AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346

UNIT Davis-Besse #1

DATE August 9, 1982

COMPLETED BY Bilal Sarsour

TELEPHONE (419) 259-5000 ext. 384

MONTH July, 1982

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

INSTRUCTIONS

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

(9/77)

OPERATING DATA REPORT

DOCKET NO. 50-346
DATE Aug. 9, 1982
COMPLETED BY Bilal Sarsour
TELEPHONE (419) 259-5000 ext. 384

OPERATING STATUS

1. Unit Name: Davis-Besse #1
2. Reporting Period: July, 1982
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	5,087	35,088
12. Number Of Hours Reactor Was Critical	0	1,708	17,938
13. Reactor Reserve Shutdown Hours	0	0	3,334.7
14. Hours Generator On-Line	0	1,707.4	16,957.6
15. Unit Reserve Shutdown Hours	0	0	1,731.4
16. Gross Thermal Energy Generated (MWH)	0	3,641,078	38,762,603
17. Gross Electrical Energy Generated (MWH)	0	1,202,294	12,884,545
18. Net Electrical Energy Generated (MWH)	0	1,124,093	12,021,378
19. Unit Service Factor	0	33.6	48.3
20. Unit Availability Factor	0	33.6	53.3
21. Unit Capacity Factor (Using MDC Net)	0	25.3	39.2
22. Unit Capacity Factor (Using DER Net)	0	24.4	37.8
23. Unit Forced Outage Rate	0	0	23.0

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: August 15, 1982
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 #1
DATE August 9, 1982
COMPLETED BY
TELEPHONE

REPORT MONTH July, 1982

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
4	82 03 13	S	744	A	4	NA	NA	NA	Unit outage which began on March 13, 1982 was still in progress through the end of July, 1982 with the auxiliary feedwater header repair underway. (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
6-Other

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)

REFUELING INFORMATION

DATE: August 9, 1982

1. Name of facility: Davis-Besse Unit 1

2. Scheduled date for next refueling shutdown: (In Progress)

3. Scheduled date for restart following refueling: August 15, 1982

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)?

   The final reload analysis for Cycle 3 has been submitted and approved by the NRC latest letter. This analysis identifies several technical specification changes relating to core operational limits and reactor protection system setpoints. An option to provide flexibility in the overall cycle length is also provided therein.

5. Scheduled date(s) for submitting proposed licensing action and supporting information. See response to No. 4 above

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.

   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) 92 - 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 1988 - assuming ability to unload the entire core into the spent fuel pool is maintained.

OPERATIONAL SUMMARY
JULY, 1982

7/1/82 - 7/31/82 The unit outage which began on March 13, 1982 was still in progress through the end of July, 1982 with the auxiliary feedwater header repair underway.

The following are the more significant outage activities performed during the month of July:

1. Auxiliary feedwater header:

   a) Completed positioning of both external headers.

   b) Completed welding of external header and risers.

   c) Completed shop fabrication of OTSG 1-1 process piping.

   d) Continued auxiliary feedwater piping and header modifications.

2. Turbine:

   a) Completed turbine alignment.

   b) Completed repair of leaks and retesting of main generator for gas leakage.

   c) Completed overspeed trip test of 1-1 and 1-2 main feedwater turbine.

   d) Completed the rework and readjustment of the main turbine thrust bearing.

3. Completed plugging of seven steam generator tubes.

4. Completed installation of reactor coolant pump seals.

5. Completed installation of pressurizer code safety valves.

1 6. Completed modification and testing of high pressure injection stop check valves HP48, 56 and 57.

7. Internal repairs were completed on both main steam isolation valves and the valves are currently being reassembled.

BMS/lmr

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 78-217

SYSTEM: Reactor

COMPONENT: Fuel Assemblies, Burnable Poison Rod Assemblies, Control Rods

CHANGE, TEST OR EXPERIMENT:

1. Remove all 68 BPRA's from the fuel assemblies that are in the reactor.

2. Install storage racks in spent fuel pool area for the burnable poison rod assemblies (BPRA's).

3. Interchange the following assemblies: N12 with H9, N4 with K8, D4 with H7, and D12 with G8.

4. Fully insert control rods F6, F10, L6, L10, H2, H14, B8, & P8 for a period of time after restart.

5. Modify control rod patching scheme in accordance with guidance from B&W.

6. Perform physics tests on new core configuration in accordance with guidance from B&W.

REASON FOR CHANGE:

The removal of the BPRA's and orifice rods at Davis-Besse Unit 1 will prevent the occurrence of a severe problem like the one at Crystal River, where two BPRA's left the fuel assemblies and caused considerable damage. Also, the other items given above will allow operation at 100% power with reasonable core operating conditions. In particular, acceptable conditions will exist for peak power and moderator temperature coefficient.

SAFETY EVALUATION:

This FCR proposes to:

1. Remove all 68 BPRA's from the fuel assemblies that are in the reactor.

2. Install storage racks in the spent fuel pool area for burnable poison rod assemblies (BPRA's).

3. Interchange fuel assemblies (N12 with H9, N4 with K8, D4 with H7, and D12 with G8).

4. Fully insert control rods (F6, F10, L6, L10, H2, H14, B8 and P8) for a period of time after restart.

5. Modify control rod patching scheme in accordance with guidance from B&W.

6. Perform physics tests on new core configuration in accordance with guidance from B&W.

The safety function of BPRA's is to provide burnable poison into the core to facilitate better control of core operations and to provide a negative beginning or life moderator temperature coefficient. The safety function of the modified ORA's is to provide a primary neutron source capturing arrangement. The safety function of the ORA's is to limit bypass flow through fuel assemblies with empty guide tubes. The safety evaluation for the above changes is contained in BAW-1489 (Revision 1, 5/26/78). As detailed and concluded in the referenced report, the above mentioned Safety Functions are not adversely affected by making these changes. Pursuant to the above, the changes proposed by this FCR do not involve an unreviewed safety question.

ms d/5

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 81-247

SYSTEM: Freeze Protection

COMPONENT: Various

CHANGE, TEST OR EXPERIMENT:

This FCR provides for an increase in the output voltage for the freeze protection circuits identified in the Thermal Manufacturing Company letter.

REASON FOR CHANGE:

This change will improve the Freeze Protection Systems by ensuring maximum output of the heater circuits, thereby reducing low alarms or freezing of lines during winter conditions.

SAFETY EVALUATION:

This FCR provides for changes to the Freeze Protection Systems. The change involves an increase of voltages to heat tracing circuits, which will increase the output of the heater circuits.

The increase of voltage will not adversely effect the function of the Freeze Protection System.

It will enhance the Freeze Protection Systems by providing an increase of output for the freeze protection circuits. It will improve system reliability by ensuring maximum heating, thereby reducing the chances for low alarms or freezing of lines.

This change will not compromise the integrity of the existing heat tracing system and will not prevent the safe shutdown of the station.

Thereby, the work authorized by this FCR does not create any new adverse environments and does not constitute an unreviewed safety question.

ms e/10

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 82-006

SYSTEM: BWST, Emergency Core Cooling System

COMPONENT: BWST

CHANGE, TEST OR EXPERIMENT:

1. This FCR is to document the safety evaluation for operation of Davis-Besse in light of a BWST water temperature increase to 120°F (beyond the 90°F value assumed in the FSAR) in conjunction with a power reduction to 90% during the period of January 13-14, 1982.

2. This FCR also documents the safety evaluation for operation of Davis-Besse at RATED THERMAL POWER for a period from January 12, 1982 to January 14, 1982 with a BWST water temperature of 120°F.

   The attached safety evaluation is not intended to be used for BWST water temperature excursions for dates other than listed above.

REASON FOR CHANGE:

To provide documentation for justification of power operation at 90% and at rated thermal power with a BWST water temperature up to approximately 120°F.

SAFETY EVALUATION:

This safety evaluation is performed to document the justification for continued power operation in light of the increase in Borated Water Storage Tank (BWST) water temperature beyond the value assumed in the accident analyses of Davis-Besse Nuclear Power Station Unit 1 (DB-1) Final Safety Analysis Report (FSAR). The safety function of the BWST is to provide a supply of borated water for injection by the ECCS in the event of a LOCA. Specifically, this safety evaluation addresses the following two modes of operation at Davis-Besse Nuclear Power Station:

1. Operation at 90% rated thermal power with a BWST water temperature of 120°F for the period of January 13 through January 14, 1982. Section I below and Attachment 1 hereto constitute the safety evaluation for this mode of operation.

2. Operation at rated thermal power with a BWST water temperature of 120°F for the period of January 12 through January 14, 1982. Section II below and Attachment 2 hereto provide the safety evaluation for this mode of Operation.

Background

On Wednesday, January 13, 1982, Nuclear Engineering was advised by the Station personnel that BWST water temperatures as high as 120°F had been

experienced on January 12 and 13, 1982. The high temperature was caused by an attempt to thaw out the frozen BWST level transmitter sensing lines. Following an investigation into the FSAR accident analyses, it was determined that the Emergency Core Cooling System (ECCS) analysis assumes a BWST water temperature of 90°F.

I. Operation at 90% Power with a BWST Water Temperature of 120°F

Subsequent to the preceding identification, Nuclear Engineering contacted the B&W ECCS analysis personnel and discussed possible resolution of the elevated BWST water temperatures. B&W indicated that the most significant contribution to core heat removal will be through the high pressure injection (HPI) pumps. For the elevated BWST water temperature condition, the low pressure injection (LPI) system is of lesser significance since it is primarily used to maintain water level above the core and the excess water inventory is discharged through the break to the containment emergency sump. As a result, effects of high BWST water temperature were considered only for the HPI pumps at first. In accordance with B&W discussions, Nuclear Engineering performed a calculation to determine the maximum safe level of power operation at which a BWST water temperature of 120°F will provide adequate emergency core cooling capability. This is described in the following paragraphs.

A. Reactor Coolant System Heat Removal

If the BWST water temperature is raised beyond 90°F, the total heat removal capacity of the ECCS is correspondingly reduced. To quantify this effect, enthalpy of water at 90° (h_{90} = 58.06 Btu/lbm) and 120°F (h_{120} = 88.00 Btu/lbm) was determined. It was then assumed that following a LOCA, the Reactor Coolant System (RCS) reaches saturation conditions at 1065 psia (552°F). This corresponds to the steam generator secondary side saturation conditions and assumes that the RCS cold leg temperature follows the steam generator saturation temperature. The enthalpy of water (reactor coolant) at this temperature (h_{sat}) was determined to be 552.4 Btu/lbm. To determine the relative change in the heat removal capability a ratio R was calculated as follows:

$$R = \frac{h_{sat} - h_{120}}{h_{sat} - h_{90}} = \frac{552.4 - 88}{552.4 - 58} = 0.939$$

This implies that the 120°F BWST water is capable of removing approximately 94% of that decay heat which is produced by operating at 100% rated thermal power. It is noted that the decay heat produced in the core is proportional to the steady state level of power operation. Based on prudent engineering judgment, Nuclear Engineering immediately instructed Station management to conservatively limit power level to 90% of rated thermal power until such time that BWST water temperature is restored to within 90°F. The above calculation possesses additional conservatisms as indicated below:

1. Latent heat of evaporation leading to formation of steam in the RCS has been neglected.

2. It is assumed that all the heat transfer is through the BWST water. No credit has been taken for the cooling via steam discharge through the postulated break.

3. No credit has been taken for heat removal through the steam generator.

Subsequent to the above limitation on power level, Nuclear Engineering further investigated the ECCS pump net positive suction head requirements and the containment energy removal effects. This is summarized in the following paragraphs. Nuclear Engineering also requested B&W to further investigate this matter in light of DB-1 ECCS analysis. The results of this investigation are summarized in Attachment 1.

B. Net Positive Suction Head Considerations

Another effect of the increase in BWST water temperature is to degrade the available net positive suction head (NPSH) for the ECCS pumps. For the low pressure injection and containment spray pumps, operation without cavitation at higher temperature has been demonstrated acceptable in FSAR (Section 6.3.2.14) when taking suction from the containment emergency sump. This bounds the 120°F suction water temperature from the BWST. In addition, calculations performed for a 30° increase in water temperature (90° to 120°F) indicate that this increase results in a loss of approximately 2.5' of available NPSH. This is acceptable in light of the ample margin existing between the required and available NPSH (See FSAR Section 6.3.2.14).

C. Effects on Containment Vessel Integrity

With an increase in BWST water temperature, the adverse impact on the containment vessel integrity is caused by two factors:

1. Containment energy removal effects owing to the higher temperature containment spray water.

2. Increase in blow down energy released to the containment through the break. This is based on the fact that the higher temperature ECCS water will eventually be released to the containment resulting in higher energy releases than those assumed for 90°F BWST water.

There is no effect of the higher temperature containment spray water on the containment temperature and pressure peak since the peak is reached before the containment sprays have an effect.

The containment energy release from the reactor coolant system, however, affects the containment pressure peak. Per the attached

B&W analysis (see Section IV of B&W analysis, Attachment 1) with operation at 90% power, the net effect of added energy to containment spray and reduced energy release from the break (due
1 to reduction in operating power) is a decrease in containment pressure and temperature response. Also, the peak containment temperature and pressure are lower than those provided in the FSAR. Based on the above, it is concluded that containment vessel integrity will not be adversely affected during a LOCA by operation at 90% power with a BWST water temperature of 120°F.

II. Operation at 100% Power with BWST Water Temperature of 120°F

Per the request of Toledo Edison, B&W performed an additional analysis to evaluate 100% full power operation in light of a BWST water temperature of 120°F. This analysis is included in Attachment 2 hereto, and is based on the worst case kw/ft actually observed during the operation between 1/12/82 to 1/19/82.

A. Net Positive Suction Head Considerations

The results of Section I.B remain valid for this case.

B. Mass and Energy Release to Containment

As described in Section I.C above, the higher temperature containment spray water does not have an adverse effect on the post-accident containment temperature and pressure peaks. However, analysis performed by B&W (see Section IV, Attachment 2) indicate an increase in the containment energy content by 1.7%. Most conservatively, this will raise the peak containment pressure from 36.95 psig to a value less than 37.55 psig (increase of less than 0.6 psi) which provides adequate margin from the maximum design pressure of 40 psig. Based on the above, it is our judgment that the containment vessel integrity would not have been adversely affected by operating at 100% power with a BWST water temperature of 120°F.

It is therefore concluded that with 120°F BWST water, 100% power and actual observed lower peaking conditions, the existing large break LOCA analyses bound the peak clad temperature. For small breaks, a decreased reactor vessel inventory will result. However, substantial margin to 10CFR 50.46 peak clad temperature would have been retained.

It is emphasized that the above determinations possess additional conservatism in that:

1. ECCS analysis requires only one train of HPI, LPI and containment spray pump to be operable. During the time period of interest above, both trains of ECCS were operable. This substantially augments the containment and core energy removal capability available during this time.

2. The ECCS analysis assumes that the core produces 1.2 times the ANS decay heat, whereas in reality the twenty percent excess decay heat is not produced.

Pursuant to the above it is concluded that:

1. The increase in BWST water temperature to 120°F with a corresponding decrease in rated thermal power to 90% (for the period of January 13-14, 1982) did not degrade the safety function of the BWST and the emergency core cooling capability of the ECCS is not compromised.

2. The consequences of operating at 100% rated thermal power with a BWST water temperature of 120°F (for the period of January 12-14, 1982) were bounded by the existing LOCA analyses in light of the low local operating peaks that were observed when the plant was in this configuration. Also, the containment integrity considerations are not significantly altered.

Subsequently, the two modes of operation described above do not involve an unreviewed safety question.

bj c/3(3-7)

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 80-217

SYSTEM: Instrument AC Power

COMPONENT: 120V AC Panels (Y1, Y2, Y3, and Y4)

CHANGE, TEST OR EXPERIMENT:

This FCR involves revising the fuse sizes between the inverter and the 120V AC distribution panels (Y1, Y2, Y3, and Y4). Also furnish information and recommendations on non-critical loads which do not warrant an uninterruptible source of power.

REASON FOR CHANGE:

The fuse sizes in Y106, Y206, Y308, and Y408 were changed from 10 amps to 15 amps. The fuse sizes in all the circuits have been reduced to values as low as practicable, also the fuses have been changed to an extremely fast acting type.

The bill of materials has been revised also, to reflect these changes and to include the requirement for fuse reducers that are needed to accommodate the smaller fuses than intended for the 60 amp fuse holders.

SAFETY EVALUATION:

This FCR involves revising fuse sizes within the 120V AC distribution panels (Y1, Y2, Y3, and Y4).

This change will not adversely effect the function of the distribution panels. It will enhance system operation by providing better coordination between the inverter and the branch circuit fuses.

All modifications are internal to the existing 120V AC distribution panels which has been reviewed by the Engineering Inspection Team (EIT).

Therefore no new adverse environmental conditions will be created by these changes. This is not an unreviewed safety question.

bj c/3(2)

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 79-201

SYSTEM: Q Motors

COMPONENT: Project Drawings

CHANGE, TEST OR EXPERIMENT:

This FCR incorporates the revision of project and vendor drawings to agree with existing and corrected Q motor nameplates per Table 1 of BT-8746.

REASON FOR CHANGE:

The design documents and existing equipment must agree with each other. This FCR is being written to preclude further problems with Q motor nameplates and design documents.

SAFETY EVALUATION:

Discrepancies were found to exist between installed Q motors and design documents. These 37 discrepancies have been evaluated as to proper motor size for the given application. In 36 cases the installed motor is correct with either the design documents or motor nameplates being incorrect. This FCR is only a change to design documents and motor nameplates and has no effect on Safety Grade Equiment.

This is not an unreviewed safety question.

bj e/2(5)

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 80-074

SYSTEM: Auxiliary Feedwater

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

CHANGE, TEST OR EXPERIMENT:

Prior to the implementation of Facility Change Request 80-074, low pressure in the suction of auxiliary feedwater pumps (AFP) 1 or 2 would automatically close the steam inlet valves to auxiliary feedwater pump turbines (AFPT) 1 or 2 without any time delay. This FCR added a 2.5 ± .12 second time delay to this circuit to prevent spurious operation. In addition the low pressure switches were reconnected as follows:

PROPOSED CHANGE:

SWITCH NUMBER	CLOSES VALVE
PSL-4930A	HV-106
PSL-4930B	HV-106A
PSL-4931A	HV-107
PSL-4931B	HV-107A

REASON FOR CHANGE:

The steam inlet valves to the auxiliary feedwater pump turbine 1 or 2 have previously spuriously closed due to low pressure dips in the auxiliary feedwater pump suction piping. The addition of this time delay in this control circuit will prevent further spurious closures. Each pressure switch was reconnected to a single valve, so that a power supply failure to valves HV-106 and HV-107 will not prevent the opening of valves HV-106A and HV-107A.

SAFETY EVALUATION:

This FCR involved two changes in the auxiliary feedwater system.

The first change entailed incorporating a time delay in automatically closing the steam inlet valves to AFPT 1 or 2 (HV-106 and HV-106A or HV-107 and HV-107A, respectively) in case of the pressure in the suction of AFP 1 or 2 detected by PSL-4930A and PSL-4930B or PSL-4931A and PSL-4931B is found to be less than 1 psig.

In the present system, if the AFP suction pressure falls below 1 psig as detected by the four pressure switches, the steam inlet valves receive automatic and instantaneous signals to close to protect the pump from loss of suction. This also could occur due to low pressure dips in the AFP suction pressure during turbine acceleration, thereby causing spurious closure of these valves.

This change will prevent this by providing a 2.5 ± .12 second time delay. The response time of starting the auxiliary feedwater system will not be changed due to this time delay. In addition, the probability of false trips of the steam inlet valves will be reduced. The time delay will not cause any damage to the AFP because it is required to be less than the time needed to damage the AFP.

The second change involves reconnecting the existing four pressure switches, PSL-4930A, 4930B, 4931A and 4931B, which detect the low pressure in the suction of AFP 1 or 2. The change requires that one switch is connected to each of the four steam inlet valves to the AFPT 1 or 2.

Prior to the implementation of this FCR, pressure switch pairs PSL-4930A and PSL-4930B, and PSL-4931A and PSL-4931B, were interlocked and received their control power from power supply of valve HV-106 or HV-107. If there were a power supply failure of HV-106 or HV-107, this would not have permitted opening the crossover steam inlet valves HV-106A or HV-107A and thus have prevented steam flow from the other steam generator. The second change will provide more diversification by way of providing separate control powers for the two steam inlet valves for each AFPT. Therefore a single power supply failure of one valve will not prevent the other valve to open.

In addition, it should be noted that in the present system, if the suction pressure to AFP falls below 2 psig, the suction is transferred from condensate storage tank to the service water system. Thus, in case a pressure switch fails and in turn fails to close the applicable steam inlet valve, the AFP will get an alternate water supply and thus not allow damage to the pump.

Pursuant to the above, no unreviewed safety question is involved.

bj e/2(3-4)

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 80-174

SYSTEM: Component Cooling Water

COMPONENT: CC-283

CHANGE, TEST OR EXPERIMENT:

Under Facility Change Request 80-174, the canopy weld on valve CC-283 was replaced with a fillet type body-bonnet seal weld. Rockwell valve Drawing #P-473665 was revised to show the alternate method of sealing the valve. The work was completed July 29, 1980.

REASON FOR CHANGE:

In order to remove the cover for valve repair, the canopy weld had to be ground out. Tolerances for rewelding the canopy could not be met to allow reassembly via a canopy weld.

SAFETY EVALUATION:

This FCR involves changing drawings to allow installation of a valve with a fillet type body-bonnet seal weld for valve CC-283. The safety function of the seal weld is to maintain pressure boundary only and can be accomplished by use of either a canopy or fillet weld. This change does not affect the safety function of the valve. This valve modification has the concurrence of the valve manufacturer, Rockwell International. An unreviewed safety question does not exist.

bj e/2(2)

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 80-060

SYSTEM: Main Steam

COMPONENT: MS101, MS100

CHANGE, TEST OR EXPERIMENT:

Additional position indication lights for MS100 and MS101 were added to panel C5717, per Facility Change Request 80-060 in a location that could be easily seen. The work was completed January 26, 1981.

REASON FOR CHANGE:

During several plant trips in the past, the main steam isolation valves (MSIV) have closed and the operators were not rapidly alerted of the event because of the location of the original position indication lights. To improve the operators knowledge of plant status, MSIV position is now
1 prominently displayed on panel C5717.

SAFETY EVALUATION:

The addition of new indicating lights for the Main Steam Isolation Valves will improve the operators' knowledge of the MSIVs position in an area of the panel most beneficial to the operator. All changes are within the cabinet, therefore, an unreviewed safety question does not exist.

bj e/2(1)

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 80-256

SYSTEM: Containment Spray (L.P. & H.P. Injection)

COMPONENT: N/A

CHANGE, TEST OR EXPERIMENT:

This FCR involves the revising of Drawing 7749-C-849, Section E, to reflect the as-built condition.

REASON FOR CHANGE:

In compliance with NCR #70-79, revise Drawing 7749-C-849, Section E, to agree with the as-built condition.

SAFETY EVALUATION:

This FCR revises Drawing 7749-C-849 to show the "as-built" configuration of a "Q" pipe anchor located in the Auxiliary Building. This structural component acts as a seismic restraint for the following piping.

Pipe	Anchor No.	System
4" - CCB-2	A-55	H.P. Injection
10" - GCB-10	A-61	L.P. Injection
8" - GCB-5	A-79	CTMT Spray
8" - HCC-38	A-80	CTMT Spray
8" - HCB-9	A-83	CTMT Spray

This piping is part of the Emergency Core Cooling System (L.P. and H.P. Injection) and the emergency heat removal system (CTMT spray).

Under LOCA conditions, the ECCS supplies cooling water to the core while the CTMT spray removes heat from the CTMT building atmosphere thereby decreasing the pressure.

bj c/3(8)

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 80-182

SYSTEM: Emergency Diesel Generator

COMPONENT: Relay CR-3X

CHANGE, TEST OR EXPERIMENT:

Facility Change Request 80-182 was implemented to eliminate the Emergency Diesel Generator (EDG) voltage frequency relay CR-3X contact 46-47 (48-49) and the EDG 4.16KV breaker auxiliary 5251 (2)a contact from the SFAS sequencer starting logic. The work was completed April 22, 1981.

REASON FOR CHANGE:

With the EDG breaker closed on the bus, the CR-3X contact referred above closes whenever there is a diesel undervoltage and/or underfrequency condition. This energizes the K6 relay and stops the sequencer if it is running.

These CR-3X contacts have failed frequently, inhibiting sequencer response.

SAFETY EVALUATION:

This FCR calls for removing the Emergency Diesel Generator (EDG) voltage frequency relay CR-3X contacts 46-47 (48-49) and the EDG 4.16KV breaker auxiliary contact 5251 (2)a 19-20 (9-10) from the SFAS sequencer start/stop logic (See E64B, sheet 17).

The CR-3X relay contacts above and the breaker contacts are in series and provide a path to energize/deenergize the K6 relay. The K6 relay is to be energized (with coincident SFAS actuation) to turn the sequencer "ON". The K6 relay is then to be deenergized to start the sequencer running.

The safety function of the CR-3X contacts referred above is to (re)energize the K6 relay on EDG undervoltage/underfrequency. Thus, if the sequencer is running, after SFAS actuation and loss of offsite power (K6 is deenergized) and an EDG undervoltage/underfrequency condition occurs, the CR-3X contacts 46-47 (48-49) will close. Since the 5251a 19-20 (9-10) contact is already closed because the EDG breaker is closed on the bus, K6 gets reenergized and the sequencer stops running. The safety function achieved by the 5251a contacts is to provide for this path for reenergization of the K6 relay when the breaker is closed on the bus. If the breaker is not closed on the bus, this path for reenergiziation of K6 relay is not available. Both these contacts thus provide for EDG undervoltage/underfrequency protection. This protection, however, is only a backup protection. The Davis-Besse diesel generators are capable of starting the largest load while carrying all other loads and capable of withstanding total load rejection without exceeding speeds (frequency) or voltages. In addition, the engine is equipped with an automatic speed governor and a motor operated speed changer device to provide for underfrequency protection.

Thus the protection provided by the above mentioned CR-3X contacts is only a backup to already existing other means.

The reliability history of these CR-3X contacts at Davis-Besse has been very poor and their failure has resulted in undesirable sequencer failures, instead of enhancing the emergency DG reliability. In addition, the operating experience with the Davis-Besse EDGs has shown that with the EDG capability at Davis-Besse and the undervoltage/underfrequency setpoint of CR-3X, these contacts have served no useful purpose. Also, as noted above, since these contacts are only for backup protection, no commitment has been made in the Davis-Besse FSAR to have the CR-3X contacts in the sequencer start/stop circuit. The advantages of removing these contacts from this circuit outweigh their presence.

Pursuant to the above, it is concluded that the changes proposed by this FCR do not involve an unreviewed safety question.

bj e/3(2-3)