AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346

UNIT Davis-Besse Unit 1

DATE June 8, 1984

COMPLETED BY Bilal Sarsour

TELEPHONE 419-259-5000, Ext. 384

MONTH May, 1984

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	817	17	0
2	814	18	266
3	815	19	652
4	814	20	727
5	816	21	801
6	812	22	800
7	811	23	806
8	811	24	804
9	817	25	801
10	813	26	798
11	814	27	799
12	809	28	800
13	778	29	807
14	811	30	810
15	816	31	809
16	47		

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)

IE 24

OPERATING DATA REPORT

DOCKET NO. 50-346
DATE June 8, 1984
COMPLETED BY Bilal Sarsour
TELEPHONE 419-259-5000, Ext. 384

OPERATING STATUS

1. Unit Name: Davis-Besse Unit 1
2. Reporting Period: May, 1984
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): 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	3,647	51,168
12. Number Of Hours Reactor Was Critical	697.5	3,082.8	30,585.3
13. Reactor Reserve Shutdown Hours	0.0	134.8	4,014.1
14. Hours Generator On-Line	693.9	3,046.1	29,197.9
15. Unit Reserve Shutdown Hours	0.0	0.0	1,732.5
16. Gross Thermal Energy Generated (MWH)	1,760,996	7,852,300	68,896,114
17. Gross Electrical Energy Generated (MWH)	576,183	2,584,199	22,876,392
18. Net Electrical Energy Generated (MWH)	543,727	2,433,290	21,431,989
19. Unit Service Factor	93.3	83.5	57.1
20. Unit Availability Factor	93.3	83.5	60.4
21. Unit Capacity Factor (Using MDC Net)	83.6	76.3	47.9
22. Unit Capacity Factor (Using DER Net)	80.7	73.6	46.2
23. Unit Forced Outage Rate	6.7	16.5	18.3

24. Shutdowns Scheduled Over Next 6 Months (Type, Date, and Duration of Each):
Refueling Outage: Scheduled Start 9/1/84, Scheduled End 11/9/84

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

DOCKET NO. 50-346
UNIT NAME Davis-Besse Unit 1
DATE June 8, 1984
COMPLETED BY Bilal Sarsour
TELEPHONE 419-259-5000, Ext. 384

REPORT MONTH May, 1984

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
3	84 05 15	F	50.1	A	1	NP-33-84-06	CJ	VALVEX	A unit shutdown was initiated due to Reactor Coolant System leakage in containment. (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

(9/77)

OPERATIONAL SUMMARY
MAY, 1984

5/1/84 - 5/13/84

Reactor power was maintained at 94% with the generator gross load at approximately 850 ±10 MWe until 0200 hours on May 13, 1984, when a manual power reduction to approximately 85% was initiated to perform turbine valve testing. (Reactor power was limited to 94% due to an inoperable main steam safety valve.)

After the completion of turbine valve testing, reactor power was slowly increased and attained approximately 94% at 0900 hours on May 13, 1984.

5/14/84 - 5/18/84

Reactor power was maintained at 94% power until 0020 hours on May 16, 1984, when a plant shutdown was initiated due to Reactor Coolant System leakage which was identified to be out of the packing of the pressurizer mini spray flow valve, RC49. The first three rings of packing had loosened allowing the leakage to occur. The valve packing has been replaced.

The reactor was critical at 0432 hours on May 18, 1984. The turbine generator was synchronized on line at 0630 hours on May 18, 1984.

5/19/84 - 5/31/84

Reactor power was slowly increased and attained approximately 94% power on May 21, 1984 and maintained at this power level for the rest of the month.

REFUELING INFORMATION DATE: May, 1984

1. Name of facility: Davis-Besse Unit 1

2. Scheduled date for next refueling shutdown: September 1, 1984

3. Scheduled date for restart following refueling: November 9, 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.

BMS/005

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 77-291

SYSTEM: Control Room Heating, Ventilating, and Air Conditioning

COMPONENT: Thermal Overload Heaters

CHANGE, TEST OR EXPERIMENT: This FCR replaced the thermal overload heaters with a Westinghouse heater FH-50, modified the control scheme to have the overload condition alarm only, and to replace the combination starter with a breaker at the motor control center. All work was completed August 12, 1982.

REASON FOR CHANGE: Problems had been experienced with the Control Room emergency condensing units on overload conditions due to undersized heaters. This change was performed to provide a more reliable design.

SAFETY EVALUATION: The equipment that was modified is safety related and should not trip on overload conditions. The safety of this system was enhanced, therefore, an unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 77-430

SYSTEM: 4.16 KV Essential Power

COMPONENT: Undervoltage Relays

CHANGE, TEST OR EXPERIMENT: The time delay settings of undervoltage relays 27A1 through 27A4 were reduced from ten to nine seconds. A bypass button was also installed to defeat the undervoltage relays during a reactor coolant pump motor or a circulating water pump motor start. A blue light within the switch indicates when the bypass condition is in effect. Work was completed September 6, 1983.

REASON FOR CHANGE: This modification insures emergency core cooling system injection within 30 seconds and protects against unnecessary actuation of the undervoltage relays on essential 4.16 KV buses during reactor coolant pump motor and circulating water pump motor starts.

SAFETY EVALUATION: The changes in the 4.16 KV protection did not adversely affect the safety analysis. To increase the margin of safety a slightly more conservative setting of 90% voltage for nine seconds was decided upon. This change has assured emergency core cooling system injection in 30 seconds. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 78-256

SYSTEM: Hydrogen Dilution System

COMPONENT: Panel C3801

CHANGE, TEST OR EXPERIMENT: Drawing E-460A, Item C, Sheet 2, was revised to provide a support detail for electrical panel C3801. The drawing was updated and verified November 30, 1983.

REASON FOR CHANGE: Bechtel Non-Conformance Report 1254 required the addition of the gussets to the subject panel. Disposition of TED NCR 78-124 was that the gussets were adequate and "use as is". These drawings were updated to show the as-built support detail.

SAFETY EVALUATION: The safety function of the supports and anchors is to provide anchorage from seismic loading for the electrical panel.

The as-built condition of the supports and anchors was addressed by Bechtel BT-8092 and found that the as-built condition of the anchors and supports for panel C3801 perform their intended safety function. This modification did not involve an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 78-265

SYSTEM: Various

COMPONENT: Valve and Motor Terminations

CHANGE, TEST OR EXPERIMENT: This FCR called for the replacement of Scotchfill putty and Scotch Brand 33 tape with Raychem Heat Shrink. Work was completed November 24, 1980.

REASON FOR CHANGE: Insufficient environmental qualification existed for Scotchfill putty and Scotch Brand 33 tape to allow its use on terminations of essential equipment.

SAFETY EVALUATION: Raychem heat shrink is environmentally qualified for essential equipment terminations. Hence, an unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-017

SYSTEM: Fire Protection

COMPONENT: Fire Detection System

CHANGE, TEST OR EXPERIMENT: Fire Detection Systems were installed in Rooms 400, 402, 404, 405, 406, 411, 427, 428, 428A, 429B, and in the annulus space in the areas of electrical penetrations. 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 this addition from creating any new adverse environments. An unreviewed safety question is not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-018

SYSTEM: Fire Protection

COMPONENT: Auxiliary Building Fire Protection

CHANGE, TEST OR EXPERIMENT: Fire Detection Systems were installed in Rooms 500, 501, 502, 503, 504, 505, 506, 507, 510, 511, 512, 513, 514, 515, and 516. All of these rooms are located on Elevation 623'-0" of the Auxiliary Building. Work was completed October 8, 1980.

REASON FOR CHANGE: This change was completed to upgrade the Fire Protection System 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 REQUEST

FCR NO: 79-070

SYSTEM: 480 Volt System Essential Power

COMPONENT: Unit Substations E1, E2, E3, F1, F2, F3, EF4, and EF6

CHANGE, TEST OR EXPERIMENT: General Electric connection diagrams were revised to agree with Bechtel Elementary Diagrams E36B and E37B. The General Electric drawings that were updated were 0122D1138, 0122D1142, 0122D1156, 0122D1160, 0122D1147, 0122D1151, 0122D1165, 0122D1174, 0122D1178, 0122D1183, 0122D1187, 0122D1192, 0122D1196, 0126D1088, 0126D1092, and 0122D1169. This was completed February 7, 1979.

REASON FOR CHANGE: General Electric connection diagrams incorrectly showed the common lead for current transformer circuit terminated on the ammeter itself instead of on the ammeter switch. These drawings now more accurately reflect conditions in the plant.

SAFETY EVALUATION: The unit substations provide electrical power to various nuclear safety related components. The current transformers provide an indication of flow of electric power from the unit substations. This drawing change has not affected the safety functions of the substation and does not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-123

SYSTEM: Borated Water Storage Tank (BWST)

COMPONENT: Instrument enclosures

CHANGE, TEST OR EXPERIMENT: This modification provided the replacement of O'Brien Model B enclosures with O'Brien Model A enclosures on instruments LIS6822, LS6823A, LT6823, LSLL6823, LSLL6823A, LT1051A, LT1051, LS1051, LT1525A, B, C, D, LT6601, LSLL6812, LS6808A, and LS6808B. Heat trace was also provided for LT1525 A and C. Work was completed October 8, 1981.

REASON FOR CHANGE: The original Model B enclosures were designed in such a way that the seal was not adequate to maintain proper temperature. The installation of Model A enclosures allows the instrument to be mounted in the enclosure using the existing instrument brackets, (reference Licensee Event Reports NP-33-78-01 and NP-33-79-32). The heat traces were added due to their omission in the original design of the freeze protection, (reference Licensee Event Reports NP-33-80-23 and NP-33-80-25).

SAFETY EVALUATION: This change facilitated maintainability of the instruments and enhanced the reliability of this system. The new enclosures are independently supported and ensure a more positive enclosure for instrument environmental control. This did not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-158

SYSTEM: Miscellaneous

COMPONENT: Valves

CHANGE, TEST OR EXPERIMENT: This FCR provided locks for valves in various systems and to control their position with AD 1839.02, Operation and Control of Locked Valves, provided seals on valves, and deleted seals from valves. Work was completed July 6, 1979.

REASON FOR CHANGE: This modification ensures that all valves on safety related systems are in their correct position to assure each system will perform its intended safety function.

SAFETY EVALUATION: This FCR insures manually operated valves are not altered so as to prevent a system from operating properly. Directed and planned position alteration can be accomplished by unlocking or breaking a valve seal. Administrative controls will be in effect whenever this occurs. This does not present an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-164

SYSTEM: Miscellaneous

COMPONENT: Anchors for hangers

CHANGE, TEST OR EXPERIMENT: Concrete expansion anchors installed for hangers in nuclear safety related piping systems were inspected and tested. This was done in accordance with Bechtel procedure PDP-1, Inspection and Testing Procedure for Concrete Expansion Anchors. All work was verified as completed on October 18, 1983.

REASON FOR CHANGE: This has verified that installed anchors meet or exceed the anchor's design loadings in response to Nuclear Regulatory Commission I.E. Bulletin 79-02.

SAFETY EVALUATION: Procedure PDP-1 controlled the removal of one bolt at a time and working on one hanger at a time in operating systems. Hanger spacing was determined by seismic loading so the temporary removal of a hanger did not affect any system's performance. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-165

SYSTEM: Auxiliary Feedwater

COMPONENT: Valves

CHANGE, TEST OR EXPERIMENT: This FCR added to Piping and Instrument Diagram M-007 the notation that shows feedwater valves 2685B&D and 2686B&D normally closed. Drawing FSK-M-EBD-14-4 was updated to show the removal of pressure differential switches (PDS) 2686A-D and FSK-M-EBD-14-5 was changed to show the removal of PDS2685A-D. This was verified January 16, 1984.

REASON FOR CHANGE: This change has made these drawings more accurately reflect conditions in the plant.

SAFETY EVALUATION: These drawing updates have made the drawings more accurate. This is not an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-215

SYSTEM: Auxiliary Feedwater

COMPONENT: Auxiliary Feed Pump Suction Lines

CHANGE, TEST OR EXPERIMENT: The original strainer baskets, S201 and S206, in the suction lines to the Auxiliary Feed Pumps were replaced by strainer baskets with 1/16" holes. This FCR also replaced strainer S257 in the common auxiliary feed pump suction line with a conical strainer. Work was completed August 19, 1981.

REASON FOR CHANGE: This modification has prevented debris from the non-nuclear safety related supply to the pumps from plugging the suction strainers and interrupting supply from the Service Water System.

SAFETY EVALUATION: These strainers were replaced in order to minimize the chance of pluggage of these strainers. This modification was approved by Byron Jackson, the pump manufacturer, and has improved the operability of the Auxiliary Feedwater System. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-317

SYSTEM: Auxiliary Feedwater

COMPONENT: Auxiliary Feed Pumps

CHANGE, TEST OR EXPERIMENT: The automatic closure of service water valves SW1382 and SW1383 and the automatic opening of auxiliary feed pump suction valves from condensate storage tank FW786 and FW790, when normal suction pressure increases above 2 psig, has been eliminated. The logic of computer points P008 and P009 was changed so that they alarm if either pressure switch detects an auxiliary feed pump suction pressure less than 2 psig. Work was completed March 5, 1983.

REASON FOR CHANGE: In the previous system, if the auxiliary feed pump suction recovered to 2 psig after falling below 2 psig, auxiliary feed pump suction was transferred back to the condensate storage tanks automatically. In the event that suction pressure oscillated near 2 psig, the valves unnecessarily cycled open and closed. With this modification, when the suction falls below 2 psig, the suction is transferred to the Seismic 1 Service Water System. This alignment will be maintained until at least one pressure switch senses normal pressure and the operator manually aligns the condensate water supply to the auxiliary feed pumps.

SAFETY EVALUATION: This change does not involve a safety issue since adequate water will be available to the auxiliary feed pumps at all times to remove decay heat from the core and to achieve safe shutdown.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-080

SYSTEM: Auxiliary Feedwater

COMPONENT: Pipe Hangers

CHANGE, TEST OR EXPERIMENT: This FCR was implemented to modif, pipe hangers 6C-EBD-14-H1, H9, H11, H14, H33, H67, H70, H80, H90, 6A-HBD-24-H2, and 6A-HBC-35-A34. Anchor A-216 was also modified and anchor A-489 was installed in place of anchor A-8. All affected hangers and anchors are shown on Bechtel isometric 7749-M-206F. Work was completed June 3, 1983.

REASON FOR CHANGE: Modifications to the original designs were required as a result of reanalysis required by I.E. Bulletins 79-02 and 79-14.

SAFETY EVALUATION: The modifications to the subject hangers and anchors have reduced the stresses to an acceptable level and increased the factor of safety. No unreviewed safety question was involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-086

SYSTEM: Hydrogen Dilution

COMPONENT: Pipe Hangers and Beam

CHANGE, TEST OR EXPERIMENT: This FCR provided for the modification of piping hangers 29-HBC-73-H8, 29-HBB-15-H7, 29-HBB-15-H9, and 29-HBC-74-H11. Plates were also added to an existing beam in the Auxiliary Building Area 7, Elevation 603'. All modified equipment are part of the Hydrogen Dilution System and are shown on isometric 7749-M-229. Work was completed June 3, 1983.

REASON FOR CHANGE: These changes were required as a result of reanalysis of pipe supports in accordance with IE Bulletins 79-02 and 79-14.

SAFETY EVALUATION: These modifications have reduced the stresses to an acceptable level and increased the factor of safety. No unreviewed safety question existed.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-090

SYSTEM: Spent Fuel Pool Cooling

COMPONENT: Pipe Hangers and Anchors

CHANGE, TEST OR EXPERIMENT: This FCR provided for the modification of pipe hangers 35-HCB-6-H3, 35-HCB-10-H1, 35-HCC-21-H12, 35-HCB-24-H9, 35-HCB-24-H8, 35-HCB-24-H5, and pipe anchors A-418, A-406, and A-206. All subject pipe supports are in the spent fuel pool cooling system and are shown on isometric 7749-M-235B. Work was completed April 22, 1983.

REASON FOR CHANGE: These changes were required as a result of reanalysis of pipe supports and anchors in accordance with IE Bulletins 79-02 and 79-14.

SAFETY EVALUATION: This modification has reduced the stresses to an acceptable level and increases the factor of safety. No unreviewed safety question existed.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-092

SYSTEM: Reactor Coolant System (RCS)

COMPONENT: Pipe Supports

CHANGE, TEST OR EXPERIMENT: Modifications were performed to pipe supports for reactor coolant drains. All piping is shown on isometric 7749-M-240C. Work was completed September 6, 1983.

REASON FOR CHANGE: These modifications have reduced the level of stress on the pipe supports as required by IE Bulletins 79-02 and 79-14.

SAFETY EVALUATION: The modifications to pipe supports in the reactor coolant drain system have reduced the level of stress to acceptable levels and increased the margin of safety. No unreviewed safety question was involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-116

SYSTEM: Component Cooling Water

COMPONENT: Heat Exchangers

CHANGE, TEST OR EXPERIMENT: Work implemented by this FCR was completed June 15, 1981. This involved the extension of the base of the fixed saddle support for the three component cooling water heat exchangers and the addition of bolts and a gusset to each base.

REASON FOR CHANGE: During seismic reevaluation for an earthquake in magnitude of 0.20G, required by Operating License NPF-3, Section 2.c.(3)(r), it was determined that the base and anchor bolts of these heat exchangers could have become overstressed.

SAFETY EVALUATION: The more recent analysis used a better model of these heat exchangers and calculated more realistic stress values. It should be noted that the vendor's original seismic analysis report used acceptable 1971 state-of-art analysis techniques. This modification has not resulted in any unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-121

SYSTEM: Pressure Relief System

COMPONENT: Relief valve line

CHANGE, TEST OR EXPERIMENT: Pipe supports on the pilot operated relief valve line 30-CCA-8-H2, H1, and H6 were modified. Work was completed February 19, 1981.

REASON FOR CHANGE: These pipe supports were modified to meet ASME III piping stress analysis.

SAFETY EVALUATION: These modifications to the pressure relief system were designed to reduce the level of stress to an acceptable level to increase the margin of safety of the system. No unreviewed safety question existed.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-141

SYSTEM: Containment Spray

COMPONENT: Hydraulic Snubbers

CHANGE, TEST OR EXPERIMENT: The equalizing lines between the two snubbers on hanger 34-HCB-4-H32 was removed. This was verified October 21, 1983.

REASON FOR CHANGE: This piping arrangement had caused problems such as venting a snubber during installation and removal. Grinnell, the manufacturer, has found this arrangement to be unnecessary.

SAFETY EVALUATION: Manifolding of hydraulic snubbers is not required and may present problems. No adverse environment was created, and no unreviewed safety question existed.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-160

SYSTEM: Reactor Coolant System

COMPONENT: Reactor Coolant Pumps (RCP)

CHANGE, TEST OR EXPERIMENT: Insulation support for RCP 1-1, 3/4" piping CCB-7, CCB-26, and MCC-56; 3/4" hangers CCB-7 and HCC-56; and 1½" hanger CCB-8 were relocated. Work was completed April 18, 1981.

REASON FOR CHANGE: During the walkdown of inaccessible piping systems, as required by I.E. Bulletin 79-14, indications were observed that the RCP wire rope whip restraints had come in contact with the small piping around the pumps.

SAFETY EVALUATION: The relocation of this piping has decreased the likelihood of the RCP rope pipe whip restraints from coming in contact with the small piping and hangers without changing the ability of the piping to perform its intended safety function. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-174

SYSTEM: Component Cooling Water System

COMPONENT: Valves

CHANGE, TEST OR EXPERIMENT: This FCR involved changing Rockwell valve drawing D-473665 to allow installation of a valve with either a canopy or fillet type seal weld. Component Cooling Water Valve 238, Component Cooling to Hydrogen Gas Analyzer 1-2 Inlet Isolation Valve From Line 2, was replaced. Work was completed October 1, 1980.

REASON FOR CHANGE: In order to remove the cover for valve repair, the canopy weld must be ground out. This modification shows an alternate method of sealing the valve by rewelding with a fillet weld. This change has been reviewed and found acceptable by the valve manufacturer, Rockwell.

SAFETY EVALUATION: Sealing the cover of the valve to the body does not affect the ability of the valve to perform its safety function. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-178

SYSTEM: Diesel Generators

COMPONENT: Emergency Diesel Generator Exhaust

CHANGE, TEST OR EXPERIMENT: This modification included the replacement of an expansion joint, the addition of new supports, and modifications to original supports of the emergency diesel generator exhaust. Work was completed September 30, 1980.

REASON FOR CHANGE: During seismic reevaluation for an earthquake in magnitude of 0.20G, required by Operating License NPF-3, Section 2.C.(3)(r), it was determined that the nozzle on the emergency diesel generator would be overstressed.

SAFETY EVALUATION: The modifications completed under this FCR have made this design conform to the unit's Final Safety Analysis Report. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-237

SYSTEM: Diesel Generators

COMPONENT: Pipe Supports

CHANGE, TEST OR EXPERIMENT: Modifications were performed to pipe supports 47-HBD-359-H1, H5, H7, H9, H13, and H15. These supports are on the emergency diesel generator exhaust lines. Work was completed January 29, 1983.

REASON FOR CHANGE: These modifications were required as a result of reanalysis of the supports in accordance with IE Bulletins 79-02 and 79-14.

SAFETY EVALUATION: These changes have reduced the stresses in the hangers, increasing their factor of safety. This did not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-030

SYSTEM: Block Walls

COMPONENT: Walls 4786 and 4906

CHANGE, TEST OR EXPERIMENT: This FCR Braced the top connection of wall 4906 with a steel angle thru-bolted to the top of the wall on the north side and provided lateral bracing near the top of wall 4786 using a steel angle and struts thru-bolted to the north side of the wall. These walls form one wall unit which separates No. 1 Electrical Isolation Room (428B) from No. 2 Electrical Penetration Room (427) at Elevation 603'. Work was completed December 9, 1983.

REASON FOR CHANGE: Work performed as a result of NRC Bulletin 80-11 had shown that during a seismic event the top connection of walls 4786 and 4906 could become overstressed.

SAFETY EVALUATION: These walls function as a fire barrier and as a support for safety related conduits which are attached to or penetrate these walls.

This modification has lowered the stresses in the concrete masonry and connections to within allowable limits. An unreviewed safety question is not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-033

SYSTEM: Block Walls

COMPONENT: Wall 3026

CHANGE, TEST OR EXPERIMENT: This FCR attached steel angles to the top of masonry wall 3026 and the floor slab above. This wall separates Switch Room 324 from passage 322 at Elevation 585'. Work was completed November 4, 1983.

REASON FOR CHANGE: Work performed as a result of NRC Bulletin 80-11 had shown that during a seismic event the top connections of adjacent walls 3016 and 3036 to become overstressed.

SAFETY EVALUATION: The safety function of wall 3026 is to act as a fire barrier. This modification has lowered the stresses in the connections of walls 3016 and 3036 to within allowable limits. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-188

SYSTEM: Plant Computer

COMPONENT: Station Annunciator

CHANGE, TEST OR EXPERIMENT: This FCR consisted of installing new conduit and cable to provide 120 VAC alternate power on the non-essential 120 VAC uninterruptible distribution panel YBU to the station annunciator circuit. The preferred power supply is the 120 VAC uninterruptible distribution panel YAU. Work was completed May 12, 1983.

REASON FOR CHANGE: This alternate power source has ensured that power is always available for the Control Room annunciators.

SAFETY EVALUATION: The installed conduit and cables does not perform a specific safety function. However, "Q" core drill and cutout reports were involved. Installation in accordance with PICA requirements ensured that no new adverse environments were created. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-287

SYSTEM: Drawings

COMPONENT: Valves

CHANGE, TEST OR EXPERIMENT: Piping and Instrument Diagrams M-004B, M-006B, M-007, M-014, M-020, M-033, and M-034 were modified to show valves Feedwater 155A, Main Steam 891A, and Auxiliary Steam 354 were installed. This change was also initiated to show that valves Decay Heat 100A, Heater Drains 118A and 119A, Feedwater 83A and 26A, Vacuum System 101A, Auxiliary Steam 319A, Containment Spray 29A, and Steam Trap 98 were not installed. This change was verified January 4, 1984.

REASON FOR CHANGE: FCRs 77-504 and 77-505 requested a second valve be added downstream of existing valves, which were leaking, due to the fact that repair welding was not permitted in the negative pressure areas in which these valves are located. The valves were no longer necessary after a procedure change allowing the welding.

SAFETY EVALUATION: Of the valves originally requested as additional downstream isolation, only three, Decay Heat 100A, Feedwater 26, and Containment Spray 29A would have been nuclear safety related. Since no physical work was done this has not affected the safety function of the main valves. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-295

SYSTEM: Reactor Protection System (RPS)

COMPONENT: Trip setpoints

CHANGE, TEST OR EXPERIMENT: The following RPS trip setpoints were changed on all four channels:

Description	New Setpoint
Hi-Flux/4 Pumps	104.15% Full Power, +0.0, -0.75
Hi-Flux/3 Pumps	79.0% Full Power, +0.0, -0.75
Hi-Temperature	617.5°F, +0.0, -0.35
Lo-Pressure	1990.0 PSIG, +5.0, -0.0
Hi-Pressure	2295.0 PSIG, +0.0, -5.0
Pressure/Temperature	$12.6\ T_{HOT} - 5627.5$

Work was completed January 22, 1981.

REASON FOR CHANGE: The instrument string error was revised by Babcock and Wilcox due to a change in analytical methodology.

SAFETY EVALUATION: The revised instrument string error from Babcock and Wilcox resulted in a reduction in the safety margins of the RPS trip setpoints. The new setpoints were calculated to restore the safety margins by adjusting according to the differences between the old and new instrument string errors.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-007

SYSTEM: Solid Waste Disposal

COMPONENT: Spent Resin Storage Tank

CHANGE, TEST OR EXPERIMENT: This FCR was implemented to relocate the Spent Resin Storage Tank (SRST) rupture disk to outside of the SRST room. This included the addition of a vacuum breaker valve and flush connection piping. The rupture disk discharge piping was routed back into the SRST room to maintain the maximum amount of shielding in the event of a rupture disk relief. New SRST level instrumentation was also provided utilizing sealed diaphragm design reference legs instead of the original dry reference leg design. Work was completed August 10, 1983.

REASON FOR CHANGE: This modification has greatly reduced ALARA concerns when a rupture disk release occurs. The improved level instrumentation has provided accurate SRST level monitoring.

SAFETY EVALUATION: The improved level instrumentation has reduced the likelihood of relieving the rupture disk because of a inadvertent overfill.

Installation in accordance with PICA and "Q" core requirements has assured that no adverse environments were created. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-043

SYSTEM: Reactor Coolant System

COMPONENT: Pressurizer Surge Line

CHANGE, TEST OR EXPERIMENT: This modification involved the replacement of the concrete anchors with 1¼" x 9" bolts in the base plate of hanger PSU-H1. This was completed July 23, 1982.

REASON FOR CHANGE: The top of the hanger plate had pulled 3/8" out of the wall. Analysis by Bechtel had determined that the pressurizer surge line could have been overstressed during an earthquake with hanger PSU-H1 inoperable. (See Licensee Event Report 82-033).

SAFETY EVALUATION: The primary safety function of the PSU-H1 installation is to provide restraint to the surge piping during dynamic loading. These changes have not affected the ability of this hanger installation to act as designed. No unresolved safety question was involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-046

SYSTEM: Auxiliary Feedwater System

COMPONENT: Steam Generators

CHANGE, TEST OR EXPERIMENT: This FCR covered the modifications required for the steam generators associated with the internal auxiliary feedwater header deficiency. This included

hole drilling through the shell and shroud of the steam generators (nominal size up to 5" diameter),

stabilization of the existing auxiliary feedwater internal headers,

removal of the existing auxiliary feedwater nozzle thermal sleeve and installation of a flanged cover on the original nozzle.

Installation of an external header with eight injection nozzles required modification of the mirror insulation. Work was completed July 22, 1982.

REASON FOR CHANGE: These changes were implemented to correct the auxiliary feedwater header deficiency. The holes are required to provide access for the stabilization of the internal header and to provide injection ports for the new auxiliary feedwater external header. Stabilization of the internal header is required to prevent internal steam generator damage. Since the internal header will no longer be used for auxiliary feedwater injection, the existing nozzle and thermal sleeve will not be used. The nozzle opening was covered by a blind flange such that it can be used as an access port. Installation of the external header has provided means for auxiliary feedwater injection. The installation of the external header has required modifications of the mirror insulation. (See Licensee Event Report 82-019 for further details.)

SAFETY EVALUATION: The safety function of the steam generators is to act as a heat sink for reactor coolant and forms a portion of the primary pressure boundary.

The safety functions of the Auxiliary Feedwater System are to:

a) provide feedwater in the event of loss of main feedwater,

b) promote natural circulation in the event of loss of forced circulation, and

c) provide a condensing surface high in the primary system and under two phase conditions in the primary system.

These modifications have not affected the safety function of the steam generators or the Auxiliary Feedwater System and does not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-089

SYSTEM: Freeze Protection

COMPONENT: Controller Setpoints

CHANGE, TEST OR EXPERIMENT: The controller setpoints on freeze protection circuits 1Q, 3Q, 4Q, 5Q, 8Q, 13Q, 15Q, 20Q, 21Q, and 24Q were raised to 60°F and the setpoints on 9Q and 12Q were raised to 65°F. These circuits are associated with the Borated Water Storage Tank system. Work was completed October 12, 1983.

REASON FOR CHANGE: This change ensures that the cover portions of the piping is maintained at the minimum desired temperature.

SAFETY EVALUATION: This FCR has increased the reliability of the subject freeze protection circuits and as a result, it has increased the reliability of the Borated Water Storage Tank and associated equipment. No unreviewed safety question was involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-090

SYSTEM: Freeze Protection

COMPONENT: Borated Water Storage Tank (BWST)

CHANGE, TEST OR EXPERIMENT: The voltage on the transformers for freeze protection circuits 4Q, 8Q, and 12Q was raised to 60 volts. This was completed September 27, 1983.

REASON FOR CHANGE: These voltage increases have improved the freeze protection of this area by increasing heat input to the system.

SAFETY EVALUATION: This FCR has increased the reliability of the Freeze Protection System, as a result increasing the reliability of the BWST and associated equipment. No unreviewed safety question was involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-107

SYSTEM: Reactor Coolant System

COMPONENT: Hydraulic Snubbers

CHANGE, TEST OR EXPERIMENT: Shims on the supports of snubbers I/F-PSH-RC02B4-H1 and I/F-PT-RC2A1-H2 were shortened, and a shim was added to the support of snubber EBB-5-19-H10. Work was completed August 9, 1982.

REASON FOR CHANGE: This modification brought the snubbers cold piston settings to within the range specified in Bechtel drawing 12501-M-618. This is corrective action for Non-Conformance Reports 427-82, 428-82, and 432-82.

SAFETY EVALUATION: The safety function of these snubbers is to prevent piping overstresses during a dynamic loading event such as a safety shutdown earthquake. Structural evaluations of these changes were made by Bechtel and found to be satisfactory. An unresolved safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-155

SYSTEM: Containment Vessel

COMPONENT: Containment Air Sampling and Isolation Valves

CHANGE, TEST OR EXPERIMENT: The limit switches for motor operated valves CV5010A, CV5010B, CV5010C, CV5010D, CV5011A, CV5011B, CV5011C, and CV5011D were rewired. Work was completed September 23, 1983.

REASON FOR CHANGE: As originally installed, the torque switch breaks the circuit on closing of the previously mentioned ball valves. This modification allows the limit switch to break the circuit during closing of the valves. This change was specified by Torrey Pines Technology's report on limitorque operated valves.

SAFETY EVALUATION: The safety function of the limit switch is to stop the motor on predetermined valve position. The safety functions of the subject valves are to provide containment isolation on a Safety Features Actuation Level 1 and to provide a path for post accident containment air sampling.

Since this FCR involved wiring changes only, the limit switch still performs its intended safety function. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 83-029

SYSTEM: Auxiliary Building Non-Radioactive Heating & Ventilating System

COMPONENT: East Blowout Panel

CHANGE, TEST OR EXPERIMENT: A blind flanged through sleeve was added to the east blowout panel on the north wall of the Auxiliary Building at four feet above 585' on panel centerline. Work was completed August 11, 1983.

REASON FOR CHANGE: This has eliminated the need for blowout panel removal for eddy current inspection of the steam generators.

SAFETY EVALUATION: The safety function of the blowout panel is to provide a negative pressure boundary. This function has been enhanced by reducing the need to remove the blowout panel. This does not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 83-056

SYSTEM: Reactor Coolant System (RCS)

COMPONENT: Piping and Instrument Diagram (P&ID) M-030A

CHANGE, TEST OR EXPERIMENT: This FCR provided a Drawing Change Notice (DCN) for P&ID M-030A to add an instrumentation line to show input from hand switch RC3B to temperature differential transmitter RC6B, temperature relay transmitters RC7 and RC3, and to hand switch RC3. This was verified December 13, 1983.

REASON FOR CHANGE: The portion of the instrumentation line added by the DCN in this work package was inadvertently left off when redrawing P&ID M-030 into M-030A and M-030B.

SAFETY EVALUATION: This FCR was a drawing change only, hence it has not affected the safety function of any related equipment. Therefore, an unreviewed safety question is not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 83-106

SYSTEM: Reactor Coolant System (RCS)

COMPONENT: Pressurizer Spray Valve, RC2

CHANGE, TEST OR EXPERIMENT: The original 1.6 HP motor, 25 ft-lb torque, and RH class insulation were replaced with a 1.0 HP motor, 15 ft-lb torque, Dings brake and RH insulation on Pressurizer Spray Valve RC2. Work was completed September 16, 1982.

REASON FOR CHANGE: The 1.6 HP motor failed, and the valve design requires less than 1.0 HP to operate the valve. The addition of the brake is to prevent the valve from overtraveling its open position or relaxing from its closed position due to the inertia of the valve or system conditions.

SAFETY EVALUATION: The safety function of valve RC2 is to close so that a boron dilution path can be provided to the pressurizer from the auxiliary spray for a post loss of coolant accident condition. The replacement of original 1.6 HP motor with a 1.0 HP motor with brake was based on the original Babcock and Wilcox/Velan design with additional substantiation from the Torrey Pines Limitorque Motor Operated Valve Study. The work authorized by this FCR did not create any new adverse environments and did not constitute an unreviewed safety question.

TOLEDO EDISON

June 8, 1984

Log No. K84-689
File: RR 2 (P-6-84-05)

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, May, 1984
<u>Davis-Besse Nuclear Power Station Unit 1</u>

Enclosed are ten copies of the Monthly Operating Report for Davis-Besse Nuclear Power Station Unit 1 for the month of May, 1984.

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

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

IE24
1/1