OPERATING DATA REPORT

DOCKET NO. 50-346
DATE September 7, 1979
COMPLETED BY Erdal Caba
TELEPHONE 419-259-5000, Ext. 236

Notes

POOR ORIGINAL

OPERATING STATUS

1. Unit Name: Davis-Besse Unit 1
2. Reporting Period: August, 1979
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): to be determined
7. Maximum Dependable Capacity (Net MWe): to be determined
8. If Changes Occur in Capacity Ratings (Items Number 3 Through 7) Since Last Report, Give Reasons:

9. Power Level To Which Restricted, If Any (Net MWe): None
10. Reasons For Restrictions, If Any:

	This Month	Yr.-to-Date	Cumulative
11. Hours In Reporting Period	744	5,831	17,596
12. Number Of Hours Reactor Was Critical	744	2,989.4	9,621.2
13. Reactor Reserve Shutdown Hours	0	1,858.2	2,648.5
14. Hours Generator On-Line	744	2,898.9	8,632.1
15. Unit Reserve Shutdown Hours	0	1,728.2	1,728.2
16. Gross Thermal Energy Generated (MWH)	2,022,114	7,131,662	17,319,232
17. Gross Electrical Energy Generated (MWH)	674,179	2,377,397	5,761,152
18. Net Electrical Energy Generated (MWH)	641,855	2,236,227	5,277,687
19. Unit Service Factor	100	49.7	50.8
20. Unit Availability Factor	100	79.4	61.9
21. Unit Capacity Factor (Using MDC Net)	to be determined		
22. Unit Capacity Factor (Using DER Net)	95.2	42.3	36.7
23. Unit Forced Outage Rate	0	2.8	19.6

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

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

	Forecast	Achieved
INITIAL CRITICALITY		
INITIAL ELECTRICITY		
COMMERCIAL OPERATION		

336149 (9/77)

7909130 438.

AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346
UNIT Davis-Besse Unit 1
DATE September 7, 1979
COMPLETED BY Erdal Caba
TELEPHONE 419-259-5000, Ext. 236

MONTH August, 1979

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	874	17	883
2	877	18	877
3	879	19	881
4	878	20	878
5	876	21	874
6	863	22	857
7	671	23	870
8	677	24	873
9	875	25	876
10	872	26	878
11	881	27	877
12	882	28	868
13	880	29	880
14	877	30	861
15	884	31	877
16	881		

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.

POOR ORIGINAL

(9/77)

936150

POOR ORIGINAL

UNIT SHUTDOWNS AND POWER REDUCTIONS

REPORT MONTH August, 1979

DOCKET NO. 50-346
UNIT NAME Davis-Besse Unit 1
DATE September 7, 1979
COMPLETED BY Erdal Caba
TELEPHONE 419-259-5000, Ext. 236

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
10	79 08 06	S	0	B	NA	NA	NA	NA	Power was reduced to perform the Unit Load Transient Test, TP 800.23.

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-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)

936151

OPERATIONAL SUMMARY FOR AUGUST, 1979

8/1/79 - 8/6/79 — The reactor power level was maintained between 99 and 100 percent with the generator gross load at 915 $\pm$ 10 MWe.

8/7/79 - 8/8/79 — The 90-40-90 percent reactor power transient testing was performed on August 7, 1979 as part of the Unit Load Transient Test, TP 800.23. The 60-30-60 transient with three reactor coolant pumps running was performed on August 8, 1979 to complete the Unit Load Transient Test. This marked the completion of the power escalation test program.

The unit experienced feedwater swings periodically from August 8, 1979 to August 13, 1979. The problem was found to be caused when the speed control for the main feed pump turbine would go high causing the Integrated Control System (ICS) to recover and bring the speed back down. The speed control problem was attributed to high temperature in the speed control cabinet for the main feed pump turbine 1-1. A fan was temporarily installed to cool the speed control cabinets.

8/9/79 - 8/21/79 — The reactor power level was maintained between 99 and 100 percent with the generator gross load at 915 $\pm$ 10 MWe. At 1113 hours on August 22, 1979, the ICS went into the tracking mode and followed a reduction in power to approximately 82%. The runback was caused when an erroneous low pressure signal from the turbine header pressure limiter transducer overrode the ICS signal and began to close the turbine control valves. The turbine load limit was reduced until it controlled header pressure and the throttle pressure limiter setpoint was increased to the maximum. The turbine header pressure limiter circuit output was grounded to assure the circuit would not affect the unit if the signal totally failed.

The unit was returned to 100% full power operation by 1515 hours on August 22, 1979.

8/23/79 - 8/31/79 — The reactor power level was maintained at 100% with the generator gross load at 915 $\pm$ 10 MWe.

On August 28, 1979, high airborne activity was noticed in all negative pressure areas. The problem was traced to the makeup pump seal drain and a temporary fix was initiated on September 1, 1979, until a more permanent solution could be found.

REFUELING INFORMATION

DATE: August, 1979

1. Name of facility: Davis-Besse Nuclear Power Station Unit 1

2. Scheduled date for next refueling shutdown: March, 1980

3. Scheduled date for restart following refueling: May, 1980

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)?

Yes, see attached

5. Scheduled date(s) for submitting proposed licensing action and supporting information. December, 1979

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.

The spent fuel pool capacity expansion program was approved by the NRC in Amendment 19 to the operating license received August 1, 1979.

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

(a) 177 (b) 0 (zero)

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 260 Increase size by 475 (735 total)

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

Date 1989 (assuming ability to unload the entire core into the spent fuel pool is maintained and the unit goes to an 18 month refueling cycle)

936153

4. The following Technical Specifications (Part A) will require revision:

2.1.1 & 2.1.2 - Reactor Core Safety Limits (and Bases)
2.2.1 - Reactor Protection System Instrumentation Setpoints (and Bases)
3.1.3.6 - Regulating Rod Insertion Limits
3.1.3.7 - Rod Program
3.2.1 - Axial Power Imbalance (and Bases)

The following Technical Specifications (Part A) may also require revision:

3.1.2.8 & 3.1.2.9 - Borated Water Sources (and Bases)
3.2.4 - Quadrant Power Tilt (and Bases)
3.2.5 - DNB Parameters (and Bases)

FACILITY CHANGE REQUESTS COMPLETED DURING AUGUST, 1979

FCR NO: 79-247

SYSTEM: Containment Air Coolers

COMPONENT: Motor power lead conduits

CHANGE, TEST, OR EXPERIMENT: This FCR was written to approve cutting the conduit between the fan casing and the motor on all three containment air coolers. Upon reinstalling the fan motors, the subject conduits were reconnected with threaded couplings. This change was done with the prior approval of the fan vendor, Joy Manufacturing Company, as well as the unit architect-engineer, Bechtel Company.

REASON FOR THE FCR: Due to the conduit configuration outside of the fan housing, the conduit could not be removed in one piece to allow the disconnection of power leads to the fan motor. The motor leads must be removed to allow the fan motors to be removed in order to repack the bearing grease.

SAFETY EVALUATION: The addition of these couplings will not affect the function of the containment air coolers, nor their seismic qualification. This is not an unreviewed safety question.

FACILITY CHANGE REQUESTS COMPLETED DURING AUGUST, 1979

FCR NO: 79-265

SYSTEM: Main Feedwater and Main Steam

COMPONENT: Seismic Supports

CHANGE, TEST, OR EXPERIMENT: On June 29, 1979, under Maintenance Work Order 79-265 the physical work for FCR 79-265 was completed. This FCR added shims to the hydraulic snubber mounting brackets of seismic snubbers SR-9A and SR-32. These changes have been documented by the unit architect-engineer, Bechtel Company by revisions to Bechtel drawings C-611, C-617 and vendor (ITT Grinnell) drawings CC-12-146-4 and CC-12-128-6.

REASON FOR THE FCR: The shims were added to extend the hydraulic snubber mounting brackets in order to bring the piston position during normal operation closer to the center of its travel.

SAFETY EVALUATION: This change adds shims to seismic supports SR-32 and SR-9A. The added shims will assure that the cold and hot piston settings for these hydraulic snubbers meet Toledo Edison acceptance criteria for piston location. Snubber reliability will be enhanced by this change. There will be no adverse effect on the safety function of the main feedwater and main steam systems. This is not an unreviewed safety question.

FACILITY CHANGE REQUESTS COMPLETED DURING AUGUST, 1979

FCR NO: 78-264

SYSTEM: Neutron Flux Monitors

COMPONENT: Amphenol connectors in penetrations associated with neutron flux monitors

CHANGE, TEST, OR EXPERIMENT: On May 16, 1979, work was completed on Maintenance Work Order 78-1346 which completed the work called for in FCR 78-264. This change installed Ray Chem type WCSF-N heat shrink tubing over all Amphenol connectors located in penetrations P1L1L, PZL4G, P3L4S, and P4L1G which are associated with neutron flux monitoring. Ray Chem type WCSF-N heat shrink tubing has been qualified for nuclear service and has been successfully tested by Ray Chem for LOCA conditions.

REASON FOR THE FCR: The connectors were covered with heat shrink tubing to ensure their hermiticity during small steam line breaks and rod ejection accidents within containment during which the reactor neutron flux monitors may be required.

SAFETY EVALUATION: The subject Amphenol connectors are used on the neutron flux monitors, which are not needed during or after a LOCA. The connectors are qualified for the design temperatures which can be expected during small steam line breaks and rod ejection accidents. The addition of the Ray Chem heat shrink tubing to these connectors will not affect their function.

936157

FACILITY CHANGE REQUESTS COMPLETED DURING AUGUST, 1979

FCR NO: 79-259

SYSTEM: Containment Air Coolers

COMPONENT: Temperature Indicating Controllers (TICs) 1356, 1357 and 1358

CHANGE, TEST, OR EXPERIMENT: On June 25, 1979, work was completed which changed the setpoints of TIC-1356, TIC-1357, and TIC-1358 from 120°F to 50°F. Prior to this change, the controllers regulated the Service Water outlet valves of their associated containment air cooler to maintain a proper containment air temperature. This change forces the valves to remain wide open during normal operation thereby eliminating the response time requirement. The Davis-Besse setpoint index was revised to reflect this change.

This change is being made on a temporary basis until FCR 79-280, which will modify the pneumatic actuators on these valves to bring their response time within the required limits, is implemented. FCR 79-280 is scheduled to be implemented during the 1980 refueling outage.

REASON FOR THE FCR: The stroke time of the service water outlet valves of the coolers was measured to be approximately 75 seconds from the fully closed position to the fully open position. Formerly, when the valves were being regulated by their associated temperature controllers to maintain proper containment air temperature, the valves may have been at times throttled to the extent that upon receipt of a Safety Features Actuation System signal, they may not have fully opened within the allowable time (see Licensee Event Report NP-32-79-10).

SAFETY EVALUATION: This FCR provides for the revision to the setpoints on TIC-1356, 1357 and 1358 from 120°F to 50°F. This change will force the service water flow control valves in the outlet of containment air coolers to a wide open position during normal operation. With the valves wide open initially, the full service water flow would occur when the service water pump is started. This FCR will not degrade the cooling function of the containment air coolers, and it is not an unreviewed safety issue.

936158

FACILITY CHANGE REQUESTS COMPLETED DURING AUGUST, 1979

FCR NO: 77-509

SYSTEM: Emergency Diesel Generators

COMPONENT: Fuel Oil Storage Tanks

CHANGE, TEST, OR EXPERIMENT: FCR 77-509 was written to review work done on November 23, 1977, under Maintenance Work Order 77-1709. Under this Maintenance Work Order, the three inch diameter, four bolt flanges on the sounding connections of both fuel oil storage tanks were modified to provide for a 1/2" pipe plug for level measuring purposes. This change has been documented in the appropriate drawings by the unit-architect engineer, Bechtel Company.

REASON FOR THE FCR: Removing the three inch diameter, four bolt, flange to measure the tank level as required to fulfill surveillance requirements was found to be extremely inconvenient.

SAFETY EVALUATION: This change, which installs a plug for measuring tank level, will have no adverse effect on the safety function of the emergency diesel generator system.

936159

FACILITY CHANGE REQUESTS COMPLETED DURING AUGUST, 1979

FCR NO: 77-358

SYSTEM: Service Water

COMPONENT: Service Water Pump Motors 1-1, 1-2 and 1-3

CHANGE, TEST, OR EXPERIMENT: This FCR was written to provide drain valves on the upper bearing oil reservoirs of the three service water pumps. On June 23, 1979, Maintenance Work Order 78-2045, which added the drain valves to the already existing oil drain piping, was completed. The affected drawings and piping class sheets were revised by the unit architect-engineer, Bechtel Company, to document the addition of these drain valves. The service water system procedure and surveillance test, as well as the capped valve procedure, have been modified to reflect the addition of these valves.

REASON FOR THE FCR: The valves were added to facilitate the changing of the motor lubricating oil. Formerly, the oil had to be drained by removing a pipe plug in the oil drain line, which lead to oil being spilled on personnel or equipment. The addition of these drain valves eliminates this oil spillage.

SAFETY EVALUATION: This change adds a drain valve in the drain line to the upper bearing oil reservo.r on service water pump motors 1-1, 1-2 and 1-3. The valve will make it easier to drain the reservoir. A pipe cap is installed downstream of the valve, and the valve is held in the closed position by a lock wire. This change will not affect the safety function of the service water pumps, but will reduce oil spills when draining the reservoir for maintenance.

FACILITY CHANGE REQUESTS COMPLETED DURING AUGUST, 1979

FCR NO: 79-172

SYSTEM: Various Systems

COMPONENT: Motor starter overload heaters

CHANGE, TEST, OR EXPERIMENT: Maintenance Work Order 79-172, completed on May 15, 1979, completed the physical work associated with FCR 79-172. This FCR replaced overload heaters on Q-listed motor starters, which were found to be not in accordance with the overload heaters size specified on the relay setting data sheet.

REASON FOR THE FCR: This FCR is part of the corrective action fcr Licensee Event Report NP-32-79-04 (see also FCR 79-163). This change replaces the overload heaters which were not in accordance with the relay setting data sheets.

SAFETY EVALUATION: The function of the overload heaters which are being changed under this FCR bear no action on the control and indication circuits. The controls of the nuclear safety related motors are not affected by the function of these heaters. However, these heaters are designed properly for the given application. No overload relay is used for tripping the essential motor operated valves and dampers. The fault protection is provided by the molded case breaker. The duty cycle of the essential motor operated valve decide the selection of the overload heaters.

It is thus concluded that the safety function of the systems is not affected by changing to the required thermal overload heater size. The changes do not involve an unresolved safety question.

936161

FACILITY CHANGE REQUESTS COMPLETED DURING AUGUST, 1979

FCR NO: 79-163

SYSTEM: Various Systems

COMPONENT: Motor starter overload heaters

CHANGE, TEST, OR EXPERIMENT: Maintenance Work Order 79-163 completed on April 24, 1979, completed the physical work associated with FCR 79-163. This FCR replaced overload heaters on Q-listed motor starters, which were found to be not in accordance with the size overload heaters size specified on the relay setting data sheet.

REASON FOR THE FCR: This FCR is part of the corrective action for Licensee Event Report NP-32-79-04 (see also FCR 79-172). This change replaces the overload heaters which were not in accordance with the relay setting data sheets.

SAFETY EVALUATION: The overload heaters being changed perform no action on the control or indication circuits. Functionally the overload heaters have no effect on the nuclear safety related motors. For essential motor operated valves and dampers, overload relays are not used for tripping. Molded case breakers provide fault protection. The operation time of the valves is considered in the selection of essential MOV overload heaters.

Therefore, the safety function of the systems is not affected by changing to the proper thermal overload heater size per notes on relay setting sheet 54A. The changes do not involve an unreviewed safety question.

936162

FACILITY CHANGE REQUESTS COMPLETED DURING AUGUST, 1979

FCR NO: 78-039

SYSTEM: Service Water (SW)

COMPONENT: Pressure switches PSH 2917, PSH 2917A, PSH 2918, PSH 2918A, PSH 2919, PSH 2919A

CHANGE, TEST, OR EXPERIMENT: The setpoint index, Bechtel drawing 7749-M-620S, was revised to document the changing of the setpoints of the above mentioned pressure switches as follows:

Pressure Switch	From	To
PSH 2917	115 psig	95 psig
PSH 2917A	125 psig	110 psig
PSH 2918	115 psig	95 psig
PSH 2918A	125 psig	110 psig
PSH 2919	115 psig	95 psig
PSH 2919A	125 psig	110 psig

These changes were made with the approval of the unit architect-engineer, Bechtel Company, and are documented in a revision to the Davis-Besse Unit 1 Setpoint Index.

REASON FOR THE FCR: The settings of these pressure switches were changed in order to allow the service water strainer blowdown valves to operate as designed. PSH 2917A, 2918A and 2919A operate to open their associated strainer blowdown valve when the service water pump discharge pressure valve increases to the switch setpoint. Since the setpoint of the relief valves are 120 psig, the former pressure switch setpoint of 125 psig caused the relief valve to actuate prior to the strainer blowdown valve. This change corrects this off design condition.

SAFETY EVALUATION: The changed setpoints have been tested and resulted in the blowdown valves opening before the relief valves lift. This is in accordance with system design. The setpoint changes will not adversely affect the safety function of the system. This is not an unreviewed safety question.

FACILITY CHANGE REQUESTS COMPLETED DURING AUGUST, 1979

FCR NO: 79-224

SYSTEM: Pressurizer Relief System

COMPONENT: Hydraulic snubbers on hanger 30-CCA-8-H12

CHANGE, TEST, OR EXPERIMENT: When the two hydraulic snubbers located on pipe hanger 30-CCA-8-H12 were replaced after undergoing functional testing on May 27, 1979, it was found that it was nearly impossible to replace and to purge air from the equalizing tubing between the two snubbers. Consulation with the snubber vendor, ITT Grinnell, revealed that replacement of the equalizing tubing was not necessary for the proper operation of the snubbers. As the tubing was not necessary, and since failure to purge the air from the tubing would not allow the snubbers to operate properly, the tubing was not replaced. This change is documented on the applicable pipe hanger drawing.

REASON FOR THE FCR: It was nearly physically impossible to replace the equalizing tubing in its prior configuration and no means for bleeding of the air from the tubing was provided. Since the tubing is not necessary for proper snubber operation, and air not purged from the tubing may impair the operation of the snubbers, the tubing was not replaced.

SAFETY EVALUATION: FCR 79-224 involves removal of a pressure equalization line on two snubbers located at hanger CCA-8-H12. Per ITT Grinnell, Providence, Rhode Island, the equalization line is not required; hence, its removal will not prevent the snubbers from performing their intended safety function. Since the line is not required for proper snubber performance, no unreviewed safety question is created.

FACILITY CHANGE REQUESTS COMPLETED DURING AUGUST, 1979

FCR NO: 79-258

SYSTEM: Startup Test Program

COMPONENT: 100% Turbine Trip Test

CHANGE, TEST, OR EXPERIMENT: Facility Change Request 79-258 was written to initiate a safety evluation to verify that the expected results of a turbine trip from 100% full power (after the addition of the Anticipatory Reactor Trip System (ARTS) do not present any unreviewed safety questions. The basis for this evaluation was reactor trip from 100% of full power which occurred on January 17, 1979. This evaluation also shows that the requirements of a 100% turbine trip in the startup test program can be satisfied by using the data from the aforementioned reactor trip from 100% of full power.

REASON FOR THE FCR: FCR 78-496 performed a safety evaluation to use the test results of the 100% of full power load rejection test to satisfy the requirement of a turbine trip test from 100% of full power. Since the addition of ARTS, which initiates a reactor trip on any turbine trip which occurs from greater than or equal to 15% of full reactor power, FCR 78-496 no longer satisfies the test program requirement for a turbine trip test from 100% of full power.

SAFETY EVALUATION: Toledo Edison Power Engineering had performed a safety evaluation (FCR 78-496) on performing the 100% load rejection test (TP 800.13) in place of the 100% turbine trip test (TP 800.14). In light of changes made to Davis-Besse Unit 1 and installation of the Anticipatory Reactor Trip System (ARTS), 100% turbine trip cannot be replaced by the 100% load rejection test. With the installation of ARTS, the reactor is tripped almost instantaneously when the turbine is tripped and the transient for all practical purposes is identical if either the reactor or the turbine is tripped first. A new safety evaluation is performed below, utilizing data for a reactor and turbine trip from 100% thermal power to substitute for the 100% turbine trip test. Specifically, the acceptance criteria for Reactor Trip Test as outlined in TP 800.14 are shown to be successfully met. The acceptance criteria for reactor trip (instead of turbine trip) were used since with ARTS, a turbine trip causes a simultaneous reactor trip. The reactor and turbine trip transient used is the January 12, 1979 event.

Brief Description of the January 12, 1979 Event: The unit was in Mode 1 with reactor power 2772 MWT and load 900 MWE (gross). The event was initiated by an accidental grounding of containment hydrogen analyzer. The ground resulted in blowing of the 200 amp internal fuse on the inverter feeding Y2 essential instrument bus. The resultant loss of Y2 bus caused a loss of power to Reactor Protection System (RPS) Channel 2, Safety Features Actuation System (SFAS) Channel 2 and Steam and Feedwater Rupture Control System (SFRCS) Channel 2. Prior to this, RPS Channel 3 was bypassed for surveillance testing resulting in average nuclear instrumentation (NI) power from Channel 1 and Channel 3 to be locked out of high auctioneer power circuit such that the NI power signal to the ICS was only the average of Channel 2 and Channel 4. With the loss of

power to RPS Channel 2 the average NI power signal to the ICS dropped to approximately 50%. The ICS, attempting to increase the power to the desired 100% called for a rod withdrawal.

With the loss of power to RPS Channel 2 the Reactor Coolant System (RCS) flow indicator to the ICS fed through RPS Channel 2 also went to zero. This caused the BTU limits of the ICS to immediately reduce main feedwater flow.

The result of the above sequence of events was a reactor trip (on high flux/delta flux/flow) and an SFRCS trip (on steam generator-2 low level). The SFRCS trip was caused by the loss of Y2 bus, resulting in ICS cutting back main feedwater to steam generators on BTU limits. The turbine tripped almost at the same time (within 210 milliseconds) the reactor tripped. For further details, see Licensee Event Report NP-33-79-13.

Acceptance Criteria for Reactor Trip Test as Outlined in TP 800.14

8.1.1 The turbine tripped when the reactor was tripped.

8.1.2 The reactor coolant system pressure remains above the high pressure injection setpoint of 1600 psig after a reactor trip.

8.1.3 None of the reactor coolant pumps trip as a result of the reactor trip.

8.1.4 The turbine bypass system setpoint is transferred to maintain header pressure at 995 ± 25 psig.

The header pressure is controlled at the steam generator outlet following a reactor trip.

8.1.5 Steam generator outlet steam pressure remains below 1155 psig after the reactor trip.

8.1.6 During the reactor trip transient, the pressurizer level must remain between ten (10) and 320 inches and be returned to above 40 inches when the plant has stabilized. (Reactor Trip only)

8.1.7 Steam Generator Level remains above the initiation point of the SFRCS and below 375 inches indication and controls at the low level limit when the plant has stabilized.

8.1.8 The steam generator feedwater temperature remains above 110°F after the reactor trip.

936166

The following discussion demonstrates that all of the above acceptance criteria were satisfactorily met.

1. The turbine tripped 0.21 sec. after the reactor trip.

2. High pressure injection setpoint of 1600 psig was never reached during the transient. The minimum RCS pressure available from the reactimeter data is 1867 psig.

3. All four reactor coolant pumps were running through the transient and none was tripped as a result of the reactor trip.

4. The steam generator outlet pressure was maintained at 995 ± 25 psig following the reactor trip excepting the initial pressure rise caused by ICS runback of main feedwater flow. This acceptance criteria is not required to be met after the SFRCS trip, since actuation of SFRCS disables ICS control of the atmospheric vent valves and the Main Steam Isolation Valves are closed.

5. The steam generator outlet steam pressure did not exceed 1155 psig after the reactor trip. The maximum steam generator pressure available on the reactimeter data is 1079 psig observed on steam generator 2.

6. During the reactor trip transient the pressurizer level remained within the range of 10-320 inches and was controlled between 88-128 inches (from control room strip chart) after the plant returned to stable conditions.

7. During the applicable period, main feedwater temperature was of the order of 450-460°F satisfactorily meeting the 110°F limit.

8. SFRCS tripped on low steam generator level. The trip was not a result of the reactor trip but was caused by the loss of Y2 bus which resulted in ICS cutting back main feedwater to the steam generators to balance the mismatch between reactor power signal to ICS and main feedwater flow. At no time did the level exceed 375" on the startup range. Also, pressure on the secondary side of either steam generator was never lost and sufficient heat sink was maintained. Since the low steam generator level trip of SFRCS was <u>not</u> a result of the reactor trip this acceptance criterion is considered to be satisfactorily met. In addition, Davis-Besse Unit 1 has successfully demonstrated this capability (steam generator levels above SFRCS trip set point and controlled at low level limit after plant stabilization) on another reactor trip from 99% full power on November 13, 1978.

936167

Pursuant to the above, it is concluded that with the change to the test program proposed in this FCR (79-258):

i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not created.

ii) A possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created.

iii) The margin of safety as defined in the bases for any Technical Specification is not reduced.

This is not an unreviewed safety question.

FACILITY CHANGE REQUESTS COMPLETED DURING AUGUST, 1979

FCR NO: 79-216

SYSTEM: Decay Heat Removal and Low Pressure Injection

COMPONENT: Hydraulic snubber on pipe hanger CCB-6-H8

CHANGE, TEST, OR EXPERIMENT: On June 29, 1979, work was completed which added an extension to the mounting of the hydraulic snubber CCB-6-H8 which is on the decay heat removal/low pressure injection system piping. The affected pipe hanger drawings have been revised to document this change.

REASON FOR THE FCR: The extension to the snubber mounting plate was added to bring the piston position during normal operation closer to the center of its travel.

SAFETY EVALUATION: This FCR involves the repositioning of a Q-listed snubber to its correct position. This change enhances the ability of the snubber to perform its safety related function. No unreviewed safety question is involved.

936169

FACILITY CHANGE REQUESTS COMPLETED DURING AUGUST, 1979

FCR NO: 79-185

SYSTEM: Reactor Coolant System (RCS)

COMPONENT: Pressurizer

CHANGE, TEST, OR EXPERIMENT: This FCR was written to authorize the temporary installation of special test apparatus and to authorize the conduct of an experiment using that apparatus to determine the effectiveness of using the pressurizer Resistance Temperature Detector (RTD) and heater bridge to measure approximate pressurizer level. The experiment was conducted under Test Procedure TP 550.03, "Pressurizer Low Level Determination Using Pressurizer RTD and Heater Bridge", which was written expressly for this purpose. The experiment was conducted on May 2 and May 3, 1979, under the direction of personnel from the Oak Ridge National Laboratory, as well as the Reactor Coolant System vendor, Babcock and Wilcox.

REASON FOR THE FCR: The purpose of the aforementioned experiment was to collect data to assist the personnel at Three Mile Island Unit 2 in measurement of pressurizer level by alternate means.

SAFETY EVALUATION: The pressurizer low level determination using pressurizer RTD and heater bridge will be accomplished in operational mode 5 and will include the following:

Phase 1 - Installation of pressurizer RTD circuitry
Phase 2 - Installation of pressurizer heater circuitry
Phase 3 - Varying pressurizer level
Phase 4 - System restoration

This experiment will be conducted within the limits of Technical Specification 3.4.9.1 and 3.4.9.2. This experiment is not described in the Davis-Besse Unit 1 FSAR, however, as the procedure limits and precautions state, the pressurizer heater bank control switch will be in the "ON" position and RCS pressure will be controlled manually with heater banks 2B, 4. The pressurizer is required by Technical Specification 3.4.4 to be operable only in Modes 1 and 2. Level will be manually maintained in the pressurizer by monitoring digital voltmeter (DVM) readings, and/or converting uncompensated pressurizer level to actual level. During the experiment, the level in the pressurizer will be maintained above the pressurizer lower delta P nozzle, to avoid having to re-vent the RCS and vent the level transmitter lower taps. The experiment will be terminated if the level goes below the low level tap. Tests by the RTD vendor and others have indicated that the RTDs can withstand the applied current without adverse effects. The instrumentation involved in this experiment is Q-listed from pressure boundary of RCS standpoint, but otherwise it is not nuclear safety related.

Based upon a detailed review of this experiment procedure, it is concluded that this experiment is not an unreviewed safety question.

936170

POOR ORIGINAL

OPERATING DATA REPORT

DOCKET NO. 50-346
DATE August 7, 1979
COMPLETED BY E. Caba
TELEPHONE 259-5000 Ext. 236

OPERATING STATUS

1. Unit Name: Davis-Besse Unit 1
2. Reporting Period: July, 1979
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): To be det.
7. Maximum Dependable Capacity (Net MWe): To be det.
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): Zero (until July 6, 1979)
10. Reasons For Restrictions, If Any: NRC OIE Bulletins and Shutdown Orders

	This Month	Yr.-to-Date	Cumulative
11. Hours In Reporting Period	744	5,087	16,852
12. Number Of Hours Reactor Was Critical	498	2,245.4	8,877.2
13. Reactor Reserve Shutdown Hours	246.0	1,858.2	2,648.5
14. Hours Generator On-Line	479.8	2,154.9	7,888.1
15. Unit Reserve Shutdown Hours	264.2	1,728.2	1,728.2
16. Gross Thermal Energy Generated (MWH)	1,230,451	5,109,548	15,297,118
17. Gross Electrical Energy Generated (MWH)	409,950	1,703,218	5,086,973
18. Net Electrical Energy Generated (MWH)	381,814	1,594,372	4,635,832
19. Unit Service Factor	64.5	42.4	48.3
20. Unit Availability Factor	100	76.3	59.9
21. Unit Capacity Factor (Using MDC Net)	To be det.		
22. Unit Capacity Factor (Using DER Net)	56.6	34.6	33.7
23. Unit Forced Outage Rate	0	3.8	21.3

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

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

	Forecast	Achieved
INITIAL CRITICALITY		
INITIAL ELECTRICITY		
COMMERCIAL OPERATION		

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