AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346

UNIT Davis-Besse Unit 1

DATE August 9, 1984

COMPLETED BY Bilal Sarsour

TELEPHONE 419-259-5000, Extension 384

MONTH July, 1984

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	800	17	797
2	801	18	800
3	797	19	800
4	796	20	797
5	797	21	798
6	799	22	766
7	771	23	788
8	634	24	790
9	765	25	796
10	791	26	798
11	794	27	800
12	795	28	799
13	794	29	792
14	794	30	797
15	792	31	809
16	793		

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.

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OPERATING DATA REPORT

DOCKET NO. 50-346
DATE August 9, 1984
COMPLETED BY Bilal Sarsour
TELEPHONE 419-259-5000, Ext. 384

OPERATING STATUS

1. Unit Name: Davis-Besse Unit 1
2. Reporting Period: July, 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	5,111.0	52,632
12. Number Of Hours Reactor Was Critical	744	4,532.0	32,034.9
13. Reactor Reserve Shutdown Hours	0.0	134.8	4,014.1
14. Hours Generator On-Line	744	4,492.9	30,644.7
15. Unit Reserve Shutdown Hours	0.0	0.0	1,732.5
16. Gross Thermal Energy Generated (MWH)	1,907,358	11,563,897	72,607,711
17. Gross Electrical Energy Generated (MWH)	619,211	3,789,759	24,081,952
18. Net Electrical Energy Generated (MWH)	586,466	3,574,536	22,573,235
19. Unit Service Factor	100	87.9	58.2
20. Unit Availability Factor	100	87.9	61.5
21. Unit Capacity Factor (Using MDC Net)	90.2	80.0	49.1
22. Unit Capacity Factor (Using DER Net)	87.0	77.2	47.3
23. Unit Forced Outage Rate	0.0	12.1	17.7

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

REPORT MONTH July, 1984

DOCKET NO. 50-346
UNIT NAME Davis-Besse Unit 1
DATE August 9, 1984
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
			NO UNIT SHUTDOWNS OR SIGNIFICANT POWER REDUCTIONS						

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
JULY, 1984

Reactor power was maintained at approximately 94% power with the generator gross load at approximately 843 ±10 MWe until 1800 hours on July 7, 1984, when a manual power reduction to approximately 76% was initiated due to low load requirements. (Reactor power was limited to 94% due to an inoperable main steam safety valve.)

Reactor power was maintained at 76% power until 0300 hours on July 9, 1984, when reactor power was slowly increased and attained approximately 94% power at 0700 hours on July 9, 1984.

Reactor power was maintained at 94% power until 0010 hours on July 22, 1984, when a manual power reduction to approximately 85% was initiated due to low load requirements and to perform turbine valve testing.

After the completion of turbine valve testing, reactor power was slowly increased and attained approximately 94% at 1100 hours on July 22, 1984, and maintained at this power level until 0100 hours on July 29, 1984, when power was reduced to approximately 87% due to low load requirements and to perform control rod drive exercise testing.

After the completion of control rod drive exercise testing, reactor power was slowly increased and attained approximately 94% and maintained at this power level for the remainder of the month.

REFUELING INFORMATION DATE: July, 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: 77-451

SYSTEM: Auxiliary Feedwater

COMPONENT: Auxiliary Feed Pump

CHANGE, TEST OR EXPERIMENT: Portions of the pneumatic speed-setting mechanism were removed from the Auxiliary Feed Pump Turbine governor by the manufacturer, Woodward Governor Company. The governor pilot valve was reinstalled and tested October 19, 1977.

REASON FOR CHANGE: This modification permits the governor pilot valve to overtravel, allowing the pivot bearing to always remain in contact with the floating lever.

SAFETY EVALUATION: This change has lessened the chance of the governor bind under certain conditions, and has lessened the possibility that the turbine will be prevented from reaching design speed. This design improvement permits the Auxiliary Feed Pump to perform its intended safety function and has lessened a potential safety hazard.

COMPLETED FACILITY CHANGE REQUEST

FCR NO. 77-460

SYSTEM: Diesel Generator

COMPONENT: Fuel Oil Tank

CHANGE, TEST OR EXPERIMENT: The new Emergency Diesel Generator Fuel Oil System was isolated from the old non-nuclear safety related fuel oil system. This was accomplished by removing piping and electrical components from a point upstream of the emergency diesel generator day tanks to a point downstream of the diesel oil transfer pumps. Work was completed December 14, 1983.

REASON FOR CHANGE: This equipment is no longer required due to the new diesel oil fuel tanks.

SAFETY EVALUATION: The fuel oil system serves as a supply and storage for the emergency diesel generators. During construction, modifications were made to the fuel oil system to split that portion supplying the EDG's from the non-nuclear safety related portion supplying such components as the auxiliary boiler, miscellaneous diesel generator, second diesel fire pump. This was accomplished by blanking off the lines. This facility change now physically removes the piping and component. Removal of these old system components has not adversely affected the function of the fuel oil system, but has increased the factor of safety by removing the unused system piping and valves. This did not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 78-211

SYSTEM: Hydrogen Dilution

COMPONENT: Valves CV 117, CV 186, CV 187, CV 191

CHANGE, TEST OR EXPERIMENT: This FCR consisted of the replacement of check valves CV117, CV186, CV187, and CV191 with swing check type valves. Work was completed January 17, 1984.

REASON FOR CHANGE: The originally installed valves had a history of sticking closed. Also, there are no means to verify that the valves had closed once flow through the valve had stopped.

SAFETY EVALUATION: This change has improved the hydrogen dilution system reliability and did not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-082

SYSTEM: Demineralized Water

COMPONENT: Pipe Supports

CHANGE, TEST OR EXPERIMENT: Work implemented by this FCR was completed March 23, 1984. This involved modifications to pipe supports 10B-HCC-50-H10, 10B-HCD-4-H56, and 10B-HCC-50-H13 which are shown on isometrics 7749-M-210C, 210E, and 210F.

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

SAFETY EVALUATION: This modification has reduced the level of stress and has increased the factor of safety. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-083

SYSTEM: Circulating Water

COMPONENT: Pipe supports and anchors

CHANGE, TEST OR EXPERIMENT: This FCR provided for the modification of pipe hangers 12B-HBD-100-H1, H2A, H4, H8, H9, and 12B-HBD-28-H1A. Anchors 189 and 182 were also modified. All subject hangers and anchors are shown on isometric 7749-M-212B. Work was completed September 10, 1982.

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

SAFETY EVALUATION: These modifications have reduced the stresses of piping hangers in the Circulating Water System to an acceptable level and has increased the factor of safety. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-084

SYSTEM: Fire Protection

COMPONENT: Pipe hangers and anchors

CHANGE, TEST OR EXPERIMENT: This FCR was written to modify pipe hangers 16-KBE-2-H55, H235, H236, H237, H238, H245-A, H246-A, H255, H258, H259, H261, H263, H265, and H295 and anchors 359 and 287. All are shown on isometrics 7749-M-216C, 216D, and 216E. Work was completed November 8, 1983.

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

SAFETY EVALUATION: These modifications to pipe supports in the Fire Protection System have reduced the level of stress to an acceptable value and have increased the margin of safety. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-085

SYSTEM: Auxiliary Steam

COMPONENT: Pipe supports and anchors

CHANGE, TEST OR EXPERIMENT: Modifications were performed to pipe supports and anchors for the piping shown on isometric 7749-M-220D. Hangers 20-HBD-86-H2 and H3 were modified and anchor 498 was installed. Work was completed August 22, 1983.

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

SAFETY EVALUATION: The modifications to pipe supports in the Auxiliary Steam System have reduced stress levels to acceptable values and have increased the margin of safety. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-094

SYSTEM: Auxiliary Building Drains

COMPONENT: Pipe hangers and anchors

CHANGE, TEST OR EXPERIMENT: This FCR was implemented to modify pipe supports in the Auxiliary Building drain system. The anchors involved were AE-1, 2, 3, 27, 29, 31, 33; A-2, 10, 11, 12, 13, 17, 440, 1/DATA POINT (DP) 100, 4/DP77, 5/DP55, 6/DP130, 16/DP200, and 16/DP242. The hangers that were modified were 46-HSC-145-H36 and H37, R-1/DP68, 4/DP85, 5/DP45, 11/DP25, 12/DP45, 12/DP50, 13/DP75, 17/DP90, 23/DP150, 33/DP40, 33/DP45, 34/DP90, 34/DP95, 36/DP165, 44/DP97, 44/DP270, 50/DP185, 51/DP173, 53/DP135, 68, 68/DP415, 69/DP485, and 70/DP373. All subject hangers and anchors are shown on isometric 7749-M-246B. Work was completed November 28, 1983.

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

SAFETY EVALUATION: The modification of pipe hangers in the Auxiliary Building drain system has reduced the stress to an acceptable level and has increased the factor of safety. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-131

SYSTEM: Reactor Coolant System

COMPONENT: Pipe supports

CHANGE, TEST OR EXPERIMENT: Restraint R3 on the pressurizer spray line and pipe hangers 30-GCC-8-H8, 30-GCC-8-H14, 30-GCC-8-H15, 30-GCC-8-M17, 30-GCA-8-H15, FSK-M-CCA-3-H3, FSK-M-CCA-4-H1, FSK-M-CCA-7-1-A4, FSK-M-CCA-7-1-A5, FSK-M-CCA-7-1-A6, FSK-M-CCA-7-1-H11, FSK-M-CCA-7-1-H16, FSK-M-HCB-34-3-A1, FSK-M-HCB-34-3-A2, and FSK-M-HCB-34-2-A17 on the Reactor Coolant System line were all modified. Work was completed October 25, 1983.

REASON FOR CHANGE: These modifications were required as a result of reanalysis in accordance with I.E. Bulletin 79-02.

SAFETY EVALUATION: This FCR modified pipe supports for the Reactor Coolant System. This has reduced the level of stress and increased the factor of safety. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-018

SYSTEM: Concrete Masonry Block Walls

COMPONENT: Wall 3167, 3177, 3187, and Door 308B

CHANGE, TEST OR EXPERIMENT: Two levels of internal horizontal bracing were added to walls 3167, 3177, and 3187 forming a cable chase in Mechanical Penetration Room #4, Room 314, Elevation 585'. Door 308B was replaced with a seismically qualified, pressure resistant door. Work was completed August 15, 1983.

REASON FOR CHANGE: Reanalysis of block walls required by Nuclear Regulatory Commission I.E. Bulletin 80-11 had shown that during a seismic event walls 3167 and 3187 could have developed loadings that would have overstressed the floor beams attached above. A postulated rupture in the main feedwater line in Mechanical Penetration Room #4 would have overstressed walls 3167, 3177, and 3187 along with exceeding allowable limits on door 308B.

SAFETY EVALUATION: The subject walls function as fire walls to protect the cables in the cable chase from fire in Room 314. In their original condition, these walls could have failed to perform this safety function under postulated seismic or compartment pressurization loadings.

These modifications have lowered the induced stresses in the walls and floor beams to allowable limits specified in Section 3.8 of the Final Safety Analysis Report.

An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-020

SYSTEM: Concrete Masonry Block Walls

COMPONENT: Walls 4107, 4117, and 4127

CHANGE, TEST OR EXPERIMENT: Steel bracing was added to masonry block wall 4107, and walls 4117 and 4127 were removed. One and one-half inch thick steel plate barriers were installed in place of wall 4117 and in front of wall 4107. Conduits and conduit supports were relocated from walls 4117 and 4127. Work was completed December 21, 1983.

REASON FOR CHANGE: Reanalysis of block walls required by Nuclear Regulatory Commission I.E. Bulletin 80-11 had shown that following a main feedwater line break in corridor 404, walls 4107, 4117, and 4127 could become overstressed due to an increase in compartment pressure and jet impingement forces.

SAFETY EVALUATION: Wall 4107 is a fire wall and part of the negative pressure boundary on elevation 603'-00". Walls 4117 and 4127 supported non-safety related conduits and enclosed a portion of main feedwater pipe in corridor 404.

If wall 4107 had failed, the negative pressure boundary would be breached at elevation 603'-00", and the wall's ability to function as a fire wall would be affected. Wall 4117 could have damaged wall 4107 following a postulated pipe break. Removal of wall 4127 had no effect on any safety related system.

This modification has lowered stresses in wall 4107 to the allowable limit specified in Section 3.8 of the Final Safety Analysis Report. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-028

SYSTEM: Concrete Masonry Block Walls

COMPONENT: Walls 4796, 4886, and 4896

CHANGE, TEST OR EXPERIMENT: Steel beams were used to brace the top connections on each side of wall 4796, 4886, and 4896. Also, steel angles and expansion anchors were used to brace the bottom connection and a section of the east edge connection on both sides of wall 4886. These walls form one wall unit which separates the Low Voltage Switchgear Room No. 428 from No. 2 Electrical Penetration Room No. 427 on Elevation 603'. Work was completed November 8, 1982.

REASON FOR CHANGE: Block wall reanalysis required by Nuclear Regulatory Commission I.E. Bulletin 80-11 had shown that during a seismic event the top connections of walls 4796, 4886, and 4896, and the lower east edge connection of wall 4886 could have 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. If a failure of the connections were to occur, fire protection provided by these walls could possibly be lost. Potential damage to the attached safety related conduits and electrical equipment in the rooms adjacent to the wall could also occur.

This modification has lowered the possible stresses in these walls to within allowable limits established in Section 3.8 of the Final Safety Analysis Report.

An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-184

SYSTEM: Reactor Coolant System (RCS)

COMPONENT: Reactor Coolant Pump Seals

CHANGE, TEST OR EXPERIMENT: The reactor coolant pump seals were modified to increase the staging flow from 1.0 to 1.5 gpm, provide a flow staging path from the first to the second stage seal cavity, install slotted carbide faces, and install modified v-cups. Work was completed October 14, 1983.

REASON FOR CHANGE: The increase in staging flow allows a higher variation in seal leakage without changing the seal cavity pressure, decreasing the tendency of the seals to sustain oscillations. This should promote longer seal life.

The staging flowpath directly from the first to the second seal cavity has increased the operating temperature in the second seal cavity and decoupled the second seal cavity from fluctuations in the component cooling water temperature.

The modification of the rotating carbide face rings has reduced the overall hydraulic pressure on the seal faces.

The v-cup performs as a secondary seal and has been modified to reduce the axial load on the rotating seal caused by friction between the shaft and the v-cup.

SAFETY EVALUATION: The reactor coolant pump seals prevent uncontrolled leakage of reactor coolant from the pump shaft area.

The increase in flow was achieved by shortening the pressure breakdown tubes and thereby decreasing their flow resistance. Adequate margin exists in the design of the reactor coolant pump seal heat exchanger to accommodate this increase in staging flow.

The modification of the rotating carbide face rings forces fluid between the faces in the areas between the slots, and the overall force that separates the faces is increased, developing greater resistance of the faces to defect.

These changes were all recommended by Babcock & Wilcox and the pump and seal vendor, Byron Jackson, as part of the package to enhance seal performance.

Pursuant to the above, an unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-091

SYSTEM: Borated Water Storage Tank (BWST)

COMPONENT: Freeze Protection

CHANGE, TEST OR EXPERIMENT: Exposed valve bonnets and yoke assemblies on freeze protection circuits 3Q, 7Q, 11Q, and 16Q were insulated with foam glass insulation. Work was completed December 14, 1983.

REASON FOR CHANGE: A large amount of heat was being lost through the valve bonnets. The installation of foam glass insulation was recommended by Energy Consultants, Inc., in their report on freeze protection systems.

SAFETY EVALUATION: The safety function of the BWST is to provide a borated water source for the Emergency Core Cooling System. This FCR has increased the reliability of the freeze protection and as a result, increases the reliability of the BWST.

An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-092

SYSTEM: Freeze Protection

COMPONENT: Various circuits

CHANGE, TEST OR EXPERIMENT: The calcium silicate insulation on freeze protection circuits 1Q, 3Q, 4Q, 5Q, 7Q, 8Q, 9Q, 11Q, 12Q, 13Q, 15Q, 16Q, 17Q, 20Q, 21Q, 24Q, 86Q, and 87Q, was replaced with foam glass insulation. These circuits are associated with the Borated Water Storage System and the Control Room Chlorine Detector System. Work was completed January 23, 1984.

REASON FOR CHANGE: Wet insulation, a significant source of failure on freeze protection circuits, was found in several locations. The replacement of the calcium silicate insulation, a highly absorbent wicking material, with foam glass insulation, rated at 0% moisture absorption, has eliminated this problem.

SAFETY EVALUATION: The safety function of the borated water source for the Emergency Core Cooling System and the safety function of the Control Room Chlorine Detection System are to isolate the Control Room Ventilation System in case of chlorine detection. This change has increased the reliability of the freeze protection system, and as a result has increased the reliability of the Borated Water Storage Tank and Control Room Chlorine Detection Systems. No unreviewed safety question was involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 83-074

SYSTEM: Auxiliary Feedwater

COMPONENT: Header

CHANGE, TEST OR EXPERIMENT: The auxiliary feedwater header riser flanges were removed and replaced. Also, material samples were taken from the header straight tees, MK-571, and end caps, MK-573. This was completed September 8, 1983.

REASON FOR CHANGE: The auxiliary feedwater header riser had failed materials physical properties tests and, therefore, had to be completely removed and replaced. The material samples were to be chemically tested to determine acceptability.

SAFETY EVALUATION: The safety function of the auxiliary feedwater header riser flanges, tees and end caps is to maintain the Auxiliary Feedwater System's pressure boundary integrity in the vicinity of the steam generators. The removal of material for compositional analysis from the tees and end caps did not exceed the allowable limits specified in the ASME code for minimum wall thickness, and, therefore, allowed the tees and end caps to perform their safety function.

This FCR has not degraded the Auxiliary Feedwater System or components from performing their safety function.

Based on the above, an unreviewed safety question was not involved.

TOLEDO EDISON

August 9, 1984

Log No. K84-1075
File: RR 2 (P-6-84-07)

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 1984
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 1984.

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

Yours truly,

Stephen M Quennoz

Stephen M. Quennoz
Assistant Station Superintendent
Davis-Besse Nuclear Power Station

SMQ/BMS/bec

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

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