3).

3. Toledo Edison will install a SUFP, associated piping, and valves, to remove the hazards to the AFW pumps before commencing Cycle 6.

PRA Based Justification for Continued Operation of the SUFP System

During SUFP operation, a portion of the SUFP discharge line and the minimum recirculation line renders itself to consideration as a high energy line. Similarly, a portion of the suction piping because of use with the deaerator storage tank water requires consideration as a moderate energy line. The immediate safety impact of lack of such high/moderate energy line considerations is on the Auxiliary Feedwater System (AFWS) since the SUFP is housed in the same room as Auxiliary Feedpump (AFP) 2. A portion of the discharge and minimum recirculation lines is routed in this same room. In addition, the suction path from the deaerator runs through both AFP rooms. With SUFP in operation while taking suction from the deaerator the following lengths of pipes may pose a challenge to the AFWS in view of the high/moderate energy line breaks. Further, the non-seismic TPCW piping, in AFP rooms (as noted below) poses a potential flooding concern.

17 feet of 4" discharge line in AFP Room 2
18 feet of 1½" minimum recirculation line in AFP Room 2
27 feet of 6" suction line in AFW Room 2
27 feet of 6" suction line in AFP Room 1
62 feet of 4" TPCW line in AFP Room 1
40 feet of 4" in AFP Room 2
24 feet of ≤2" in AFP Room 2

The SUFP may be operated continuously for a period of approximately 72 hours every reactor shutdown/startup. In a year with a fuel reload, the SUFP system may be operated for as long as 168 hours continuously during zero power physics testing. Based on the expected number of reactor trips in a year (conservatively assuming ten trips per non-refueling year and eight trips in a refueling year) the maximum number of hours per year that the SUFP system would be in operation for any year is of the order of 744 hours.

The overall figure of merit for any one train of the Davis-Besse AFWS is of the order of 10^{-2} per year as deduced from the EDS (now Impell) AFWS PRA study (EDS Report No. 02-1040-0195, Revision 1) submitted to the NRC in December 1981. This implies that one train of the Davis-Besse AFWS will be unavailable with a frequency of 10^{-2} per year for all initiating events which may require availability of AFWS.

Assuming the duration of SUFP operation per year presented above, the total worst case probability of any break (whether high energy, moderate energy or seismic) in the unanalyzed piping which may challenge the availability of an AFW train is of the order of 3.2×10^{-6} per year. The worst case probability is for AFW train 2 because of significantly larger overall length of piping in this room. This evaluation conservatively assumes that any rupture of this non-seismic piping will flood the room to the extent of causing train inoperability with a probability of unit.

Since the SUFP system failure as postulated above poses a challenge to the AFW train at a frequency of 3.2×10^{-6} per year, the probability of such SUFP system ruptures/breaks leading to inoperability of an AFW train is

insignificant as compared to other failures that may render the AFW system inoperable. It is, therefore, concluded that the above issue poses a very minimal risk to the accomplishment of the safety function of the AFWS and an extremely negligible risk to public health and safety. Continued operation of the SUFP system in the mode evaluated above is, therefore, adequately justified.

Exhibit 3

CONFIRMATORY ACTION LETTER

RIII-CAL-85-03

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

APR 26 1985

RMS ... A85-319B

T. D. Murray
UIS Consultants
W. A. Johnson
STA
SAR-Up
Site NL
SMQuennoz
MLStewart
JKWood
CNRB
SRB

RECEIVED
APR 29 1985
TOLEDO EDISON

Docket No. 50-346

Toledo Edison Company
ATTN: Mr. Richard P. Crouse
Vice President
Nuclear
Edison Plaza
300 Madison Avenue
Toledo, OH 43652

Gentlemen:

This refers to the discussion between Mr. J. Wood of your staff and Mr. D. Danielson of my staff on April 26, 1985, regarding activities you will take as a result of the damage recently sustained by several auxiliary steam supply feed pump turbine (AFPTSS) piping restraints on Davis-Besse Unit 1. With regard to the matters discussed, we understand you will take the following actions:

1. Complete by May 10, 1985, piping system operability stress analyses local to the following damaged restraints utilizing loads calculated to have been necessary to cause the degraded conditions of the restraints:

   - AFPTSS Train No. 1-1: 3A-EBD-19-H144
     3A-EBD-19-H145
   - AFPTSS Train No. 1-2: 3A-EBD-19-H14
     3A-EBD-19-H58

   The maximum piping transient stress (St) will be combined with applicable primary pipe stresses utilizing the following formula:

$$Sp + Sw + \sqrt{Se^2 + St^2} \leq 2.4\ Sh$$

   Where Sp = pressure stress at operating condition
   Sw = stress due to piping and attachment dead weights
   Se = stress calculated based on plant safe shutdown earthquake (SSE) condition
   Sh = ANSI B31.1 pipe stress allowable for hot operating condition

Item 1 - 5/10
Item 2 - 5/24
Item 3 - 6/5

1-1154

SBW 4/30/85

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2. Complete by May 24, 1985, torque tests for concrete expansion anchor bolts installed on all snubbers and rigid supports and restraints shown on the following drawings:

   - HL-203F, "Main Steam System, Supply to Auxiliary Feed Pump Turbine No. 1-1", Revision 0.
   - HL-203H, "Main Steam System, Supply to Auxiliary Feed Pump Turbine No. 1-2 and Exhaust", Revision 1.

3. Initiate by June 5, 1985 an inservice testing program for the AFPTSS piping system to determine the cause and nature of the system transient that resulted in the degradation of the restraints identified during previous inspections by Toledo Edison Company (TECo). The program will include load and displacement measurements under cold and hot operating conditions for the following piping alignments:

   - Steam supply to auxiliary feed pump turbine (AFPT) No. 1-1 from Steam Generator (SG) No. 1-1.
   - Steam supply to AFPT No. 1-2 from SG No. 1-2.
   - Steam supply to AFPT No. 1-1 from SG No. 1-2 through crossover leg.
   - Steam supply to AFPT No. 1-2 from SG No. 1-1 through crossover leg.

   Prior to initiation of the program, TECo will:

   a. Develop test procedures including identification of instrumentation locations on isometric drawings and delineation of personnel interface and responsibilities during the tests.

   b. Implement calibration procedures for all instrumentation including the strain gauges installed on snubbers and rigid restraints.

   c. Establish evaluation criteria for load and displacement measurements based on the calculated SSE values.

   An inservice test that requires steam flow to AFPT Nos. 1-1 and 1-2 from SG Nos. 1-1 and 1-2 utilizing both main and crossover legs will not be conducted until previous tests performed under this program have been evaluated, necessary precautionary measures have been established, and test provisions and measures have been reviewed by Region III.

4. Conduct piping system surveillance during the monthly auxiliary feed pump tests and the quarterly system response tests performed under normal plant operating conditions and during testing conducted as a result of plant shutdowns or maintenance activities.

5. Report to Region III verbally within 2 working days, with a followup written report within 72 hours, any defective structures and piping suspension system components discovered during the activities described in Items 1 through 4 above or following any actuation of the auxiliary feedwater system. Such reports will not relieve TECo of the obligation to submit any reports required by 10 CFR 50.72.

Please inform us immediately if your understanding of these actions are different from that stated above.

Charles E. Norelius
for James G. Keppler
Regional Administrator

cc: S. Quennoz, Station
Superintendent
DMB/Document Control Desk (RIDS)
Resident Inspector, RIII
Harold W. Kohn, Ohio EPA
James W. Harris, State of Ohio
Robert H. Quillin, Ohio
Department of Health
R. Baer, IE
F. Cherney, MEB, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

Exhibit 4

NOV 2 1 1984

Docket No. 50-346
EA 84-95

Toledo Edison Company
ATTN: Mr. John P. Williamson
Chairman and Chief Executive Officer
Edison Plaza
300 Madison Avenue
Toledo, OH 43652

Gentlemen:

This refers to the safety inspection conducted by Messrs. W. G. Rogers and D. C. Kosloff of the Region III staff during the period June 11 through July 27, 1984 of activities at the Davis-Besse Nuclear Power Station authorized by Operating License No. NPF-3. The results of the inspection were discussed on July 13, 1984 during an Enforcement Conference held in the Region III office between Mr. R. P. Crouse and others of your staff and Mr. C. E. Norelius and other members of the NRC staff and on October 2, 1984 during a meeting between Mr. W. A. Johnson and others of your staff and Messrs. R. C. DeYoung and J. G. Keppler of the NRC. The following violations were identified during the inspection.

On May 7, 1984, both Control Room Emergency Ventilation System (EVS) chiller control switches were discovered in the "off" position. This rendered both Control Room EVS trains inoperable. Your program failed to recognize the technical specification requirements for the operability of the equipment and your program failed to ensure that procedures were followed to verify the operability of the equipment.

On November 1, 1983, one of the two ventilation fans for the Number One Emergency Diesel Generator was removed from service. You failed to recognize that removal of this ventilation fan from service represented a change in the facility as described in the Updated Safety Analysis Report (USAR). This change affected the design basis requirements for equipment operability. In addition, the required review in accordance with 10 CFR 50.59 was not conducted.

On December 19, 1982, you initiated a Facility Change Request that was implemented on May 24, 1983 that changed the position of the suction valve to the startup feed pump to the open position instead of closed as required by the design basis analysis for flood protection. On May 14, 1984, you determined one auxiliary feedwater pump was inoperable as this valve was open contrary to USAR requirements. You immediately closed the suction valve and modified procedures to control the opening and closing of this valve. During recovery activities following a unit trip on June 25, 1984, the suction valve was routinely used for unit startup. On July 1, 1984, you again discovered the suction valve was open rather than

CERTIFIED MAIL
RETURN RECEIPT REQUESTED

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closed. A review of these occurrences determined that an adequate 10 CFR 50.59 review was not conducted, that approved procedures for operating the system were not followed, and that operators failed to implement the corrective actions you initiated following the discovery of this problem on May 14, 1984. In addition, a recent Performance Appraisal Inspection identified additional deficiencies with regard to the conduct of reviews in accordance with the requirements of 10 CFR 50.59. This inspection also identified two examples when on March 8, 1984 and May 4, 1984, lead shielding was hung on decay heat piping and no safety evaluations in accordance with 10 CFR 50.59 were performed.

These events indicate the need for significant improvement in your ability: 1) to recognize the design basis and technical specification requirements for equipment operability and to ensure that these requirements are met when equipment is removed from service and 2) to ensure that procedures which define requirements for equipment operability are followed.

These events also indicate the need to ensure that adequate corrective actions are taken to preclude repetition of identified deficiencies. During the September 23, 1982, Systematic Assessment of Licensee Performance (SALP), we identified a weakness in your ability to recognize design basis requirements for equipment operability. The NRC Region III staff restated this concern during an Enforcement Conference on March 9, 1983 and again during the October 28, 1983 SALP. As a result of the March 9, 1983 Enforcement Conference, you committed to implement a Comprehensive Corrective Action Program to address these and other concerns. You also assured us that other administrative measures were being implemented to deal with these problems. However, your corrective actions have been ineffective as evidenced by your failures to recognize design basis requirements for safety-related equipment/systems.

To emphasize the need for the licensee: (1) to recognize the importance of design basis and technical specification requirements for equipment operability and to ensure that these requirements are met when equipment is removed from service, (2) to ensure that procedures which define the requirements for equipment operability are followed, (3) to ensure that appropriate reviews are conducted in accordance with the requirements of 10 CFR 50.59, and (4) to ensure that adequate corrective actions are taken to preclude repetition of identified problems, I have been authorized, after consultation with the Deputy Director, Office of Inspection and Enforcement, to issue the enclosed Notice of Violation and Proposed Imposition of Civil Penalties in the cumulative amount of Ninety Thousand Dollars ($90,000) for the violations described in the enclosed Notice. The violations have been categorized in the aggregate as two Severity Level III problems in accordance with the General Policy and Procedure for Enforcement Actions, 10 CFR Part 2, Appendix C, and the Policy as revised, 49 FR 8583 (March 8, 1984).

The base civil penalty for Item I is $50,000. The base civil penalty for Item II is $40,000 because two of the violations identified occurred prior to the revisions to the recent Enforcement Policy.

You are required to respond to the enclosed Notice and you should follow the instructions specified therein when preparing your response. Your response should specifically address the corrective actions you will take to increase management involvement and oversight and to reduce personnel errors. Your reply to this letter and the results of future inspections will be considered in determining whether further enforcement action is warranted.

In accordance with 10 CFR 2.790, "Rules of Practice," a copy of this letter and the enclosure will be placed in the NRC Public Document Room.

The responses directed by this letter and the accompanying Notice are not subject to the clearance procedure of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

Sincerely,

James G. Keppler

James G. Keppler
Regional Administrator

Enclosures:
1. Notice of Violation and Proposed Imposition of Civil Penalties
2. Inspection Report No. 50-346/84-15(DRP)

cc w/encls:
T. D. Murray, Station Superintendent
Harold W. Kohn, Ohio EPA
James W. Harris, State of Ohio
Robert H. Quillin, Ohio Department of Health

Distribution
PDR
LPDR
NSIC
SECY
ACRS
CA
JTaylor, IE
JAAxelrad, IE
BBeach, IE
JLieberman, ELD
Enforcement Coordinators
 RI, RII, RIII, RIV, RV
VStello, DED/ROGR
FIngram, PA
JKeppler, RIII
SConnelly, OIA
BHayes, OI
HDenton, NRR
RStark, NRR
JCrooks, AEOD
NGrace, IE
Project Manager, NRR
Reisdent Inspector
State Attorney General
IE:ES File
IE:EA File
EDO Rdg File
DCS

IE:ES	ELD	RIII	ES:	IE:OD
BBeach	JLieberman	JKeppler	JAAxelrad	JTaylor
11/14/84	11/8/84	11/14/84	11/14/84	11/14/84
RIII	RIII	RIII	RIII	RIIII
Shafer/jp	Norelius	Schultz	Davis	Keppler
11/19/84		11-20-84		

NOTICE OF VIOLATION
AND
PROPOSED IMPOSITION OF CIVIL PENALTIES

Toledo Edison Company
Davis-Besse Nuclear Power Station

Docket No. 50-346
License No. NPF-3
EA 84-95

An inspection conducted during the period June 11 through July 27, 1984 at the Davis-Besse Nuclear Station identified a number of violations of NRC requirements. These violations relate to the licensee's inability to recognize the importance of design basis and technical specification requirements for equipment operability, to ensure procedures which define requirements for equipment operability are followed, and to ensure that the requirements of 10 CFR 50.59 are satisfied. The violations involving equipment operability also relate to the licensee's failure to take effective corrective actions once problems have been identified.

To emphasize the need for the licensee: (1) to recognize the importance of design basis and technical specification requirements for equipment operability and to ensure that these requirements are met when equipment is removed from service, (2) to ensure that procedures which define the requirements for equipment operability are followed, (3) to ensure that appropriate reviews are conducted in accordance with the requirements of 10 CFR 50.59, and (4) to ensure that adequate corrective actions are taken to preclude repetition of identified problems, I propose to impose civil penalties in the cumulative amount of $90,000. The base civil penalty for Item I is $50,000. The base civil penalty amount for Item II is $40,000 because two of the violations occurred prior to the revisions to the recent Enforcement Policy.

In accordance with the General Policy and Procedure for Enforcement Actions, 10 CFR Part 2, Appendix C, and the Policy as revised, 49 FR 8583 (March 8, 1984) and pursuant to Section 234 of the Atomic Energy Act of 1954, as amended, 42 U.S.C. 2282, PL 96-295, and 10 CFR 2.205, the particular violations and the associated civil penalties are set forth below.

I.A. Technical Specification 3.7.6.1, "Control Room Emergency Ventilation System," requires that two independent control room emergency ventilation systems shall be operable. A system is considered operable when it is capable of performing its specified function(s).

Technical Specification 6.8.1.a requires that written procedures be established, implemented and maintained covering the activities specified in Appendix A of Regulatory Guide 1.33, November 1972. Appendix A specifies typical safety-related activities that should be covered by written procedures. This includes procedures for operation of the control room emergency ventilation systems.

Administrative Procedure (AD) 1839.00, "Station Operations," requires that, prior to removal of safety-related equipment from service, operability of redundant safety-related equipment must be verified by inspection. In addition, this procedure requires that the

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applicable technical specification action statements be evaluated prior to the removal of the safety-related equipment from service.

Contrary to the above, both trains of the Control Room Emergency Ventilation System were removed from service on April 23, 1984 through May 7, 1984 without verifying the operability of the redundant equipment or evaluating applicable technical specification action statements. This rendered the Control Room Emergency Ventilation System inoperable in violation of technical specification requirements.

B. Technical Specification 6.8.1 requires that written procedures be be established, implemented and maintained covering the activities specified in Appendix A of Regulatory Guide 1.33, November 1972. The activities specified in Appendix A, Section A, Administrative Procedures, include procedure adherence, shift and relief turnovers and log entries. Appendix A, Section C, "Procedures for Startup, Operation, and Shutdown of Safety Related PWR Systems," list the feedwater system as requiring instructions for energizing startup and shutdown of the system.

Contrary to the above, on June 24, 1984, the licensee failed to start the startup feed pump in accordance with the applicable sections of the approved procedures (SP 1105.27 and SP 1106.27) for operation of the startup feed pump; failed to log the starting of the startup feed pump in the reactor operator's log; and improperly initialed the trip recovery procedure (PP 1102.03) indicating that the startup feed pump was started per an approved procedure (SP 1106.27). In addition, on June 25, 1984, the licensee failed to shutdown or restore the startup feed pump to normal in accordance with the applicable sections of the approved procedures (SP 1105.27 and SP 1106.27); improperly initialed the plant startup procedure (PP 1102.2) indicating the startup feed pump was stopped per the approved procedure (SP 1106.27); failed to perform an adequate turnover regarding the status of the startup feed pump system; and failed to properly sign off the completion of Section 8 of the startup procedure (PP 1102.2).

Collectively, these two violations have been evaluated as a Severity Level III problem. (Supplement I)
(Cumulative Civil Penalties $50,000 assessed equally among the violations.)

II. 10 CFR 50.59(a)(1) states that the license may make changes in the facility as described in the safety analysis report...without prior Commission approval provided that the proposed change...does not involve a change in the technical specifications incorporated in the license or an unreviewed safety question.

10 CFR 50.59 requires that the licensee maintain records of changes in the facility to the extent that such changes constitute changes to the facility as described in the safety analysis report. These records shall include a written safety evaluation which provides the bases for the determination that the change does not involve an unreviewed safety question.

Contrary to the above, in the following instances, the licensee made changes in the facility as described in the safety analysis report without preparing a written safety evaluation of whether the change involved a change in the technical specifications or an unreviewed safety question.

(1) On November 1, 1983, the licensee removed one of two Emergency Diesel Generator (EDG) ventilation supply fans from service without preparing a written safety evaluation and without realizing this action represented a change in the facility as described in the Updated Safety Analysis Report (USAR). The USAR describes the EDG ventilation supply as containing two 50% capacity fans.

(2) On December 19, 1982, the licensee initiated a Facility Change Request (FCR) that was implemented on May 24, 1983 which changed the position (to open) of the Startup Feedwater Pump (SUFP) suction valve during power operation without preparing a written safety evaluation. The USAR describes the valve as closed during power operation.

(3) On March 8, 1984 and May 4, 1984 lead shielding was hung on decay heat system piping changing the loading of the safety system as described in the FSAR and without preparing a written safety evaluation.

Collectively, the above violations have been evaluated as a Severity Level III problem (Supplement I).
(Cumulative Civil Penalties - $40,000 assessed equally among the violations.)

Pursuant to the provisions of 10 CFR 2.201, Toledo Edison Company is hereby required to submit to the Deputy Director, Office of Inspection and Enforcement, U. S. Nuclear Regulatory Commission, Washington, DC 20555 and a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region III, 799 Roosevelt Road, Glen Ellyn, IL 60137, within 30 days of the date of this Notice a written statement or explanation, including for each alleged violation; (1) admission or denial of the alleged violation; (2) the reasons for the violation, if admitted; (3) the corrective steps which have been taken and the results achieved; (4) the corrective steps which will be taken to avoid further violations; and (5) the date when full compliance will be achieved. Consideration may be given to extending the response time for good cause shown. Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Within the same time as provided for the response required above under 10 CFR 2.201, Toledo Edison Company may pay the civil penalties in the amount of $90,000 or may protest imposition of the civil penalties in whole or in part, by a written answer. Should Toledo Edison Company fail to answer within the time specified, the Deputy Director, Office of Inspection and Enforcement will issue an order imposing the civil penalties proposed above. Should Toledo Edison Company elect to file an answer in accordance with 10 CFR 2.205 protesting the civil penalties, such answer may: (1) deny the violation listed in the Notice, in whole or in part; (2) demonstrate extenuating circumstances; (3) show error in this Notice; or (4) show other reasons why the penalties should not be imposed. In addition to protesting the civil penalties, in whole or in part, such answer may request remission or mitigation of the penalty. In requesting mitigation of the proposed penalties, the five factors contained in Section V(b) of 10 CFR Part 2, Appendix C, as revised 49 FR 8583 (March 8, 1984) should be addressed. Any written answer in accordance with 10 CFR 2.205 should be set forth separately from the statement or explanation in reply pursuant to 10 CFR 2.201, but may incorporate statements or explanations by specific reference (e.g., citing page and paragraph numbers) to avoid repetition. Toledo Edison Company's attention is directed to the other provisions of 10 CFR 2.205, regarding the procedures for imposing a civil penalty.

Upon failure to pay any civil penalty due, which has been subsequently determined in accordance with the applicable provisions of 10 CFR 2.205, this matter may be referred to the Attorney General, and the penalty unless compromised, remitted, or mitigated, may be collected by civil action pursuant to Section 234c of the Act, 42 U.S.C. 2282.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Keppler

James G. Keppler
Regional Administrator

Dated at Glen Ellyn, Illinois
this 21st day of November 1984

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-346/84-15(DRP)

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

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

Facility Name: Davis-Besse 1

Inspection At: Oak Harbor, OH

Inspection Conducted: June 11 through July 27, 1984

Inspectors: W. Rogers

D. Kosloff

Approved By: I. N. Jackiw, Chief
Projects Section 2B

9-10-84
Date

Inspection Summary

Inspection on June 11 through July 27, 1984 (Report No. 50-346/84-15(DRP))
Areas Inspected: Special inspection of the circumstances surrounding three events: the discovery of both control room emergency ventilation systems being incapable of performing their air conditioning function; removal of an emergency diesel generator ventilation fan from service without declaring the diesel inoperable; and inoperable auxiliary feed pump due to an open startup feed pump suction valve. The inspection involved 30 inspector-hours onsite by two NRC inspectors including 4 inspector-hours onsite during off-shifts.
Results: Five items of noncompliance were identified (both trains of the control room emergency ventilation system made inoperable; emergency diesel ventilation supply fan taken out-of-service rendering the diesel generator inoperable; one auxiliary feedwater pump inoperable due to an open startup feed pump suction valve; procedures for startup feed pump and shift turnover not adhered to; improper 10 CFR 50.59 determination that changing the position of a SUFP valve did not constitute a change in the facility).

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DETAILS

1. Persons Contacted

T. Murray, Station Superintendent
B. Beyer, Assistant Station Superintendent
S. Quennoz, Assistant Station Superintendent
D. Miller, Operations Engineer
L. Simon, Operations Supervisor
J. Faris, Administrative Coordinator

The inspectors also interviewed other licensee employees, including members of the technical, operations, maintenance, I&C, training and health physics staff.

Enforcement Conference on July 13, 1984

Toledo Edison Personnel

R. P. Crouse, Vice President, Nuclear Mission
T. D. Murray, Station Superintendent
J. Helle, Engineering Division Director
T. Myers, Nuclear Services Director
J. Lingenfelter, Technical Engineer
R. Peters, Nuclear Licensing Manager

NRC Personnel

C. E. Norelius, Director, Division of Reactor Projects
W. D. Shafer, Chief, Projects Branch 2
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2. Control Room Emergency Ventilation System Inoperable

a. Background Information

Davis-Besse Technical Specification Limiting Condition for Operation 3.7.6.1 requires two independent control room emergency ventilation systems (CREVS) to be operable in Modes 1, 2, 3 and 4. Both independent systems are the same with redundant 100% capacity capable of performing the two safety functions associated with a CREVS. The two safety functions are: (1) Maintain the ambient air temperature below the maximum allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and (2) Maintain the control room habitable for operations personnel during and following all credible accident conditions.

A CREVS is composed of three subsystems. The first subsystem circulates air through the control room via a 3300 cfm centrifugal fan and associated ventilation ducting. The second subsystem cools the air passing through the first subsystem by a cooling coil located in the ductwork of the first subsystem. The cooling medium is freon R-12 which is supplied to the cooling coil via a compressor and associated piping. The third subsystem cools the freon in the second subsystem. This is accomplished by either an air-cooled condensing unit or a service water cooled heat exchanger, depending upon the outside temperature conditions.

b. Event

At 0930 on May 7, 1984, the licensee was preparing to perform the 15 minute flow test required every 31 days by Technical Specification surveillance requirement 4.7.6.1.b. The licensee's procedure for this test is ST 5076.01, Control Room Emergency Ventilation Monthly Test. The first prerequisite in ST 5076.01 is to verify that the "on-off" switch powering the freon compressor on the second subsystem is in the "on" position. The operator performing the prerequisite observed the switch to be in the "off" position.

The operator also observed that the control switch to the other freon compressor was also in the "off" position. The operator immediately notified the shift supervisor of the situation. The shift supervisor directed that the switches be repositioned to the "on" position and ST 5076.01 be performed on both independent CREVSs. The shift supervisor then logged that Technical Specification 3.0.3 was invoked for two inoperable CREVSs. ST 5076.01 was successfully completed on both ventilation systems, the systems declared operable and the unit removed from Technical Specification 3.0.3 requirements within an hour. Technical Specification 3.7.6.1, Control Room Emergency Ventilation System, requires that two independent control room emergency ventilation systems be operable in Modes 1, 2, 3 and 4. Technical Specification 6.8.1.a requires that procedures be established, implemented and maintained covering the applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972. Administrative procedures delineating responsibilities for plant operation and shutdown are listed in Appendix "A" of Regulatory Guide 1.33.

The licensee's Administrative Procedure AD 1839.00.9, Station Operations, requires that during removal from service of a system or component, the operability of redundant safety-related equipment shall be verified by inspection and an evaluation be made of Technical Specification Action Statements. Between April 23, 1984 and May 7, 1984, the air conditioning portions of both trains of the Control Room Emergency Ventilation System (CREV) were removed from service without complying with AD 1839.00.9. This resulted in both trains of the Control Room Emergency Ventilation System being inoperable. Failure to follow procedures and ensure operability of both CREVs is a violation (346/84-15-01A).

c. Licensee Followup of the Event

A deviation report was written on this event and an investigation into the event was initiated. The deviation report is the licensee's mechanism for reporting conditions adverse to quality under Criterion XVI of 10 CFR 50 Appendix B. The licensee issued LER 84-005 on June 6, 1984 documenting the event and the results of the investigation. The LER attributed the apparent cause of the occurrence to personnel not returning the switches to the "on" position following preventive maintenance on the systems. The switches were being positioned to the "off" position and then back to the "on" position to check proper freon compressor performance under the statement "Check pump down system" on the instructions for performing preventative maintenance attached to the maintenance work order.

d. NRC Followup

Upon notification the inspector began an inspection into the circumstances surrounding the event. Following a review of the system drawings the inspector requested that the licensee perform a safety analysis assuming both CREVSs were incapable of performing their air-conditioning function to determine the safety significance associated with this condition. This request was made on June 7, 1984, during an exit interview for IE Report 84-06.

The inspector continued the review of the event based on the information supplied in LER 84-005. The inspector interviewed the personnel involved with the ventilation system preventative maintenance program. Based upon those discussions and record review of when preventative maintenance and surveillance testing were accomplished it became apparent that the positioning of both control switches to the "off" position could not have been done under the approved preventative maintenance program. The last preventative maintenance performed on a CREVS was on April 4, 1984. ST 5076.01 had since been performed for CREVS #1 on April 9, 1984 and on April 23, 1984 for CREVS #2. The inspector found no indication that the switches were in the "off" position during those tests. Therefore, the switches were repositioned sometime between April 23, 1984, and May 7, 1984. The inspector reviewed maintenance work orders assigned to the CREVS startup system number for that time period and could not find any maintenance work that would account for the control switches being in the "off" position. The licensee was informed of the inaccuracy of attributing the event to the preventative maintenance personnel and is revising the LER. This is considered an open item (346/84-15-02) until the LER is revised.

During the review of the preventative maintenance program the inspector noted that during the quarterly preventative maintenance activities the CREVSs were not being declared inoperable even though the air-conditioning portion of the CREVSs were being disabled.

The licensee had established administrative controls in AD 1844.00, Maintenance, to keep this from occurring. These controls were accomplished by the completion of an attachment to the maintenance work order entitled "Tech Spec Equipment Operability Checkoff List" by the maintenance staff. This checkoff list required a written determination as to whether the maintenance activity authorized by the maintenance work order affected operability of any Technical Specification equipment. The maintenance work order and the checkoff list was then reviewed by the shift supervisor for concurrence of the maintenance staff's operability determination. The operability determination associated with the quarterly preventative maintenance on the CREVSs was being made by the maintenance staff and concurred with by the shift supervisor as not affecting system operability on the Tech Spec Equipment Operability Checkoff List.

The inspector also noted that the only formalized training requirements, as delineated in the AD 1828 series on training of the maintenance staff, was General Orientation Training. This training did not cover Technical Specifications and the Updated Safety Analysis Report operability requirements of safety systems.

e. Safety Significance Assessment

The analysis requested by the inspector (reference 2.d above) and discussed in the July 13, 1984, enforcement conference was presented to the inspector in a meeting with the Engineering Division Director on July 19, 1984. The analysis stated "...it is felt that if a situation were to occur where the emergency ventilation system was 'inoperable' due to the compressors not functioning, it would be recoverable in sufficient time so as not to affect the operability of control/monitoring equipment and/or the safety of the plant."

The inspector reviewed the licensee's analysis against the two safety functions assigned in the bases of Technical Specifications for CREVS. The inspector concluded that the safety function of the habilitability of the control room for all creditable accident functions was not affected by the loss of the freon compressors since the isolation of outside air to the control room was not affected. Based on the licensee's "after the fact" analysis, the inspector concluded that the safety function of maintaining the control room temperature below maximum instrumentation/equipment ratings, though degraded, would have been minimized through reasonable operator action.

3. Emergency Diesel Generator Ventilation Fan Taken Out of Service

While reviewing the licensee's safety tag log and the jumper/lifted wire log on November 1, 1983, the inspector observed that an emergency diesel generator (EDG) #1 ventilation supply fan had been taken out of service at 0600 and returned to service at 1055. The inspector determined that

this maintenance activity made the diesel generator inoperable based on a review of Section 9.4.2.1.2 and Table 9.4-4 of the Update Safety Analysis Report (USAR). The USAR states that the two supply fans associated with one EDG are each 50% capacity fans. The unit was in Mode 1 for all of November 1, 1983.

Technical Specification 3.8.1.1 requires two operable EDGs in Mode 1, 2, 3 and 4. If an EDG becomes inoperable the action statement requires the licensee to demonstrate the operability of the offsite power sources by performing a breaker alignment and power availability check, and demonstrate the operability of the unaffected EDG. These actions are required to be performed within one hour of the EDG being declared inoperable.

Since the maintenance staff and shift supervisor had determined that the EDG would be operable during the maintenance activity, the affected EDG was not declared inoperable. The failure to recognize that the maintenance activity made the EDG inoperable is considered an example of an item of noncompliance against Technical Specifications 3.8.1.1 (346/84-15-01b).

After the event was brought to the licensee's attention by the inspector an analysis of the EDG ventilation requirements was performed. The analysis concluded that only one of the two supply fans was required if the ambient outside temperature was less than 68°F. During the time the supply fan was out of service the highest ambient outside temperature was 59°F. A 10 CFR 50.59 review was not conducted to determine the acceptability of this analysis. This is considered an open item (346/84-15-03).

The licensee requested general ventilation requirements for equipment operability from their architect-engineer after this event occurred. The architect-engineer provided a list to the shift supervisors identifying general ventilation systems required for operability of safety related equipment. In addition, licensee management developed administrative controls requiring their concurrence prior to placing these systems in an abnormal configuration.

The inspector reviewed procedure, SP 1107.11, Emergency Diesel Generator Operating Procedure and noted that the procedure did not reflect the requirement for two ventilation supply fans to be operable. Also, a licensee review of the procedure was conducted on August 26 and October 17, 1983 without identifying this deficiency. The inspector ascertained that the cognizant individual responsible for the above review was not aware of the Updated Safety Analysis Report (USAR) requirement.

In the response to IE Report 83-01 the licensee committed to increase emphasis on design assumptions by providing procedure reviews with the related USAR sections for their use during annual procedure reviews. Selection of the USAR sections was to be by a computer program that correlates USAR sections to inputted keywords. The keyword computer index utilized by the licensee did not reference the ventilation USAR section when the "EDG" keyword was inputted.

4. <u>Auxiliary Feed Pump Inoperable</u>

a. <u>Background Information</u>

The Startup Feed Pump (SUFP) system is a system which provides secondary cooling during plant startup and shutdown. The system is composed of discharge piping, a pump, suction piping and manual valves. The discharge piping connects to the common main feedwater piping upstream of where the main feedwater piping splits to each of the steam generators. The pump is electric driven and its maximum heat removal capacity is 1-2% reactor power. The suction piping for the SUFP goes through the two auxiliary feed pump rooms to two water sources, the deaerator storage tank and the condensate storage tank.

On May 14, 1984, the licensee determined that one auxiliary feedwater pump (AFWP) was inoperable because Figure 10.4-12 of the Updated Safety Analysis Report (USAR) was not being complied with. This USAR requires the startup feedwater pump (SUFP) suction isolation valves from the deaerator tank (FW 32) and from the condensate storage tank (FS 91) to be closed to prevent flooding of the auxiliary feedwater pump rooms during a medium energy pipe break.

One suction valve (FW 32) to the startup feed pump mentioned in the USAR was being maintained open per the Startup Feed Pump Operating Procedure SP1106.27, and the Turbine Plant Cooling Water Operating Procedure SP 1104.39. Therefore, the auxiliary feedwater pumps were not being protected from flooding in the event of a medium energy pipe break.

To ensure operability of the AFWP and to ensure compliance to Figure 10.4-12 of the USAR, the licensee closed the valve in question, removed the fuses for the SUFP breaker and wrote temporary modifications (T-Mods) to all affected procedures to maintain the valve closed except when the startup feed pump was in service.

In addition, on June 14, 1984 the licensee determined that the original Safety Analysis Report did not encompass all the break spectrums associated with the startup feedpump piping. The licensee had not taken into account a high energy break of the discharge piping. The licensee shut the discharge valve and changed applicable procedures to reflect this condition.

b. <u>Event</u>

On June 24, 1984, the plant tripped during surveillance testing of the control rod drive trip breakers due to a personnel error. During the recovery from the reactor trip the licensee placed the startup feedwater pump (SUFP) in service. The SUFP was shut down following plant startup activities on June 25, 1984, however, the SUFP's suction valve (FW 32) was again left open. The valve was found open by an equipment operator on July 1, 1984. This rendered an

auxiliary feed pump inoperable from June 25 to July 1, 1984 in excess of the 78 hours allowed by the Technical Specifications for Mode 1. The plant was in Mode 1 during the June 25 to July 1, 1984 time period. This is considered an item of noncompliance for failure to meet a Limiting Condition for Operation (346/84-15-01C).

c. Followup of Event

The inspector interviewed the investigating personnel and the personnel involved in the startup and shutdown of the startup feed pump. Based on these inputs the inspector ascertained that the Startup Feedwater Operating Procedure was not properly used to start the pump and the procedure was not used to shut the pump down. Evidently, when it came time to shut down the SUFP the assistant shift supervisor provided the equipment operator with a list of valves he wished repositioned after shutting down the pump. One of the valves on the piece of paper was FW32, the SUFP suction valve. The equipment operator repositioned all the valves except FW32. Prior to the repositioning of this valve the equipment operator was called away to the switchyard. The shift supervisor was informed that the valve had not been repositioned. The need to close FW32 was lost during the next shift turnover. This is considered an example of an item of noncompliance (346/84-15-1C) for failure to conduct an adequate turnover.

During the next shift, the shift supervisor directed an operator to check some of the startup feed valves for proper position. The operator reported that all valves were properly positioned and erroneously identified normally closed valve FW 33 as FW 32.

The inspector performed a record review of applicable logs and procedures. The results of that review were:

(1) Sections of SP 1105.27 and 1106.27 (startup and shutdown of the SUFP) were not signed off for starting and shutting down the SUFP as required. Prerequisites, action steps and valve checklist steps were not signed.

(2) Trip Recovery Procedure PP 1102.03 step 4.2.2. as amended by T-mod 8048, was initialed annotating that the SUFP was started per SP 1106.27.

(3) Plant Startup Procedure PP 1102.2, step 8.1.4, as amended by T-Mod 8047, was initialed annotating that the SUFP had been stopped per SP 1106.27.

(4) Temporary Modifications (T-Mods) for Procedures PP 1102.03 and SP 1106.27 were still attached even though other more recent T-Mods had deleted these T-Mods. This is normal practice on T-Mods that have been authorized for use but had not been approved by the onsite safety review committee.

(5) Temporary modification 8057 as written was inadequate to start the SUFP and assure proper operation. One step in the procedure instructs the operator to start the SUFP even before the installation of the power fuses to the pump's breaker. Other steps in the procedure are not referenced as required to be performed after the pump is sta-ted.

(6) Completion of PP 1102.02 section 3, Zero to 25% Power Operations, was not signed off.

(7) The reactor operator's log for June 24, 1984, does not reflect when the SUFP was put in service.

(8) The procedures for restoring the startup feed pump's suction and discharge valves, the startup feed pump's lube oil cooling and the startup feed pump's pump seal cooling did not require independent verification.

Items (1), (2), (3), (6), (7) and (8) are examples of an item of noncompliance for failure to properly implement procedures (346/84-15-01C). Item (5) is considered an example of an item of noncompliance for failure to maintain an adequate procedure (346/84-15-01C).

The inspector performed a historical review of revisions and reviews of SP 1106.27, Startup Feed Pump Operating Procedure. The results of that review were:

(1) Since the beginning of plant operation, SP 1106.27 required that valve FW 32 be open.

(2) SP 1106.27 had an annual review on July 7, 1983 by a co-op student and on November 23, 1983, by a shift supervisor. the Technical Section provided an USAR review package for SP 1106.27 which did not include any of USAR section 3.6. requirements for the position of the suction valve.

(3) USAR section 3.6, Protection Against Dynamic and Environmental Effects Associated with Postulated Rupture of Piping, was not keyworded in the licensee's computer data bank.

(4) On December 19, 1982, the station operations department initiated Facility Change Request (FCR) 82-176 requesting valve FW 32 be shown open instead of closed on design document, P&ID M-006B. The FCR was implemented on May 24, 1983. The licensee's engineering staff determined that this FCR did not constitute a change to the facility as described in the Safety Analysis Report even though Figure 10.4-12 of that report showed valve FW 32 closed. As a result, the licensee did not perform an adequate 10 CFR 50.59(b) safety evaluation. Changing

the position of valve FW 32 constituted an unreviewed safety question requiring prior NRC approval before implementation. Had the licensee realized the safety significance, it is reasonable to conclude that the licensee would have directed closure of FW 32 at that time.

Item (4) is considered an example of an item of noncompliance for failure to perform an adequate 10 CFR 50.59 review (346/84-15-1D).

5. Enforcement Conference

On July 13, 1984, an Enforcement Conference was held at the NRC regional office to discuss the circumstances surrounding the mispositioning of the freon compressor control switches. Licensee representatives in attendance are denoted in paragraph 1. The meeting started with opening remarks from the NRC and a presentation of past events leading to and continuing to be a concern of the NRC in the area of the licensee's inability to recognize design basis requirements for operability of safety-related equipment. The licensee made a presentation on their short term corrective action of requiring a senior reactor operator to review maintenance work orders for operability requirements before submission to the shift supervisor. Potential long term corrective action was also presented dealing with key senior experienced licensee personnel reviewing the design basis of all safety-related equipment and identifying all components necessary for operability.

A general discussion then took place as to whether previous corrective actions in this area should have prevented the event. The discussion then centered on the safety significance associated with the freon control switches being in the "off" position. The licensee indicated that 20 to 30 minutes would be available to the operator to take corrective action and that this time frame was adequate to determine the mispositioning of the switches and reposition them.

A discussion then ensued as to the CREVS function and its impact on station operations. The licensee stated that the reason for the switches being placed in the "off" position was still under investigation. The meeting concluded with the NRC stating that further internal discussion would have to be pursued to: (1) determine if the safety significance of the event would constitute escalated enforcement for violation of a Limiting Condition for Operation based on review of the licensee's analysis and (2) determine if the event occurred due to lack of adequate management controls in an area where inadequacies had been previously identified and corrective actions implemented.

Exhibit 5

3.6.2.7.2.12 Startup Feed Pump System (reference Figure 10.4-12)

The startup feed pump is used for steam generator warmup and may be used for normal station shutdown. It is located in the same room as one of the auxiliary feed pumps in the auxiliary building. A rupture of the suction pipe from the condensate storage tank would cause flooding. In addition, a rupture of the pump seal and lube oil cooler piping from the turbine building cooling water system could cause flooding.

The following changes were made to this system:

a. The suction piping from the Auxiliary Building wall to the normally closed pump suction isolation valve was upgraded to Seismic Class I.

b. Normally closed isolation valves are installed in the cooling water supply and return piping to the coolers. These valves are located in the Turbine Building.

3.6.2.7.2.13 Circulating Water System (reference Figures 10.4-4 and 10.4-5)

During normal station operation, four circulating water pumps, each with a capacity of 120,000gpm, are in operation. Assuming a complete rupture of the main condenser circulating water expansion joint (108 inch I.D. pipe) on the inlet side of the condenser, two circulating pumps on one train would run out. In addition, water from the cooling tower and piping flows back through the rupture. The condenser pit (El. 567) floods, inundating the main feed pump turbine hydraulic and lube oil pump motors at El. 573. This results in a trip of the main feed pump turbines. At about El. 575 the condensate pump motors is flooded, causing the trip of these pumps. There is no other major equipment of consequence below El. 585. No essential components are located in the Turbine Building. There are no paths for water to escape below El. 585. See Figures 3.6-20, 3.6-21, and 3.6-22.

Above El. 585, water would flow into the circulating pump house and leakage paths through the railroad door and other external doors. Water also enters the following essential areas:

a. Service Water tunnel access stairwell.

b. Auxiliary feed pump room access stairwell in the auxiliary building.

c. Component Cooling Water equipment room through doorways in the Auxiliary Building.

d. High voltage switchgear rooms through doorways into the auxiliary building.

See Figures 3.6-3 and 3.6-23.

Assuming a complete rupture of the expansion joints, the pumps on the affected line run out to about 149,000gpm each where cavitation occurs due to

Exhibit 6

TOLEDO EDISON

May 7, 1985

Log No. K85-726
File: RR 2 (NP-33-84-09)

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

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Gentlemen:

Enclosed is Revision 2 to Licensee Event Report 84-009. The revisions to the report are indicated by a "2" in the left margin of each page.

Please replace your previous copy of this report with the attached revision.

Yours truly,

Stephen M. Quennoz /wgo

Stephen M. Quennoz
Plant Manager
Davis-Besse Nuclear Power Station

SMQ/ljk

Enclosure

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Regional Administrator,
USNRC Region III

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DB-1 NRC Resident Inspector

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JCS/001

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NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Davis-Besse Unit 1	0 5 0 0 0 3 4 6	1 OF 0 3

TITLE (4)

Potential Piping Breaks, Startup Feedwater Pump Piping

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 6	1 6	8 4	8 4	0 0 9	0 2	0 5	0 7	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 4

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	XX 50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER	
	AREA CODE	
F. R. Miller/P. H. Straube	4 1 9	2 4 9 - 5 3 7 2

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
B	B A	T R B	T 1 4 7	N					

SUPPLEMENTAL REPORT EXPECTED (14)

		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

During the conceptual design stage of Facility Change Request 83-159 and work on Non-Conformance Report 84030, it was discovered that an unanalyzed situation existed in Auxiliary Feedwater Pump Rooms 237 and 238 due to potential pipe break effects from non-seismic piping located in these rooms. This non-seismic piping is associated with the Startup Feedwater Pump. Auxiliary Feedwater Pump Room 238 (Auxiliary Feedwater Pump 1-2) could have been affected by a high energy pipe break in the startup feedwater pump discharge piping or by a moderate energy break in the Startup Feedwater Pump suction piping. Auxiliary Feedwater Pump Room 237 (Auxiliary Feedwater Pump 1-1) could have been affected by a moderate energy pipe break in the startup feedwater pump suction piping.

Both rooms would also be affected by flooding in the event that the non-seismic turbine plant cooling water piping serving the startup feedwater piping coolers would rupture.

Description of Occurrence: During the review of the Auxiliar Feedwater Pump Room Environmental Requirements, it was discovered that the Station was operating in a condition outside of that addressed in the Updated Safety Analysis Report (USAR).

The discharge piping from the SUFP is a high energy line during operation of the SUFP according to the criteria in USAR Section 3.6. This criterion stipulates that a line outside Containment shall be classified as high energy if operating conditions subject the piping to more than 275 psig and 200°F. A high energy system which is in service more than six hours is analyzed for pipe rupture. The original design criteria for this piping assumed that operational time would be less than six hours at a time and that suction would be taken from the Condensate Storage Tank. The operation of the SUFP has exceeded the six hour limit, and suction has been taken from the Deaerator Storage Tank. This subjects the SUFP discharge line to all the requirements of high energy piping.

The suction piping to the SUFP is a moderate energy line according to the criterion in USAR Section 3.6. This criteria stipulates that a line outside Containment operating above 275 psig or 200°F is a moderate energy line. Postulation of critical piping cracks at the most adverse location must be considered for this piping.

The non-seismic turbine plant cooling water lines serving the SUFP are neither high nor moderate energy lines, but require postulation of rupture during a seismic event. These lines are routed through both rooms 237 and 238 and would subject either room to flooding in the event of a pipe rupture.

Designation of Apparent Cause of Occurrence: Three conditions caused the potential concerns explained previously. The USAR did not recognize the fact that the SUFP suction could be normally lined up to the Deaerator Storage Tank rather than the Condensate Storage Tank. Modifications to the SUFP suction piping did not recognize that moderate energy fluid was being introduced into Rooms 237 and 238. Systems were used in a manner different from FSAR/USAR commitments.

Analysis of Occurrence: The potential for pipe whip and jet impingement in Room 238 and flooding and high temperature in Room 237 or 238 caused an unanalyzed condition in either room. If a pipe break would have occurred, this condition could have affected the operation of AFP 1-1 or 1-2.

Corrective Action: Corrective action was taken when these conditions were discovered by isolating AFP Rooms 237 and 238 from flooding, fluid jet impingement, and high temperature concerns. To accomplish this, FW106 (SUFP discharge line, FW32 (suction line from deaerator), CW 196 and 197 (turbine plant cooling water lines), all located outside AFP rooms 237 and 238, and FW 91 (suction line from the CST) located in the AFP room 238, were closed. The affected procedures were revised to reflect these changes, and the valves may now only be opened within the limits of the procedures.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Davis-Besse Unit 1	0 5 0 0 0 3 4 6	8 4	— 0 0 9 —	0 2	0 3 OF 0 3

TEXT (If more space is required, use additional NRC Form 366A's) (17)

After these actions were taken, AFP Rooms 237 and 238 are no longer subjected to the potential concerns explained previously when the SUFP is not running. The concerns still exist during the limited amount of time that the SUFP is in operation.

2 A request for permission to operate the SUFP during unit startup and shutdown was approved by the NRC in Amendment 83 as License Condition 2.c(3)(t).

If the SUFP is operated, interim corrective action will be taken to place an operator at the SUFP. Upon indication of a pipe leak or break, the operator would either stop the SUFP locally or contact the Control Room to stop the SUFP. He would then close the isolation valves. This would minimize any flooding and high temperature effects. This action is being taken since it is assumed that the piping will develop leaks through pipe cracks before a complete piping rupture would occur.

In addition, the SUFP suction and discharge piping was hydrotested to original
2 hydrotest pressures. This verified that the piping is in sound condition from a pressure retention standpoint.

Long term corrective actions will be taken to implement permanent hardware modifications which will resolve all high energy, moderate energy, and flooding concerns within AFP Rooms 237 and 238. Long term corrective action under FCR 85-025 will be
2 to install a new SUFP outside of the AFP rooms before commencing Cycle 6 operation.

Failure Data: There have been no previous similar reported occurrences.

Report No: NP-33-84-09 DVR No(s): 84-080