TENNESSEE VALLEY AUTHORITY

NUCLEAR POWER GROUP
SEQUOYAH NUCLEAR PLANT

MONTHLY OPERATING REPORT

TO THE

NUCLEAR REGULATORY COMMISSION

APRIL 1989

UNIT 1

DOCKET NUMBER 50-327

LICENSE NUMBER DPR-77

UNIT 2

DOCKET NUMBER 50-328

LICENSE NUMBER DPR-79

Submitted by: S. J. Smith
S. J. Smith, Plant Manager

IE24

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0288f

OPERATIONAL SUMMARY

PERFORMANCE SUMMARY

April 1989

The following summary describes the significant operational activities for the month of April. In support of this summary, a chronological log of significant events is included in this report.

Unit 1

Unit 1 operated the entire month at approximately 100 percent power. It generated 842,260 MWh of electrical power with a capacity factor of 99.02 percent. This marks a milestone for the unit for reliable day-to-day service. Not since March 1985 has the unit accomplished a capacity of 99 percent or greater (unit 1 - 99.9 percent, March 1985, and unit 2 - 99.3 percent, April 1985). This is equivalent to generating 1,171 MWh every hour throughout the month of April.

The unit has been in continuous operation for 77 days as of April 30, 1989. Since restart on November 10, 1988, a total of 3,315,110 MWh has been produced.

Unit 2

On April 11, 1989, at 2024 (EDT), the unit was taken critical. On April 14, 1989, at 1408 (EDT), unit 2 began generating electrical power, signaling the completion of the cycle 3 refueling/modification outage. The outage duration was approximately 87 days. A total of 17,610 MWh was generated for the month, comprising a capacity factor of 2.07 percent.

Three reactor trips occurred during unit restart because of erratic behavior of the feedwater controls and associated equipment. This accounts for the small amount of electrical generation and capacity. Unit 2 returned to service on April 26, 1989, at 1627 (EST). The unit continued in service for the remainder of the month, except for a small portion of time dedicated for a turbine overspeed test.

SIGNIFICANT OPERATIONAL EVENTS

Unit 1

Date	Time (EST)	Event
04/01/89	0001	100 percent reactor power, 1,185 MW_e.
	0714	RPI H-8 is inoperable.
	1356	Control rod H-8 RPI D bank operable after maintenance.
	1958	B-2 intermediate string heaters isolated on high-high level.
	1959	Reduced turbine load because of secondary-side flow swings and decreasing S/G levels. Reactor at 100 percent, 1,180 MW_e.
	2016	Began draining heater B-2 in preparation for placing back in service.
	2029	Began increasing turbine load.
04/02/89	0126	Began placing intermediate pressure heater B string in service.
	0358	B string of intermediate pressure heaters are back in service.
	0500	Reactor at 100 percent, 1,188 MW_e.
04/03/89	0151	RPI B-4 is reading approximately 18 steps different from its group demand steps counter, declared inoperable.
	1130	RPI B-4 rod is operable after maintenance and PMT were performed.
	1534	Reactor at 100 percent, 1,184 MW_e.
04/15/89	1444	RPI E-13 on S/D bank D inoperable.
	1532	RPI E-13 operable, but being monitored.
04/24/89	0259	Isolate waterbox 1B-2 for maintenance. Reactor at 99 percent, 1,175 MW_e.
	1230	Waterbox 1B-2 back in service.
	1528	Reactor at 100 percent, 1,180 MW_e.
04/26/89	0117	Diluted RCS to increase T_{avg}.

SIGNIFICANT OPERATIONAL EVENTS

Unit 1

Date	Time (EST)	Event
04/27/89	1625	RPI E-13 inoperable, greater than 12 steps out.
04/28/89	1030	RPI E-13 remains inoperable. Detection circuitry problems.
04/29/89	0038	RPI L-13 inoperable. Indicator showing rod at the bottom of the core.
	0517	Maintenance complete on RPI L-13. RPI operable.
	1336	RPI F-14 inoperable. Indicator showing rod at the bottom of the core.
	1819	RPI F-14 operable after maintenance.
04/30/89	2400	Reactor at 100 percent, 1,172 MW_e.

Unit 2

Date	Time (EST)	Event
04/01/89	0001	Mode 5, 178.1°F at 350 lb/in^2. Activities continue for start-up.
04/05/89	0415	Began heatup in preparation for mode 4 entry.
	0751	Entered mode 4.
	0915	Mode 4, RCS at 222°F, 365 lb/in^2.
	2038	Started RCS heatup to approximately 335°F.
	2225	RCS temperature at approximately 335°F.
04/06/89	1630	Entered mode 3.
	1652	356°F, pressure 550 lb/in^2.
	2319	Terminated heatup, holding RCS temperature at 450°F for performance of RCS RTD cross calibration verification.
04/07/89	1300	Began low-power physics testing.
04/08/89	1303	Resume temperature increase.
04/09/89	0117	Began pulling S/D bank A.
04/10/89	0708	RPI F-2 on control bank D inoperable.

SIGNIFICANT OPERATIONAL EVENTS

Unit 2

Date	Time (EST)	Event
04/10/89 (cont.)	1338	RPI F-2 returned to normal after maintenance.
	1525	Began pulling S/D rods.
	1540	Stopped pulling S/D bank A from core, RPI D-2 indicating approximately 20 steps below other RPIs. Maintenance initiated to repair RPI.
	2039	Began pulling S/D rods in preparation for mode 2.
	2043	Shutdown bank A pulled.
	2052	Shutdown bank B pulled.
	2058	Shutdown bank C pulled.
04/11/89	0019	RCS 545°F at 2,335 lb/in^2.
	1537	All S/D banks fully withdrawn.
	1545	Entered mode 2.
	1651	Began RCS dilution at various times for criticality.
	2024	Unit 2 reactor critical.
04/12/89	0100	Low-power physics testing in progress.
	0403	RPI B-12 inoperable.
	0855	Low-power physics testing in progress.
	1600	Shutdown rod bank A, RPI B-12 operable after maintenance, but RPI is being monitored.
04/13/89	0330	Low-power physics testing complete.
	1024	Reactor power at 1 percent.
	1250	Entered mode 1, reactor at 5 percent and increasing.
	1500	Reactor at 20 percent power.
	1646	Reactor at 20 percent power. Holding power for turbine balancing.
	1847	Holding status because of excessive RCS leakage.

SIGNIFICANT OPERATIONAL EVENTS

Unit 2

Date	Time (EST)	Event
04/13/89 (cont.)	2121	Started boration for controlled power reduction to 8 percent to investigate RCS leakage. Power level currently 18 percent.
04/14/89	0029	Reactor at 8 percent power.
	0210	RCS leakage found and resolved.
	0345	Started dilution of RCS for power increase.
	0515	Completed dilution of RCS.
	0943	Started power increase to 20 percent.
	1213	Reactor at 23 percent.
	1227	Rolling main turbine.
	1408	Unit 2 main generator online.
	1620	Began power increase to 30 percent.
	1847	Reactor at 30 percent, 284 MW_e.
	2211	Started turbine load decrease. Maintaining reactor power at 30 percent for turbine overspeed test.
	2344	Turbine taken offline for overspeed test.
04/15/89	0009	Reactor trip on S/G low-low level, loop 4.
	0011	Started emergency boration, T_{avg} less than 540°F.
	0749	Mode 3, RCS at 547°F, 2,230 lb/in2.
	1140	Began RCS dilution for criticality.
	1457	Completed dilution.
	1635	All S/D banks fully withdrawn.
	2159	Unit entered mode 2, began control rod withdrawal.
	2240	RPI H-12 out greater than 12 steps, declared inoperable. Initiated maintenance for repair.

SIGNIFICANT OPERATIONAL EVENTS

Unit 2

Date	Time (EST)	Event
04/15/89 (cont.)	2301	Unit 2 reactor is critical.
	2321	Reactor at 1 percent power.
04/16/89	0040	Entered mode 1, 5 percent power.
	0048	Reactor trip, began emergency boration, T_{avg} less than 540°F.
	0057	Terminate emergency boration.
	0126	T_{avg} increasing to 547°F.
	0713	Maintenance on various RPIs begins.
	1941	Initiated RCS cooldown to 400°F.
	2307	Terminated cooldown at 450°F.
04/17/89	0817	Mode 3, 452°F.
	1030	Began heatup.
	1736	RCS at 498°F, 2,085 lb/in^2.
04/18/89	0109	Began pulling S/D banks.
	0148	Entered mode 2.
	0341	Reactor critical.
04/19/89	0058	Reactor power increase started.
	0145	Entered mode 1, 5 percent power.
	0444	Turbine trip because of high-high level in S/G 3. Reactor at 18 percent power.
	0447	Reactor trip, low-low S/Gs 1 and 2.
	0451	Began emergency boration, T_{avg} is 540°F.
	0456	Terminated emergency boration.
	0659	Began pulling S/D banks.
	0714	Shutdown bank withdrawal completed.

SIGNIFICANT OPERATIONAL EVENTS

Unit 2

Date	Time (EST)	Event
04/20/89	0020	RCS temperature 547°F, pressurizer 2,230 lb/in^2.
04/24/89	2214	Began dilution of RCS.
	2228	Completed dilution
04/25/89	0750	Mode 3, 547°F.
	2323	Verified all S/D rods are fully withdrawn.
	2325	Entered mode 2.
04/26/89	0009	Unit 2 reactor critical.
	0755	Mode 2, 547.4°F. Maintenance on secondary equipment in progress.
	1934	Reactor at 2 percent power. Maintenance continues on feedwater system.
04/27/89	0116	Started reactor power increase.
	0423	Entered mode 1, 5 percent power.
	0800	Began diluting RCS at various intervals for power ascension. Reactor at 10 percent power.
	1446	Rolling main turbine.
	1627	Online.
	1900	Reactor at 15 percent power, 105 MW_e.
04/28/89	0130	Began power increase to 20 percent.
	0431	Reactor power at 20 percent, 164 MW_e.
	0550	Reactor power at 23 percent, and holding for preconditioning, 184 MW_e.
	1905	RPI K-14 is inoperable, out greater than 12 steps.
	2009	Removing main generator from service for overspeed test.
	2324	Offline.

SIGNIFICANT OPERATIONAL EVENTS

Unit 2

Date	Time (EST)	Event
04/29/89	0056	Test complete, generator online.
	0220	RPI K-14 operable after maintenance, but RPI is being monitored.
	0757	Mode 1, 23 percent power, 190 MW_e. Holding power for fuel conditioning and thermal power verification.
04/30/89	1700	Began power ascension.
	2400	Reactor at 37 percent power, 338 MW_e. Power ascension in progress.

FUEL PERFORMANCE

Unit 1

The core average fuel exposure accumulated during April was 1,144 MWd/MTU, with a total accumulated core average fuel exposure of 4,644 MWd/MTU.

Unit 2

The core average fuel exposure accumulated during April was 46 MWd/MTU, with a total accumulated core average fuel exposure of 46 MWd/MTU.

SPENT FUEL PIT STORAGE CAPABILITIES

The total storage capability in the SFP is 1,386 bundles. However, there are six cell locations that are incapable of storing spent fuel. Four locations (A10, A11, A24, and A25) are unavailable because of a suction-strainer conflict, and two locations (A16 and A21) are unavailable because of an instrumentation conflict. Presently, there is a total of 428 spent-fuel bundles stored in the SFP. The remaining storage capacity is 952 bundles.

PORVs AND SAFETY VALVES SUMMARY

No PORVs or safety valves were challenged during the month of April.

SPECIAL REPORTS

There were no special reports for the month of April.

LERs

The following LERs were transmitted to the Nuclear Regulatory Commission in April 1989.

LER	Description of Event
1-89007	On March 19, 1989, with unit 1 in mode 1 (100 percent power) and unit 2 in mode 5, an A-train CRI occurred. At approximately 1125 (EST), an A-train CRI signal, as indicated on MCR panel 0-M-27B (window 20), was received in the control room. The unit 1 ASOS responded to the CRI alarm. The ASOS had knowledge that maintenance was being performed on Control Building fresh air intake duct smoke detectors 0-XS-31A-3 and 0-XS-31A-4, and could have initiated the CRI. Subsequently, the ASOS suspended the maintenance activity on the smoke detectors. The ASOS verified that no other conditions were present, such as high radiation or a safety injection signal, and initiated realignment of the control room ventilation system to normal operation in accordance with SOI-30.1B, "Control Building and Control Room Heating, Air Conditioning and Ventilation System." The Control Building ventilation system was returned to normal operation by 1310 (EST). Further investigation of the event revealed that before maintenance on the smoke detectors was authorized, the smoke detector circuit was deenergized by opening breaker No. 7 on 120-V ac preferred power rack 1-M-7, located in the control room. However, the 120-V ac power supply to an electrical interlock on the smoke-detector circuit was not deenergized. Subsequently, while reterminating wires on the smoke detector (0-XS-31A-3, A-train), terminals 7 and 8 were accidently shorted. This energized relay SDA3 and completed the required logic for the CRI. To prevent recurrence, this event will be reviewed with Electrical and Instrument Maintenance engineers, WR planners, appropriate craft personnel, and Work Control Group personnel to emphasize the need to deenergize electrical circuits while performing maintenance on electrical equipment.
1-89008	On March 20, 1989, at 1450 (EST), with unit 1 in mode 1 (100 percent power) and unit 2 in mode 5, both trains of CREVS were declared inoperable; and it was discovered that LCO 3.0.3 had been inadvertently entered earlier in the day at 0820 (EST), as a result of tornado dampers that isolate fresh air intake to the MCR having been closed. The dampers were closed to support replacement of smoke detectors. This condition could result in the loss of suction-flow path from the outside atmosphere to the Control Building emergency pressurizing fans if required upon receipt of an accident signal. Loss of suction to the pressurization fans could preclude the system from performing its design function to maintain the MCR habitability area at greater than or equal to 0.125-inch positive static pressure during accident periods. The cause of this event is attributed to an incomplete evaluation of the effect of closing the tornado dampers by a licensed operator. Upon discovery of this condition, immediate corrective actions were to open the tornado dampers and exit LCO 3.0.3. Both trains of CREVS were returned to operable status

LER	Description of Event
1-89008 (cont.)	at 1451 (EST), on March 20, 1989. In addition to the disciplinary action taken, long-term corrective actions include reviewing this event with Operations personnel; emphasizing the need to do a thorough review of all available drawings and information before deciding to operate equipment; replacing an ineffective placard on containment/auxiliary vent board 1A1 with one that more adequately details the consequences of closing the tornado dampers; and revising SOI-30.7, "On Site Electrical Power Systems Board Rooms Heating, Venting, Cooling," and SOI-30.1, "Control Building and Control Room Heating, Air Conditioning and Ventilation Systems," to include warnings about effects on the CREVS when tornado dampers are closed.
1-89009	On March 19, 1989, at approximately 0600 (EST), with unit 1 in mode 1 (100 percent power) and unit 2 in mode 5, it was discovered that the switch on the local control panel for the CO_2 fire-suppression system that protects the computer room was in the OFF position, thus making the system inoperable. The Fire Operations Shift Supervisor was immediately notified, and a fire operator was dispatched to investigate the condition. A check to determine if the system should be inoperable revealed that no log entries were made to remove the CO_2 system for the computer room from service, and the system was immediately returned to operable status. Subsequent to returning the system to normal, an investigation ensued to determine how the switch was left in the OFF position. It was revealed by discussion with one of the fire operators that he had isolated the CO_2 system for the unit 2 auxiliary instrument room on March 18, 1989, to facilitate work activities. Upon arriving at the work area, the fire operator recalled that he had opened the door for the control panel for the CO_2 system that protects the unit 2 auxiliary instrument room, as well as the adjacent control panel for the computer room, because of a lack of identification external to the panels. The control panels for the CO_2 systems are controlled to allow limited access. Because of the controls in place to limit access, and the fact that the fire operator recalled opening the CO_2 control panel for the computer room, it is concluded that the fire operator inadvertently placed, and left, the switch in the OFF position. A contributing cause of this event is attributed to the lack of identification tags on the control panels that apparently confused the fire operator as to which control panel corresponded to the unit 2 auxiliary instrument room. To prevent recurrence, this event will be reviewed with the fire operators to provide a lesson learned on how inattention to detail can lead to rendering a system inoperable. Additionally, identification tags will be placed on the outside of the CO_2 control panels.
1-89010	On March 29, 1989, at 1830 (EST), with unit 1 in mode 1 (100 percent power) and unit 2 in mode 5, a review of SI packages determined that SI-307.2, "Degraded Voltage Relay Response Time Test and Timer Calibration," was out of its 18-month TS required frequency. This instruction satisfies SR 4.3.2.3 by performing a response-time test of the 6900-V shutdown board 2A-A and 2B-B degraded voltage timers

LER	Description of Event
1-89010 (cont.)	and relays. Two auxiliary relays (DS and DV) associated with degraded voltage logic on 6900-V shutdown board 2B-B had not been tested within frequency. The scheduled performance date of SI-307.2 had been based on an incomplete package performed on November 10, 1987, by the EM Group. The November package had been incorrectly designated as a basis for scheduling the next performance of the SI and led to the auxiliary relays not being tested within the required frequency. Unit 1 entered the action statements of TS LCOs 3.3.2.1 and 3.8.2.1 at 1830 (EST). Unit 2 was in mode 5; therefore, these LCOs were not applicable. At 2130 (EST), SI-307.2 was initiated to test the 6900-V shutdown board 2B-B degraded voltage relays and was successfully completed with no deficiencies at 0030 (EST), on March 30, 1989. As a corrective measure, the EM supervisor discussed this event with personnel involved in reviewing the SI package. As an enhancement, SI-1, "Surveillance Program," will be revised to update all SI data package coversheets to include the scheduling questions in the text. The person who signs for review and approval of the test package will verify the scheduling questions were correctly answered.
1-88033 (Rev. 2)	This revision provided additional information concerning the corrective action taken by TVA on the generation of an unplanned reactor trip signal because of an instrument failure during the performance of SI-94.22, "Channel Calibration of delta T/T_{avg} Channel II, Rack 6 (T-68-25)," on October 4, 1988. Immediate corrective action was to initiate a WR to diagnose the erratic behavior of 1-TI-68-2A. Troubleshooting concluded 1-TM-68-2E was unreliable; and a similar instrument was procured, calibrated, and installed to allow a failure analysis on the removed module. Based on the inconclusiveness of the Foxboro failure analysis and TVA's inhouse investigation, there is no further action associated with this event recommended at this time.
1-89004 (Rev. 1)	This revision provided NRC with an update of the causes and corrective actions taken as a result of this event.
2-89002	On March 25, 1989, at 1059 (EST) and 1549 (EST), with unit 2 in mode 5, two separate unplanned reactor-trip signals were generated from the safety-injection logic during the performance of IMI-99 RT-601A, "Response Time Test For Turbine Trip." Before these events, on March 23, 1989, the A-train SSPS had been removed from service and the input error inhibit switch located on the SSPS logic test panel was placed in the INHIBIT position, disabling inputs from the relay contacts to the logic; however, this condition does not block the permissives. Also, the mode selector switch was placed in TEST, which unblocked several of the at-power trips and ESF functions. On March 25, 1989, performance of IMI-99 RT-601A was commenced, and the input error inhibit switch was returned to NORMAL. The blocks for the at-power trips and ESF actuations had not been reinstated; and,

Description of Event

LER	
2-89002 (cont.)	since one of the ESF signals results from low-pressurizer pressure, an SI signal was generated, resulting in an unplanned reactor trip at 1059 (EST). A second reactor trip was generated at 1549 (EST), by an SI signal from high-steam flow coincident with low-low T_{avg}. Investigation revealed that one flow transmitter had been removed from service for maintenance and a second flow transmitter drifted high. With the RCS temperature well below the setpoint for low-low T_{avg}, the coincident logic necessary for an SI signal was complete. Neither event resulted in an SI because the SI system had been adequately removed from service. The cause of the first reactor trip was a deficient procedure, because the blocks for the reactor trips and ESF functions were not reinstated. The cause of the second event was not promptly addressing the generic ramification of the condition that caused the first event. To prevent recurrence, IMI-99 RTI-601A will be revised to reinstate blocks before returning the input error inhibit switch to NORMAL. Additionally, other response time verification tests associated with SSPS will be reviewed to ensure that this condition does not exist.
2-89003	On April 1, 1989, at 0150 (EST), Operations personnel noted an unusual odor adjacent to the 120-V ac Class 1E vital inverter 2-IV. Initial investigation found a smoking capacitor, and the unit 1 SRO was immediately notified of the situation. The SRO was directed to transfer the power supply for 120-V ac vital instrument power board 2-IV from the inverter (normal supply) to the alternate/maintenance power supply in accordance with SOI-250, "Low Voltage AC/DC Electrical Systems." The alternate power supply for board 2-IV is 120-V ac instrument power distribution panel 2B. When attempting the transfer, the SRO discovered the inverter output frequency would not synchronize with the distribution board 2B output frequency by the absence of a sync light on vital board 2-IV. At 0246 (EST), the inverter ac output breaker was opened to perform the transfer and resulted in a loss of power to board 2-IV. While attempting to deenergize the inverter, popping sounds from inside the inverter were heard, and a significant amount of smoke began emitting from the inverter. During the review of the MCR boards for loss of equipment, it was discovered that a CVI had occurred; and RHR suction valve 2-FCV-74-2 had closed, which isolated RHR flow. Immediately, RHR train 2A pump was secured. At 0256 (EST), vital board 2-IV was energized from distribution panel 2B. This caused 2-FCV-74-2 to reopen, and RHR train A pump was started, reestablishing RHR flow for unit 2. The root cause of this event was the inability to manually transfer the power supply for vital instrument board 2-IV from the normal supply to the alternate/maintenance supply without momentarily losing power to the board. As immediate corrective actions, the unit 2 vital inverters were verified to synchronize with the vital instrument board alternate supply; and Operations will perform by May 31, 1989, the appropriate portions of SOI-250 to verify that the synchronization signal, necessary for inverter transfer, is present.

LER	Description of Event
2-88041 (Rev. 1)	This revision provides information on the completed corrective action of this event, which was a result of insufficient seismic qualification of instrument cabinets containing class 1E devices. On December 9, 1988, Sequoyah Site NE personnel identified that as-installed instrument cabinets supplied by the Bailey Meter Company did not meet seismic qualification requirements and issued CAQR SQP880601. The seismic qualification of the cabinets requires that two restraint bars be installed on each row of instrument modules in the cabinets. Instead, as-installed cabinets had only one restraint bar installed on each row of instrument modules. In November 1979, the Bailey Meter Company notified TVA that two restraint bars are necessary to effectively restrain instrument modules in the event of a seismic occurrence. This modification was not implemented in 12 instrument cabinets. To ensure the seismic qualification, a temporary alteration installed a 3/64-inch diameter aircraft cable across the front face of the instrument modules. The aircraft cable provided a stable and adequate restraint until permanent seismic restraints were installed. TVA has now installed permanent seismic restraint bars in the affected instrument cabinets as specified by DCN M01020A for unit 1 on February 12, 1989, and DN M01021A for unit 2 on March 23, 1989. Aircraft cable across the front face of the instrument modules, which was installed as immediate corrective action, was removed upon installation of the permanent seismic restraint bars.

RADWASTE SUMMARY

April 1989

1. Total volume of solid waste shipped offsite:

   A. Dry active waste: 915.2 ft^3 Activity: 8.088 curies

   B. Spent resins, sludges, bottoms: 158.5 ft^3
   Activity: 7.1774 curies

   Shipped: Barnwell, Inc. - April 10, 1989 (2)
   April 20, 1989 (1)

2. Radwaste onsite and awaiting shipment:

   A. Resin in storage: 744.0 ft^3

   B. Estimate resin that will be generated: 225.0 ft^3

   C. Dry active waste awaiting shipment: 1035.0 ft^3

   1 - Dry active waste
   2 - Spent resin

OFFSITE DOSE CALCULATION MANUAL CHANGES

No changes were made to the Offsite Dose Calculation Manual for the month of April.

OPERATING STATISTICS
(NRC REPORTS)

OPERATING DATA REPORT

DOCKET NO. 50-327
DATE MAY 05, 1989
COMPLETED BY D.C.DUPREE
TELEPHONE (615)843-67??

OPERATING STATUS

1. UNIT NAME: SEQUOYAH NUCLEAR PLANT, UNIT 1 — NOTES:
2. REPORT PERIOD: APRIL 1989
3. LICENSED THERMAL POWER(MWT): 3411.0
4. NAMEPLATE RATING (GROSS MWE): 1220.6
5. DESIGN ELECTRICAL RATING (NET MWE): 1148.0
6. MAXIMUM DEPENDABLE CAPACITY (GROSS MWE): 1183.0
7. MAXIMUM DEPENDABLE CAPACITY (NET MWE): 1148.0
8. IF CHANGES OCCUR IN CAPACITY RATINGS(ITEMS NUMBERS 3 THROUGH 7)SINCE LAST REPORT, GIVE REASONS: ____________
9. POWER LEVEL TO WHICH RESTRICTED, IF ANY(NET MWE): ________
10. REASONS FOR RESTRICTIONS, IF ANY: ____________

	THIS MONTH	YR.-TO-DATE	CUMULATIVE
11. HOURS IN REPORTING PERIOD	719.00	2879.00	68664.00
12. NUMBER OF HOURS REACTOR WAS CRITICAL	719.00	2830.25	27654.69
13. REACTOR RESERVE SHUTDOWN HOURS	0.00	0.00	0.00
14. HOURS GENERATOR ON-LINE	719.00	2804.66	26868.54
15. UNIT RESERVE SHUTDOWN HOURS	0.00	0.00	0.00
16. GROSS THERMAL ENERGY GENERATED (MWH)	2445906.48	9413478.21	86990415.08
17. GROSS ELECTRICAL ENERGY GEN. (MWH)	842260.00	3247870.00	29383146.00
18. NET ELECTRICAL ENERGY GENERATED (MWH)	813470.00	3135491.00	28056919.00
19. UNIT SERVICE FACTOR	100.00	97.42	39.13
20. UNIT AVAILABILITY FACTOR	100.00	97.42	39.13
21. UNIT CAPACITY FACTOR(USING MDC NET)	98.55	94.87	35.59
22. UNIT CAPACITY FACTOR(USING DER NET)	98.55	94.87	35.59
23. UNIT FORCED OUTAGE RATE	0.00	2.58	54.93

24. SHUTDOWNS SCHEDULED OVER NEXT 6 MONTHS (TYPE, DATE, AND DURATION OF EACH): ____________

25. IF SHUTDOWN AT END OF REPORT PERIOD, ESTIMATED DATE OF STARTUP: ____________

NOTE THAT THE YR.-TO-DATE AND CUMULATIVE VALUES HAVE BEEN UPDATED.

SEQUOYAH NUCLEAR PLANT
AVERAGE DAILY POWER LEVEL

DOCKET NO. : 50-327
UNIT : ONE
DATE : MAY 08,1989
COMPLETED BY : D.C.DUPREE
TELEPHONE : (615)870-6722

MONTH: APRIL 1989

DAY	AVERAGE DAILY POWER LEVEL (MWe Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe Net)
01	1140	16	1137
02	1140	17	1137
03	1139	18	1135
04	1138	19	1131
05	1137	20	1127
06	1137	21	1135
07	1139	22	1138
08	1137	23	1132
09	1132	24	1132
10	1137	25	1131
11	1136	26	1120
12	1136	27	1128
13	1139	28	1130
14	1137	29	1129
15	1137	30	1130

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-327
UNIT NAME Sequoyah One
DATE May 5, 1989
COMPLETED BY D. C. Dupree
TELEPHONE (615) 843-6722

REPORT MONTH April 1989

No.	Date	Type[1]	Duration (Hours)	Reason[2]	Method of Shutting Down Reactor[3]	Licensee Event Report #	System Code[4]	Component Code[5]	Cause & Corrective Action to Prevent Recurrence
									No activities for April.

[1]F: Forced
S: Scheduled

[2]Reason:
A-Equipment Failure (Explain)
B-Maintenance or Test
C-Refueling
D-Regulatory Restriction
E-Operator Training & License Examination
F-Administrative
G-Operational Error (Explain)
H-Other (Explain)

[3]Method:
1-Manual
2-Manual Scram.
3-Automatic Scram.
4-Cont. of Existing Outage
5-Reduction
9-Other

[4]Exhibit G-Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File (NUREG-0161)

[5]Exhibit I-Same Source

OPERATING DATA REPORT

DOCKET NO. 50-328
DATE MAY 05, 1989
COMPLETED BY D.C.DUPREE
TELEPHONE (615)843-6722

OPERATING STATUS

1. UNIT NAME: SEQUOYAH NUCLEAR PLANT, UNIT 2 — NOTES:
2. REPORT PERIOD: APRIL 1989
3. LICENSED THERMAL POWER(MWT): 3411.0
4. NAMEPLATE RATING (GROSS MWE): 1220.6
5. DESIGN ELECTRICAL RATING (NET MWE): 1148.0
6. MAXIMUM DEPENDABLE CAPACITY (GROSS MWE): 1183.0
7. MAXIMUM DEPENDABLE CAPACITY (NET MWE): 1148.0
8. IF CHANGES OCCUR IN CAPACITY RATINGS(ITEMS NUMBERS 3 THROUGH 7)SINCE LAST REPORT, GIVE REASONS: ____________
9. POWER LEVEL TO WHICH RESTRICTED, IF ANY(NET MWE): ______
10. REASONS FOR RESTRICTIONS, IF ANY: ____________

		THIS MONTH	YR.-TO-DATE	CUMULATIVE
11.	HOURS IN REPORTING PERIOD	719.00	2879.00	60624.00
12.	NUMBER OF HOURS REACTOR WAS CRITICAL	222.47	651.72	27838.36
13.	REACTOR RESERVE SHUTDOWN HOURS	0.00	0.00	0.00
14.	HOURS GENERATOR ON-LINE	88.00	516.97	27108.94
15.	UNIT RESERVE SHUTDOWN HOURS	0.00	0.00	0.00
16.	GROSS THERMAL ENERGY GENERATED (MWH)	78400.23	1112587.67	82936685.73
17.	GROSS ELECTRICAL ENERGY GEN. (MWH)	17610.00	357880.00	28046200.00
18.	NET ELECTRICAL ENERGY GENERATED (MWH)	-1750.00	314101.00	26705197.00
19.	UNIT SERVICE FACTOR	12.24	17.96	44.72
20.	UNIT AVAILABILITY FACTOR	12.24	17.96	44.72
21.	UNIT CAPACITY FACTOR(USING MDC NET)	0.00	9.50	38.37
22.	UNIT CAPACITY FACTOR(USING DER NET)	0.00	9.50	38.37
23.	UNIT FORCED OUTAGE RATE	77.51	36.98	49.48

24. SHUTDOWNS SCHEDULED OVER NEXT 6 MONTHS (TYPE, DATE, AND DURATION OF EACH): ____________

25. IF SHUTDOWN AT END OF REPORT PERIOD, ESTIMATED DATE OF STARTUP: ____________

NOTE THAT THE YR.-TO-DATE AND
CUMULATIVE VALUES HAVE BEEN UPDATED.

SEQUOYAH NUCLEAR PLANT
AVERAGE DAILY POWER LEVEL

DOCKET NO. : 50-328
UNIT : TWO
DATE : MAY 08,1989
COMPLETED BY : D.C.DUPREE
TELEPHONE : (615)843-6722

MONTH: APRIL 1989

DAY	AVERAGE DAILY POWER LEVEL (MWe Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe Net)
01	-21	16	-28
02	-12	17	-29
03	-21	18	-33
04	-14	19	-28
05	-27	20	-30
06	-29	21	-32
07	-29	22	-28
08	-30	23	-29
09	-26	24	-30
10	-26	25	-32
11	-30	26	-29
12	-30	27	6
13	-31	28	110
14	69	29	154
15	-29	30	253

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-328
UNIT NAME Sequoyah Two
DATE May 5, 1989
COMPLETED BY D. C. Dupree
TELEPHONE (615) 843-6722

REPORT MONTH April 1989

No.	Date	Type[1]	Duration (Hours)	Reason[2]	Method of Shutting Down Reactor[3]	Licensee Event Report #	System Code[4]	Component Code[5]	Cause & Corrective Action to Prevent Recurrence
1	890118	S	325.13	C	4				Cycle 3 refueling outage.
2	890414	S	.42	B	9				Offline for turbine overspeed test, reactor at 30 percent power.
3	890415	F	304.92	A	3				Reactor tripped because of the erratic swinging of the feedwater regulator valves while turbine was being removed from service to perform overspeed test. The reactor was at 30 percent power.
4	890428	5	1.53	B	9				Offline for turbine overspeed test. Reactor at 23 percent.

[1]F: Forced
S: Scheduled

[2]Reason:
A-Equipment Failure (Explain)
B-Maintenance or Test
C-Refueling
D-Regulatory Restriction
E-Operator Training & License Examination
F-Administrative
G-Operational Error (Explain)
H-Other (Explain)

[3]Method:
1-Manual
2-Manual Scram.
3-Automatic Scram.
4-Cont. of Existing Outage
5-Reduction
9-Other

[4]Exhibit G-Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File (NUREG-0161)

[5]Exhibit I-Same Source

OPERATING STATISTICS
(TVA REPORTS)

TVA 7382A (ONP-4-88)

NUCLEAR PLANT OPERATING STATISTICS

Sequoyah Nuclear Plant

Period Hours 719 | Month April 19 89

	Item No.	Unit No.	UNIT ONE		UNIT TWO		PLANT
Generation	1	Average Hourly Gross Load, kW	1,171,433		200,114		1,065,514
	2	Maximum Hour Net Generation, MWh	1,156		300		1,432
	3	Core Thermal Energy Gen, GWD (t)[2]	101.9128		4.1000		106.0128
	4	Steam Gen. Thermal Energy Gen., GWD (t)[2]	102.2991		4.1168		106.4159
	5	Gross Electrical Gen., MWh	842,260		17,610		859,870
	6	Station Use, MWh	28,790		19,360		48,150
	7	Net Electrical Gen., MWh	813,470		- 1,750		811,720
	8	Station Use, Percent	3.42		109.94		5.60
	9	Accum. Core Avg. Exposure, MWD/Ton[1]	4,644		46		4,689
	10	CTEG This Month, 10^6 BTU	8,347,879		335,840		8,683,719
	11	SGTEG This Month, 10^6 BTU	8,379,523		337,213		8,716,736
	12						
Factors & Use	13	Hours Reactor Was Critical	719.0		222.47		941.47
	14	Unit Use, Hours-Min.	719:00		88:00		807:00
	15	Capacity Factor, Percent	99.02		2.07		50.55
	16	Turbine Avail. Factor, Percent	100.0		54.38		77.19
	17	Generator Avail. Factor, Percent	100.0		55.37		77.69
	18	Turbogen. Avail. Factor, Percent	100.0		54.38		77.19
	19	Reactor Avail. Factor, Percent	100.0		57.39		78.69
	20	Unit Avail. Factor, Percent	100.0		12.24		56.12
	21	Turbine Startups	0		3		3
	22	Reactor Cold Startups	0		1		1
	23						
Efficiency	24	Gross Heat Rate, Btu/kWh	9,910		19,070		10,100
	25	Net Heat Rate, Btu/kWh	10,260		N/A		10,700
	26	Gross heat Rate, Btu/kWh (w/oil)					10,100
	27	Net Heat Rate, Btu/kWh (w/oil)					10,700
Temp & Press	28	Throttle Pressure, psig	849.1		940.1		859.1
	29	Throttle Temperature, °F	526.8		538.8		528.1
	30	Exhaust Pressure, InHg Abs.	2.2		5.1		2.5
	31	Intake Water Temp., °F	56.8		61.7		57.3
	32						
Flows	33	Main Feedwater, M lb/hr	14.9		3.1		13.6
	34						
	35						
	36						
Misc.	37	Full Power Capacity, EFPD	404.86		411.60		816.46
	38	Accum. Cycle Full Power Days, EFPD	121.29		1.20		122.49
	39	Oil Fired for Generation, Gallons					1,056
	40	Oil Heating Value, Btu/Gal.					138,000
	41	Diesel Generation, MWh					16
	42						

Station Data	Max. Hour Net Gen.			Max. Day Net Gen.		Load Factor, %
	MWh	Time	Date	MWh	Date	
43	1,432	2400	4-30-89	32,496	4-30-89	47.04

Remarks: [1]For BFNP this value is MWD/STU and for SQNP and WBNP this value is MWD/MTU.

[2](t) indicates Thermal Energy.

Date Submitted MAY 12 1989 Date Revised

S. J. Smith
Plant Manager

TVA 6560C (PP-3-76)

UNIT OUTAGE AND AVAILABILITY

Sequoyah Nuclear Plant

Unit No. One

Licensed Reactor Power 3,411 MW(th)

Generator Rating 1220.5 MW(e)

Design Gross Electrical Rating 1,183 MW

Month/Year April 1989

Period Hours 719

Day	Time Unit Available						Time Not Available								Unit				OUTAGE CAUSE	METHOD OF SHUTTING DOWN REACTOR	UNIT STATUS DURING OUTAGE	CORRECTIVE ACTION TAKEN TO PREVENT REPETITION
	Total		Gen.		Not Used		Turbine		Gen.		Reactor		Unit		Time Out		Time In					
	Hrs	Min	Hrs	Min	Hrs	Min	Hrs	Min	Hrs	Min	Hrs	Min	Hrs	Min	Hrs	Min	Hrs	Min				
1	24	00	24	00																		
2	23	00	23	00																		
3	24	00	24	00																		
4	^	^	^	^																		
5																						
6																						
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24																						
25																						
26																						
27																						
28																						
29	v	v	v	v																		
30	24	00	24	00																		
31																						
Total	719	00	719	00																		

TVA 6560C (PP-3-76)

UNIT OUTAGE AND AVAILABILITY

Sequoyah Nuclear Plant

Unit No. Two

Licensed Reactor Power 3,411 MW(th)

Generator Rating 1220.5 MW(e)

Design Gross Electrical Rating 1,183 MW

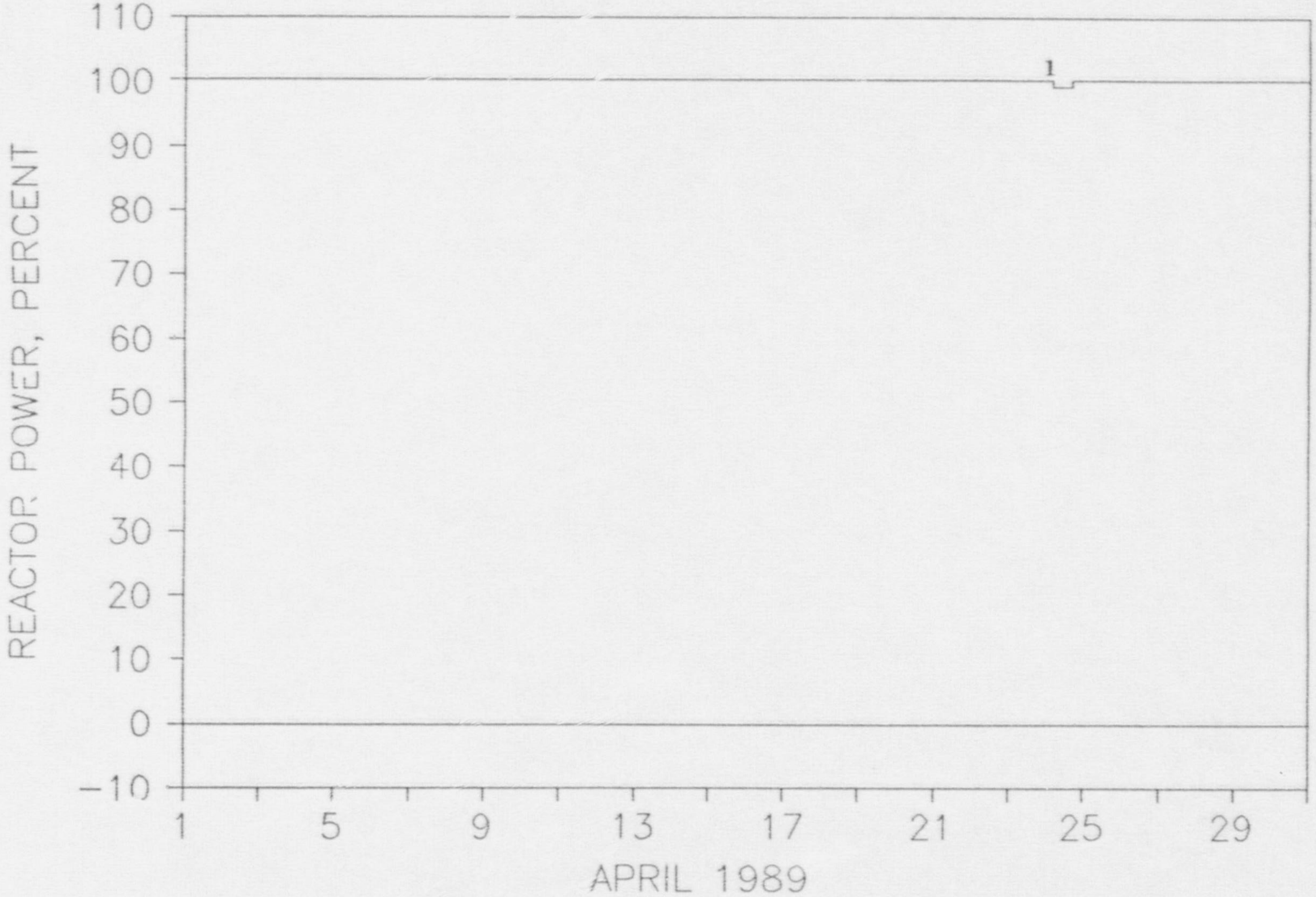
Month/Year April 1989

Period Hours 719

Day	Time Unit Available: Total Hrs	Total Min	Gen. Hrs	Gen. Min	Not Used Hrs	Not Used Min	Time Not Available: Turbine Hrs	Turbine Min	Gen. Hrs	Gen. Min	Reactor Hrs	Reactor Min	Unit Hrs	Unit Min	Unit Time Out Hrs	Time Out Min	Unit Time In Hrs	Time In Min	OUTAGE CAUSE	METHOD OF SHUTTING DOWN REACTOR	UNIT STATUS DURING OUTAGE	CORRECTIVE ACTION TAKEN TO PREVENT REPETITION
1							24	00	24	00	24	00	24	00								
2							23	00	23	00	23	00	23	00								
3							24	00	24	00	24	00	24	00								
4							^	^	^	^	^	^	^	^								
5																						
6																						
7																						
8																						
9																						
10											v	v										
11											20	24										
12							v	v	v	v			v	v								
13							24	00	24	00	05	13	24	00								
14	09	36	09	36			10	01	09	52	04	11	14	24	23	44	14	08	Turbine overspeed test	Reactor @ 30%	Mode 1	
15							00	09			01	42	24	00								
16											18	54	24	00								
17											16	13	^	^								
18																						
19											00	46										
20																						
21							05	42														
22																						
23																						
24																						
25													v	v								
26													24	00								
27	07	33	07	33									16	27			16	27				
28	23	47	23	47			00	13					00	13	23	47			Turbine overspeed test	Reactor @23%	Mode 1	
29	23	04	23	04			00	56					00	56			00	56				
30	24	00	24	00																		
31																						
Total	88	00	88	00			328	01	320	52	306	23	631	00								

REACTOR HISTOGRAM

SEQUOYAH UNIT ONE

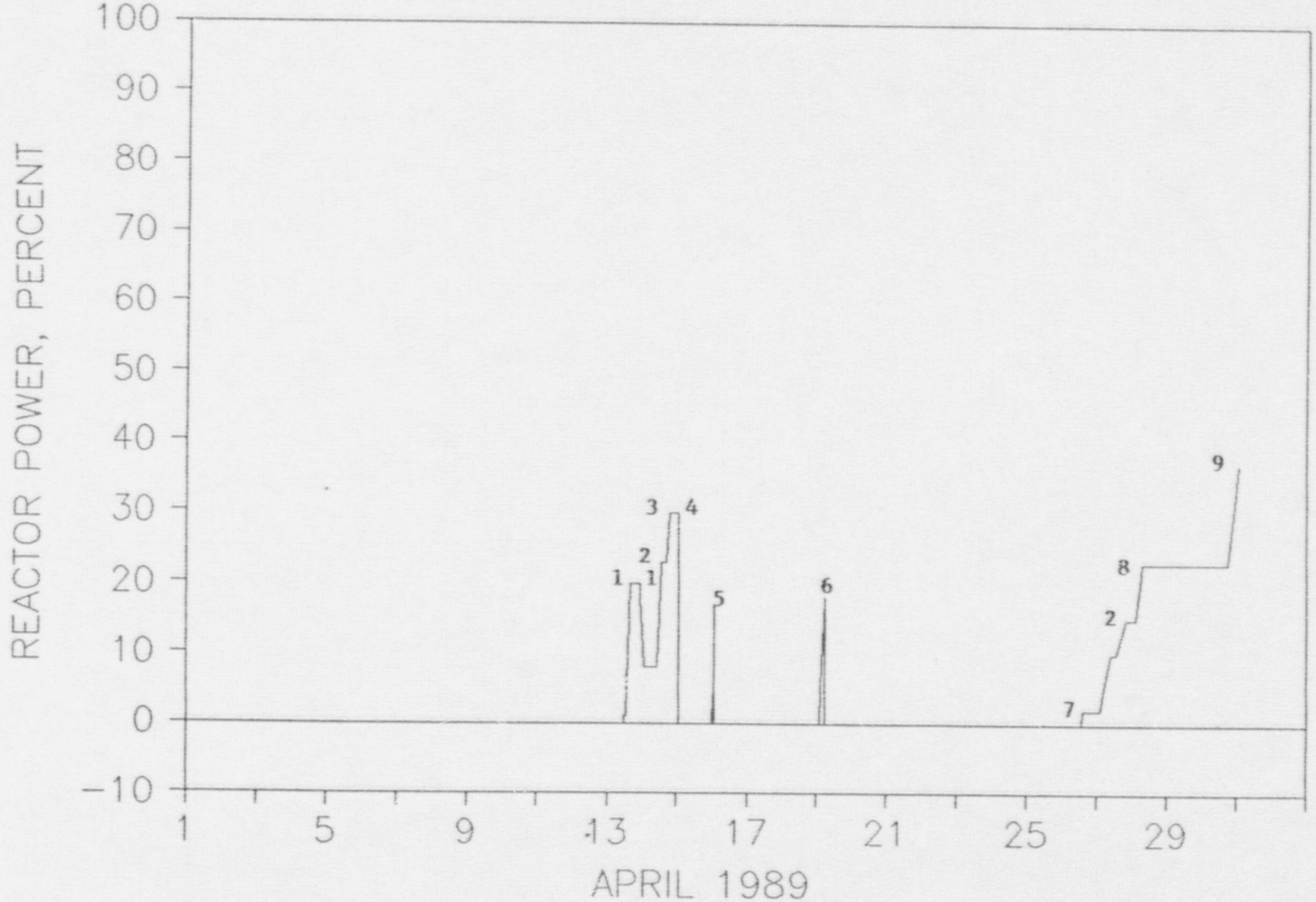


UNIT 1 REACTOR HISTOGRAM ANALYSIS

Unit 1

1. Condenser waterbox 1B-2 out of service for maintenance.

REACTOR HISTOGRAM
SEQUOYAH UNIT TWO



UNIT 2 REACTOR HISTOGRAM ANALYSIS

Unit 2

1. Holding/reduce power because of excessive RCS leakage.
2. Tying online.
3. Holding for turbine overspeed test.
4. Reactor trip, low-low level on S/G 4 (04/15/89).
5. Reactor trip, low-low level on S/G 1 (04/16/89).
6. Reactor trip, low-low level on S/G 2 (04/19/89).
7. Holding power at 2 percent power, maintenance on the feedwater system.
8. Holding for fuel conditioning and thermal power verification test.
9. Power ascension.

SUMMARY OF MAINTENANCE ACTIVITIES

MAINTENANCE SUMMARY
(ELECTRICAL)

02/13/89, 03/24/89	2-FCV-062-0022, During unit 2 refueling outage, a special inspection indicated RCP 2 seal return flow control valve was not operating. The solenoid coil housing and coil retainer were missing, allowing air to leak from the solenoid preventing the control valve from operating. Also, the solenoid coil was installed up-side-down. The solenoid was disassembled and reassembled correctly. The problem persisted. Replaced solenoid, reterminated wiring and returned valve to service. (WR B256476).
03/31/89, 04/02/89	2-FCV-001-0183-A, During unit 2 refueling outage, a visual alarm indicated steam generator 3 blowdown isolation valve throttled. This is a generic Masoneilan stem rotation problem. Removed stem nut and applied thread locker. Reinstalled nut and staked threads. Returned valve to service. (WR B256280).

MAINTENANCE SUMMARY
(INSTRUMENTATION)

05/04/88, 03/23/89 — 2-XE-068-0366,
During routine observation the RCS acoustic valve positioner accelerometer special element was not emitting a proper signal. The element was found to have a broken hardline cable. A new hardline cable was installed with Raychem, and the element was functionally tested. (WR B785232).

06/17/88, 03/07/89 — 2-PT-068-0301,
During unit shutdown the RCS pressurizer relief tank pressure transmitter required replacing because of instrument drift problems. Cause was determined to be the age of equipment and wear and because of drift. Installed new transmitter, rechecked calibration and found in tolerance. Performed postmaintenance test and returned to service. (WR B295731).

11/03/88, 02/25/89 — 1-PDI-030-0133,
During unit operation the ventilation system containment annulus delta P pressure differential indicator was found out of tolerance while performing SI-193, "Containment Building and Auxiliary Building Ventilation Systems - Units 0, 1 and 2." Root cause is unknown. It may have been because of the age of equipment. Recalibrated to desired tolerance and returned to service. No PRO report was necessary. (SI-193).

12/27/88, 04/06/89 — 2-PT-068-0066-E,
During unit 2 cycle 3 outage while performing SI-484, "Periodic Calibration of Reactor Vessel Level Instrumentation (RVLIS) and RCS Wide Range Pressure Channels (P-403, P-406) (Refueling Outage) 10 CFR 50.49, - Units 1 and 2," the RCS loop 4 hot leg pressure transmitter was found out of operational tolerance. The cause of failure was unknown but may have been because of transmitter drift or equipment age. The transmitter was recalibrated and returned to service during the performance of SI-484. EQ maintenance was performed in accordance with QMDS requirements.

12/27/88, 04/06/89 — 2-PT-068-0069-D,
During unit 2 cycle 3 outage while performing SI-484, the RCS wide-range pressure loop 3 hot leg pressure transmitter was found out of operational tolerance. The cause of failure was unknown but may have been because of drift or equipment age. The transmitter was recalibrated and returned to service during the performance of SI-484. EQ maintenance was performed in accordance with QMDS requirements.

01/20/89, 03/30/89	2-LT-063-0060, During unit outage the SIS accumulator tank 4 level transmitter was found out of tolerance while performing SI-161, "Channel Calibration of S.I.S. Accumulator Tank Water Level and Pressure Instrumentation - Units 1 and 2." The root cause is unknown. It may have been because of aging of equipment and/or cycling. Recalibrated to desired tolerance and returned to service. PRO 2-89-56 was initiated and submitted for failure analysis. (SI-161).
01/20/89, 03/30/89	2-LS-063-0060B, During unit outage the SIS accumulator tank 4 low level switch was found out of tolerance while performing SI-161. The root cause is unknown. It may have been because of aging and cycling of equipment. Recalibrated to desired tolerance and returned to service. PRO 2-89-56 was initiated and submitted for failure analysis. (SI-161).
01/20/89, 03/30/89	2-LT-063-0082, During unit outage the SIS level transmitter was found out of tolerance while performing SI-161. The root cause was unknown. It may have been because of age and/or cycling. Recalibrated to desired tolerance and returned to service. PRO 2-89-56 was initiated and submitted for failure analysis. (SI-161).
01/20/89, 03/30/89	2-LT-063-0089, During unit outage the SIS accumulator tank 3 level transmitter was found out of tolerance while performing SI-161. The root cause is unknown. It may have been because of age of equipment. Recalibrated to desired tolerance and returned to service. PRO 2-89-56 was initiated and submitted for failure analysis. (SI-161).
01/20/89, 03/30/89	2-LT-063-0099, During unit outage the SIS accumulator tank 2 level transmitter was found out of tolerance while performing SI-161. The root cause was unknown. It may have been because of the age of equipment. Recalibrated to desired tolerance and returned to service. PRO 2-89-56 was initiated and submitted for failure analysis. (SI-161).
01/20/89, 03/30/89	2-LS-063-0099B, During unit outage the SIS accumulator tank 2 level switch was found out of tolerance while performing SI-161. The root cause is unknown. It may have been because of aging of equipment. Recalibrated to desired tolerance and returned to service. PRO 2-89-56 was initiated and submitted for failure analysis. (SI-161).

01/20/89, 03/30/89	2-LT-063-0109, During unit outage the SIS level transmitter was found out of tolerance while performing SI-161. The root cause was unknown. It may have been because of equipment aging and/or cycling. Recalibrated to desired tolerance and returned to service. PRO 2-89-56 was initiated and submitted for failure analysis. (SI-161).
01/20/89, 03/30/89	2-LT-063-0119, During unit outage the SIS accumulator tank 1 level transmitter was found out of tolerance while performing SI-161. The root cause is unknown. It may have been because of age of equipment. Recalibrated to desired tolerance and returned to service. PRO 2-89-56 was initiated and submitted for failure analysis. (SI-161).
01/20/89, 03/30/89	2-LT-063-0129, During unit outage the SIS accumulator tank 1 level transmitter was found out of tolerance while performing SI-161. The root cause is unknown. It may have been because of aging of equipment. Recalibrated to desired tolerance and returned to service. PRO 2-89-56 was initiated an submitted for failure analysis. (SI-161).
01/21/89, 03/16/89	2-L[illegible]-003-0038-E, During unit 2 cycle 3 outage while performing SI-94.4, "Reactor Trip/Engineered Safety Feature/Accident Monitoring Instrument Steam Generator Level Channel Calibrations (18 Months) Units 1 or 2," the main and auxiliary feedwater system steam generator 1 level transmitter was found out of operational desired values. The transmitter was out of tolerance because of transmitter drift. This is a generic problem to Barton 764 transmitters and has been identified. Trending data is being evaluated. The transmitter was recalibrated and returned to service during the performance of SI-94.4. Applied lubricant and installed inner and outer o-rings and torqued cover screws. (SI-94.4).
01/21/89, 03/16/89	2-LT-003-0039-F, During unit 2 cycle 3 outage while performing SI-94.4, the main and auxiliary feedwater system steam generator 1 level transmitter was found out of operational desired values. The transmitter was out of tolerance because of transmitter drift. This is a generic problem to Barton 764 transmitters and has been identified. Trending data is being evaluated. The transmitter was recalibrated and returned to service during the performance of SI-94.4. PRO 2-89-35 was initiated, but after evaluation the problem was determined not reportable. Applied lubricant and installed inner and outer o-rings and torqued cover screws. (SI-94.4).

01/21/89, 03/16/89
2-LT-003-0097-G,
During unit 2 cycle 3 outage while performing SI-94.4, the main and auxiliary feedwater system steam generator 3 level transmitter was found out of operational desired values. The transmitter was out of tolerance because of transmitter drift. This is a generic problem to Barton 764 transmitters and has been identified. Trending data is being evaluated. The transmitter was recalibrated and returned to service during the performance of SI-94.4. Applied lubricant and installed inner and outer o-rings and torqued cover screws. (SI-94.4).

01/21/89, 03/16/89
2-LT-003-0106-E,
During unit 2 cycle 3 outage while performing SI-94.4, the main and auxiliary feedwater system steam generator 4 level transmitter was found out of operational desired values. The transmitter was out of tolerance because of transmitter drift. This is a generic problem to Barton 764 transmitters and has been identified. Trending data is being evaluated. The transmitter was recalibrated and returned to service during the performance of SI-94.4. Applied lubricant and installed inner and outer o-rings and torqued cover screws. (SI-94.4).

01/25/89, 02/25/89
2-TS-001-0018B-B,
During unit 2 cycle 3 outage while performing SI-580, "Periodic Calibration of Main Steam System (Refueling Cycle), - Units 1 and 2," the main steam system flow to AFWP turbine isolation high temperature switch was found out of operational setpoint tolerance. The cause of failure was unknown but may have been because of setpoint drift or equipment age. The switch was recalibrated and returned to service during SI-580. EQ maintenance was performed in accordance with QMDS requirements. (SI-580).

01/27/89, 02/10/89
2-PT-063-0086,
During unit operation the SIS accumulator tank 3 pressure transmitter output reading was 94mA with no pressure applied. Defective amplifier probably because of age and/or wear. Replaced amplifier and powered up transmitter. Calibrated and returned to service. (WR B757658).

01/27/89, 03/06/89
2-PT-068-0068-F,
During unit 2 cycle 3 outage while performing SI-199, "Periodic Calibration of Reactor Coolant System Instrumentation (Refueling Cycle), - Units 1 and 2," the RCS loop 4 hot leg pressure transmitter was found out of operational tolerance. The cause of failure was unknown but may have been because of transmitter drift. The transmitter was recalibrated and returned to service during the performance of SI-199. EQ maintenance was performed in accordance with QMDS requirements. (SI-199).

01/28/89, 03/02/89	2-LT-068-0325C, During unit outage the RCS pressurizer level transmitter was found out of calibration while performing SI-88.2, "Remote Shutdown Monitoring Instrumentation - Pressurizer Level Channel Calibration (Refueling Outage), - Unit 2." The root cause is unknown. It could have drifted out of tolerance from age of equipment and/or wear from cycling. Recalibrated to desired tolerance, performed postmaintenance test and returned to normal. (SI-88.1).
01/28/89, 03/21/89	2-LT-068-0326C, During unit outage the RCS pressurizer level transmitter was found out of calibration while performing SI-88.2. The root cause is unknown. It could have drifted out of tolerance from age of equipment and/or wear from cycling. Recalibrated to desired tolerance, performed postmaintenance test and returned to service. (SI-88.1).
02/02/89, 03/10/89	2-PT-068-0322-G, During unit 2 cycle 3 outage while performing SI-94.88, "Channel Calibration of Pressurizer Pressure Channel 4, Rack 12 Loop P-68-322 (P-458) - Unit 2," the RCS pressurizer pressure transmitter was found out of operational tolerance. The cause of failure was unknown but may have been because of transmitter drift or equipment age. The transmitter was recalibrated and returned to service during the performance of SI-94.88. EQ maintenance was performed in accordance with QMDS requirements.
02/03/89, 03/08/89	2-PT-068-0334-E, During unit 2 cycle 3 outage while performing SI-94.86, "Channel Calibration of Pressurizer Pressure Channel 2, Rack 5 Loop P-68-334 (P-456) - Unit 2," the RCS pressurizer pressure transmitter was found out of operational tolerance. The cause of failure was unknown but may have been because of transmitter drift. The transmitter was recalibrated and returned to service during SI-94.86. Lubricant was applied to the new inner and outer o-rings then installed. The cover was torqued in accordance with QMDS requirements. (SI-94.86).
02/06/89, 03/14/89	2-LS-068-0339D, During unit outage the RCS pressurizer level switch was found tripping at different setpoints while performing SI-94.1, "Reactor Trip Instrumentation Refueling Outage Channel Calibration (RCS PRZ Press and Levels), - Units 1 and 2." The root cause is unknown. It may have been because of cycling fatigue and/or a deficiency in design. Tried a different wiring scheme and calibrated switch. Nuclear Engineering is evaluating the problem and will provide a recommendation at a later date. Returned to service. (WR B769345).

02/09/89, 03/08/89	2-PT-068-0340-D, During unit 2 cycle 3 outage while performing SI-94.85, "Channel Calibration of Pressurizer Presure Channel 1, Rack 1 Loop P-68-340 (P-455) - Unit 2," the RCS pressurizer pressure transmitter was found out of operational tolerance. The cause of failure was unknown but may have been because of equipment aging. The transmitter was recalibrated and returned to service during the performance of SI-94.85. Lubricant was applied on new inner and outer o-rings then installed. The cover was torqued in accordance with QMDS requirements. (SI-94.85).
02/10/89, 03/22/89	2-FT-001-0021A-D, During unit 2 cycle 3 outage while performing SI-98.1, "Channel Calibration for Engineered Safety Feature Instrumentation (Steam Flow & Pressure), - Units 1 and 2," the main steam system steam generator 3 main steam header flow channel 1 transmitter was found out of operational tolerance. The cause of failure was unknown but may have been because of transmitter drift. The transmitter was recalibrated and returned to service during the performance of SI-98.1. EQ maintenance performed in accordance with QMDS requirements. (SI-98.1).
02/13/89, 03/08/89	2-PT-068-0323-F, During unit 2 cycle 3 outage while performing SI-94.87, "Channel Calibration of Pressurizer Pressure Channel 3, Rack 9 Loop P-68-323 (P-457) - Unit 2," the RCS pressurizer pressure transmitter was found out of operational tolerance. The cause of failure was unknown but may have been because of transmitter drift because of equipment age. The transmitter was recalibrated and returned to service during SI-94.87. Lubricant was applied to new inner and outer o-rings then installed. The cover was torqued in accordance with QMDS requirements. (SI-94.87).
02/13/89, 03/08/89	2-PM-068-0323B-F, During unit outage the RCS pressurizer pressure modifier was found out of tolerance while performing SI-94.87. The root cause is unknown. It may have been because of the age of the equipment and/or cycling fatigue. Recalibrated controller to desired tolerance and returned to service. No PRO was initiated. (SI-94.87).
03/02/89, 03/28/89	2-FM-003-0155A, During unit outage the main and auxiliary feedwater system auxiliary feedwater to steam generator 2 flow modifier was found out of tolerance while performing SI-97.2, "Calibration of Auxiliary Feedwater Flow Rate for Remote Shutdown and Accident Monitoring (550 Days), Unit 2." The modifier was out of calibration, possibly because of the age of equipment and/or wear. Recalibrated to desired tolerance and returned to service. (SI-97.2).

03/02/89, 03/28/89	2-FM-003-0170A, During unit outage the main and auxiliary feedwater system steam generator 4 flow modifier was found out of tolerance while performing SI-97.2. The modifier was out of calibration, possibly because of age of equipment and/or wear. Recalibrated to desired tolerance and returned to service. (SI-97.2).
03/03/89, 03/06/89	2-LT-063-0082, During unit outage the CVCS level transmitter was found with the "strain gage" wire broken while performing SI-161. The root cause is unknown. It could have been from age of equipment, vibration and/or fatigue. Replaced "strain gage" and calibrated to desired tolerance. (WR B769963).
03/11/89, 03/27/89	2-LT-068-0339-D, During unit 2 cycle 3 outage while performing SI-94.1, the RCS pressurizer level channel 1 transmitter was found out of operational tolerance. The cause of failure was unknown but may have been because of setpoint drift due to equipment age. The transmitter was recalibrated and returned to service during the performance of SI-94.1. Lubricant was applied to new inner and outer o-rings and installed. The cover was torqued in accordance with QMDS requirements.
03/15/89, 03/17/89	2-PC-068-0340A, During unit refueling the RCS pressurizer pressure controller was found out of tolerance while performing SI-94.90, "Channel Calibration of Pressurizer Pressure Control and Operational Test of PORV's PCV-68-340a (PCV-455A) and (PCV-456) - Unit 2." The root cause is unknown. It may have been because of age of equipment and/or cycling fatigue. Recalibrated the controller to desired tolerance and returned to service. (SI-94.90).
03/16/89, 03/23/89	2-FM-002-0035A, During unit outage the condensate system gland seal heat exchanger recirculating flow control valve was found modulating open. Modifier was moving around from vibration. Reinstalled a new support stand adjacent to valve to reduce vibration. Mechanical maintenance is working WR B795767 on valve. (WR B795768).
03/16/89, 03/16/89	2-LM-003-0172-A, During unit outage the main and auxiliary feedwater system steam generator 3 level modifier input resistor was found cracked which caused the output to read low. This was attributed to the age of equipment and heat stress. Replaced the input resistor and the 100 microfarads capacitor. Bench calibrated and returned to service. (WR B757383).

03/21/89, 03/21/89	2-FCV-062-0093A, During unit outage the CVCS charging header flow controller would not stroke valve properly. The cam was worn out from age and use. Replaced cam and calibrated controller. Verified stroke and returned to service. (WR B757196).

MAINTENANCE SUMMARY
(MECHANICAL)

MECHANICAL MAINTENANCE
MONTHLY REPORT FOR APRIL 1989

Unit 1

1. Completed monthly inspection on D/G 1A-A and 1B-B.
2. Replaced fittings on SI pump 1B-B.
3. Flushed oil system on HDTP 3.
4. Replaced BAE rupture disc A.
5. Completed repairs on various valves (systems 27, 61, and 90).

Unit 2

1. Repacked AFW 2A-A and 2B-B.
2. Reinstalled MFP turning gear motor 2B-B.
3. Rebuilt bus duct fan 2A and 2B.
4. Replaced and adjusted packing in condenser vacuum pumps.
5. Balanced RCPs.
6. Replaced V-belts on the east main steam vault air handling unit.
7. Completed repairs on various valves (systems 1, 3, 43, 62, 70, and 74).

Unit 0

1. Rebuilt control air compressor D.
2. Completed repair on waste gas compressor A.
3. Rebuilt control air compressor B.
4. Rebuilt Auxiliary Building floor/equipment sump pump.
5. Completed work on floor drain collector tank.
6. Rebuilt CDWE pump.
7. Replaced refueling water purification filters.
8. Completed repairs on various valves (systems 27 and 67).

Other

1. Completed closure of various CAQRs, etc.

08/12/85, 12/28/88	1-FCV-062-0070-A, With unit 1 shutdown and during routine observation, the CVCS reactor coolant loop 3 letdown flow control valve was discovered to be leaking through when closed to isolate letdown. The cause was determined to be dirty internals and steam cut on plug and seat. Bolting is also too short. Installed new plug, stem and seat with new gaskets and replaced four studs with new studs. (WR B529490).
09/10/87, 02/23/89	2-FCV-072-0023-A, With unit 2 in operation, the containment spray pump A flow control valve from containment sump was discovered to have lots of boron buildup around bonnet of the valve. Packing had been adjusted until there was no adjustment left. New packing was installed. MOVATS to be performed on preventive maintenance 3851. (WR B276615).
09/14/87, 10/05/87	1-FCV-062-0093-A, During routine observation and with unit 1 shutdown, the CVCS charging header flow control valve was discovered to be allowing flow with valve closed and in manual. Erosion of plug and cage had been caused by operating condition in extended mode 5 outage. Fabricated and installed new stem, plug and cage. (WR B288515).
01/12/88, 02/24/89	2-FCV-062-0083, During unit 2 shutdown the CVCS RHR letdown flow control valve was discovered leaking during routine observation. This was attributed to normal wear. Stem threads were staked, new packing was installed and nut was tightened. (WR B257608).
01/14/88, 07/15/88	1-VLV-063-0640-S, During unit 1 shutdown the RHR heat exchanger discharge check valve was discovered with excessive boron buildup on the valve body. This was found during incidental observation and was because of aging of bonnet gasket. Installed new bonnet gasket and new studs and retorqued valve. (WR B281341).
03/16/88, 02/03/89	2-FCV-062-0084, During unit 2 shutdown the CVCS charging flow to RCS spray valve was discovered during routine observation to be leaking borated water. There was no adjustment left on the packing. Cleaned boron off of valve and installed new packing. (WR B238217).
03/16/88, 02/02/89	2-FCV-062-0085-B, During unit 2 shutdown the CVCS charging flow RCS cold leg loop 1 valve was discovered during routine observation to be leaking borated water. No adjustment was left on packing. Cleaned boron off of valve, added packing, and adjusted. (WR B238217).

03/16/88, 02/24/89 — 2-VLV-062-0564-S,
While performing SI-146.1, "Reactor Coolant System Leak Test, Unit 2," during unit 2 shutdown, the CVCS number 1 seal bypass isolation valve was leaking borated water. No adjustment was left on packing. New packing was installed. (WR B223346).

03/16/88, 02/24/89 — 2-VLV-062-0566-S,
While performing SI-146.1 during unit 2 shutdown, the CVCS seal water injection isolation valve was leaking borated water. No adjustment was left on packing. New packing was installed. (WR B223346).

03/16/88, 02/24/89 — 2-VLV-062-0596-S,
While performing SI-146.1 during unit 2 shutdown, the CVCS number 1 seal bypass isolation valve was discovered leaking borated water. No adjustment was left on packing. New packing was installed. (WR B223346).

03/16/88, 02/24/89 — 2-VLV-062-0608-S,
While performing SI-146.1 during unit 2 shutdown, the CVCS number 1 seal leakoff bypass valve was leaking borated water. No adjustment was left on packing. New packing was installed. (WR B223346).

03/16/88, 02/01/89 — 2-VLV-062-0660-S,
While performing SI-146.1 during unit 2 shutdown, the alternate charging check valve was discovered leaking borated water. This valve is in the CVCS. The bonnet stud was bent and bonnet seal was old. A new bonnet seal-ring and stud were installed. (WR B279034).

03/16/88, 01/31/89 — 2-VLV-062-0717-S,
During unit 2 shutdown and performance of SI-146.1, the CVCS alternate charging check valve was discovered leaking borated water because of age. A new bonnet seal-ring was installed. (WR B279034).

03/16/88, 03/12/89 — 2-VLV-068-0535-S,
The RCS loop 3 cold leg manifold isolation valve was discovered leaking borated water during performance of SI-146.1. No adjustment was left on packing. Cleaned stuffing box and installed new packing. (WR B279038).

03/16/88, 01/30/89 — 2-VLV-063-0583-S,
During unit 2 shutdown while performing SI-146.1, the SIS boron injection valve cold leg loop 2 was found to be leaking borated water. No adjustment was left on packing. Boron was cleaned frum valve, and new packing was installed. (WR B279040).

04/08/88, 02/16/89	2-VLV-063-0610-S, By routine observation during unit 2 shutdown, the SIS accumulator 1 fill isolation valve was discovered leaking borated water. No adjustment was left on packing, so new packing was installed. (WR B279524).
04/26/88, 03/05/89	2-FCV-063-0111, By routine observation during unit 2 shutdown, the SIS check valve leak test isolation valve was discovered leaking through. Cause was unknown. No apparent problem was found when valve was disassembled. A new bonnet gasket and packing were installed. (WR B271239).
07/15/88, 08/13/88	2-TCV-067-0158, During unit 2 operation the ERCW shutdown board room air conditioner condensate supply control valve failed closed and will not open. This valve needs to be open to perform SI-566, "ERCW Flow Verification Test, - Units 0, 1 and 2." Valve had sludge above the top surface of the top of the diaphragm. Valve was dirty and corroded. Installed new diaphragm and o-rings in valve. Installed 1/2" plug where control tubing normally is. Did not install pilot valve internals. (WR B261668).
08/15/88, 03/04/89	2-FCV-001-0015-A, During unit 2 shutdown and routine observation, the main steam system AFW pump turbine steam supply valve from steam generator 1 was discovered to be leaking. No adjustment was left on packing. Cleaned stuffing box and installed new packing. (WR B789196).
08/15/88, 03/04/89	2-FCV-001-0016-A, During unit 2 shutdown and routine observation, the main steam system AFW pump turbine steam supply valve from steam generator 4 was discovered to be leaking. No adjustment was left on packing. Cleaned stuffing box and installed new packing. (WR B789196).
09/27/88, 11/23/88	1-VLV-070-0696A-A, During unit 1 shutdown and routine observation, the CCS return valve from the RCP oil cooler was discovered to be leaking. The packing had no more adjustment left. Installed new packing, cleaned all parts and lubricated. (WR B789865).

Mechanical Maintenance Report

09/27/88, 11/23/88 — 1-VLV-070-0696B-A,
During unit 1 shutdown the CCS return valve from RCP motor 2 oil cooler was discovered leaking during routine observation. Packing had no more adjustment left. Installed new packing, cleaned all parts and lubricated. (WR B789865).

10/24/88, 03/27/89 — 1-VLV-001-0512,
During unit 1 shutdown while performing SI-111, "Testing and Setting of Main Steam Safety Valves, - Units 1 and 2," the main steam safety valve was discovered leaking past the seat. Leakage was because of scale in system. Also, there was nozzle damage during disassembly of valve. Installed new nozzle and lapped disc insert and nozzle. This valve was not reinstalled; it was returned to power stores to become a spare. A new valve was installed. (WR B769839).

01/19/89, 02/04/89 — 2-FCV-062-0070-A,
During unit 2 cycle 3 refueling outage, routine observation discovered that the CVCS reactor coolant loop 3 letdown flow control valve was not opening because of a blown diaphragm. Operator age and condition may have contributed. Installed new diaphragm and replaced packing in operator. (WR B797646).

01/19/89, 03/05/89 — 2-FCV-063-0080-A,
During unit 2 cycle 3 refueling outage, the SIS accumulator tank 3 flow isolation valve was discovered to be leaking borated water. Packing had been adjusted down. Cleaned stuffing box and added packing to valve and adjusted. (WR B238800).

01/19/89, 01/30/89 — 2-VLV-063-0582-S,
During unit 2 cycle 3 refueling outage, routine observation of the SIS boron injection valve cold leg loop 1 detected leakage. No adjustment was left on packing. Cleaned boron from valve and repacked valve. (WR B238799).

01/23/89, 02/20/89 — 2-FCV-068-0305-A,
During unit 2 cycle 3 refueling outage, the pressurizer relief tank was routinely observed increasing in pressure. The RCS nitrogen manifold flow control valve was leaking through because of normal wear. Installed air jumper, rebuilt valve per maintenance instruction (MI) 11.7.2, "Air Operated Grinnell Diaphragm Valve Rebuilding With Air Operator Model No. 3225 (Air To Open) For All Systems." Removed air jumper and reinstalled air line. (WR B238868).

01/27/89, 03/09/89	2-FCV-067-0088-B, During unit 2 cycle 3 refueling outage while performing SI- 158.1, "Containment Isolation Valve Leak Rate Test, - Units 1 and 2," the ERCW lower containment cooler A discharge isolation valve outside of containment failed leak rate test. There was a cut on the rubber seat and limitorque was out of adjustment. Cleaned seat with fine sandpaper. Replaced grease in limitorque operator and MOVATS test was performed on WR B292072. (WR B753972).
01/31/89, 02/02/89	2-VLV-061-0692, During unit 2 cycle 3 refueling outage, the ice condenser system glycol bypass check valve failed leak rate test while performing SI-158.1. Seating surface had trash on it, and the spring was bad. Lapped seat, installed new spring and holder assembly. (WR B271353).
01/31/89, 02/02/89	2-VLV-061-0745, During unit 2 operation while performing SI-158.1, the ice condenser glycol bypass check valve failed leak rate test because of trash on seat. Disassembled top of valve, removed spring, installed red rubber disc to block valve closed. Lapped seat and installed new spring and holder assembly. Reassembled valve. (WR B271354).
02/02/89, 03/25/89	2-VLV-067-0585B-B, During unit 2 cycle 3 refueling outage while performing SI-158.1, the ERCW return upper containment cooler check valve failed leak rate test because of rust on seat and disc. Cleaned body and internals, lapped seat and disc and installed spring. (WR B271355).
02/06/89, 02/15/89	2-VLV-072-0513-B, During unit 2 cycle 3 outage while performing SI-164, "Testing Setpoint of Safety Relief Valves (ASME Section XI Category C Valves) Unit 0, 1 and 2," the containment spray system pump suction relief valve was discovered leaking through because of dirty internals. Cleaned internals, lapped seating surface and performed SI-164 as post maintenance test and it was acceptable. (WR B784038).
02/24/89, 03/06/89	2-VLV-063-0643-S, During unit 2 cycle 3 refueling outage, the RHR heat exchanger discharge check valve was discovered to be leaking borated water because of age. Installed new bonnet gasket and cleaned boron off valve body. (WR B759785).

Mechanical Maintenance Report

02/27/89, 03/14/89

2-VLV-068-0530-S,
During unit 2 cycle 3 refueling outage, the RCS loop 3 hot leg manifold isolation valve was discovered to be leaking borated water. Cause was attributed to age. Installed new bonnet gasket, replaced valve and cleaned boron from valve. (WR B233093).

03/03/89, 03/04/89

2-VLV-070-0553A-A,
During unit 2 cycle 3 outage, the CCS return component cooling pump mechanical seal cooler water flow was not being detected on the flow indicator (2-FI-70-146). This was found during routine observation and was caused by scale and corrosion from piping internals. Removed valve internals, cleaned all parts. Lubed and reassembled valve. Torqued to 400 foot pounds and installed new diaphragm. No maintenance was performed on flow indicator. (WR B28[illegible]854).

03/14/89, 03/22/89

2-VLV-062-0660-S,
During unit 2 cycle 3 refueling outage, the CVCS alternate charging check valve was discovered to be leaking at the bonnet. There were scuff marks on internal parts and one on seating surface. There was also minor pitting on bonnet. Cleaned valve internals, lightly polished seating surfaces on valve bonnet and valve body. Reinstalled valve bonnet with new seal ring. (WR B275583).

03/16/89, 03/19/89

2-FCV-002-0035A,
During unit 2 cycle 3 refuel outage, the condensate system gland seal heat exchanger recirculating flow control valve was discovered to be leaking from valve body. The body of the valve had a crack in it. Removed valve from system, welded body and reinstalled valve. (WR B795767).

03/17/89, 03/22/89

2-VLV-067-0575A-A,
During unit 2 cycle 3 refuel outage, the ERCW return level control valve cooler check valve was discovered leaking through during routine observation. The seating surface was dirty. Disassembled valve, installed rubber blocks, reinstalled top of valve. Disassembled valve, removed rubber block, cleaned seating surface and reassembled valve. (WR B775973).

03/19/89, 03/24/89

2-VLV-067-0573A-A,
During unit 2 cycle 3 outage while performing SI-156, "Containment Integrated Leak Rate Test - Units 1 and 2," the ERCW return level control valve cooler valve was discovered leaking through because of dirty internals. Lapped seat and nozzle, cleaned internals and tested valve with water. (WR B775733).

Mechanical Maintenance Report

03/20/89, 03/21/89	2-TRB-001-0034, During unit 2 cycle 3 refuel outage, the main feedwater pump 2A turbine was discovered during routine observation to have a broken ruptured disc. This was attributed to normal wear. Installed new ruptured disc. (WR B769462).
03/22/89, 03/25/89	2-PMP-002-0020, During unit 2 cycle 3 refueling outage, the condensate system hotwell pump 2C was discovered during routine observation to be leaking at the seal. The retainer ring was bent out of place. A new mechanical seal was installed. (WR B256624).
03/26/89, 03/29/89	2-VLV-062-0649-S, During unit 2 cycle 3 refueling outage while performing SI-632.4.3, "Auxiliary Building Chemical and Volume Control System Unit 2 Train A External Leakage - Unit 2," the CVCS seal water heat exchanger relief valve was leaking. The seat was slightly dirty, and the outlet was full of water. Cleaned internals, lightly lapped seats and installed new bellows assembly. (WR B775280).
04/02/89, 04/03/89	2-FCV-001-0182-B, During unit 2 cycle 3 refueling outage while performing SI-166.6, "Testing of Category 'A' and 'B' Valves After Maintenance or Upon Release From a Hold Order - Units 1 and 2," the main steam system steam generator 2 blowdown isolation valve inside containment failed stroke time. The stem nut was loose. This is a generic Masoneilan valve rotation problem. Tightened nut and installed locktite. (WR B256255).
04/03/89, 04/04/89	2-FCV-062-0093, During unit 2 cycle 3 refueling outage, the CVCS charging header flow control valve was suspected to have a packing leak. This was found during routine observation of temperature readings on the packing leakoff line. The cause was attributed to the fact that the packing had no adjustment left. Installed new packing. (WR B256257).
04/18/89, 04/18/89	2-PMP-003-0118, During unit 2 cycle 3 refueling outage, the MDAFWP 2A-A was discovered during incidental observation to be leaking at the outboard end of pump. No adjustment was left on packing. Cleaned stuffing box and installed new packing (WR B797429).

04/19/89, 04/20/89	2-PMP-003-0128, During the restart of unit 2 following an automatic trip because of feedwater level control, MDAFWP 2B-B was discovered during incidental observation with no packing adjustment left. The packing blew out before the pump was removed from service for repacking. Low flow characteristic as a result of plant conditions caused the pump discharge check valve to chatter. This valve chatter caused higher than normal pressures in the outboard stuffing box causing the packing to fail. Installed new packing and added oil to inboard and outboard bearings. Added 12 ozs. of oil. (WR B282552).

MAINTENANCE SUMMARY
(MODIFICATIONS)

Major Capital Projects:

PN7108: ECN 6720 - Crane Consistency Program

Unit 1 polar crane limit switch weights remain to be painted. Completion is scheduled for unit 1 cycle 4 (U1C4).

Completed cranes - unit 1 Turbine Building 15-ton crane, unit 2 polar crane, unit 2 Turbine Building 200-ton crane, unit 2 Turbine Building 15-ton crane, Service Building 5-ton crane, and Turbine Building 10-ton hatchway crane.

Remaining listed cranes are to be modified after unit 2 cycle 3 (U2C3) - Auxiliary Building 125-ton crane, waste packaging crane, railroad bay crane, and unit 1 Turbine Building 200-ton crane.

PN7130: ECN 6180 - Postaccident Monitoring

Work is currently being held for material.

PN7132: DCN 0026 - Sewage Treatment Facility and Civil Upgrade

All work is complete.

PN7161: ECN 5855 - Replacement of Doors A56 and A57

WP 09679 remains on hold and is partially complete.

Other Items:

ECN 5111 - Provide Permanent Power to Manholes 42-46

Work has stopped because of lack of funding. The cable has been run from breaker 4E at the 480-V common board in the Turbine Building through manhole 1 to manhole 42. All material was purchased before the job was stopped for lack of funding. Work is 45 to 50 percent complete.

ECN 5503 - Evacuation Alarms O&PS/Fire Detection O&PS

WP 12482 - Work has stopped because of lack of funding.

ECN 5552 - Condensate Demineralizer Modifications and High Crud Filter

WP 5552-03 - Fieldwork is complete.

Other Items (cont.):

ECN 5609 - Alteration to the Makeup Water Treatment Plant

WP 12387 - Work is 90 percent complete. WP is now in work.

WP 12576 - Work is complete. WP revision is required before closing.

WP 12633 - Work is approximately 90 percent complete because of redesigns.

WP 12665 - Work is field complete. WP is in final closure.

WP 12682 - WP is 80 percent complete. Awaiting installation of a pump for the alum sludge pond.

WP 12684 - WP is field complete. Equipment calibration and functional tests remain.

WP 12731 - WP is approximately 97 percent complete. MODS needs approval of FCR 8211 by Systems Engineering.

ECN 5626 - Containment Ladders, Unit 1

MODS needs additional design information to complete. NE needs to issue all drawings listed on this ECN. Work has not begun because of this holdup.

ECN 5841 - Hot Shop Fire Protection/Evacuation Alarm

WP 12360 is field complete. Awaiting drawings to be updated.

ECN 5911 - Waste Disposal Piping Addition

System tie-ins are in progress and scheduled to be completed by May 15, 1989.

ECN 5916 - Replacement of Cask Decon Collector Tank Pumps

Fabrication and installation of waste disposal piping is in progress.

ECN 5935 - Correct Power Block Lighting Deficiencies

WP 12437 is complete. WP 12275 is complete. WP 5935-01 is field complete. Waiting on secondary drawings to be revised.

WP 5935-02 is field complete.

ECN 5977 - Install Steam Generator Blowdown Demineralizer

System tie-ins are planned for refueling outage.

Other Items (cont.):

ECN 6357 - ERCW Roof Access and Rails for Security Equipment

Original design for WP 12238 was rejected by Operations. NE to rework design to comply with Operations' needs and attempt to salvage existing work.

ECN 6388 - Hydrogen Monitors in Switchyard

WP 12223 - Work has been stopped because of a lack of funding.

ECN 6429 - Component Cooling Heat Exchanger Replacement

Work on heat exchanger B is complete, except for insulation and painting. Work on heat exchanger C will resume as soon as funding is approved.

ECN 6815 - 500-kV Switchyard Addition

A 30-liter air replenishing tank had to be purchased to replace a defective tank on PCB 5018. This tank is scheduled to be delivered by May 15, 1989. Receipt of this tank will allow continuation of PCB testing. Software changes remain to be done for the data logger. Testing remains for the switchyard data acquisition. A decision has been made to retire the 161-kV equipment in bay 20 by leaving it in place. This decision will require revision of design drawings by means of an FCR. FCR 8087 has been submitted to NE to revise the drawings. Awaiting FCR approval.

DCN 214 - AFW Tap Rotation

Work is complete, PMT required.

ECN 7328 - Installation of Backflush Line Spent Resin Storage Tank Drain Line

WP is in review cycle.

ECN 7349 - Removal of Temporary Bull Hose from CDWE to Floor Drain Collector Tank

Fabrication of new piping is in progress and is scheduled to be completed May 19, 1989.

DCN 341 - Modify Pressurizer Safety Valves to Accept Steam Trim and Install Loop Seal Drains - Unit 1

All work is complete except touchup paint and grouting of one baseplate (to be done during U1C4 outage).

Other Items (cont.):

DCN 550 - Modify Pipe Support - System 68, Unit 1

All work is complete except touchup paint (to be done during U1C4 outage).

DCN 703 - Modify RCP Seals - System 68, Unit 2

All work is complete on RCP Nos. 3 and 4.

DCN 704 - Modify Pipe Support - System 68, Unit 1

All work is complete except touchup paint (to be done during U1C4 outage).

DCN 943 - Addition of Stiffeners to Penetrations - System 61, Unit 2

All work is complete.

DCN 1045 - Modify Pressurizer Safety Valve Discharge Pipe Support - Unit 1

All work is complete except for touchup paint (to be done during U1C4 outage).

GLOSSARY

GLOSSARY OF VARIOUS ABBREVIATIONS

1. ABGTS - Auxiliary Building Gas Treatment System
2. ABSCE - Auxiliary Building Secondary Containment Enclosure
3. AB(I) - Auxiliary Building (Isolation)
4. AFW - Auxiliary Feedwater
5. AOI - Abnormal Operating Instruction
6. ASOS - Assistant Shift Operations Supervisor
7. AUO - Assistant Unit Operator
8. BAE - Boric Acid Evaporator
9. BAT - Boric Acid Storage Tank
10. BIT - Boron Injection Tank
11. CAQR - Condition Adverse to Quality Report
12. CAR - Corrective Action Report
13. CCP - Centrifugal Charging Pump
14. CCS - Component Cooling System
15. CCW - Component Cooling Water
16. CDWE - Condensate Demineralizer Waste Evaporator
17. CRI - Control Room Isolation
18. CREVS - Control Room Emergency Ventilation System
19. CSS(CS) - Containment Spray System
20. CVCS - Chemical Volume and Control System
21. CVI - Containment Ventilation Isolation
22. D/G(s) - Diesel Cenerator(s)
23. DCN - Design Change Notice
24. DCR - Design Change Request
25. DR - Discrepancy Report
26. ECCS - Emergency Core Cooling System
27. ECN - Engineering Change Notice
28. EGTS - Emergency Gas Treatment System
29. EM - Electrical Maintenance
30. EMI - Electromagnetic Interference
31. EQ - Environmentally Qualified/Environmental Qualification
32. ERCW - Essential Raw Cooling Water
33. E/ES - Emergency Instruction
34. ESF - Engineered Safety Feature
35. ESFA - Engineered Safety Feature Actuation
36. FCR - Field Change Request
37. FCV - Flow Control Valve
38. FDCT - Floor Drain Collector Tank
39. FDS - Flow Differential Switch
40. FIC - Flow Indicating Controllers
41. FSAR - Final Safety Analysis Report
42. FS - Flow Switch
43. FWI - Feedwater Isolation
44. GOI - General Operating Instruction
45. GPM - Gallons Per Minute
46. HDTP - Heater Drain Tank Pump
47. HO - Hold Order
48. IM - Instrument Mechanic/Instrument Maintenance
49. IMI - Instrument Maintenance Instruction
50. LCV - Level Control Valve

51. LER - Licensing Event Report
52. LCO - Limiting Condition for Operation
53. LOCA - Loss Of Coolant Accident
54. LS - Level Switch
55. M&TE - Measuring and Test Equipment
56. mA - Milliampere
57. MAST - Maximum Allowable Stroke Time
58. MCR - Main Control Room
59. MDAFWP - Motor-Driven Auxiliary Feedwater Pump
60. MFI - Main Feedwater Isolation
61. MWF - Main Feedwater
62. MFWRV - Main Feedwater Regulating Valves
63. MFP - Main Feedwater Pump
64. MI - Maintenance Instruction
65. MODS - Modifications
66. MOV - Motor Operated Valve
67. MSI - Main Steam Isolation
68. MSIV - Main Steam Isolation Valve
69. MSR - Moisture Separator Reheaters
70. NE - Nuclear Engineering (formerly Division of Nuclear Engineering)
71. NIS - Nuclear Instrumentation System
72. NMUDI - New Makeup Deionized System
73. NSS - Nuclear Security Service
74. NSSS - Nuclear Steam Supply Systems
75. O&PS - Office and Power Stores Building
76. PM - Preventive Maintenance
77. PMT - Postmodification Test
78. PORC - Plant Operations Review Committee
79. PORV - Power-Operated Relief Valve
80. PRO - Potential Reportable Occurrence
81. PDS - Pressure Differential Switch
82. QMDS - Qualification Maintenance Data Sheet
83. RCS/(P) - Reactor Coolant System/(Reactor Coolant Pump)
84. RHR - Residual Heat Removal
85. RM - Radiation Monitor (RAD Monitor/RAD MON)
86. RPI - Rod Position Indicator
87. RWST - Refueling Water Storage Tank
88. SCR - Significant Condition Report
89. S/D - Shutdown
90. SFP - Spent Fuel Pit
91. S/G(s) - Steam Generator(s)
92. SI - Surveillance Instruction/or Safety Injection
93. SMI - Special Maintenance Instruction
94. SOS - Shift Operations Supervisor
95. SOI - System Operating Instruction
96. SQN - Sequoyah Nuclear Plant
97. SR - Surveillance Requirement/Source Range
98. SSPS - Solid State Protection System
99. TACF - Temporary Alteration Control Form
100. TI - Technical Instruction

<u>GLOSSARY OF VARIOUS ABBREVIATIONS</u>

101. TS(s) - Technical Specification(s)
102. TVA - Tennessee Valley Authority
103. UHI - Upper Head Injection
104. UO/(S)RO - Unit Operator/(Senior) Reactor Operator
105. VLV - Valve
106. WP - Workplan
107. WR - Work Request

GLOSSARY OF VARIOUS SYSTEMS OF SEQUOYAH NUCLEAR PLANT

SYSTEM CODE	SYSTEM TITLE
1	Main Steam System (Turbine) (MSR)
2	Condensate System (FW Heaters)
3	Main and Auxiliary Feedwater System
5	Extraction Steam System
6	Heater Drains and Vents System
14	Condensate Demineralizer
15	Steam Generator Blowdown System
24	Raw Cooling Water System
27	Condenser Circulating Water System
30	Ventilating System
35	Generator Cooling Systems
36	Feedwater/Secondary Treatment System
37	Gland Seal Water System
46	Main/Auxiliary Feedwater Control System
47	Turbogenerator Control System
54	Injection Water System
58	Generator Bus Cooling System
61	Ice Condenser System
62	Chemical and Volume Control System
63	Safety Injection System
64	Ice Condenser Containment System
65	Emergency Gas Treatment System
67	Essential Raw Cooling Water System
68	Reactor Coolant System (Steam Generator)
70	Component Cooling System
74	Residual Heat Removal System
82	Standby Diesel Generator System
87	Upper Head Injection System
90	Radiation Monitoring System
268	Hydrogen Mitigation System

OPERATIONAL MODES

MODE	% RATED THERMAL POWER	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 5%	$\geq$ 350°F
2. STARTUP	$\leq$ 5%	$\geq$ 350°F
3. HOT STANDBY	0	> 350°F
4. HOT SHUTDOWN	0	350°F $> T_{avg}$ > 200°F
5. COLD SHUTDOWN	0	$\leq$ 200°F
6. REFUELING	0	< 140°F

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TENNESSEE VALLEY AUTHORITY

5N 157B Lookout Place

MAY 12 1989

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

SEQUOYAH NUCLEAR PLANT (SQN) - APRIL 1989 MONTHLY OPERATING REPORT

Enclosed is the April 1989 Monthly Operating Report as required by SQN Technical Specification 6.9.1.10.

If you have any questions concerning this matter, please call R. R. Thompson at (615) 843-7470.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

M J Ray for

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