IOWA ELECTRIC LIGHT AND POWER COMPANY

Duane Arnold Energy Center

1978

Annual Report of Facility Changes, Tests and Experiments

Section A

PLANT DESIGN CHANGES

This section has been prepared in accordance with the requirements of 10 CFR, Part 50.59(b).

A. Plant Design Changes

This section contains brief descriptions of and bases for plant design changes accomplished during the calendar year 1978, and summaries of the safety evaluations for those changes, pursuant to the requirements of 10 CFR, Part 50.59 (b).

DCR No. 69 RE 4131 A & B

Description of Change: the cables were separated. The detectors were remounted on a new base plate of 1/4" steel plate with steel plate welded between the detectors.

Reason for Change: The equipment was not properly separated, nor was it indicated as seismic Class I on the P & IDs.

Safety Evaluation: This design change upgraded the installation of RE 4131 A & B to meet FSAR commitments.

DCR No. 107 Special Test Equipment

Description of Change: The ECCS test switch was changed to provide two (2) off positions.

Reason for Change: The test complexity and chances for error during surveillance testing were reduced.

Safety Evaluation: The change made no reflection upon the safety of the test switch or any related component.

DCR No. 141 Rev. 1 Provide Air Filters in Diesel Air Start System

Description of Change: A Fram filter, Model #1MCD2-57PLO, was installed in each of the four air lines upstream of the air solenoid valves on diesel generator units 1G-21 and 1G-31. Additionally, the necessary isolation valves required to maintain the systems Seismic Class I Criteria were installed. The hanger detail PS-1, Det. "U" on FSK-3431B and FSK-3432B was removed and a PS-2, detail "A" at the location shown on FSKs 4930B, 4981B, 4982B, and 4983B was added.

Reason for Change: The change was necessitated due to clogging of the solenoid valves in the starting air lines from compressors 1K-10A through D. The final stress analysis dictated the addition of 4 supports below the tanks, and removal of two hangers, which provided the support and thermal growth required.

Safety Evaluation: The change increased the starting reliability of the standby diesel generators by eliminating/reducing solid contaminates in the air that impaired the operation of the air start solenoid valves. The additional components in this change meet the original quality/design criteria, hence no degradation of the system. This change also did not pose additional hazards or considerations that were not already addressed in the Safety Analysis Report. This change did not interface with another safety related function.

DCR No. 161 Closed Radwaste Drains

Description of Change: A cap was welded on an open 2" MRD closed radwaste equipment drain line approximately one foot south of "A" core spray pump in the SE corner of the room.

Reason for Change: The previously installed screwed cap did not satisfy the governing piping specification.

Safety Evaluation: The change did not interface with safety related equipment. The change had no safety significance.

DCR No. 343 CRD

Description of Change: Automatic valves, which close when the pump is stopped, were installed in the cooling water lines to 1P-209 A, B.

Reason for Change: The change was made to prevent condensation from forming in the bearing house.

Safety Evaluation: The components added by this change were neither part of nor did they interface with any safety related system.

DCR No. 413 Off Gas Stack Rod Monitor

Description of Change: A standby vacuum pump was added and the related drawings were corrected to the as-built status.

Reason for Change: The reliability of the sampling system was increased. The drawings were updated to reflect this change.

Safety Evaluation: This change did not affect or interface with a safety system. It was to aid normal operation by providing a standby vacuum pump for the stack radiation monitoring.

DCR No. 444 Reactor Building H & V

Description of Change: A wind shield was provided for OPS-7638. An up-to-date description of the previously existing device was provided.

Reason for Change: The previously existing equipment provided operation which appeared to be susceptible to false pressure sensing under variable wind conditions. The change provided a shield to limit the effect of wind velocity on sensing of atmospheric pressure.

Safety Evaluation: The instrument involved in this DCR was not part of a safety system and did not represent a safety concern. The change did not interface with any safety system and was only intended to enhance the operation of the building H & V system.

DCR No. 449 Fuel Handling

Description of Change: A section of woven wire cloth was installed directly under the 1 HP bridge drive motor. The woven wire fit the ID of the drive motor enclosure and was secured by 4 - 1/4 x 1" lg. countersunk screws with hex nut and flat washers.

Reason for Change: The installation of the woven wire screen under the bridge drive motor reduced the possibility of objects being dropped into the water while maintenance is being performed on the unit.

Safety Evaluation: This design change significantly reduced the possibility of a tool or device being dropped into the fuel pool area while maintenance was performed on the bridge drive motor. Additionally, this change did not present hazards or considerations not already addressed by the safety analysis report and did not communicate with a safety related function.

DCR No. 481 Reactor Building H & V

Description of Change: Isolation valves were added to instrument piping; to DPTs 7613, 7614, and 7615.

Reason for Change: The change was required in order to isolate the instruments to perform surveillance tests. The installation included fittings necessary to provide connection of test equipment.

Safety Evaluation: This change did not present a safety question since the change did not involve any safety related equipment. It provided a piping configuration which allowed for surveillance testing and did not modify the original design intent for the instruments concerned.

DCR No. 489 Containment Atmospheric Monitoring

Description of Change: The check valve internals of V-81-33 and V-81-34 were removed to allow proper drainage. Drain traps were added to lines of panels 1C-219A and 1C-219B.

In addition, heat tape was added to the internal sample lines in panels 1C-219A and 1C-219B between the inlet and outlet drain trap take off points.

Reason for Change: Large quantities of water were collecting in the radiation detector chambers causing instrument damage and excessive filter changes.

Safety Evaluation: The design change involved additions and modifications which duplicated existing drain configurations presenting no unreviewed safety question. Materials and arrangement for the drain trap added to the discharge line were the same as the existing trap on the inlet line. Check valves V-81-33 and V-81-34 were used to isolate the draw traps during maintenance. Isolation, however, was possible by closing the other drain trap valves since flow rates were low and did not require continuous drainage. Therefore removal of the check valve internals did not degrade the system.

DCR No. 532 Recirc System

Description of Change: Relays B31A-K45A & B and associated wiring were added in panel 1C-18.

Reason for Change: In the event the loss of one feedwater pump is coincident with a low level alarm, it is necessary to reduce reactor power to a level within the capacity of the remaining pump. Moving the runback limiter to the output of the speed controllers ensured the runback will be rapid enough to avoid a level 3 scram.

Safety Evaluation: This design change did not interface with any safety related components. The implementation of this change enhanced the plant safety and reliability in the event of a loss of one feedwater pump with a coincident low level alarm occurring.

DCR No. 533 Uninterruptable AC Control

Description of Change: The control relay (CR) of transfer switch 11/22 was replaced with an undervoltage relay.

Reason for Change: The control relay furnished with the transfer switch did not drop out until an approximate 50% voltage condition was reached. The original relay was replaced with an undervoltage relay which drops out at 70% of the rated voltage.

Safety Evaluation: The transfer of uninterruptible AC from normal to emergency source was made more reliable. As the transfer switch 11/22 was not safety related and did not interface with any safety related equipment, the change had no safety signficance.

DCR No. 541 Area Radiation Monitors

Description of Change: The resistors on component board A, sensor, and converter of area radiation monitors were changed.

Reason for Change: The change of the resistors was due to the change in specification of G-M tubes by the vendor. This modification was made to insure GM tubes operate properly at high radiation levels.

Safety Evaluation: This change did not interfere with any safety related components and did not present significant hazard considerations described or implicit in the safety analysis report. The implementation of this change enhanced plant safety due to increased reliability of the area radiation monitors.

DCR No. 592 Emergency Service Water System

Description of Change: Eight pressure differential indicators were permanently installed in the emergency service water system, four in each ESW loop.

Reason for Change: This change was made so Surveillance Test Procedure 48C001 could be run without switching gages from loop to loop.

Safety Evaluation: The change did not present any significant hazards or considerations not described or implicit in the Safety Analysis Report. The change facilitated surveillance testing by attaching permanent instrumentation to previously existing instrumentation. Since the instruments did not perform any new function nor modify the original design function of the ESW systems, no new considerations were involved.

DCR No. 595 Off Gas

Description of Change: The calibration and zero gas bottles were relocated to the ground floor of the off gas retention building.

Reason for Change: The movement of the bottles up and down stairs was a personnel safety hazard.

Safety evaluation: The change relocated components which neither interfaced with nor were part of any safety related system.

DCR No. 596 Containment Atmospheric Control

Description of Change: A CRW drain was added near the containment nitrogen compressor skid and its receiver. The drain trap discharges from the moisture separator (IE-71) dryer (IE-72), and receiver (IT-128) were directed to CRW. The CRW drain was installed per FCR #2082 and core drill #217.

Reason for Change: The existing drain traps discharged to the floor, where the condensate drained to an open floor drain about 10 feet north of the equipment. Since this water was condensed from the drywell atmoshphere, the drains were piped to the CRW to prevent future contamination problems.

Safety Evaluation: The addition of the CRW drain did not affect the operation of any safety related system.

DCR No. 597 Condenser Air Removal

Description of Change: An annular flow element was installed.

Reason for Change: This change provided for permanent flow instrumentation to facilitate condenser leakage monitoring.

Safety Evaluation: This change did not present significant hazards or considerations not implied or described in the Safety Analysis Report. The purpose of the change was to facilitate condenser leakage monitoring.

DCR No. 601 N_2 Vaporizer Freeze Protection

Description of Change: Freeze protection was provided for the N_2 vaporizer system. Piping drains and heat tracing to instrument tubing were provided.

Reason for Change: Piping and instrumentation has frozen. This change enhanced the operation of the system.

Safety Evaluation: The additions did not present significant hazard considerations not described or implicit in the Safety Analysis Report. This addition did not change the intended design function of this system. The additions also did not interface with any safety related systems.

DCR No. 617 Neutron Monitoring System

Description of Change: Metal oxide varistors were added across coils of K2, K5, K7A through D, K8A through D and K9 A through D, relays on the SRM drive control, and across phase leads of all SRM drive motors.

Reason for Change: Noise problems in the IRM channels generated by the SRM drives were eliminated. The varistors suppressed noise spikes generated by the drive motors and relay coils which had caused at least one spurious scram.

Safety Evaluation: This change did not interface with safety related equipment. No potential hazard was introduced by this design change.

DCR No. 619 Primary Containment

Description of Change: LT-4363B and LT-2325 were replaced with the same model instruments having tighter range. The instruments were calibrated as required. In addition, the agastat time delay relay was replaced with a Potier & Brumfield sold state time delay relay type R15-30A-115-X2E1.

Reason for Change: The change was made to improve the accuracy of the torus level alarm/recorder system to meet NRC requirements. The relay change was made to upgrade the present time delay relay to extend the life of the relay.

Safety Evaluation: This work did not change any of the previously approved design functions of this non-safety related equipment. This change improved the overall accuracy of the water level alarm/recorder system for the torus and upgraded the quality of the previously existing time delay relay to extend its life due to severe service.

DCR No. 638 MD Stop Valve Test Solenoid Valve Modification

Description of Change: This modification changed the orifices in test solenoid valves and changed the actuation point of fast acting solenoids for MSVs.

Reason for Change: The modification was made to minimize team pressure spikes during turbine main stop valve testing.

Safety Evaluation: No significant safety considerations were presented by this modification since it did not affect nor interface with any safety related equipment. This change was made to enhance operability only.

DCR No. 645 SBLC Continuity Monitoring

Description of Change: Two series connected 10 Ω resistors were removed. Two sets of series connected 20 Ω resistors were wired and installed. Minor changes were made in wiring in TB BB on 1C05 PNL.

Reason for Change: The change was made to reduce the possibility of a resistor burning out when a squibb is fired. In addition, the continuity loop was extended to annunciate resistor failures.

Safety Evaluation: The change improved the standby liquid control continuity monitoring circuit as the continuity monitor loop was extended to include resistors and failure of any of the resistors would be annunciated in the control room. The change also improved reliability of the system due to the following facts:

1. The possibility of resistors burning out when squibb is fired would be reduced. The power dissipated through a resistor in the modified control circuit was made half that of the previously installed resistors.

2. A resistor failure in the modified design would jeopardize only one squibb valve firing control circuit as compared to both in the previous design.

The change did not affect any previously approved FSAR Safety Analaysis of the System.

DCR No. 668 Add Filter in CRD Pump Suction

Description of Change: A redundant filter (1F-15B) was installed at the CRD pump suction, adjacent to the existing filter (1F-15A). The new filter was Cuno Model No. 19SL3. The associated piping (4" - HBO-201 and 3/4" - HBD-201) and valves (V-17-108 through V-17-112) were added. The design documents were updated to reflect as-built.

Reason for Change: Previously, only one suction filter was installed in the line. When the suction filter required changing, the filter bypass was used. Due to the bypass piping and filter piping configurations, crud was released which plugged the CRD discharge filters. When the high pressure discharge filters were changed, they required a much more expensive filter and a contamination filter. With the redundant filter at the pump suction, this plugging of the high pressure discharge filters was eliminated.

Safety Evaluation: The change did not present significant hazards or considerations not described or implicit in the safety analysis report since the change did not interface with any safety or safety-related equipment. The portion of the CRD hydraulic system affected by this change was non-nuclear and seismic Class II.

DCR No. 676 Nitrogen Supply From the Liquid Supply Tank

Description of Change: A low pressure alarm on PNL 1C35 was added in the control room for the nitrogen supply from the liquid supply tank.

Reason for Change: The change was made to alert the operator of low N_2 pressure from the liquid supply tank to avert closing of out-board MSIVs (closing on low N_2 pressure).

Safety Evaluation: The alarm unit which was generically seimically qualified was securely mounted in the lower portion of PNL 1C35 where no safety related equipment was located. If during a seismic event the mounting fails, the unit would not fall on safety related equipment.

DCR No. 682 Reactor Building Air Header Isolation

Description of Change: The push button type hand switches HS3039, HS3034, and HS3035 were converted to key lock type switches.

Reason for Change: This change was made to prevent unauthorized persons from closing the subject hand switches.

Safety Evaluation: Hand switches HS3034, HS3035 and HS3039 were not safety related. This change did not interface with safety related equipment. No potential hazard was introduced by this change.

DCR No. 691 CRD Hydraulic

Description of Change: The previously existing pressure switch, PS 1842, was replaced with an improved switch.

Reason for Change: The new pressure switch was provided with a narrower dead band capable of being reset in the desired range.

Safety Evaluation: This pressure switch was not safety related. This design change neither affected safety related systems nor created a safety hazard. The new pressure switch has equal or better qualification.

DCR No. 695 Torus Vacuum Breakers

Description of Change: Hand switches HS-4304 and HS-4305 were changed from monentary contact push button type switches to key lock maintained contact switches. Connection diagrams Aped-H11-67, sheet 2 and sheet 4 were changed to reflect the existing condition.

Reason for Change: The change was made to allow hand switches HS-4304 and 4305 to remain in test position during refueling outage surveillance test STP 47A010. The subject hand switches were connected between contacts 1 and 2 instead of 3 and 4. The related internal connection diagrams Aped-H11-67 sheet 2 and sheet 4 required updating.

Safety Evaluation: The subject hand switches are turned to 'test' position only during the performance of surveillance test STP 47A010. These switches were keylocked and the key is only removable in 'normal' position. These features prevent the switches from being left in the 'test' position after the completion of testing. Should the switches be inadvertently switched to the 'test' position the other self-actuated check valve in the same channel would still prevent air from blowing from the torus area to the reactor building. A primary containment isolation signal would bypass the subject hand switches and ensure the proper control valves were in the closed position. The change exceeded requirements of

Regulatory Guide 1.47 for "Bypass and Inoperable Status Indication for Nuclear Power Plant Safety System." No unreviewed safety hazard was created by this design change.

DCR No. 697 Fire Protection System

Description of Change: FI-3300 was modified to provide a scale range of 0 to 4000 GPM with a calibration range of 0 to 600 in water.

Reason for Change: The change was required due to a proposed technical specification change. The previously installed flow indicator was incapable of measuring the flow rate required by the tech spec change.

Safety Evaluation: This system was not required for safe shutdown of the plant. The change did not alter the operation of the system and did not affect the original safety analysis.

DCR No. 718 Reactor Water Cleanup (RWCU) Valve V-27-01 Repair

Description of Change: The pipe between the weldolet and the first elbow and the valve V-27-01 was replaced with new pipe and valve.

Reason for Change: This change was necessitated because a thru-wall crack on the upstream weld and a crack on the downstream weld of the RWCU manual valve V-27-011.

Safety Evaluation: The repair did not present significant hazards or considerations not described or implicit in the safety analysis report, since the original design criteria were not altered. The new pipe and weld metal were on the type (316L) that has historically shown a high resistance to stress corrosion cracking in the as installed condition and should improve overall system reliability. Bechtel Power Corp. reviewed the stress report to verify the replacement met the original design criteria.

DCR No. 721 Packing Leak-Off Lines From MSIVs

Description of Change: For each of the inboard MSIV leak-off lines there exists a pair of flanges downstream of the leak-off valve (6" downstream of the leak-off valve for valves V-14-51 and V-14-52, 9" downstream for valves V-14-53 and V-14-54). The "handle blank" was placed between the two flanges on each leak-off line. For each outboard MSIV leak-off lines there exists a pair of flanges downstream of the leak-off valve (approximately 15" downstream of the leak off valve for valves V-14-47 and V-14-50 and approximately 17" downstream for valves V-14-41 and V-14-44). The "handle blank" was placed between the two existing flanges on each leak off line.

Reason for Change: The packing leak-off lines from the MSIVs (both sets of four) leaked through the leak-off line valves. Since the valves of these leak-off lines were operated in the closed position during operation, the blanking off of these lines did not affect system operation. The replacement of the leak-off lines valves would have been more expensive than blanking off the lines.

Safety Evaluation: The change did not present significant hazards or considerations not described or implicit in the Safety Analysis Report. Since the valves on the MSIV leak-off lines were operated in the closed position during operation, blanking off these lines gave increased assurance any valve leaning during operation or abnormal conditions would not violate the primary system isolation boundary.

DCR No. 723 Install Cap/Plug in RWCU Line

Description of Change: A 3/4" half coupling, pipe and cap was added above the penetration of 2"-DCA-7.

Reason for Change: Due to the leak across valve V-27-15, the connection was added above the penetration of 2" DCA-7 in order to drain the water so that the final weld for DCR-718 could be performed.

Safety Evaluation: The change did not present significant hazards or considerations not described or implicit in the safety analysis report. The change was in accordance with the original design criteria. No design function was changed or affected.

DCR No. 736 Modify DAEC Fuel Grapple

Description of Change: The fuel grapple was modified per General Electric SIL #181: to install a new redundant hook fuel grapple head. Each hook has its own air cyclinder actuator. Independent position sensors were installed for each grappling hook and wired in series to display a single indication of hook position on the grapple control console. The hook position indicator was interlocked with the fuel grapple hoist to prevent lifting of more than approximately 500 pounds if both hooks were not fully closed.

Reason for Change: The change was necessary to prevent fuel bundles from being dropped.

Safety Evaluation: Modification of the fuel grapple further reduced the chance of dropping a fuel bundle by:

1. Using a redundant hook with an independent air actuator.

2. Using a position sensor which would not indicate a properly picked up load unless both hooks engage.

3. Preventing the hoist from lifting any load of more than 500 pounds unless both hooks are completely closed.

DCR No. 745 Fire Protection System

Description of Change: The previously existing 4 inch Viking alarm valve was replaced with a 6 inch Viking alarm valve.

Reason for Change: The change was made to reduce the pressure demand of the fire pumps in order to meet tech specs.

Safety Evaluation: This system was not required for safe shutdown of the plant. The change did not alter the original safety analysis.

DCR No. 753 Recirculation System

Description of Change: Capacitors were installed across each MVI converter input for recirc. system speed control loops.

Reason for Change: The change was made to filter the noise picked up in the recirc system speed control loops as recommended by General Electric.

Safety Evaluation: The affected components were not safety related items. No safety function was involved with this design change. No unreviewed safety question was created by this design change. The addition of the capacitors improved the stability and reliability of the subject speed control loops.

DCR No. 760A Pump House Hatch Enlargement/RHR Strainer Installation

Description of Change: The pump house hatch, PR-3, was enlarged. An increase of 4½" on each side of the openings narrow dimension was made.

Reason for Change: DCR 760 provided for the installation of new RHR and ESW service water strainers. Because the new RHR service water strainer was larger than any previously existing entrance way into the pumphouse, there existed a requirement to enlarge one of the entrance ways into the pumphouse. This requirement was met by enlarging the pumphouse hatch, PR-3.

Safety Evaluation: The change did not present significant hazards or considerations not described or implicit in the Safety Analysis Report. The purpose of the change was to allow the transfer of replacement RHR service water strainers into the pumphouse. The enlarged hatch design was engineered to assure this section of the pumphouse would remain a seismic Class I structure as stated in the Safety Analysis Report.

DCR No. 766 Drywell Sump Level Switches

Description of Change: All low current capacity reed switches in the drywell sump level indication circuitry have been replaced with higher current capacity "pip" switches.

Reason for Change: The changes were made due to frequent malfunctions of the level indication circuitry.

Safety Evaluation: The change had no effect on the safe shut down of the plant and does not increase the possibility of a hazardous condition to the plant.

DCR No. 800A RPV Recirculation Inlet Line Nozzle Safe End Replacement

Description of Change: This design change provided the cutting sequence and dimensional tolerances necessary to remove the previously existing eight recirculation inlet nozzles safe ends, along with a section of the previously existing thermal sleeves. This modification consisted of making three cuts on each of the previously existing nozzle safe end/pipe sections. The following pipe cuts were weld prepped:

1. Weld prep at the nozzle stub
2. Weld prep at the existing thermal sleeve
3. Weld prep nozzle side of safe end extension piece
4. Weld prep riser elbow
5. New thermal sleeve adapter prep
6. The riser pipe elbow side of the section of pipe containing the new extension piece

Reason for Change: This modification was made to permit the installation of new safe ends and adapters to the existing thermal sleeves.

Safety Evaluation: Since the fuel was stored in the spent fuel pool, the following conclusions were drawn based on the analysis given supra.

Safety analysis for normal reactor operation was not necessary.

The nozzle safe end repair did not create the possibility for an accident or malfunction of a different type than that evaluated previously in the FSAR. The possibility of the occurrence of an accident or malfunction previously analyzed in the FSAR was not increased. The severity of the consequences of any accident or malfunction or equipment previously analyzed in the FSAR was not increased.

The probability of increased corrosion due to the additional air in the reactor vessel was evaluated and it was concluded that there were no adverse effects.

Any possibility of damage to the reactor internals or the vessel flange surface by foreign matter was precluded by the replacement of the vessel head.

Therefore there was no unreviewed safety questions concerning the effect of the repair outage and no adverse effect on the health and safety of the public.

Section B

SPECIAL TESTS

This section has been prepared in accordance with the requirements of 10 CFR, Part 50.59(b)

B. SPECIAL TESTS

This section contains summaries of those special tests conducted at the plant during the calendar year 1978. The Special Test Procedures, which governed the performance of these tests, were reviewed by the DAEC Operations Committee and were found not to present any unreviewed safety questions.

SpTP No. 44 Vessel Draining Procedure for Radiation Level Measurements

The purpose of this procedure was to provide a method for draining the reactor vessel after all fuel was unloaded, in order to obtain radiation measurements in the drywell and refueling floor at predetermined vessel water levels. Radiation measurement data was used to verify proper shielding design for the recirculation system inlet nozzle safe end replacement program.

This Special Test was performed on July 20 and 21.

SpTP No. 45 Reactor Vessel Water Level Control During Recirculation Return Nozzle Safe End Repair

The purpose of this procedure was to define a method of maintaining the reactor vessel water level at a predetermined value during the recirculation system inlet nozzle safe end replacement program.

This Special Test was initiated on August 14 and was in continuous use until December 16.

SpTP No. 46 Flushing Following Recirc System Pipe Repair

This purpose of this procedure was to flush the reactor recirculation loops in accordance with CCP-1 after completion of weld repair in the Reactor Vessel Recirculation System. This procedure provided permanent documentation for this portion of the work.

This Special Test was performed on December 31.

Section C

EXPERIMENTS

This section has been prepared in accordance with the requirements of 10 CFR, Part 50.59(b)

C. EXPERIMENTS

During the calendar year 1978, there were no experiments reportable pursuant to the requirements of 10 CFR, Part 50.59(b).

IOWA ELECTRIC LIGHT and POWER COMPANY

Duane Arnold Energy Center

1977

Information for 1977 Annual Operating Report

Section A

NARRATIVE SUMMARY

This section has been prepared in accordance with the requirements of Appendix A to Operating License DPR-49 for the Duane Arnold Energy Center.

A. Narrative Summary

This section contains a chronological narrative summary of plant operations for the calendar year 1977.

1/1 At the beginning of this report period, the plant was operating at 277 MWe. Reactor power was being decreased in preparation for performing a rod sequence exchange.

1/6 Fuel preconditioning at rated flow was completed, and the plant was operating at 463 MWe.

1/7 During normal operation, the "B" drywell particulate monitor failed downscale as the result of condensate in the detector. The condensate was removed and the monitor was returned to service, and a design change was initiated. (RO 77-3)

1/9 The "B" drywell particulate monitor again failed downscale. Repair was made and the unit was returned to service. (RO 77-4)

1/10 The voltage regulator on the uninterruptible MG set was repaired and the uninterruptible MG set was returned to service.

1/10 Commenced load decrease from 455 MWe to permit repair of a faulty alarm card for TIS 4443-1, -2, and -4. Following the repair, the plant load was increased again.

1/11 The plant load was increased to 456 MWe.

1/12 The valve time for main steam line drain isolation valve MOV 4423 failed to meet the Technical Specifications requirement of less than 15 seconds. A Deviation Report was initiated.

1/12 During the performance of STP 42A004, 12 of 16 main steam line high flow PDIS's were found to be out of calibration. As a result, the requirement to have at least two operable channels per steam line was not met on three main steam lines. The instruments were recalibrated. (RO 77-5)

1/14 After notifying the Control Center, the plant load was reduced to 328 MWe.

1/15 Commenced fuel preconditioning ramp and increased the plant load to 462 MWe.

1/17 A low flow alarm was received from the offgas stack radiation monitors. A subsequent investigation revealed that the exhaust line from the sampler was frozen. The line was thawed out and heat-traced, and the monitors were source-checked and returned to service. (ETSV 77-1)

1/18 An inspection of the HPCI room revealed that hydraulic snubber HBB-6-SS-22 on the HPCI turbine exhaust line had a broken piston and that hydraulic snubber GBD-29-SS-12 on the auxiliary boiler-to-HPCI line had broken loose from its stationary mounting point. The HPCI system was declared inoperable while HBB-6-SS-22 was replaced and GBD-29-SS-12 was repaired. (RO 77-6)

1/19 The inspection of the HPCI system piping and snubber repairs was completed. The quarterly surveillance test to demonstrate operability of the HPCI system was satisfactorily run and the system was declared operational again.

1/22 The plant load was reduced, by adjusting recirculation flow, to 407 MWe to permit CRD exercise.

1/23 Following CRD exercise, the plant load was returned to 463 MWe.

1/28 Cooling tower fans "P" and "K" tripped due to high vibration levels.

1/29 All of the cooling tower fans vibration switches were reset.

1/29 As the result of a gas leak on the truck and the accumulation of fumes in the airlock, the truck, loaded with radioactive material, was removed from the Reactor Building railroad airlock to outside of the Reactor Building.

1/31 During an inspection, hydraulic snubber HBB-25-SS-178 on the fuel pool-to-RHR line was found to be pulled out from the wall. One concrete fastener was replaced and other necessary repairs were made to return the snubber to an operable condition. (RO 77-8)

2/1 The plant was operating at 450 MWe.

2/1 A high stator inlet temperature alarm was received. An investigation of the problem revealed that temperature controller TC 3607 was oscillating, causing rapid fluctuations in the stator cooling water temperature and an above-normal stator inlet temperature. The system was realigned and control valve CV 3607 was failed open to permit adequate cooling water flow until repairs could be made.

2/2 The stator cooling system was returned to normal operation.

2/2 The plant load was reduced to 328 MWe to perform CRD exercise and rod pulls.

2/3 The CRD exercise and rod pulls were completed, and the fuel preconditioning ramp was commenced.

2/4 The plant output was increased to 419 MWe, at which point, further increases in the reactor power were delayed pending replacement of a bad detector in the No. 1 tip machine.

2/4 The "B" RHR pump was placed in operation to permit an Engineering inspection of the performance of the hydraulic snubber HBB-25-SS-178.

2/7 A spurious signal from the vent shaft radiation monitor was received, causing secondary containment isolation to occur. The trip was reset and the SBGT system secured.

2/7 The No. 1 tip machine was repaired and commenced increasing reactor power at the fuel preconditioning rate.

2/8 The plant load was increased to 466 MWe.

2/8 345 KV substation breakers 0710 and 4730 tripped as the result of a neutral overcurrent and were later reset.

2/11 During the performance of STP 42B010, the "A" diesel generator and the "A" core spray pump were inadvertently started. Due to a leaking bypass test valve, the "A" core spray system piping was not completely filled, and the resulting water hammer caused the clutch housing of MOV 2115 to fracture. The "A" core spray system was determined to be inoperable pending a full evaluation of the system integrity. (RO 77-9)

2/13 A buildup of ice caused the roof of the "A" Cooling Tower Breaker House to collapse. Measures were taken to protect the electrical equipment housed in the building.

2/17 During the performance of STP 42B012, the "B" ADS timer setting exceeded its allowable range. The timer was recalibrated and functionally tested. (RO 77-11)

2/17 STP 45A001 was performed on the "A" core spray system with satisfactory results and the system was declared operable.

2/19 During surveillance testing, three hydraulic snubbers in the RHR suction line were found to have missing or loose concrete anchors. The snubbers were repaired and a check of the other snubbers outside of the containment was initiated. (RO 77-13)

2/20 During the inspection of the other snubbers outside of the containment, five additional snubbers were found to have loose anchors or anchor bolts. These snubbers were repaired and a design review was initiated to investigate the cause of the problem. (RO 77-13)

2/25 Well pump 1P-58C was secured because its motor was running hot, and well pump 1P-58B was placed in service.

2/26 During reduction of the plant load for CRD exercise, the No. 3 tip machine malfunctioned. An investigation revealed that the cable had slipped from the reel and had two severe bends in it. Repair activities were initiated.

2/26 A new clutch housing was installed on MOV 2115 and the valve was functionally tested.

2/27 Repair activities on the No. 3 tip machine were completed.

3/1 The plant was operating at 445 MWe.

3/1 Further repairs were made on the No. 3 tip machine and the tip machine was declared operable.

3/3 The 24 v DC and 250 v DC undervoltage relays were replaced and functionally tested to clear a previously reported alarm indication problem. (RO 77-7)

3/7 During the performance of STP 42B029, the HPCI test switch failed to give an alarm on 1C-03. This problem was the result of a bad connection on the annunciator card.

3/8 The site fire alarm proved to be unsatisfactory during a test of the site fire and evacuation alarms. A Deviation Report and Maintenance Action Request were initiated to correct the problem.

3/9 The problem with the site fire alarm was corrected and both the site fire and evaluation alarms were satisfactorily tested.

3/11 Commenced power reduction from 450 MWe at 1150 hours in preparation for shutdown for refueling.

3/11 During the performance of STP 47E001, MOV 8401 A did not completely close. In addition, temperature element TE 8416A in the "A" MSIV leakage control system failed downscale, rendering that portion of the MSIV leakage control system inoperable. (RO 77-19)

3/12 MOV 8401A and those valves in the "A" portion of the MSIV leakage control system were manually closed.

3/12 The in-plant electrical loads and all 4160 v switchgear were shifted to the startup transformer at 0221 hours.

3/12 At 0308 hours, the reactor power had been reduced to ten percent.

3/12 The turbine was tripped at 0343 hours.

3/12 The turbine was placed on the turning gear at 0528 hours.

3/12 All control rods were fully inserted at 1121 hours.

3/13 The removal of the drywell head from the drywell was completed at 1700 hours.

3/14 The turbine was disengaged from the turning gear and all lubricating oil systems were secured at 0745 hours.

3/15 Removal of the reactor head from the RPV was started at 0638 hours.

3/15 Removal of the steam dryer assembly from the RPV was started at 1034 hours.

3/15 Flooding of the refueling cavity was completed at 1553 hours.

3/15 The plant mode was switched from cold shutdown to refuel at 2110 hours.

3/16 The initial movement of fuel was started at 1457 hours.

3/18 The initial movement of fuel was completed at 1534 hours and the draining of the refueling cavity was commenced. The plant was returned to the cold shutdown mode.

3/19 The RPV service platform and the steam line plugs were installed at 0630 hours.

3/22 During an in-service ultrasonic inspection of the "A" recirculation loop bypass pipe, a possible small hairline crack on the inner surface of the pipe-to-elbow weld was indicated. This problem was referred to Engineering to determine a method for repairing the weld. (RO 77-22)

3/24 The movement of fuel was resumed at 2150 hours.

3/26 The second movement of fuel was completed at 2358 hours.

3/27 During an inspection of the drywell, thirteen mechanical snubbers were found to be "locked up" and were replaced with like-for-like spares. In addition, a visual inspection of those hydraulic snubbers listed in Tables 4.6.3 and 4.6.4 of the Technical Specifications noted eleven units as being inoperable as the result of a loss of oil in their accumulators. Subsequent testing yielded fifteen additional hydraulic snubbers which failed to meet the specified performance criteria. Investigations into the causes of these problems were initiated. (RO 77-23, 77-24)

3/28 During an inspection of the suppression chamber, the "A" RHR system suppression chamber suction strainer was found to be torn and deformed. Measures to fabricate a replacement were initiated. (RO 77-25)

4/1 The plant remained in the cold shutdown mode.

4/4 During the performance of sensor checks, the Reactor Building stack monitor sample pumps were found to be not operating. The sample pumps had been removed from service for breaker maintenance and did not automatically restart when the power was restored. (RO 77-33)

4/6 The plant was switched to the refuel mode.

4/7 The plant was returned to the cold shutdown mode at 2013 hours, and again switched to the refuel mode at 2051 hours.

4/9 Following two mode switches between the cold shutdown and refuel modes, the plant was returned to the cold shutdown mode at 2031 hours.

4/10 The plant was switched to the refuel mode.

4/11 The plant was switched between the refuel and cold shutdown modes four times.

4/12 The plant was returned to the cold shutdown mode at 1641 hours and again switched to the refuel mode at 2121 hours.

4/13 The plant was returned to the cold shutdown mode at 1658 hours and again switched to the refuel mode at 2026 hours.

4/16 The general service water, electric fire, diesel fire, and jockey fire pumps were removed from service to permit divers to enter the circulating water pit for mud removal. Fire watches were maintained until the circulating water pit was cleared and the various pumps were returned to service.

4/17 The plant was placed in the cold shutdown mode while measures were taken to improve the water clarity in the refueling cavity.

4/18 The plant was switched to the refuel mode.

4/18 During surveillance testing, radiation monitor RC 4448D did not respond properly to a test source. The main steam line radiation monitor was repaired, functionally tested, and returned to service. (RO 77-35)

4/19 The plant was returned to the cold shutdown mode.

4/20 The plant was switched between the refuel and cold shutdown modes three times.

4/20 The LPRM replacement was completed.

4/21 The general service water and fire pumps were removed from service to permit draining of the circulating water pit. The various pumps were returned to service following the completion of the work in the circulating water pit.

4/21 The plant was returned to the cold shutdown mode.

4/22 The plant was placed in the refuel mode at 0713 hours and then returned to the cold shutdown mode at 1806 hours. The plant was again placed in the refuel mode at 2107 hours.

4/23 The plant was returned to the cold shutdown mode.

4/26 The drilling of fuel assemblies was commenced at 0500 hours.

4/27 The hydrostatic test of the weld repair made on the "A" recirculation loop bypass pipe was satisfactorily completed.

4/29 The plant was placed in the refuel mode to complete surveillance testing and then was returned to the cold shutdown mode.

4/30 The drilling fuel assemblies was completed at 0415 hours.

5/1 The plant remained in the cold shutdown mode.

5/2 The plant was switched to the refuel mode and then was returned to the cold shutdown mode. The plant was later switched to the refuel mode to complete loading of the core.

5/3 At 1716 hours, following the completion of core loading, the plant was returned to the cold shutdown mode. The remaining control rod checks were delayed while the SBLC concentration was brought into specification.

5/3 At 2147 hours, the plant was switched to the refuel mode in preparation for performing core verification.

5/4 Core verification was completed at 0500 hours, and the plant was returned to the cold shutdown mode at 0727 hours.

5/4 The SBLC concentration was brought into specification and the remaining control rod checks were completed. At 1909 hours, the plant was placed in the refuel mode.

5/5 The steam separator was installed in the RPV at 0230 hours, and the steam dryer was installed in the RPV at 0620 hours. Friction testing was commenced at 0710 hours.

5/6 Friction testing was completed at 0638 hours.

5/6 At 0705 hours, the plant was returned to the cold shutdown mode to perform surveillance testing, and then was switched to the refuel mode at 1253 hours. The plant was again returned to the cold shutdown mode at 1603 hours.

5/7 The reactor head was placed upon the RPV, but not tensioned down, at 0530 hours.

5/7 A fire occurred in tie breaker B52-3401, but the damage was confined to the breaker and its cubicle. (RO 77-36)

5/8 At 1135 hours, the reactor head tensioning was completed.

5/9 At 0944 hours, the reactor vessel operational leakage test was commenced.

5/9 At 1835 hours, the turbine was placed on the turning gear.

5/9 At 2042 hours, the reactor vessel operational leakage test was satisfactorily completed.

5/10 At 0120 hours, the drywell head was set in place.

5/10 At 0830 hours, the turbine was disengaged from the turning gear, and the lift pumps and EHC system were subsequently secured.

5/10 During the performance of STP 48A002, the standby diesel generator 1G-21 failed to supply power to bus 1A4 when required. It was determined that two contacts in the standby transformer to bus 1A4 breaker 152-401 did not close to permit the standby diesel generator 1G-21 breaker 152-411 to close, and this problem was corrected. (RO 77-37)

5/11 At 1413 hours, the plant was placed in the startup mode, and then was returned to the cold shutdown mode at 1500 hours. At 1627 hours, the plant was placed in the refuel mode, and then was switched to the startup mode at 1743 hours to perform the pre-startup surveillance package.

5/12 During the performance of STP 48A001, the standby diesel generator 1G-31 failed to pick up its rated load of 2850 KW. An adjustment was made to its governor and the standby diesel generator subsequently demonstrated satisfactory performance. (RO 77-43)

5/12 Tie breaker B52-3401 was functionally tested and performed satisfactorily. Following resolution of the problem with the seismic suitability of the replacement breaker, the reactor startup was resumed.

5/12 At 1830 hours, the reactor was critical.

5/14 At 0203 hours, the turbine-generator was placed on line, and the plant was subsequently placed in the run mode at 0204 hours.

5/15 As a result of an inadequate nitrogen supply available in storage, the inerting of the torus was not completed within 24 hours after going to the run mode. An orderly shutdown was initiated, however, this operation was secured when a liquid nitrogen shipment was received and the inerting of the torus was completed. (RO 77-38)

5/17 After pulling the control rods to achieve an approved rod pattern, an MCPR violation was found to have occurred. The control rods were inserted to bring the MCPR back within the limits specified by the Technical Specifications (RO 77-41)

5/18 After pulling the control rods to achieve an approved rod pattern, an MCPR violation was found to have occurred. The control rods were inserted to bring the MCPR back within the limits specified by the Technical Specifications. (RO 77-42)

5/20 A faulty control power relay in the breaker for the RCIC steam supply valve MOV 2404 rendered the valve inoperable, and as a result, the RCIC system was declared inoperable.

5/21 The breaker for MOV 2404 was repaired and the RCIC system was declared operable.

5/21 At 2250 hours, an orderly load reduction from 459 MWe was commenced to permit taking the turbine off line to investigate and repair the No. 9 bearing on the exciter which was exhibiting high vibration levels.

5/22 The turbine-generator was taken off line at 0719 hours.

5/22 At 0748 hours, the plant was switched to the startup mode.

5/22 At 0851 hours, while raising the reactor water level by manually regulating the feedwater flow with the block valves, a cold water transient caused a high flux reactor scram. (RO 77-53)

5/22 At 1036 hours, the turbine was disengaged from the turning gear to commence repair activities.

5/22 Following completion of the repairs, the turbine was placed on the turning gear at 1145 hours.

5/22 At 2250 hours, reactor startup was commenced.

5/23 At 0013 hours, the reactor was critical.

5/23 At 0324 hours, the plant was switched to the run mode.

5/23 At 0332 hours, the turbine was rolled. Due to high vibration levels in the No. 9 bearing, the turbine was tripped to permit reworking of the generator-exciter coupling.

5/23 At 0520 hours, the plant was switched to the startup mode.

5/23 At 1649 hours, the turbine was placed on the turning gear.

5/23 At 2242 hours, the plant was placed in the run mode and the turbine was rolled.

5/23 The turbine-generator was placed on line at 2310 hours.

5/27 With the plant operating at 458 MWe, a reactor shutdown was commenced at 0954 hours to investigate unidentified leakage in the drywell.

5/27 The turbine-generator was taken off line at 1318 hours.

5/27 At 1442 hours, while manually regulating the feedwater flow with the block valves, a cold water transient caused a high flux reactor scram (RO 77-53A)

5/27 The plant was switched to the refuel mode at 1711 hours to perform surveillance testing.

5/28 At 0109 hours, following the replacement of a seal in the "B" feedwater check valve, the plant was placed in the startup mode and reactor startup was commenced.

5/28 At 0249 hours, the reactor was critical.

5/28 At 0528 hours, the plant was placed in the run mode, and the turbine-generator was placed on line at 0635 hours.

5/28 During the performance of STP 43A002, 22 CRD accumulator alarms were received. Investigation of the problem revealed that a fuse in the accumulator panel had blown. Replacement fuses also opened until a ground short in the level switch on CRD 30-27 was discovered and repaired. The reactor power was maintained at a constant level until the problem had been corrected.

6/1 The plant was operating at 459 MWe.

6/4 The plant load was reduced to 272 MWe to permit General Electric personnel to conduct turbine vibration testing. After maintaining load at this level for a period of 45 minutes, the plant load was increased to 425 MWe. A second load reduction to 278 MWe was executed to permit additional vibration testing. The load was then increased to 425 MWe and the vibration testing was completed. The load was subsequently reduced to perform CRD exercise and rod pulls.

6/4 Following the performance of the CRD exercise and rod pulls, the plant load was increased and the No. 2 tip machine began to malfunction.

6/5 After pulling the control rods to achieve an approved rod pattern and then maintaining the reactor power level while the repairs were being made to the No. 2 tip machine, an MCPR violation was found to have occurred. The control rods were inserted to bring the MCPR back within the limits specified by the Technical Specifications. (RO 77-49)

6/5 Commenced fuel preconditioning ramp and increased reactor power.

6/7 The detector in the No. 2 tip machine was replaced and the No. 2 tip machine was returned to service.

6/10 The plant load was reduced from 462 MWe by recirculation flow to permit the performance of condenser leak testing.

6/11 The plant load was reduced to and maintained at 300 MWe, however, the plant load was later increased to permit closure of the hi-load valves.

6/12 As MOV 4151 was being adjusted to regulate the offgas flow, the loop seal at CV 1379 was blown out, resulting in a high Reactor Building stack release. This condition remained undiscovered for approximately two hours and a possible violation of the Environmental Technical Specifications was indicated. However, when the release calculations were checked and revised, it was determined that a violation had not occurred.

6/12 Both drywell oxygen analyzers failed and were subsequently repaired without adversely affecting plant operation.

6/12 The condenser leak testing was completed and the plant load was subsequently increased. No significant leaks were found.

6/13 During the performance of STP 42B022, the RCIC steam line high flow instrumentation, PDIS 2441 and PDIS 2442, tripped at valves exceeding the limits specified by the Technical Specifications. The switches were recalibrated and functionally tested. (RO 77-52)

6/15 The HPCI system was taken out of service to perform maintenance on the turbine governor.

6/19 The plant load was reduced from 454 MWe to 325 MWe to perform CRD exercise and OD-002. The plant load was then increased again.

6/19 It was reported that the motor windings on the HPCI recirculation valve MOV 2318 were burned. The motor was subsequently removed and sent offsite for repair.

6/20 The motor from HPCI-to-condensate storage tank recirculation valve MOV 2316 was removed and installed on MOV 2318. MOV 2316 was left in the fully-open position, and its breaker was racked out.

6/21 The HPCI system was returned to service and demonstrated operable by STP 45D001.

6/21 The "C" well pump tripped. Upon restarting the pump, it exhibited excessively high levels of vibration and there was no flow indication. It was suspected that the pump had a broken shaft and arrangements were made to pull and inspect the pump.

6/21 The inspection of the snubbers per STP 46H001 was started.

6/23 The "C" well pump was repaired and returned to service.

6/27 The inspection of the snubbers per STP 46H001 was completed.

6/28 During the performance of STP 47A006, the pressure differential switches which control the torus-to-reactor building vacuum breakers tripped at values exceeding the limits specified by the Technical Specifications. PDS 4304 and PDS 4305 were recalibrated and functionally tested. (RO 77-55)

6/30 Smoke was detected coming from a lighting transformer in the Control Room. The transformer was deenergized and the fire extinguished. The transformer was then disconnected and the Control Room lighting was restored.

7/1 The plant was operating at 467 MWe.

7/1 Due to leaks at the dispensers and the apparent lack of flow, the chlorination system was taken out of service.

7/2 The chlorination system was repaired and returned to service.

7/4 The plant load was reduced from 466 MWe to 358 MWe to perform CRD exercise.

7/4 The seal water line to the "B" RWCU pump seal water cooler was found to be broken. The pump was secured and the reactor water cleanup system was isolated. The "A" pump, which had been disassembled, was reassembled and placed in service.

7/4 Following the RWCU system repair, the plant load was increased to 464 MWe.

7/4 As a result of an SJAE loop seal being blown out, above-normal radiation levels were detected in the main stack. The radiation was reduced to a normal level when MOV 1379 was closed.

7/5 While investigating a high vibration alarm on reactor feed pump 1P-1A, it was noted that the pump motor was running hot. The motor temperature continued to increase and measures were taken to provide additional cooling to the motor. In order to remove 1P-1A from service, a reactor power reduction to 60 percent was commenced.

7/6 1P-1A was removed from service, and the motor cooler was backwashed and the oil cooler was cleaned. The pump was returned to service and the plant load was increased to 465 MWe.

7/8 The "A" RWCU pump tripped and would not restart. Since the "B" RWCU pump was also out of service, the cleanup flow was routed through the "A" pump to the main condenser.

7/13 During the performance of STP 42B017, temperature switch TS 2742B failed, isolating the RWCU system. The switch was repaired and flow was reestablished in the system.

7/13 The RCIC system was isolated to repair and recalibrate the RCIC steam line high flow switches, PDIS 2442 and PDIS 2441, one of which was found to be malfunctioning during the performance of STP 42B022. Following the completion of the repairs, the RCIC system was demonstrated operable and returned to service. (RO 77-56)

7/14 The river water temperature exceeded 89.9^{O}F for a period of approximately 4 hours and 20 minutes. Surveillance testing had previously determined that the "A" ESW pump was inoperable at river water temperatures greater than 89.9^{O}F, and the "A" ESW system was secured. The "B" ESW system remained operable. The "A" ESW system was returned to service following a drop in the river water temperature. (RO 77-57)

7/15 The river water temperature again rendered the "A" ESW system inoperable for a period of approximately 4 hours. (RO 77-62)

7/16 During surveillance testing, supply breaker 1B3401 for essential bus 1B34A tripped when the hand switch for MOV 2004 was closed. Since an overcurrent trip had occurred, breaker 1B4401 from the redundant power supply did not close automatically to pick up 1B34A and 1B44A, thus rendering the MSIV-LCS system and both LPCI loops inoperable. The breaker trip setpoints were found to be incorrect and were corrected to permit completion of STP 45A002. (RO 77-58)

7/18 During the performance of STP 47E001, MSIV-LCS MOV 8401C failed to open. Its breaker was found to have tripped on thermal overload, and was subsequently reset. The valve then operated properly, but the breaker again tripped upon valve closure. The valve torque switch was adjusted and the valve operated properly. (RO 77-79)

7/18 As a result of the plant being struck by lightning, the computer, "C" well flow indication, and the offgas stack flow indication were lost. The computer was reinitialized and a Deviation Report was filed on the offgas stack flow recorder FR 4133. (ETSV 77-3)

7/19 The offgas stack flow recorder FR 4133 was repaired and returned to service.

7/19 While investigating an EHC sump low level alarm, a severe oil leak was discovered on the No. 3 intercept valve. A plant load reduction was commenced to repair the leak, but the leak was reduced to a drip, and the plant load was then maintained at 425 MWe.

7/20 The plant load was increased to 471 MWe.

7/22 At 0000 hours, a plant load reduction was commenced in preparation for performing rod sequence exchanges.

7/22 At 0550 hours, with the reactor at a 45 percent power level the rod sequence exchanges were commenced.

7/22 During surveillance testing, the "B" ADS timer setpoint was found to have exceeded the limits specified by the Technical Specifications. The timer was replaced with a like-for-like spare. (RO 77-59)

7/22 At 1330 hours, the rod sequence exchanges and control rod pulls were completed, and an increase in plant load was commenced.

7/26 At 0640 hours, the plant was operating at 494 MWe.

7/26 During the performance of STP 42A005, one subchannel of the main steam line area high temperature switch TIS 4445 tripped at a value which exceeded the limits specified by the Technical Specifications. The subchannel was recalibrated and functionally tested. (RO 77-61)

7/29 The offgas stack monitor vacuum pump was found to be inoperable, hence, the offgas stack effluent was not being sampled. The vacuum pump was later repaired and returned to service. (ETSV 77-4)

8/1 The plant was operating at 491 MWe.

8/3 At 1521 hours, during surveillance testing, a technician disturbed the reactor high pressure switches for A2 and B2 causing an automatic scram to occur.

8/3 At 1629 hours, the "A" recirculation pump motor high/low oil level alarm was received. The drywell was de-inerted, and entry was made to check the motor and add oil to the upper motor bearing.

8/4 The plant was placed in the startup mode and, at 0303 hours, the reactor was critical.

8/4 At 0527 hours, the plant was switched to the run mode.

8/4 At 0550 hours, the turbine-generator was placed on line.

8/4 During the performance of STP 42G001, reactor high pressure recirculation pump trip pressure switch PS 4593D tripped at a value which exceeded the limits specified by the Technical Specifications. The switch was recalibrated and functionally tested. (RO 77-63)

8/6 The CAD system nitrogen volume was found to have decreased to a level below the minimum volume specified by the Technical Specifications. The nitrogen vendor was unable to resupply the system in time to prevent the system volume from dropping below the minimum required volume of 50000 SCF. (RO 77-64)

8/8 The CAD system nitrogen volume was restored.

8/13 At 2257 hours, a reduction in plant load from 528 MWe was commenced in preparation for performing CRD exercise and installing the MSR instrumentation package.

8/14 At 1936 hours, installation of the MSR instrumentation package was completed, and a reactor power increase from 62 percent was started.

8/15 During surveillance testing, the suppression pool area high ambient temperature switch TS 2526B tripped at a value significantly lower than its specified setpoint. The switch was recalibrated and functionally tested. (RO 77-66)

8/16 The torus water temperature recorders, TR 4386A and TR 4386B, were found to have readings which differed by more than 5° F, and a Deviation Report was subsequently initiated.

8/22 At 2230 hours, the plant load was reduced from 530 MWe to 230 MWe due to problems with maintaining condenser vacuum. The vacuum was maintained above the trip point while an investigation was made. At 2240 hours, the condenser vacuum was restored and the reactor power was increased.

8/24 During the daily instrument checks, the torus water temperature transmitter, TT 4324, was found to be indicating erratically and was replaced with a like-for-like spare. The redundant temperature transmitter, TT 4325, was found to be drifting and was recalibrated. (RO 77-67)

8/27 The exhaust valves on the "B" air compressor and a solenoid valve on the "C" air compressor were replaced. Also, a leaking nipple on the "C" air compressor was repaired. Both compressors were then returned to service.

8/28 During an operability test of the "A" standby filter unit, the supply fan, 1V-SF-30A, was found to have a loose impeller, which rendered the unit inoperable. The fan was repaired and returned to service. (RO 77-76)

9/1 The plant was operating at 500 MWe.

9/2 During the performance of STP 41A004, the IRM channel "A" downscale trip was found to be inoperable. The channel was repaired, functionally tested, and returned to service. (RO 77-73)

9/2 Following replacement of the charcoal filters, the "B" standby filter unit was returned to service.

9/2 At 1620 hours, an orderly shutdown from 505 MWe was commenced in preparation for performing an inspection of the hydraulic snubbers in the drywell.

9/2 At 2339 hours, the reactor was manually scrammed.

9/3 At 0708 hours, the hydraulic snubber inspection per STP 46H001 was started.

9/3 At 0742 hours, the plant was placed in the cold shutdown mode.

9/4 During the inspection, five hydraulic snubbers were found to be inoperable. The failures were attributed to leaking seals which resulted in low oil levels in the snubber accumulators. All five hydraulic snubbers were replaced with mechanical snubbers. (RO 77-70)

9/5 The plant was placed in the hot shutdown mode at 1710 hours, and then was switched to the startup mode at 1832 hours.

9/6 At 0035 hours, the reactor was critical.

9/6 Following repairs on the offgas inlet valve, MOV 4151, which failed to open, the plant was placed in the run mode at 1015 hours, and turbine roll was commenced.

9/6 Due to an inability to maintain condenser vacuum, the reactor power was reduced to avoid a reactor scram. The reactor power was increased again at 1800 hours following an improvement in the condenser vacuum.

9/11 At 0826 hours, the reactor scrammed as the result of an instantaneous 10% increase in flow demand on the "A" recirculation pump M-G set. The step increase occurred as the M-G set scoop tube, which had been locking up, was being reset. The plant was subsequently placed in the hot shutdown mode.

9/11 At 1036 hours, the plant was placed in the startup mode.

9/11 At 1605 hours, the reactor was critical.

9/11 At 1858 hours, the plant was switched to the run mode, and at 1915 hours, the turbine-generator was placed on line.

9/11 While experiencing problems with the offgas system, the turbine tripped at 2000 hours due to a loss of condenser vacuum.

9/12 The condenser vacuum improved as the hotwell temperature was reduced, and the turbine-generator was again placed on line at 0107 hours.

9/12 The "A" recirculation pump M-G set scoop tube was removed from service to investigate and repair the control problem.

9/14 The "A" recirculation pump M-G set scoop tube was reset and the plant load was subsequently increased from 211 MWe.

9/15 During the performance of STP 41A004, the IRM channel "B" downscale trip was found to be inoperable. The channel was repaired, functionally tested, and returned to service. (RO 77-75)

9/17 As the result of a broken detector, the No. 3 tip machine failed.

9/17 As the result of a severe electrical storm, the 125 v DC system experienced problems, the "A" well and stack dilution flow indications were lost, and the computer failed. Lightning also started a fire at the transformer at the sewage plant. The fire was subsequently extinguished by the heavy rains.

9/18 The detector on the No. 3 tip machine was replaced, and the tip machine was returned to service. As a result of problems with the index channel reference on the No. 2 and No. 3 tip machines, the plant load was maintained at 350 MWe until repairs could be made.

9/19 The No. 2 and No. 3 tip machines were repaired and returned to service, and the plant load was increased.

9/21 During the performance of STP 45D001, the HPCI turbine would not develop sufficient speed for the HPCI pump to deliver rated flow and discharge pressure. As the result of an excessive oil demand for the turbine and pump bearings, insufficient oil pressure was developed to open the turbine stop valve enough. The HPCI system was declared inoperable. (RO 77-77)

9/21 The offgas stack flow recorder, FR 4133, was found to have a bad transmitter. The flow recorder was repaired and returned to service. (ETSV 77-5)

9/21 The No. 1 tip machine malfunctioned again. As a result, the plant load was maintained at 484 MWe until repairs could be made.

9/22 At 1000 hours, the main transformer differential relay tripped during a phase fault relay installation causing a generator lock out and reactor scram to occur.

9/22 At 1330 hours, the plant was placed in the startup mode and the reactor startup was commenced.

9/22 At 1451 hours, the reactor was critical.

9/22 The plant was placed in the run mode at 1708 hours, and the turbine-generator was placed on line at 1735 hours.

9/23 The No. 1 tip machine was repaired and returned to service.

9/23 The HPCI system was repaired and demonstrated operable.

9/26 During the performance of STP 45D001, the HPCI turbine again failed to develop sufficient speed upon a fast start. The HPCI system was again declared inoperable.

9/27 In the course of an unrelated inspection, damage was noted on the HPCI-to-RHR piping insulation. This damage apparently resulted from the hydraulic shock in the HPCI steam supply piping which occurred on 6/11/74. Subsequent engineering evaluations and non-destructive examinations of the affected line sections revealed no adverse effects. (RO 77-78)

9/28 The HPCI system was demonstrated operable in accordance with the provisions of STP 45D001.

9/30 The Control Building emergency air supply fan 1V-SF-30A was found to be operating with high levels of vibration and noise. The fan was secured, and an investigation revealed that the fan impeller had moved from its normal position on the shaft. (RO 77-76)

10/1 The plant was operating at 505 MWe.

10/3 Repairs were completed on 1V-SF-30A, and following a ten-hour operability demonstration as prescribed by STP 47B008, the fan was returned to service.

10/6 During surveillance testing, the RWCU system return line isolation valve MOV 2740 failed to close within the ten-second time limit specified by the Technical Specifications. The valve-open limit switch was adjusted and the valve was satisfactorily tested. (RO 77-83)

10/6 While investigating a low level alarm on the standby diesel generator fuel storage tank, 1T-35, the direct reading level indicator was determined to be incorrectly calibrated due to incorrect as-built design data being used in the initial instrument calibration. The level indicator was subsequently recalibrated and other corrective actions were initiated. (RO 77-80)

10/11 During normal operation, the "A" standby filter unit did not automatically start when the outside air inlet temperature dropped below 38° F. A subsequent check indicated that the unit would not have automatically started upon a high radiation signal. Flow transmitter FT 7120 was found to be out-of-calibration which prevented the permissive start signal. The flow transmitter for the "B" unit, as well as that for the "A" unit, was recalibrated. (RO 77-82)

10/14 At 0458 hours, as the result of high vibration levels on the No. 9 and No. 10 turbine bearings, a reactor scram occurred.

10/14 At 0757 hours, the plant was switched from the hot shutdown mode to the refuel mode to perform the SRM functional trip test.

10/14 The turbine was disengaged from the turning gear at 1130 hours to permit inspection of the generator alterex bearings. The turbine was again placed on the turning gear at 1220 hours.

10/14 The plant was placed in the startup mode at 1340 hours, and the reactor startup was commenced.

10/14 At 1502 hours, the turbine was disengaged from the turning gear.

10/14 At 1654 hours, the reactor was critical, but at 1656 hours, the reactor was taken subcritical because it was reported that the outboard alterex bearing was in need of repair.

10/16 At 0040 hours, the turbine was again placed on the turning gear to allow maintenance personnel to check the couplings. The turbine was secured from the turning gear at 0045 hours, and when the repair work was completed at 0330 hours, the turbine was placed on the turning gear.

10/16 At 0636 hours, the reactor was critical.

10/16 The plant was placed in the run mode at 1106 hours, and the turbine-generator was placed on line at 1240 hours.

10/19 The plant load was maintained at 342 MWe for approximately 11 hours while the No. 3 tip machine was repaired.

10/27 During normal operation with a plant load of 500 MWe, the "A" recirculation pump experienced a reduction in flow accompanied by increases in the "A" M-G set field voltage and generator winding temperature. The plant load was decreased and the "A" M-G set was removed from service to investigate the problem. The investigation revealed that the primary-side fuses to the "A" M-G set voltage regulator reference transformer were opened. The fuses were replaced, the "A" M-G set was returned to service, and the plant load was increased. (RO 77-84)

10/28 Similar problems with the "A" M-G set were again experienced. The plant load was decreased and the "A" M-G set was removed from service. The fuses and the reference transformer were replaced with like-for-like spares, and the "A" M-G set was returned to service. (RO 77-85)

10/29 Due to seal leakage, the jockey fire pump was secured.

11/1 The plant was operating at 534 MWe.

11/1 At 1321 hours, during the performance of STP 42F001, a false high reactor water level signal tripped the turbine, which subsequently caused the reactor to scram.

11/1 At 1521 hours, the plant was placed in the startup mode, and at 2148 hours, the reactor was critical.

11/2 At 0001 hours, the plant was placed in the run mode, and at 0024 hours, the turbine-generator was placed on line.

11/2 A leak in the GSW system piping to the "B" condensate pump oil cooler was discovered. In order to make repairs on the line, the reactor power was reduced to 52 percent, and the "B" condensate pump and the "B" reactor feed pump were secured. The repairs were completed, the two pumps were returned to service, and the reactor power was again increased.

11/3 The "A" standby diesel generator was secured for maintenance. Following the completion of maintenance, 1G-31 was demonstrated operable.

11/3 During surveillance testing, the torus water temperature recorder, TR 4386A, was found to have failed downscale. The associated temperature transmitter, TT 4325, was replaced with a like-for-like spare, calibrated, and returned to service. (RO 77-87)

11/8 During the performance of STP 42B017, LPCI loop selection logic PDIS 4641 would not reset after being tripped. The PDIS microswitch was found to be slightly out of adjustment which did not allow the microswitch contacts to open properly. The microswitch was adjusted, and the PDIS was functionally tested and returned to service. (RO 77-88)

11/13 During the performance of control rod exercise, control rod 34-07 would insert, but would not withdraw. This problem was attributed to dirty contacts in the K1 and K2 relays associated with the timer and was corrected.

11/17 In order to prepare for the performance of Special Test Procedure No. 40, "Recirculation 4-Inch Bypass Line Removal Operation Demonstration Test", at 1759 hours, the plant load was reduced from 517 MWe. The 80-percent load line required for the test was achieved at 2035 hours. The test was completed at 2153 hours.

11/17 An orderly shutdown for the purpose of inspecting the hydraulic snubbers in the drywell was commenced at 2200 hours.

11/18 At 0252 hours, the turbine was tripped, and at 0322 hours, the plant was switched to the startup mode.

11/18 At 0437 hours, the turbine was placed on the turning gear.

11/18 At 0525 hours, the plant was placed in the hot shutdown mode.

11/18 At 1305 hours, the turbine was secured from turning gear to uncouple the exciter, and was again placed on the turning gear at 1345 hours. At 1554 hours, the turbine was again secured from the turning gear for maintenance.

11/19 Special Test Procedure No. 41, "Isolated Operation of the Control Rod Drive Return Line to Vessel," was commenced at 0735 hours. As part of the test, the plant was placed in the run mode at 1242 hours. At 1323 hours, the plant was returned to the shutdown mode and further testing was postponed when the full-in indication on all the control rods was lost. The full-in indication was restored at 1754 hours, and the special test was resumed.

11/20 The plant was placed in the startup mode at 2347 hours.

11/21 At 0305 hours, the reactor was critical.

11/21 At 1045 hours, the plant was switched to the run mode, and at 1149 hours, the turbine-generator was placed on line.

11/21 At 1220 hours, when the nitrogen supply to MSIV 4421 was valved in, the nitrogen control pressure on the other MSIV's was reduced, causing the valves to drift and the reactor to scram.

11/21 At 1351 hours, the reactor startup was commenced, and at 1427 hours, the reactor was critical.

11/21 At 1658 hours, the plant was switched to the run mode, and at 1721 hours, the turbine-generator was placed on line.

11/21 At 2138 hours, the "A" feedwater regulating valve failed to control the reactor water level in either the automatic or manual modes and the "B" reactor feed pump failed to start, causing a reactor scram.

11/21 At 2234 hours, the plant was switched from the shutdown mode to the startup mode.

11/22 At 0227 hours, the reactor was critical.

11/22 At 0518 hours, the plant was placed in the run mode, and at 0558 hours, the turbine-generator was placed on line.

11/22 Following a loss of service air, the "A" and "B" feedwater regulating valves received lockout signals. However, the "B" feedwater regulating valve lockout relay hung up and the coil burned out, and the valve would not lockout.

11/23 During the performance of STP 45D001, the HPCI injection valve MOV 2312 tripped when cycled closed although there was no indication of full closure. The motor thermal overloads were subsequently reset and the same result was observed when the valve was cycled again. Subsequent testing, however, proved proper operation of the valve. The cause of the problem was unknown, but additional surveillance testing on the valve was ordered. (RO 77-90)

11/27 Following a power increase in accordance with approved procedures, an MCPR violation was found to have occurred. The reactor power was immediately reduced to bring the MCPR back within the limits specified by the Technical Specifications. (RO 77-91)

11/28 During the performance of Special Test Procedure No. 22, "Installation and Removal of LPRM Signal Monitoring Equipment", nonconservative errors of approximately 13 percent were induced in APRM channels A, B, C, and D. The problem occurred when the test instrumentation was improperly connected. Upon identification of the problem, the special test was secured. (RO 77-92)

12/1 The plant was operating at 505 MWe.

12/6 Although the Reactor Building ventilation shaft radiation monitor, RM 7606B, was not found to be upscale or to have failed, a Group 3 isolation was initiated despite acceptable radiation levels being present. In order to clear the isolation and work on RM 7606B, the radiation monitor was jumpered out in accordance with the provisions of STP 42D002.

12/7 Radiation monitor RM 7606B was demonstrated operable and returned to service.

12/8 The "A" recirculation pump M-G set scoop tube was locked out when the M-G set speed was found to be fluctuating and abnormal movement was observed in the scoop tube positioner. Upon locking out the scoop tube, the positioner oscillations cleared, but the speed fluctuations continued.

12/12 During the performance of STP 42B007, RCIC system steam line high flow switch PDIS 2441 tripped at a value exceeding the limits specified by the Technical Specifications. The switch was recalibrated and functionally tested. (RO 77-93)

12/13 During the performance of STP 42B022, RCIC system steam line high flow switch PDIS 2442 tripped at a value exceeding the limits specified by the Technical Specifications. The switch was recalibrated and functionally tested. (RO 77-93)

12/15 During a routine plant inspection, RHR system cross-tie valve MOV 2010 was found to be in the closed position despite an open indication in the Control Room. The normally-open valve could not be either manually or remotely opened, and an inspection of the motor operator internals indicated damage to the worm and worm gear. The LPCI system was declared inoperable and the appropriate surveillance testing was initiated. (RO 77-94)

12/16 MOV 2010 was manually opened and locked in the open position. The LPCI system was then demonstrated operable.

12/20 During the performance of STP 45D001, three fast start attempts were required before the HPCI system discharge flow rate satisfied the Technical Specifications requirement of 3000 gpm within 25 seconds after initiation. Although the HPCI system lube oil throttling valve settings and instrument settings were checked and a few minor adjustments made, no definitive cause for this problem could be determined. (RO 77-95)

12/22 Following three satisfactory fast starts of the HPCI system and the completion of the operability demonstration per STP 45D001, the HPCI system was declared operable.

12/22 Testing was completed on all MOV's to ensure their operability and proper remote valve position indication. This testing was prompted by the failure of MOV 2010.

12/27 Two fast start attempts were required before the HPCI system discharge flow rate satisfied the Technical Specifications requirement. The HPCI system was then declared inoperable and the appropriate surveillance testing was initiated. (RO 77-96)

12/27 During normal operation, the RWCU system automatically isolated upon a high temperature differential in the RWCU pump room. When the isolation signal cleared, RWCU inlet isolation valve MOV 2700 was cycled to the open position, but it did not remotely indicate full open. Investigation of the problem revealed that the motor thermal overloads had tripped. When repeated, the valve again tripped near the end of its opening cycle. The valve was then declared inoperable. (RO 77-97)

12/29 During operability testing of the HPCI system, an exhaust diaphragm high pressure alarm was received. The HPCI system testing was secured pending investigation of the alarm condition.

12/29 During surveillance testing, the main steam line tunnel high ambient temperature indicating switch, TIS 4443, was observed to have unusually low readings on channels 2, 3, and 4. (RO 77-98)

12/30 An investigation revealed that channel 1 of TIS 4443 had actually failed, causing undetermined power supply problems which affected the indications on channels 2, 3, and 4. Temperature element TE 4443A was subsequently removed from service and the remaining channels resumed normal indication.

12/30 The HPCI system was successfully fast started three times and was once again declared operable. Investigation of the problem revealed that the most probable cause was wear in the turbine-driven oil pump. The capacity of the pump at low turbine speed apparently is insufficient to fully open the turbine stop valve and thus allow the HPCI turbine to reach rated speed. As a temporary measure, the HPCI auxiliary oil pump cutoff pressure setpoint was raised slightly to extend the pump run time following a HPCI system start, thus assisting the shaft-driven oil pump in maintaining the system oil pressure until such time that the HPCI turbine has attained rated speed. New pump internals were subsequently ordered.

12/31 At the close of this report period, the plant was operating at 490 MWe. Following CRD exercise and control rod pulls, the reactor power was being increased to the 100 percent level.

Section B

MAJOR SAFETY-RELATED MAINTENANCE

This section has been prepared in accordance with the requirements of Appendix A to Operating License DPR-49 for the Duane Arnold Energy Center.

B. Major Safety-Related Maintenance

This section summarizes, in tabular form, the major safety-related maintenance items accomplished during the calendar year 1977.

MAJOR SAFETY-REL[illegible] MAINTENANCE

DATE	SYSTEM NO.	SYSTEM	COMPONENT	DESCRIPTION OF MAINTENANCE PERFORMED
8/11/77	10	River Water Supply	River Water Supply Pump 1P-117D	The pump impeller was reset to correct a low discharge flow rate condition and a coupling key was replaced.
10/19/77	10	River Water Supply	Self-Cleaning Strainer 1S-85A	The shear pin was replaced.
10/31/77	10	River Water Supply	Self-Cleaning Strainer 1S-85A	The strainer lower bearing and shear pin were replaced, and the strainer shaft was remachined.
11/7/77	10	River Water Supply	Self-Cleaning Strainer 1S-85A	The shear pin was replaced.
11/21/77	10	River Water Supply	Self-Cleaning Strainer 1S-85A	The shear pin was replaced with a brass rod.
4/13/77	16	Residual Heat Removal Cooling Water	RHR Heat Exchanger 1E-201B	The heat exchanger was opened up for routine maintenance, cleaned, inspected and closed up.
4/20/77	16	Residual Heat Removal Cooling Water	RHR Heat Exchanger 1E-201A	The heat exchanger was opened up for routine maintenance, cleaned, inspected, and closed up.
6/10/77	16	Residual Heat Removal Cooling Water	RHR Service Water Pump 1P-22C	The pump impeller was reset to meet surveillance testing requirements.
7/5/77	16	Residual Heat Removal Cooling Water	RHR Service Water Pump 1P-22D	The pump impeller was reset to meet surveillance testing requirements.
3/21/77	24	Standby Diesel Generators	Cooler 1E-53B	The heat exchanger was opened up for routine maintenance, cleaned, inspected, and closed up.
4/7/77	24	Standby Diesel Generators	Cooler 1E-53A	The heat exchanger was opened up for routine maintenance, cleaned, inspected, and closed up.
3/31/77	24	Standby Diesel Generators	Standby Diesel Generator 1G-21	The annual inspection was performed.
3/31/77	24	Standby Diesel Generators	Standby Diesel Generator 1G-21	The camshafts were replaced.
3/30/77	24	Standby Diesel Generators	Standby Diesel Generator 1G-21	The air start check valves were removed, disassembled, inspected, reassembled, and reinstalled.
4/20/77	24	Standby Diesel Generators	Standby Diesel Generator 1G-31	The annual inspection was performed.
4/9/77	24	Standby Diesel Generators	Standby Diesel Generator 1G-31	The camshafts were replaced.
4/7/77	24	Standby Diesel Generators	Standby Diesel Generator 1G-31	The flywheel-end main bearing was replaced.
4/15/77	24	Standby Diesel Generators	Standby Diesel Generator 1G-31	The main thrust bearing was replaced.
4/19/77	24	Standby Diesel Generators	Standby Diesel Generator 1G-31	The crankshaft deflection was realigned.

MAJOR SAFETY-RELATED [illegible]ENANCE (Cont'd)

DATE	SYSTEM NO.	SYSTEM	COMPONENT	DESCRIPTION OF MAINTENANCE PERFORMED
8/18/77	24	Standby Diesel Generators	Starting Air Compressor 1K-10C	The unloader valve was rebuilt.
10/28/77	24	Standby Diesel Generators	Starting Air Compressor 1K-10A	Automatic compressor start pressure switch PS 3224A was replaced.
3/28/77	49	Residual Heat Removal	Core Spray and RHR Fill Pump 1P-70	The inboard and outboard motor bearings were replaced.
1/31/77	49	Residual Heat Removal	Hydraulic Snubber HBB-25-SS-178	The snubber wall mounting was repaired.
2/19/77	49	Residual Heat Removal	Hydraulic Snubbers HBB-24-SS-228 and HBB-24-SS-229	The two snubber wall mountings were repaired.
2/20/77	49	Residual Heat Removal	Hydraulic Snubber GBB-4-SS-211	The snubber anchor plate was modified and the wall mounting was repaired.
5/10/77	49	Residual Heat Removal	Recirculation Injection Loop A Valve MOV 2003	The valve internals were inspected, the valve was lapped, and a new valve stem, which had been damaged, was installed.
4/24/77	49	Residual Heat Removal	HPCI-to-RHR Cross-Tie Valve MOV 2298	The motor operator gearbox was repaired.
4/4/77	49	Residual Heat Removal	Pipe Hanger GBB-23-H152	The hanger rod, which had buckled, was replaced and the hanger was reset.
4/4/77	49	Residual Heat Removal	Pipe Restraint GBB-23-SR-160	The restraint was relocated and remounted to the wall.
4/4/77	49	Residual Heat Removal	Pipe Hanger GBB-23-H153	The hanger rod, which had buckled, was replaced and the hanger was reset.
4/4/77	49	Residual Heat Removal	Pipe Hanger GBB-23-H150	The hanger rod, which had buckled, was replaced and the hanger was reset.
5/2/77	49	Residual Heat Removal	Core Spray and RHR Fill Pump 1P-70	The pump-motor coupling was repaired.
4/7/77	49	Residual Heat Removal	Core Spray and RHR Fill Pump 1P-70	The pump bearings and seal were replaced.
6/17/77	50	Reactor Core Isolation Cooling	RCIC Turbine Steam Inlet Valve MOV 2404	The valve torque switch was replaced and adjusted to the recommended setting.
6/21/77	50	Reactor Core Isolation Cooling	RCIC Turbine Steam Inlet Valve MOV 2404	The shear pin in the gear of the valve torque switch was replaced.
7/12/77	50	Reactor Core Isolation Cooling	Pressure Differential Switch PDIS 2441	Setpoint locking screws were added to the switch.
7/14/77	50	Reactor Core Isolation Cooling	Pressure Differential Switch PDIS 2442	Setpoint locking screws were added to the switch.

DATE	SYSTEM NO.	SYSTEM	COMPONENT	DESCRIPTION OF MAINTENANCE PERFORMED
11/2/77	50	Reactor Core Isolation Cooling	RCIC Turbine Steam Inlet Valve MOV 2404	The valve torque switch was replaced and adjusted to the recommended setting.
5/3/77	30	H/V-Control Building	Control Building Chilled Water Pump 1V-CP-30A	The inboard motor bearing was replaced.
1/26/77	30	H/V-Control Building	Control Building Chiller 1V-CH-1A	The compressor, dryers, and filters were replaced.
8/11/77	30	H/V-Control Building	Control Building Chillers 1V-CH-1A and 1V-CH-1B	The temperature load controllers on both units were replaced per DCR No. 551.
5/27/77	30	H/V-Control Building	Control Building Chillers 1V-CH-1A and 1V-CH-1B	The condenser tubes on both units were cleaned.
8/11/77	30	H/V-Control Building	Control Building Chiller 1V-CH-1B	The compressor was replaced.
9/2/77	30	H/V-Control Building	Control Building Standby Filter Unit 1V-SFU-30B	The charcoal filters were renewed.
9/15/77	30	H/V-Control Building	Control Building Standby Filter Unit 1V-SFU-30A	The charcoal filters were renewed.
8/28/77	30	H/V-Control Building	Control Building Emergency Air Supply Fan 1V-SF-30A	The fan impeller was adjusted to provide adequate clearance from the housing.
9/15/77	30	H/V-Control Building	Control Building Emergency Air Supply Fan 1V-SF-30A	The fan impeller was adjusted.
10/3/77	30	H/V-Control Building	Control Building Emergency Air Supply Fan 1V-SF-30A	The fan-motor coupling was adjusted to reposition the fan impeller.
8/5/77	34	H/V-Reactor Building	Isolation Dampers 1V-AD-19A and 1V-AD-19B	Solenoid valves SV 7639A and SV 7639B were replaced per an Engineering request.
5/2/77	51	Core Spray	"B" Core Spray Test Bypass Valve MOV 2132	The valve seat and disc were lapped to permit full valve closure and reduce leakage through the valve.
2/17/77	51	Core Spray	"A" Core Spray Pump Discharge Valve MOV 2115	The motor operator bearing housing, which was damaged during a water hammer incident, was replaced.
4/19/77	51	Core Spray	"A" Core Spray Test Bypass Valve MOV 2112	The valve seat and disc were lapped to minimize leakage through the valve.
2/26/77	51	Core Spray	"A" Core Spray Pump Discharge Valve MOV 2115	The motor operator clutch housing, which was cracked during a water hammer incident, was replaced.
4/12/77	52	High Pressure Coolant Injection	HPCI Pump Recirculation Valve MOV 2318	The valve disc was lapped to minimize leakage through the valve.

DATE	SYSTEM No.	SYSTEM	COMPONENT	DESCRIPTION OF MAINTENANCE PERFORMED
1/18/77	52	High Pressure Coolant Injection	Hydraulic Snubber HBB-6-SS-22	The snubber, which was found to be broken, was replaced.
6/17/77	52	High Pressure Coolant Injection	HPCI Turbine 1S-201	The mechanical hydraulic overspeed trip was replaced.
4/16/77	52	High Pressure Coolant Injection	HPCI Turbine Barometric Condenser 1E-202	The barometric condenser was opened up for routine maintenance, cleaned, inspected, and closed up.
4/3/77	52	High Pressure Coolant Injection	HPCI Turbine 1S-201	As the result of abnormal wear, the turbine governor drive gears were replaced.
5/10/77	52	High Pressure Coolant Injection	Drain Valve CV 2211	The valve was lapped to correct an LLRT failure.
5/10/77	52	High Pressure Coolant Injection	Drain Valve CV 2212	The valve was lapped to correct an LLRT failure.
6/21/77	52	High Pressure Coolant Injection	Suppression Pool Pump Suction Valve MOV 2322	The motor was removed to repair a ground and replaced.
6/21/77	52	High Pressure Coolant Injection	Main Steam Supply Control Valve MOV 2202	The motor was removed to repair a ground and replaced.
6/20/77	52	High Pressure Coolant Injection	HPCI Pump Recirculation Valve MOV 2318	The motor was removed to repair a ground and replaced.
7/14/77	52	High Pressure Coolant Injection	HPCI-to-Condensate Storage Tank Recirculation Valve MOV 2316	The motor for this valve was previously cannibalized for use on MOV 2318. The motor originally installed on MOV 2318 was repaired and subsequently installed on this valve.
9/23/77	52	High Pressure Coolant Injection	HPCI Turbine 1S-201	Hydraulic control system pressure switch PS 2270 was found to have a ruptured bellows unit and was replaced.
11/20/77	52	High Pressure Coolant Injection	HPCI-to-Condensate Storage Tank Recirculation Valve MOV 2315	A ground was found in the motor and the motor was replaced.
3/26/77	53	Standby Liquid Control	Globe Valve V-26-14	The valve was lapped to minimize leakage through the valve from the demineralized water system.
6/9/77	54	Emergency Service Water	ESW Pump 1P-99A	The pump impeller was reset to correct a low discharge pressure condition.
4/10/77	55	Control Rod Drive-Hydraulic	CRD 26-03	CRD 26-03 was replaced with a rebuilt drive.
4/8/77	55	Control Rod Drive-Hydraulic	CRD 26-19	CRD 26-19 was replaced with a rebuilt drive.
4/7/77	55	Control Rod Drive-Hydraulic	CRD 26-15	CRD 26-15 was replaced with a new or rebuilt drive.
4/10/77	55	Control Rod Drive-Hydraulic	CRD 18-11	CRD 18-11 was replaced with a rebuilt drive.

MAJOR SAFETY-RELATED MAINTENANCE (Cont'd)

DATE	SYSTEM NO.	SYSTEM	COMPONENT	DESCRIPTION OF MAINTENANCE PERFORMED
4/8/77	55	Control Rod Drive-Hydraulic	CRD 30-39	CRD 30-39 was replaced with a rebuilt drive.
4/8/77	55	Control Rod Drive-Hydraulic	CRD 14-31	CRD 14-31 was replaced with a rebuilt drive.
4/12/77	55	Control Rod Drive-Hydraulic	CRD 22-19	CRD 22-19 was replaced with a rebuilt drive.
4/7/77	55	Control Rod Drive-Hydraulic	CRD 34-15	CRD 34-15 was replaced with a new or rebuilt drive.
4/9/77	55	Control Rod Drive-Hydraulic	CRD 14-27	CRD 14-27 was replaced with a rebuilt drive.
4/9/77	55	Control Rod Drive-Hydraulic	CRD 06-23	CRD 06-23 was replaced with a rebuilt drive.
4/9/77	55	Control Rod Drive-Hydraulic	CRD 26-07	CRD 26-07 was replaced with a rebuilt drive.
4/12/77	55	Control Rod Drive-Hydraulic	CRD 30-27	CRD 30-27 was replaced with a rebuilt drive.
5/2/77	55	Control Rod Drive-Hydraulic	CRD Water Filter 1F-201A	A leaking union was replaced.
3/29/77	55	Control Rod Drive-Hydraulic	Cooling Water Bypass Valve V-17-46	The valve was lapped and the valve stem, which was bent, was replaced.
4/13/77	55	Control Rod Drive-Hydraulic	Control Valve CV 1804A	The valve seat and disc were lapped to correct an LLRT failure.
4/25/77	55	Control Rod Drive-Hydraulic	CRD Water Pumps Flow Controller FC 1814	A connection was resoldered to re-establish reset control.
5/5/77	55	Control Rod Drive-Hydraulic	CRD Module 42-27	A defective float level switch was replaced.
5/26/77	55	Control Rod Drive-Hydraulic	CRD 38-11	The rod select switch was replaced.
5/31/77	56	Control Rod Drive-Electrical	CRD 30-27	A shorting ground in the level switch for CRD 30-27 was corrected.
3/16/77	58	Reactor Protection and Steam Leak Detection	Pressure Differential Switches PDIS 4432 A/B/C/D, PDIS 4434 A/B/C/D, PDIS 4436 A/B/C/D, and PDIS 4438A/B/C/D	Setpoint locking screws were added to the various switches.
9/16/77	58	Reactor Protection and Steam Leak Detection	Pressure Switches PS 4545 and PS 4548	The instrument isolation valves, which were leaking by, were replaced.
2/15/77	61	Reactor Water Cleanup	RWCU System Return Flow Indicator FI 2748	The flow indicator was installed per DCR No. 453 to satisfy a Technical Specification requirement.
5/12/77	61	Reactor Water Cleanup	RWCU System Return Valve MOV 2740	The valve yoke was replaced on the valve stem to restore operability.

DATE	SYSTEM NO.	SYSTEM	COMPONENT	DESCRIPTION OF MAINTENANCE PERFORMED
11/20/77	61	Reactor Water Cleanup	RWCU System Return Check Valve V-27-11	The check valve hinge pin and gasket were replaced to eliminate leakage from the valve.
3/23/77	62	Nuclear Boiler	Pressure Differential Switches PDIS 2139 and PDIS 2119	Setpoint locking screws were added to the switches.
5/12/77	62	Nuclear Boiler	Hydraulic Snubbers DBA-5-SS-31, DBA-5-SS-37, DBA-5-SS-38, and DBA-5-SS-47	The hydraulic snubber units were replaced with mechanical units.
2/16/77	64	Reactor Recirculation	Pressure Differential Switches PDIS 4641 through 4644	Setpoint locking screws were added to the various switches.
8/19/77	64	Reactor Recirculation	Pressure Switch PS 4637	The instrument isolation valve, which was leaking by, was replaced.
5/10/77	64	Reactor Recirculation	Excess Flow Check Valve XFV 4504	A shorting ground was corrected.
4/27/77	64	Reactor Recirculation	"A" Recirculation Loop Bypass Pipe	An indicated weld defect in the pipe-to-elbow weld was repaired.
5/13/77	64	Reactor Recirculation	Pressure Differential Switches PDIS 4642 through 4644	A microswitch in each instrument was replaced, and the instruments were recalibrated.
12/15/77	64	Reactor Recirculation	Jet Pump Flow Transmitter FT 4517	A new flow transmitter was installed and calibrated.
8/10/77	70	Standby Gas Treatment	Isolation Dampers 1V-AD-52A and 1V-AD-52B	Solenoid valves SV 7636A and SV 7636B were replaced per an Engineering request.
4/7/77	70	Standby Gas Treatment	Control Valve CV-4309	The valve seat and gate were lapped to correct an LLRT failure.
3/29/77	73	Containment Atmosphere Control	Sample Select Valves SV 8115A and SV 8116A	The solenoid valves, which were leaking by, were replaced.
4/12/77	73	Containment Atmosphere Control	Sample Select Valve SV 8114A	The solenoid valve, which was leaking by, was replaced.
5/3/77	73	Containment Atmosphere Control	Sample Select Valves SV 8114B, SV 8115B, and SV 8116B	The solenoid valves, which were leaking by, were replaced.
4/22/77	73	Containment Atmosphere Control	Isolation Valves CV 4304 and CV 4305	Solenoid valves SV 4304 and SV 4305 were replaced per an Engineering request.
4/4/77	73	Containment Atmosphere Control	Isolation Valves CV 4378A and CV 4378B	The valve seats and gates were lapped to correct LLRT failures.
4/13/77	73	Containment Atmosphere Control	Purge Outlet Bypass Valve CV 4310	The valve seat and disc were lapped to correct an LLRT failure.

MAJOR SAFETY-RELATE[illegible]TENANCE (Cont'd)

DATE	SYSTEM NO.	SYSTEM	COMPONENT	DESCRIPTION OF MAINTENANCE PERFORMED
4/12/77	73	Containment Atmosphere Control	Purge Outlet Valves CV 4302 and CV 4303	The valves were opened up and cleaned to correct an LLRT failure.
4/12/77	73	Containment Atmosphere Control	Sample Return Valve SV 8117A	The solenoid valve, which was leaking by, was replaced.
5/3/77	73	Containment Atmosphere Control	Sample Return Valve SV 8117B	The solenoid valve, which was leaking by, was replaced.
11/23/77	73	Containment Atmosphere Control	Pump Back Compressor 1K-18B	The compressor valves were rebuilt.
1/25/77	78	Nuclear Instrumentation	No. 3 Tip Machine	The programmer board was replaced to restore tip machine operability.
2/7/77	78	Nuclear Instrumentation	No. 1 Tip Machine	The detector was replaced to restore tip machine operability.
6/7/77	78	Nuclear Instrumentation	No. 2 Tip Machine	The detector was replaced to restore tip machine operability.
9/18/77	78	Nuclear Instrumentation	No. 3 Tip Machine	The detector was replaced to restore tip machine operability.
9/15/77	78	Nuclear Instrumentation	"B" IRM	The power supply was replaced.
9/23/77	78	Nuclear Instrumentation	No. 1 Tip Machine	The transfer cable, which had broken, was replaced.
4/24/77	83	Main Steam - Automatic Blowdown	Feedwater Stop-Check Valve MOV 4442	The valve was opened up and cleaned to correct an LLRT failure.
4/3/77	83	Main Steam - Automatic Blowdown	MSIV-LCS Valves MOV 8401 A/B/C/D, MOV 8402 A/B/C/D, and MOV 8403 A/B/C/D	The open torque switches on the various valves were jumpered out per DCR No. 611.
3/29/77	83	Main Steam - Automatic Blowdown	MSIV-LCS Valves MOV 8401 B/C/D and MOV 8402 B/C/D	The valve seats and discs were lapped to minimize leakage through the valves.
4/5/77	83	Main Steam - Automatic Blowdown	MSIV-LCS Valves MOV 8403 A/B/C/D	The valve seats and discs were lapped to minimize leakage through the valves.
3/30/77	83	Main Steam - Automatic Blowdown	MSIV-LCS Valves MOV 8401A and MOV 8402A	The valve seats and discs were lapped to minimize leakage through the valves.
4/18/77	83	Main Steam - Automatic Blowdown	MSIV-LCS Valve MOV 8402B	The valve was lapped to correct an LLRT failure.
5/4/77	83	Main Steam - Automatic Blowdown	Safety Relief Valves PSV 4400, PSV 4401, PSV 4402, PSV 4405, PSV 4406, and PSV 4407	New safety relief valves were installed and the necessary modifications were made per DCR No. 670.
5/11/77	83	Main Steam - Automatic Blowdown	Safety Relief Valve PSV 4403	The safety relief valve was modified per DCR No. 670.