AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346
UNIT Davis-Besse Unit 1
DATE August 9, 1985
COMPLETED BY Morteza Khazrai
TELEPHONE (419) 249-5000 Ext. 290

MONTH July 1985

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	0	17	0
2	0	18	0
3	0	19	0
4	0	20	0
5	0	21	0
6	0	22	0
7	0	23	0
8	0	24	0
9	0	25	0
10	0	26	0
11	0	27	0
12	0	28	0
13	0	29	0
14	0	30	0
15	0	31	0
16	0		

INSTRUCTIONS

On this format, list the average daily unit power level in MWe-Net for each day in the reporting month. Compute to the nearest whole megawatt.

(9/77)

IE24
1/1

OPERATING DATA REPORT

DOCKET NO. 50-346
DATE August 9, 1985
COMPLETED BY Morteza Khazrai
TELEPHONE (419) 249-5000
Ext. 290

OPERATING STATUS

1. Unit Name: Davis-Besse Unit 1
2. Reporting Period: July 1985
3. Licensed Thermal Power (MWt): 2772
4. Nameplate Rating (Gross MWe): 915
5. Design Electrical Rating (Net MWe): 906
6. Maximum Dependable Capacity (Gross MWe): 904
7. Maximum Dependable Capacity (Net MWe): 860
8. If Changes Occur in Capacity Ratings (Items Number 3 Through 7) Since Last Report, Give Reasons:
   To obtain an acceptable offset from Reactor Protection System setpoint limit

Notes

9. Power Level To Which Restricted, If Any (Net MWe):
10. Reasons For Restrictions, If Any:

	This Month	Yr.-to-Date	Cumulative
11. Hours In Reporting Period	744.0	5,087.0	61,392.0
12. Number Of Hours Reactor Was Critical	0.0	2,845.6	35,877.1
13. Reactor Reserve Shutdown Hours	0.0	44.7	4,058.8
14. Hours Generator On-Line	0.0	2,730.5	34,371.8
15. Unit Reserve Shutdown Hours	0.0	0.0	1,732.5
16. Gross Thermal Energy Generated (MWH)	0.0	6,312,178	81,297,600
17. Gross Electrical Energy Generated (MWH)	0.0	2,087,278	26,933,622
18. Net Electrical Energy Generated (MWH)	0.0	1,942,921	25,233,177
19. Unit Service Factor	0.0	53.7	56.0
20. Unit Availability Factor	0.0	53.7	58.8
21. Unit Capacity Factor (Using MDC Net)	0.0	44.4	47.8
22. Unit Capacity Factor (Using DER Net)	0.0	42.2	45.4
23. Unit Forced Outage Rate	100.0	33.2	18.9

24. Shutdowns Scheduled Over Next 6 Months (Type, Date, and Duration of Each):

25. If Shut Down At End Of Report Period, Estimated Date of Startup: Unknown
26. Units In Test Status (Prior to Commercial Operation):

	Forecast	Achieved
INITIAL CRITICALITY		
INITIAL ELECTRICITY		
COMMERCIAL OPERATION		

DOCKET NO.	50-346
UNIT NAME	Davis-Besse Unit 1
DATE	August 9, 1985
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TELEPHONE	(419) 249-5000, Ext. 290

UNIT SHUTDOWNS AND POWER REDUCTIONS

REPORT MONTH July 1985

No.	Date	Type[1]	Duration (Hours)	Reason[2]	Method of Shutting Down Reactor[3]	Licensee Event Report #	System Code[4]	Component Code[5]	Cause & Corrective Action to Prevent Recurrence
7	85 06 09	F	744	A	4	LER 85-013	JK	SC	The unit remained shutdown following the reactor trip on June 9, 1985. See Operational Summary for further details.

[1] F: Forced
S: Scheduled

[2] Reason:
A-Equipment Failure (Explain)
B-Maintenance or Test
C-Refueling
D-Regulatory Restriction
E-Operator Training & License Examination
F-Administrative
G-Operational Error (Explain)
H-Other (Explain)

[3] Method:
1-Manual
2-Manual Scram
3-Automatic Scram
4-Continuation from Previous Month
5-Load Reduction
9-Other (Explain)

[4] Exhibit G - Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File (NUREG-0161)

[5] Exhibit I - Same Source

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OPERATIONAL SUMMARY
JULY 1985

The unit remained shutdown the entire month of July following the reactor trip on June 9, 1985.

Investigations of the causes of the event and corrective actions continue. See NUREG 1154 for further details.

REFUELING INFORMATION   DATE: July 1985

1. Name of facility: Davis-Besse Unit 1

2. Scheduled date for next refueling shutdown: Spring, 1986

3. Scheduled date for restart following refueling: Summer, 1986

4. Will refueling or resumption of operation thereafter require a technical specification change or other license amendment? If answer is yes, what in general will these be? If answer is no, has the reload fuel design and core configuration been reviewed by your Plant Safety Review Committee to determine whether any unreviewed safety questions are associated with the core reload (Ref. 10 CFR Section 50.59)?

   Ans: Expect the Reload Report to require standard reload fuel design Technical Specification changes (3/4.1 Reactivity Control Systems and 3/4.2 Power Distribution Limits).

5. Scheduled date(s) for submitting proposed licensing action and supporting information: Winter, 1985

6. Important licensing considerations associated with refueling, e.g., new or different fuel design or supplier, unreviewed design or performance analysis methods, significant changes in fuel design, new operating procedures.

   Ans: None identified to date.

7. The number of fuel assemblies (a) in the core and (b) in the spent fuel storage pool.

   (a) 177   (b) 204 - Spent Fuel Assemblies

8. The present licensed spent fuel pool storage capacity and the size of any increase in licensed storage capacity that has been requested or is planned, in number of fuel assemblies.

   Present: 735   Increase size by: 0 (zero)

9. The projected date of the last refueling that can be discharged to the spent fuel pool assuming the present licensed capacity.

   Date: 1992 - assuming ability to unload the entire core into the spent fuel pool is maintained.

BMS/005

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 78-477

SYSTEM: Borated Water Storage Tank

COMPONENT: LT-1525A and LT-1525C

CHANGE, TEST OR EXPERIMENT: FCR 78-477 installed a permanent shelter to protect Borated Water Storage Tank (BWST) level transmitters LT-1525A and LT-1525C and their associated source lines. This shelter was designed per Seismic Category I criteria. Work was completed February 15, 1985.

REASON FOR CHANGE: This structure was installed to prevent the BWST level transmitters, LT-1525A and LT-1525C, from freezing during the winter months. The original heat tracing on the associated lines to the transmitters and in the transmitter housings was inadequate for freeze protection, requiring temporary sheltering to prevent freezing.

SAFETY EVALUATION SUMMARY: The safety function of the BWST is to provide a water source for the Emergency Core Cooling System and the Containment Spray System during a Loss of Coolant Accident. Level transmitters LT-1525A and LT-1525C are used to measure the water level in the BWST. The above change was made to allow the intended safety function of the BWST to be satisfied. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-178

SYSTEM: Reactor Coolant System

COMPONENT: Reactor Coolant Pumps

CHANGE, TEST OR EXPERIMENT: This FCR performed a 10CFR50.59 review and changed the interlock setting for starting the fourth Reactor Coolant Pump (RCP). Originally, the interlock setting required that reactor power be less than 22% before restarting the fourth RCP. This FCR review resulted in the interlock being changed to a value that would allow the fourth RCP to be started provided that reactor power was less than 60% full power. Work was completed August 13, 1979.

REASON FOR CHANGE: This change was the result of a concern that starting the fourth RCP would cause a pressure spike in the Reactor Coolant System (RCS) that would result in a high RCS pressure trip of the Reactor Protection System and the reactor. The analysis done by Babcock and Wilcox for the effects of starting the pump at different power levels for both the beginning of life and end of life core conditions led them to recommend the interlock settings be changed to 60% full power.

SAFETY EVALUATION SUMMARY: This FCR does not constitute an unreviewed safety question because:

1) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, previously evaluated in the FSAR, has not been increased.

2) The possibility of an accident or malfunction of a different type other than any evaluated previously in the FSAR has not been created.

3) The margin of safety as defined in the basis for any Technical Specification has not been reduced.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-221

SYSTEM: Steam and Feedwater Rupture Control System

COMPONENT: Startup Feedwater Pump

CHANGE, TEST OR EXPERIMENT: This FCR made two changes to the Steam and Feedwater Rupture Control System (SFRCS). First, a logic change was made to the SFRCS to allow for the feeding of the steam generators with the Startup Feedwater Pump (SUFP). Second, this FCR provided a means to manually open the following valves if actuation of SFRCS has closed these valves.

Valve No.	Description
SP7A	Main Feedwater 2 Startup Control Valve
SP7B	Main Feedwater 1 Startup Control Valve
FW 601	Main Feedwater 2 Stop Valve
FW 612	Main Feedwater 1 Stop Valve

Originally, the actuation of SFRCS closed the valves listed above. This FCR allows the valves to be opened if SFRCS signals are blocked.

Logic modifications emphasized the following features:

1) The valves listed above will close on an SFRCS trip.

2) These valves cannot be opened following an SFRCS trip without first blocking the SFRCS signal to them.

3) Momentary contact block switches were provided to override the SFRCS signal to these valves.

4) Blocking will be functional only when an SFRCS trip is present. The block signal is automatically removed when the SFRCS trip resets so that additional SFRCS trips will reclose the valves.

To control steam generator pressure after an SFRCS actuation, SFRCS signals to the atmospheric vent valves, ICS11A and ICS11B, were blocked. This modification emphasized the previous logic features in addition to the following:

1) The control of atmospheric vent valves can be handled by the Integrated Control System.

2) Instrument air will be required for the operation of these valves.

Work was completed August 9, 1982.

REASON FOR CHANGE: This FCR was implemented to provide a way to feed the steam generators with the SUFP to remove core decay heat in the event that both the Main Feed Pumps and the Auxiliary Feed Pumps are lost. This change will also provide a capability for steam generator pressure control after an SFRCS actuation.

SAFETY EVALUATION SUMMARY: The purpose of the SFRCS is to protect the nuclear steam supply system by automatically initiating the Auxiliary Feedwater System in the event of one or more of the following incidents:

- a main feedwater line rupture
- a main steam line rupture
- a loss of both Main Feed Pumps
- a loss of four Reactor Coolant Pumps

The changes made by this FCR will enhance the operation of the SFRCS. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-311

SYSTEM: Safety Features Actuation System

COMPONENT: Various solenoid valves

CHANGE, TEST OR EXPERIMENT: FCR 79-311 allowed for the replacement of various solenoid valves associated with the Safety Features Actuation System (SFAS). The following valves were replaced under this FCR:

Valve No.	Function
SV-229B	Pressurizer Quench Tank Outlet Isolation Valve
SV-235B	Pressurizer Quench Tank Sample Isolation Valve
SV-1719A	Containment Ventilation Header Isolation Valve
SV-1773A	Reactor Coolant Drain Tank Header Isolation Valve
SV-5006	Containment Purge Inlet Isolation Valve
SV-5007	Containment Purge Outlet Isolation Valve
SV-6831A	Demineralized Water to Containment Isolation Valve

Work associated with the replacement of these valves was completed November 30, 1981.

REASON FOR CHANGE: Upon review of all environmentally qualified solenoid valves provided by the Automatic Switch Company (ASCO), the above valves were found to have no qualification data. This data was not available when the original valves were purchased. Therefore, to have the proper documentation of these SFAS operated solenoid valves, the valves were replaced with duplicate ASCO valves with the proper documentation provided to Toledo Edison.

SAFETY EVALUATION SUMMARY: The purpose of the solenoid valves listed above is to isolate Containment upon an SFAS actuation. The safety function of the SFAS is to automatically prevent or limit fission products and energy release from the core, to isolate the containment vessel, and to initiate operation of the Engineered Safety Features equipment in the event of a Loss of Coolant Accident (LOCA). By replacing the solenoid valves, the safety function of SFAS was not adversely affected. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-074

SYSTEM: Auxiliary Feedwater

COMPONENT: PSL-4930A&B and PSL-4931A&B

CHANGE, TEST OR EXPERIMENT: FCR 80-074 was implemented to increase the time delay to close the steam inlet valves to the Auxiliary Feed Pump Turbines (AFPT) when there is low pressure in the suction lines of the Auxiliary Feed Pumps (AFPs). Valves HV-106 and HV-106A (for AFPT 1) and HV-107 and HV-107A (for AFPT 2) had a time delay of 0.5 seconds. The time delay has been increased to 2.5 seconds. This FCR also reconnected pressure switches PSL-4930A, 4930B, 4931A, and 4931B. Each pressure switch was connected to each of the four steam inlet valves to the AFPTs. The switches were connected as listed:

Switch No.	Valve
PSL-4930A	MS-106
PSL-4930B	MS-106A
PSL-4931A	MS-107
PSL-4931B	MS-107A

These pressure switches will cause the steam inlet valves to close at 1 psig. Work was completed December 18, 1980.

REASON FOR CHANGE: Prior to this FCR the steam inlet valves to the AFPTs have closed incorrectly due to low pressure dips in the AFP suction piping. The increasing of the time delay will prevent the spurious closure of the valves. The reconnection of the pressure switches to the steam inlet valves was performed to ensure a power supply failure would not prevent the valves to both the AFPTs from closing. Under the new connection, the power supply to HV-106 and HV-107 could fail with HV-106A and HV-107A being maintained.

SAFETY EVALUATION SUMMARY: The safety function of the Auxiliary Feedwater System is to provide feedwater to the steam generators for the removal of decay heat in the absence of main feedwater and to promote natural circulation in the Reactor Coolant System on a loss of all four Reactor Coolant Pumps. The changes made by this FCR do not reduce the safety function of this system. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-093

SYSTEM: Service Water

COMPONENT: Various pipe supports and anchors

CHANGE, TEST OR EXPERIMENT: FCR 80-093 was implemented to modify various pipe anchors and supports needed to maintain the integrity of the Service Water System. Following is a list of the modified components. Work involved with this FCR was completed August 26, 1983.

Anchors

A-135	A-136	A-137	A-138	A-139	A-140
A-141	A-144	A-161	A-217	A-262	A-370

Snubbers

41-HBC-34-H38	41-HBC-36-H11	41-HBC-43-H6	41-HBC-46-H44
41-HBC-35-H36	41-HBC-36-H12	41-HBC-45-H1	41-HBC-46-H45
41-HBC-35-H48	41-HBC-37-H20	41-HBC-45-H40	41-HBC-47-H26
41-HBC-36-H4	41-HBC-37-H26	41-HBC-45-H43	41-HBC-47-H29
41-HBC-36-H5	41-HBC-37-H37	41-HBC-45-H44	41-HBC-47-H32
41-HBC-36-H6	41-HBC-42-H2	41-HBC-46-H38	41-HBC-47-H35
41-HBC-36-H7	41-HBC-42-H9	41-HBC-46-H41	41-HBC-47-H38
41-HBC-36-H9	41-HBC-42-H12	41-HBC-46-H42	41-HBC-47-H47
41-HBC-36-H10	41-HBC-43-H5	41-HBC-46-H43	41-HBC-47-H49
41-HCB-47-H52	41-HCB-96-H4	41-HCB-194-H3	41-HBD-96-H5
41-HBD-98-H3	41-HBD-191-H1	41-HBD-194-H4	

REASON FOR CHANGE: The pipe supports and anchors modified were required as a result of re-analysis of the anchor/support in accordance with IE Bulletins 79-02 and/or 79-14.

SAFETY EVALUATION SUMMARY: The safety function of Service Water is to supply cooling water to the component cooling heat exchangers, the containment air coolers, and the turbine plant cooling water heat exchangers during normal operation. By upgrading the listed Service Water System pipe anchors and supports, the safety function of the system is not decreased but enhanced. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-119

SYSTEM: Main Steam and Core Flood

COMPONENT: MS-106, MS-106A, MS-107, MS-107A, CF01A, and CF01B

CHANGE, TEST OR EXPERIMENT: FCR 82-119 reset the torque switch settings for MS-106, MS-106A, MS-107, MS-107A, CF1A, and CF1B. The new torque switch settings are 2.5 for the open dial setting and 1.5 for the close dial setting. Work was completed November 28, 1984.

REASON FOR CHANGE: This FCR is to implement the torque switch settings specified in the Torrey Pines Technology Report to improve the operation of the valves listed above.

SAFETY EVALUATION SUMMARY: The safety function of MS-106, MS-106A, MS-107, and MS-107A is to provide steam to or isolate steam to the Auxiliary Feedwater Pump Turbine on a Steam and Feedwater Rupture Control System trip. The core flood valves, CF1A and CF1B, are isolation valves that isolate the Core Flood Tanks from that system's piping. The new settings will enhance the operation of these valves. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-108

SYSTEM: Safety Features Actuation System
Steam and Feedwater Rupture Control System

COMPONENT: Various logic drawings

CHANGE, TEST OR EXPERIMENT: This FCR updated logic drawings E-16, E-17, E-18, E-19, M-050, and M-051. These drawings are associated with the Safety Features Actuation System (SFAS) and the Steam and Feedwater Rupture Control System (SFRCS). Work was completed December 4, 1984.

REASON FOR CHANGE: These drawings are being updated to represent the actual, correct plant conditions. Deficiencies were found to exist in these drawings following a review performed on Facility Change Requests in an effort to have all drawings accurately updated.

SAFETY EVALUATION SUMMARY: The safety function of the SFAS is to automatically prevent or limit fission products and energy release from the core, to isolate the containment vessel, and to initiate operation of the Engineered Safety Features equipment in the event of a Loss of Coolant Accident.

The safety function of the SFRCS is to automatically initiate the Auxiliary Feedwater System in the event of a main feedwater line rupture, a main steam line rupture, the loss of both main feedwater pumps, or the loss of all four reactor coolant pumps.

This FCR will not decrease the safety function of either safety system. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-134

SYSTEM: Reactor Coolant System

COMPONENT: Cold leg nozzle plugs

CHANGE, TEST OR EXPERIMENT: This FCR was implemented to perform a 10CFR50.59 review for the use of cold leg nozzle plugs during the 1984 Refueling Outage while performing maintenance activities within the Reactor Coolant System (RCS). Work involved with this FCR was completed January 11, 1985.

REASON FOR CHANGE: The use of the cold leg plugs and their possible failure could drain the refueling canal and the upper portion of the reactor vessel with the potential loss of decay heat removal. Therefore, a 10CFR50.59 review was necessary prior to using the cold leg nozzle plugs.

SAFETY EVALUATION SUMMARY: The safety function of maintaining proper water levels in the refueling canal is to provide adequate shielding from and cooling for the fuel assemblies within the reactor vessel and while those assemblies are being moved during Mode 6. No movement of fuel was permitted during these maintenance activities, and the pressure boundry for the section of the cole leg below the high pressure injection nozzle was kept intact. Therefore, based on these observations, the use of the RCS cold leg nozzle plugs did not present an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-174

SYSTEM: Fire Barriers

COMPONENT: One-hour rated fire barrier over conduit 36591A

CHANGE, TEST OR EXPERIMENT: FCR 84-174 installed a temporary one-hour rated fire barrier over conduit 36591A in Room 322. This barrier is to remain in place until a permanent modification which complies with 10CFR50, Appendix R is implemented. This conduit contains cables required for Reactor Coolant System pressure and temperature and steam generator pressure and level instrumentation. Work involved with FCR 84-174 was completed December 23, 1984.

REASON FOR CHANGE: This change is the result of the Toledo Edison Appendix R non-compliance "Seriousness Assessment" meeting held September 24, 1984. In this meeting, it was determined that conduit 36591A should be protected with a temporary one-hour fire barrier until a permanent modification is implemented.

SAFETY EVALUATION SUMMARY: The safety function of the cables contained in the conduit is to provide an indication of several plant parameters at the auxiliary shutdown panel. Therefore, the addition of a temporary fire barrier will maintain system integrity until a permanent one-hour fire barrier can be installed. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-194

SYSTEM: Reactor Vessel Internal and Lifting Facilities

COMPONENT: Fuel Assembly, S/N NJØ1CR

CHANGE, TEST OR EXPERIMENT: FCR 84-194 replaced the fuel assembly holddown spring on fuel assembly S/N NJØ1CR. The original holddown spring was of the B6 type. This was replaced with the B7 type holddown assembly. Work was completed February 13, 1985.

REASON FOR CHANGE: The holddown spring for assembly S/N NJØ1CR was found broken during an inspection of the fuel assembly during the 1984 Refueling Outage.

SAFETY EVALUATION SUMMARY: The safety function of the holddown spring is to hold down the fuel assembly and to allow for differential expansion between the fuel assembly and the reactor vessel internals. Originally, the holddown spring was type B6 which was constructed of Inconel X-750. The new holddown spring is type B7 which is fabricated of Inconel 718. Because of the material it is composed of, use of the type B7 holddown spring will provide approximately a 20% higher yield stress. Replacement of the type B6 fuel assembly holddown spring with a type B7 spring will reduce the potential for breakage of the spring due to its higher fatigue capacity. Therefore, since the safety function of the holddown spring is not reduced, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-218

SYSTEM: Startup Feed Pump and Auxiliaries System (045-02)

COMPONENT: FW-32

CHANGE, TEST OR EXPERIMENT: FCR 84-218 modified drawing 7749-M-006B to show valve FW-32, the suction from the Deaerator Storage Tank, normally closed. This drawing change was completed February 8, 1985.

REASON FOR CHANGE: This change was made to the drawing so valve FW-32 could be shown in its normal, correct position per the Updated Safety Analysis Report (USAR).

SAFETY EVALUATION SUMMARY: The safety function of FW-32 on drawing 7749-M-006B is to isolate the startup feed pump suction line from the Deaerator Storage Tank before it enters the Auxiliary Feedwater Pump (AFP) Rooms 237 and 238. This eliminates the moderate energy concerns and their potential effects on the AFP and associated components of the Auxiliary Feedwater System. The moderate energy concern is introduced when FW-32 is opened for a limited amount of time during plant startup. This valve is closed any other time. Therefore, since the startup feed pump system is not in operation for most of the plant operating time, the probability of introducing moderate energy effects into Rooms 237 and 238 are greatly reduced.

The drawing modification to show FW-32 normally closed on drawing 7749-M-006B does not present an unreviewed safety question since FW-32 in a normally closed position drastically decreases the time period that the AFP rooms are subjected to moderate energy concerns and, therefore, greatly reduces the probability of introducing moderate energy effects into AFP Rooms 237 and 238.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 85-004

SYSTEM: Reactor Coolant System

COMPONENT: Pressurizer

CHANGE, TEST OR EXPERIMENT: This FCR revised drawing 7749-M-190-HBW-135-6 to reflect the correct, actual condition of the plant. The drawing revision involved correcting the drawing to show the placement of shims behind a baseplate on support PS-H36, which is associated with the pressurizer. Work was completed April 1, 1985.

REASON FOR CHANGE: This change was the result of Non-conformance Report 84-178. The non-conformance report was initiated after the performance of Surveillance Test ST 5044.01, Inspection of Safety Related Hydraulic Snubbers. During this test, the baseplate for PS-H36 was 1/4 inch away from the wall. To correct this, shims were added behind a baseplate on support PS-H36. The drawing was updated to reflect the correct, actual condition of the plant.

SAFETY EVALUATION SUMMARY: The safety function of the pressurizer is to maintain pressure in the Reactor Coolant System during all phases of plant operation. This FCR did not create any adverse environment to the pressurizer, but instead it represents the actual condition of the plant. Therefore, an unreviewed safety question does not exist.

TOLEDO EDISON

August 9, 1985

Log No. K85-1193
File: RR 2 (P-6-85-07)

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

Mr. Norman Haller, Director
Office of Management and Program Analysis
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Haller:

Monthly Operating Report, July 1985
Davis-Besse Nuclear Power Station Unit 1

Enclosed are ten copies of the Monthly Operating Report for Davis-Besse Nuclear Power Station Unit 1 for the month of July 1985.

If you have any questions, please feel free to contact Morteza Khazrai at (419) 249-5000, Extension 290.

Yours truly,

Louis F. Storz

Louis F. Storz
Plant Manager
Davis-Besse Nuclear Power Station

LFS/MK/ljk

Enclosures

cc: Mr. James G. Keppler, w/1
Regional Administrator, Region III

Mr. James M. Taylor, Director, w/2
Office of Inspection and Enforcement

Mr. Walt Rogers, w/1
NRC Resident Inspector

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