AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346
UNIT Davis-Besse #1
DATE November 9, 1983
COMPLETED BY Bilal Sarsour
TELEPHONE (419) 259-5000 ext. 384

MONTH October, 1983

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	115	17	368
2	0	18	367
3	0	19	373
4	184	20	393
5	272	21	391
6	272	22	390
7	267	23	505
8	42	24	624
9	249	25	646
10	367	26	631
11	364	27	650
12	366	28	648
13	504	29	109
14	626	30	652
15	40	31	785
16	216		

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

OPERATING DATA REPORT

DOCKET NO. 50-346
DATE November 9, 1983
COMPLETED BY Bilal Sarsour
TELEPHONE (419) 259-5000 ext. 384

OPERATING STATUS

1. Unit Name: Davis-Besse #1
2. Reporting Period: October, 1983
3. Licensed Thermal Power (MWt): 2772
4. Nameplate Rating (Gross MWe): 925
5. Design Electrical Rating (Net MWe): 906
6. Maximum Dependable Capacity (Gross MWe): 918
7. Maximum Dependable Capacity (Net MWe): 874
8. If Changes Occur in Capacity Ratings (Items Number 3 Through 7) Since Last Report, Give Reasons:

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	7,295	46,056
12. Number Of Hours Reactor Was Critical	698.3	5,377.7	26,273.2
13. Reactor Reserve Shutdown Hours	45.7	515.2	3,879.3
14. Hours Generator On-Line	643.3	5,185.3	24,944.9
15. Unit Reserve Shutdown Hours	0.0	0.0	1,732.5
16. Gross Thermal Energy Generated (MWH)	985,280	12,525,609	57,898,370
17. Gross Electrical Energy Generated (MWH)	304,074	4,145,224	19,251,242
18. Net Electrical Energy Generated (MWH)	27[illegible],989	3,903,341	18,018,781
19. Unit Service Factor	86.5	71.1	54.2
20. Unit Availability Factor	86.5	71.1	57.9
21. Unit Capacity Factor (Using MDC Net)	41.8	61.2	44.8
22. Unit Capacity Factor (Using DER Net)	40.4	59.1	43.2
23. Unit Forced Outage Rate	13.5	10.0	18.6

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:
26. Units In Test Status (Prior to Commercial Operation):

	Forecast	Achieved
INITIAL CRITICALITY		
INITIAL ELECTRICITY		
COMMERCIAL OPERATION		

UNIT SHUTDOWNS AND POWER REDUCTIONS

REPORT MONTH October, 1983

DOCKET NO. 50-346
UNIT NAME Davis-Besse Unit 1
DATE November 9, 1983
COMPLETED BY Bilal Sarsour
TELEPHONE 419-259-5000, Ext. 384

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
9	83 10 02	F	24.0	A	3	N/A	EB	TRANSF	The reactor tripped on low Reactor Coolant System (RCS) pressure due to an attempted plant runback with controls in manual after receiving a main transformer danger alarm. During the startup, the reactor tripped from approximately 1% power due to improper operation of the #2 Startup Feedwater Valve.
10	83 10 03	F	23.7	A	3	N/A	HH	INSTRU	The reactor tripped on low RCS pressure when the main feedwater control valve went full open causing an overfeed condition. The erroneous valve actuation resulted from an improperly implemented facility modification to the Integrated Control System (ICS).

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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UNIT SHUTDOWNS AND POWER REDUCTIONS

REPORT MONTH October, 1983

DOCKET NO. 50-346
UNIT NAME Davis-Besse Unit 1
DATE November 9, 1983
COMPLETED BY Bilal Sarsour
TELEPHONE 419-259-5000, Ext. 384

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
11	83 10 08	F	16.3	A	5	N/A	HA	PIPEXX	The turbine generator was taken off line to repair a high pressure turbine casing drain steam leak but the reactor stayed critical.
12	83 10 15	F	22.8	A	3	NP-33-83-77	HH	INSTRU	During the power reduction, the reactor tripped by the Anticipatory Reactor Trip System (ARTS) generated by trips in the Steam and Feedwater Rupture Control System (SFRCS) due to problems with the main feed pump turbine control system.
13	83 10 29	F	13.9	A	5	N/A	HA	INSTRU	The turbine generator was taken off line to repair Turbine Control Valve #3, but the reactor stayed critical. 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
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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

(9/77)

OPERATIONAL SUMMARY
October, 1983

10/1/83 - 10/2/83

Following the completion of the refueling outage on September 30, 1983, reactor power was slowly increased and attained 40% full power on October 2, 1983.

At 0140 hours on October 2, 1983, a rapid power reduction was initiated due to a main transformer trouble alarm received in the Control Room. At 0145 hours on October 3, 1983, and during the power reduction, a combination of existing conditions and events resulted in a mismatch of reactor power and feedwater flow that resulted in a reactor trip on low Reactor Coolant System (RCS) pressure from approximately 20% full power. The cause of the transformer difficulties was due to a defective contact in the main generator exciter field breaker interlock. This caused the transformer cooling system to fail to operate in automatic.

The reactor was critical at 1416 hours on October 2, 1983. At 1511 hours on October 2, 1983, while the unit was at approximately 1% power, a reactor trip occurred due to improper operation of the #2 Startup Feedwater Valve. The valve did not open until the steam generator level was below the low level limit, after which the valve opened excessively. This reduced the feedwater flow to Steam Generator #1, which resulted in two Steam and Feedwater Rupture Control System (SFRCS) half trips, and the Anticipatory Reactor Trip System (ARTS) automatically tripped the reactor.

The reactor was critical at 1845 hours on October 2, 1983.

10/3/83 - 10/7/83

The turbine generator was synchronized on line at 0143 hours on October 3, 1983.

Reactor power was slowly increased and attained approximately 28% full power at 0400 hours on October 3, 1983.

At 0446 hours on October 3, 1983, while the unit was at approximately 28% full power, the main feedwater control valve went full open, causing an overfeed condition which resulted in a reactor trip on low RCS pressure. The problem was found to be in the Integrated Control System (ICS).

The reactor was critical at 2328 hours on October 3, 1983.

The turbine generator was synchronized on line at 0428 hours on October 4, 1983.

The reactor power was slowly increased to 40% of full power which was attained at 2000 hours on October 4, 1983. Physics testing at the 40% power level was completed at 1200 hours on October 7, 1983.

10/8/83 - 10/13/83

Reactor power was maintained at approximately 40% power until 0720 hours on October 8, 1983, when the turbine was taken off line to repair high pressure turbine casing drain steam leak, but the reactor stayed critical at approximately 7% power.

The turbine generator was synchronized on line at 2337 hours on October 8, 1983.

The reactor power was slowly increased and attained approximately 50% power on October 9, 1983.

Reactor power was limited to approximately 50% power due to main feed pump turbine control problems.

Reactor power was maintained at 50% power until 0700 hours on October 13, 1983, when reactor power was increased to approximately 75% full power.

10/14/84 - 10/24/83

Reactor power was maintained at approximately 75% power until 2200 hours on October 14, 1983, when a manual power reduction was initiated by the operator due to another high pressure turbine casing drain line steam leak.

At 0427 hours on October 15, 1983, during the power reduction, a reactor trip occurred from approximately 31% of full power. Problems with the main feed pump turbine control system had caused feedwater oscillations that resulted in an ARTS trip generated by trips in the SFRCS. The ARTS trip automatically tripped the reactor.

The reactor was critical at 1530 hours on October 15, 1983. The turbine generator was synchronized on line at 0313 hours on October 16, 1983.

Reactor power was slowly increased and attained approximately 75% power at 0700 hours on October 24, 1983.

10/24/83 - 10/28/83

Reactor power was maintained at approximately 75% power until 0232 hours on October 26, 1983, when an automatic runback to 60% occurred because of the loss of Control Rod Group 3 out limit.

Reactor power was slowly increased to approximately 75% of full power which was attained at 0700 hours on October 26, 1983. Physics testing at the 75% power level was completed at 0930 hours on October 28, 1983.

10/29/83 - 10/31/83

Reactor power was maintained at approximately 75% of full power until 0422 hours on October 29, 1983, when the turbine generator was taken off line to repair control valve #3, but the reactor stayed critical.

The turbine generator was synchronized on line at 1813 hours on October 29, 1983.

The reactor power was slowly increased and attained approximately 90% power at 2000 hours on October 30, 1983, and maintained at this power level for the rest of the month.

REFUELING INFORMATION DATE: October, 1983

1. Name of facility: Davis-Besse Unit 1

2. Scheduled date for next refueling shutdown: August 3, 1984

3. Scheduled date for restart following refueling: October 26, 1984

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: July, 1984

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) 140 - 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: 1993 - assuming ability to unload the entire core into the spent fuel pool is maintained.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 77-107

SYSTEM: Steam and Feedwater Rupture Control System (SFRCS)

COMPONENT: Steam Generator Level Transmitters

CHANGE, TEST OR EXPERIMENT: Trip setpoints for the Steam Generator level transmitters, LSL-SP9A6, SP9A7, SP9A8, SP9A9, SP9B6, SP9B7, SP9B8, and SP9B9, were changed from 24" water ± 5" to 23" water ± 2". This change was verified February 22, 1980.

REASON FOR CHANGE: Technical Specifications require the water level to be ≧20" above the lower tubesheet. The previous setpoint would have allowed the water to get as low as 19" above the lower tubesheet. The new setpoint allows a 1" margin from the minimum specified in the Technical Specifications.

SAFETY EVALUATION: This change has assured that the Steam Generator levels will remain within the limits set forth in the Technical Specifications.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 77-276

SYSTEM: Communications

COMPONENT: N/A

CHANGE, TEST OR EXPERIMENT:

Work implemented by this FCR was completed June 15, 1981. This involved the addition of a visual page system and a visual alarm system in the Emergency Diesel Generator rooms, day tank rooms and the Diesel Fire Pump room. The visual page system is activated by pushing and holding the button located next to the page telephone console in the Control Room. This actuates rotar amber lights in these rooms. The visual alarm system's rotary red lights are actuated by the fire alarm, containment evacuation alarm or initiate emergency procedures alarm to warn an operator in these rooms.

REASON FOR CHANGE:

An operator could not be paged in the Emergency Diesel Generator rooms prior to this addition when the diesel was running due to the high noise level.

SAFETY EVALUATION:

This change has not affected the safety function of the Emergency Diesel Generator system because this modification was only to the communications systems in these rooms. This FCR was safety related only because PICAs were involved. Installation in accordance with PICA has precluded the creation of any adverse environments. This was not an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 78-528

SYSTEM: Radiation Monitoring

COMPONENT: RE2387 and RE2389

CHANGE, TEST OR EXPERIMENT: Cooling ductwork was extended from RE2004, Safety Features Actuation System (SFAS) Channel 1 Containment Radiation Monitor, to RE2387, Containment Wide Range Radiation Monitor, and from RE2007, SFAS Channel 4 Containment Radiation Monitor, to RE2389, Containment Wide Range Radiation Monitor. This work was completed August 15, 1980.

REASON FOR CHANGE: The ambient air around these monitors was hotter than specified in design specifications. Cooling these monitors has reduced their failure rate which had been about once every six months.

SAFETY EVALUATION: The two 10 inch diameter stainless steel duct runs were subject to PICA requirements to assure that they have not created any new adverse environments to nearby safety related equipment. No unreviewed safety question exists.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 79-014

SYSTEM: Fire Protection

COMPONENT: Auxiliary Building Fire Detection

CHANGE, TEST OR EXPERIMENT:

Fire detection systems were installed in rooms 101, 105, 115, 113, 110 and 124, all of which are on elevation 545'-00" of the Auxiliary Building. Work was completed October 30, 1980.

REASON FOR CHANGE:

This change was completed to upgrade the Fire Protection System in order to comply with commitments made in the Fire Hazard Analysis Report.

SAFETY EVALUATION:

Installation in accordance with the core drill report and PICA has precluded these portions from creating any new adverse environments. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 79-020

SYSTEM: Fire Protection

COMPONENT: Fire detection

CHANGE, TEST OR EXPERIMENT:

Fire detection systems were added in rooms 216, 218, 214, 215, 220, 317 and 410. All rooms are located in containment. Work was completed July 18, 1980.

REASON FOR CHANGE:

This modification was completed to upgrade the Fire Protection System in order to comply with commitments made in the Fire Hazard Analysis Report.

SAFETY EVALUATION:

Installation in accordance with the core drill report and PICA has precluded these portions from creating any new adverse environments. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 79-044

SYSTEM: Fire Protection

COMPONENT: Sprinkler System

CHANGE, TEST OR EXPERIMENT:

A sprinkler system was added to Room 314, the No. 4 Mechanical Penetration room, elevation 585'-00". Work was completed February 26, 1981.

REASON FOR CHANGE:

This change was completed to upgrade the Fire Protection System in order to comply with commitments made in the Fire Hazard Analysis Report.

SAFETY EVALUATION:

Installation in accordance with core drill reports and PICA has precluded these portions from creating any new adverse environments. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 79-047

SYSTEM: Fire Protection

COMPONENT: Sprinkler System

CHANGE, TEST OR EXPERIMENT:

A sprinkler system was added to passage 310 and hatch 313, both on elevation 585'-0". Work was completed July 23, 1980.

REASON FOR CHANGE:

This change was required by commitments made in the Fire Hazard Analysis Report.

SAFETY EVALUATION:

This work is non-nuclear safety related except for a core drill cut out. Installation in accordance with the "Q" core drill report and PICA has precluded those portions from creating any new adverse environment. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 79-048

SYSTEM: Fire Protection

COMPONENT: Sprinkler System

CHANGE, TEST OR EXPERIMENT:

A sprinkler system was added to Room 304, the east-west corridor on elevation 585'-00". Work was completed November 11, 1980.

REASON FOR CHANGE:

This change was completed to upgrade the Fire Protection System in order to comply with commitments made in the Fire Hazard Analysis Report.

SAFETY EVALUATION:

This work is non-nuclear safety related except for 2 core drill cutouts. Installation in accordance with the "Q" core drill report and PICA has precluded these portions from creating any new adverse environments. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 79-049

SYSTEM: Fire Protection

COMPONENT: Sprinkler System

CHANGE, TEST OR EXPERIMENT:

A sprinkler system was added for the radwaste exhaust equipment and Main Steam exhaust fan room, room 561 on elevation 623'-00". Work was completed September 28, 1980.

REASON FOR CHANGE:

This change was completed to upgrade the Fire Protection System in order to comple with commitments made in the Fire Hazard Analysis Report.

SAFETY EVALUATION:

This work is non-nuclear safety related except for 2 core drill cutouts. Installation in accordance with the core drill report and PICA has precluded these portions from creating any new adverse environments. An unreviewed safety question is not involved.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 79-053

SYSTEM: Fire Protection

COMPONENT: Sprinkler System

CHANGE, TEST OR EXPERIMENT:

A sprinkler system was added to the clean waste receiver tank area, Room 124 on elevation 565'-00". Work was completed June 5, 1980.

REASON FOR CHANGE:

This change was completed to upgrade the Fire Protection System in order to comply with commitments made in the Fire Hazard Analysis Report.

SAFETY EVALUATION:

Installation in accordance with core drill report and PICA has precluded these portions from creating any new adverse environments. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 79-063

SYSTEM: Fire Protection

COMPONENT: Hose Station

CHANGE, TEST OR EXPERIMENT:

A hose station was added in the Diesel Generator Room 319, on elevation 585'-00". Work was completed February 26, 1981.

REASON FOR CHANGE:

This change was completed to upgrade the Fire Protection System in order to comply with commitments in the Fire Hazard Analysis Report.

SAFETY EVALUATION:

This work is non-nuclear safety related except for the installation of the core drill. Installation in accordance with the core drill procedure has precluded the creation of any adverse environments. This was not an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 79-238

SYSTEM: Fire Protection

COMPONENT: Sprinkler System

CHANGE, TEST OR EXPERIMENT:

This FCR was implemented to add a sprinkler system in storage room 405 at elevation 603'. All work was completed August 25, 1980.

REASON FOR CHANGE:

This change was required by commitments made in the Fire Hazard Analysis Report.

SAFETY EVALUATION:

Work required by this FCR is non-nuclear safety related except for 1 core drill cut out through a negative pressure boundary. Installation in accordance with the "Q" core drill report and PICA requirements precluded the creation of any adverse environments.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 79-239

SYSTEM: Fire Protection

COMPONENT: Sprinkler System

CHANGE, TEST OR EXPERIMENT:

A sprinkler system was added to room 427, No. 2 Electrical Penetration Room on elevation 603'-0". Work was completed October 8, 1980.

REASON FOR CHANGE:

This change was completed to upgrade the Fire Protection System in order to comply with commitments made in the Fire Hazard Analysis Report.

SAFETY EVALUATION:

Installation in accordance with core drill report has precluded these portions from creating any new adverse environments. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-240

SYSTEM: Fire Protection

COMPONENT: Sprinkler System

CHANGE, TEST, OR EXPERIMENT:

A sprinkler system was added in No. 1 Electrical Penetration Room 402, Elevation 603'-0". Work was completed September 28, 1980.

REASON FOR CHANGE:

This modification was completed to upgrade the fire protection system in order to comply with commitments made in the Fire Hazard Analysis Report.

SAFETY EVALUATION:

This work was non-nuclear safety related except for a core drill. Installation in accordance with the core drill report and PICA has precluded those portions from creating any new adverse environments. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 79-325

SYSTEM: Communications

COMPONENT: N/A

CHANGE, TEST OR EXPERIMENT:

New speakers for the "Gaitronics" were added in the Boric Acid Evaporator rooms, 234 and 235. Work was completed April 28, 1980.

REASON FOR CHANGE:

NRC Bulletin 79-18 requires the communications and alarm system to be clearly audible throughout the entire plant. The alarms could not be heard in these rooms since they had no speakers and have a high backround noise level.

SAFETY EVALUATION:

The conduit for this addition was routed through an existing penetration which was resealed with silicone foam. The implementation of this FCR has had no effect on plant safety. This was not an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 82-093

SYSTEM: Reactor Coolant System (RCS)

COMPONENT: Pressurizer Surge Line

CHANGE, TEST OR EXPERIMENT:

This change involved the modification of the gaps between the pressurizer surge line and pipe whip restraints SL3, SL4, SL6 and SL7. All work was completed and Bechtel drawing 7749-C-189 was updated July 31, 1983.

REASON FOR CHANGE:

Toledo Edison Non-conformance Report 397-82 noted that certain gaps between the pressurizer surge line and the slims on the pipe whip restraints SL1 through SL8 were different than shown on drawing 7749-C-189. The as measured gaps were analyzed and it was determined that four gaps needed to be changed and that the others were acceptable but the design drawing needed to be changed.

SAFETY EVALUATION:

The function of pipe whip restraints is, in the event of a pipe rupture, to prevent the ruptured pipe from whipping around and damaging other safety related piping and equipment. The energy that a pipe whip restraint is designed to withstand is a function the gap between the pipe and the restraint. Gaps that were larger than shown on Bechtel drawing 7749-C-189 were modified to conform to the design drawing gaps that were smaller than shown on the drawing are adequate for pipe whip design. These smaller gaps were also renewed for seismic and thermal movement and found to be acceptable. This modification has not created an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-127

SYSTEM: 480 Volt Essential AC Power

COMPONENT: Heater Breakers BE1223 and BF1217

CHANGE, TEST OR EXPERIMENT: This FCR implemented the replacement of existing 200 amp trip units on essential heater breakers BE1223 and BF1217 with 250 amp trip units. These breakers were installed and trip points were verified on August 31, 1982.

REASON FOR CHANGE: The current through the essential heater breakers BE1223 and BF1217 is approximately 160 amps, which is normal for a full heater bank. With a trip setpoint of 200 amps, the normal current, rather than an overcurrent, was causing these units to trip. The non-essential heater breakers were replaced with 250 amp trip units by FCR 78-430.

SAFETY EVALUATION: The safety function of the Class 1E breakers is to timely isolate the non-1E pressurizer heaters from the 1E power supply in case of electrical fault downstream from the breaker. Replacement of these trip units enhances this function rather than adversely affects it. Therefore, this is not an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 83-016

SYSTEM: Main Steam

COMPONENT: MS611, MS603

CHANGE, TEST OR EXPERIMENT:

This FCR was implemented to change the torque switch settings for motor operated valves MS611, Steam Generator (SG) 1-1 drain line to low pressure condenser motor isolation valve and MS603 SG 1-2 drain line to high pressure condenser motor isolation valve. The new settings for both MS611 and MS603 are 3.0 to open and 1.5 to close. Work was completed January 29, 1983.

REASON FOR CHANGE:

Torrey Pines Technology had recommended these torque switch settings in their limitorque motor operated valve study to reduce the likelihood of failure of these valves.

SAFETY EVALUATION:

The safety function of these valves is to provide assurance that the containment atmosphere is isolated from the outside atmosphere in the event of a release of radioactive material to the containment atmosphere or pressurization of containment. The modification has enhanced the reliability of these valves. The safety function of valves MS611 and MS603 were not adversely affected, therefore, no unreviewed safety question existed.

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TOLEDO EDISON

November 9, 1983

Log No. K83-1583
File: RR 2 (P-6-83-10)

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, October 1983
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 October, 1983.

If you have any questions, please feel free to contact Bilal Sarsour at (419) 259-5000, Extension 384.

Yours truly,

Terry D. Murray/SMQ

Terry D. Murray
Station Superintendent
Davis-Besse Nuclear Power Station

IE24
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TDM/BMS/ljk

Enclosures

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

Mr. Richard DeYoung, Director, w/2
Office of Inspection and Enforcement

Mr. Walt Rogers, w/1
NRC Resident Inspector