TENNESSEE VALLEY AUTHORITY

6N 38A Lookout Place

August 9, 1989

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

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT UNIT 2 - DOCKET NO. 50-328 - FACILITY OPERATING LICENSE DPR-79 - LICENSEE EVENT REPORT (LER) 50-328/89008

The enclosed LER provides details concerning a reactor trip resulting from a dropped rod because of a spurious control signal fault. Also included in this report are details concerning a second reactor trip signal generated as a result of failure to comply with procedural requirements. This event is being reported in accordance with 10 CFR 50.73, paragraph a.2.iv.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

J. L. Bryant

for J. R. Bynum, Vice President
Nuclear Power Production

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IE22

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Sequoyah Nuclear Plant, Unit 2	0 5 0 0 0 3 2 8	1 OF 1 0

TITLE (4): Sequoyah Unit 2 reactor trip because of a suspe[illegible] dropped rod.

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 7	1 0	8 9	8 9	0 0 8	0 0	0 8	0 9	8 9		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

	20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER (Specify in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
	20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
J. W. Proffitt, Jr., Compliance Licensing Engineer	AREA CODE 6 1 5 8 4 3 - 7 4 6 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
B	I G	J X	W 1 2 0	NO					
B	I G	A M P	W 1 2 0	NO					

SUPPLEMENTAL REPORT EXPECTED (14)

		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On July 10, 1989, at 1134 EDT, with Unit 2 in Mode 1 (100 percent reactor power, approximately 2,235 psig, and Tavg at 578 degrees F), a reactor trip occurred. The trip occurred on high negative flux rate as noted on the first-out annunciator. Plant shutdown proceeded in an orderly manner consistent with procedures with no overcooling transient. A Posttrip Review Team was formed that conducted personnel interviews and developed recommendations for immediate corrective/investigative actions. These actions are detailed in the report and include troubleshooting on the rod control system as well as other deficiencies noted by the operators.

Personnel statements, strip chart recorders, troubleshooting work requests, previous trip reports, Westinghouse Owner's Group trip data base, and a trip modeling routine of the Watts Bar simulator were part of the resources utilized in the team's evaluation. In addition, plant management requested Westinghouse to assist in providing expertise in troubleshooting the rod control system. The team's conclusion is that the trip was a dropped rod event because of a spurious control signal fault. Neither the trip itself nor the team's recommended restart plan pose any compromise to the safe operation of the unit.

At 1857 with Unit 2 in Mode 3 (0 percent reactor power, 2235 psig, and 547 degrees F), a second reactor trip signal was generated by a source range spike approximately seven hours after the high flux trip. The source range channel had previously been declared inoperable because of noise on the channel but had not been bypassed in accordance with procedural requirements. The unit was stable and in Mode 3 with all rods on bottom when the signal occurred. A training letter was issued to all licensed personnel and shift technical advisors on July 21, 1989, addressing the failure to comply with Abnormal Operating Instruction 4, "Nuclear Instrumentation Malfunctions," when the N31 source range monitor was declared inoperable.

DESCRIPTION OF EVENT

On July 10, 1989, with Unit 2 in Mode 1 (100 percent reactor power, at approximately 2230 psig and Tavg at 578 degrees F), a reactor trip occurred at 1134 EDT. The trip resulted from high negative flux rate on the power range channels (EIIS Code IG) as indicated by the first-out reactor trip annunciator. After the trip, Operations personnel responded using Emergency Procedures E-0, "Reactor Trip or Safety Injection," and ES-0.1, "Reactor Trip Response," and General Operating Instruction (GOI) 3, "Plant Shutdown from Minimum Load to Cold Shutdown." Manual control of the auxiliary feedwater (AFW) was taken when Tavg was less than 547 degrees F. Under manual control, the turbine-driven AFW pump was ramped to minimum speed, and the motor-driven AFW valves were placed in manual bypass to control flow to approximately 110 gallons per minute (gal/min) to each steam generator. Loops 1 and 4 motor-driven level control valves (LCVs) performed as expected. However, the balance of plant (BOP) operator reported that loops 2 and 3 motor-driven LCVs did not respond as expected, and he took manual control to close the valves using the main control room (MCR) handswitch. The reactor coolant system (RCS) cooled down to approximately 543 degrees F following the trip. The plant was brought to a stabilized condition with no adverse effect on the plant or public safety. NRC was notified in accordance with 10 CFR 50.72, paragraph b.(2)(ii). Subsequent to the trip when the source range channels were placed into service, source range channel N31 was noted to have erratic readings indicating signal noise and declared inoperable at 1204 EDT. A work request (WR) was generated to investigate the source of the erratic readings. During the investigation of the source range noise problem, it was noted that intermediate range channel N36 had a degraded power supply. A WR was written to replace the power supply.

At approximately 1857 EDT, during the performance of Surveillance Instruction (SI) 603, "High Flux Adjustment After Shutdown (Source Range Drawer)," a spike occurred on source range channel N31, which resulted in a reactor trip signal. The unit was stable in Mode 3 with all rods on bottom when the signal occurred. The source range channel had been placed in bypass by instrument maintenance (IM) personnel so that a trip signal could not be generated during the performance of SI-603. Following a portion of the surveillance on source range monitor N31, the IMs placed the source range to the as-found position (normal alignment) as directed by the procedure. A few moments later, the spike occurred. NRC was notified of the event in accordance with 10 CFR 50.72, paragraph b(2)(ii).

As the unit was being stabilized following the trip at 1134 EDT, the Posttrip Review Team was put into place and an assessment begun. As detailed in the Sequence of Events portion of this report, two maintenance activities were in progress at the time of the trip that were reviewed for possible impacts on the event. At the time of the trip, the assistant shift operations supervisor (ASOS) stopped these activities until a complete review of their impact could be done.

IM was performing SI-90.72, "Reactor Trip Instrumentation Functional Test of Delta T/Tavg Channel IV, Rack 13, (T-68-67)," which is a functional test of delta T/Tavg bistable setpoints. Channel IV bistables had been placed in a tripped condition after verification that the other channels were not in a tripped state. Two bistables on

DESCRIPTION OF EVENT (Continued)

Channel IV had been tested, and the mechanics had stopped work while waiting for ambient calibration temperatures to return into specification. It was the assessment team's conclusion that the scope of components involved in SI-90.72, and the way it was being conducted, could not have resulted in the high flux rate trip.

Also at the time of the trip, Electrical Maintenance personnel were in containment to lubricate lower compartment cooler fan motors consistent with environmental qualification requirements (PM-1802). It was the team's conclusion that this maintenance activity had no impact on the unit trip.

During the posttrip interviews with personnel involved with the transient, several equipment anomalies were noted.

- o Loops 2 and 3 AFW LCVs went full open when placed in automatic posttrip with level greater than 33 percent--the valves should have remained closed.
- o Generator hydrogen cooler temperature control valve would not control in automatic and had to be placed in manual to control temperature.
- o Condensate booster pump B suction valve would not close when the pump was stopped.
- o Water hammer-type noise was heard by MCR operators and Turbine Building ASOS on secondary side.
- o Rod bottom light C-11 burned out bulb (replaced by unit operator).
- o Source range noise problem posttrip.
- o Intermediate range power supply fluctuation posttrip.

A review of recent activities associated with rod position indicators (RPIs) and rod control (EIIS Code AA) indicates that two RPI anomalies occurred on July 8, 1989, and are mentioned here for discussion. At 1630 EDT, RPI C-11 drifted low and brought in a rod deviation alarm on panel M-4 and the P250 plant process computer. The RPI immediately returned to normal indication, and no limiting condition for operation (LCO) entry was required. WR B790863 was initiated to investigate the perturbation. At 1830 EDT, the RPI for E-3 drifted low, and again, no LCO entry was required. WR B790864 was written to investigate. On June 26, 1989, WR B265269 was written when rod control handswitch 2HS-85-5110 would not step out control bank D in the manual mode. A replacement card was installed that day, restoring the control system to operability. These events are not considered to be contributors to the reactor trip experienced on July 10, 1989.

In order to evaluate nuclear recorder (NR)45 delta flux trace from the reactor trip, it was decided to use the simulator to recreate rod drop events. The Watts Bar Nuclear Plant simulator (WBS) was chosen because of its superiority in neutron kinetic modeling. Additionally, the Watts Bar computer software was readily adaptable to enable simulation of any control/shutdown rod drop. This allowed simulation of dropped rods in core location E-3 and C-11 in shutdown bank C, which had previously shown erratic RPI readings at Sequoyah.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

DESCRIPTION OF EVENT (Continued)

Since the WBS has Watts Bar Unit 1 Cycle 1 data for fuel loading, enrichments, and rod worths, a review was performed of the Sequoyah Unit 2 Cycle 4 nuclear parameters and operation package and the Watts Bar Reactor Core Model neutronics input to identify a point in core life when the shutdown bank rod worths were approximately equal. As a result, middle-of-life (MOL) conditions were chosen. The WBS was initialized using initial condition No. 14 (100 percent power, equilibrium Xenon, MOL) with essentially all rods out.

Delta flux from Nuclear Instrumentation System (NIS) channels N42 and N43 were selected on NR45, and four channels of delta flux and various rod positions were trended using the simulator computer. Five different cases were run:

- o Base Case - Manual Reactor Trip
- o One Dropped Rod - Core Location C-11
- o One Dropped Rod - Core Location E-3
- o Two Dropped Rods - Core Location C-11 and E-3
- o Four Dropped Rods - All of Shutdown Bank C

The results show that:

- o The NR45 trace from a manual reactor trip (all rods released simultaneously) is distinctively different from a negative rate trip caused by a dropped rod(s) event.

- o A single dropped rod will generate a negative rate trip.

- o The Sequoyah NR45 trace of delta flux has the same pattern qualitatively as the WBS traces from the rod(s) drop event.

The Sequoyah unit trip data base for Units 1 and 2 were reviewed to identify similar events. Two previous rod drop events were identified and are discussed individually.

1. Unit 1 Reactor Trip No. 23 - November 22, 1980

   This was a planned trip as a part of the Unit 1 initial startup test program. Startup test 9.5 (SU-9.5) was performed in order to verify that a reactor trip would occur as the result of dropping two rod control cluster assemblies (RCCAs). Rods P-4 and D-2 were chosen based on their proximity to excore detectors. The test was performed at a power level of 50 percent rated thermal power.

   The results of the test showed that a reactor trip would occur. The sequence of events for this trip is very similar to the sequence of events for the current reactor trip under investigation. This supports the conclusion that the unit 2 trip on July 10, 1989 was the result of dropped rod(s).

DESCRIPTION OF EVENT (Continued)

2. Unit 1 Reactor Trip No. 39 - March 17, 1981

   This was an unplanned trip as the result of RCCA N-5 inadvertently dropping into the core at a power level of 80 percent. Rod N-5 was identified as the dropped rod during subsequent investigation (i.e., N-5 would not move on demand).

   The cause of the dropped rod was determined to be a connector problem on the control rod drive stack. The connector was replaced, and as a result of this problem and related problems associated with the connectors on top of the coil stacks, all the connectors on the coil stacks have subsequently been removed and connections hardwired.

   The sequence of events for this trip is similar to the sequence of events for the current reactor trip under investigation. This supports the conclusion that the Unit 2 trip on July 10, 1989, was the result of dropped rod(s).

The initial troubleshooting of the rod control system on WR B757364 consisted of measuring resistance of various components. The first set of readings was taken without removing the fuses to isolate the individual components. The second measurements were taken with the fuses removed to ensure isolation. The resistance readings were taken on all movable and stationary gripper and lift coils, coil protection fuses, stationary gripper blocking diodes, and the stationary gripper sampling resistors with acceptable results. Further troubleshooting and testing will continue on WR B757364 with the rod control system energized.

SI-11, "Reactivity Control Systems Moveable Control Assemblies," is a functional test of the rod control system that exercises both shutdown and control rod banks (10 steps up and down) to verify operability of the "group demand rod position indicators." SI-11 was run on July 10, 1989, and had no adverse indications. It was run again on July 11, 1989, current traces were taken on all 53 rods, and an anomaly was seen on rod C-9. Stationary gripper current readings were taken across a test resistor with satisfactory results. A modified version of SI-11 was run on July 12, 1989, which moved the rods 228 steps and revealed the same anomaly on C-9 rod on shutdown bank B. Stationary gripper current readings were again taken with like results.

As part of the troubleshooting effort, Westinghouse Electric Corporation has assisted in providing technical support onsite and in Pittsburgh.

A listing of reactor trips initiated from negative rate flux signals was obtained from the Westinghouse Owner's Group (WOG) and reviewed in an attempt to identify problem areas experienced by other plants. A total of 49 negative rate reactor trips were identified in the WOG data base. The most common cause of dropped RCCAs was lightning strikes on plant structures followed by fuse problems (blown, defective, or degraded) and circuit card or module failures. This data was factored into the troubleshooting action plans.

DESCRIPTION OF EVENT (Continued)

An independent review of the reactor trip was conducted on July 11, 1989, by Sequoyah training personnel at the request of plant management. Their review, based primarily on the data collected from the posttrip interviews, strip charts, and the sequence of events record, concluded that the trip was most likely caused by the dropping of one or more RCCAs.

Plant management requested that Watts Bar instrument engineers evaluate the delta T/Tavg testing that was in progress at the time of the trip. Their review independently concluded that the work being performed in Rack 13 of Protection Set IV did not cause or contribute to the reactor trip. Included is a logic figure illustrating the fact that there is no tie between delta T/Tavg signal and a high flux rate trip.

Subsequent to the trip, when the source range channel N31 was noted to have erratic readings from noise, a work order was generated to investigate the source of the erratic readings, and N31 was declared inoperable at 1204 EDT.

At approximately 1857 EDT, during the performance of SI-603, a spike occurred on source range channel N31 with all rods on bottom when the signal occurred. The second reactor trip signal was generated from this spike on N31 source range channel. The channel had not been bypassed after being declared inoperable. The second reactor trip signal would not have been generated if the instructions of Abnormal Operating Instruction (AOI) 4, "Nuclear Instrumentation Malfunctions," had been complied with. The source range channel had been placed in bypass by IM personnel (so that a trip signal could not be generated) during the performance of SI-603. Following a portion of the surveillance on source range monitor N31, the IMs placed the source range to the as-found position (normal alignment) as directed by the procedure.

Troubleshooting of the noise problem on source range channel N31 by WR B882987 was started on the evening shift on July 10, 1989. A temporary capacitor was installed between the preamp case and ground, resulting in no noticeable change in the noise level. The capacitor was removed and replaced with a wire jumper, again without results. The source range was deenergized, and the signal and high-voltage cables disconnected then shorted to remove any static charge. Resistance measurements were taken between the outer shield, the inner shield, and the center conductor-to-ground on all the cables. The outer shield-to-ground on the detector cable measured 1.6 ohms--all others had infinite resistance. Noise was observed on N31 with the detector cable disconnected at the preamp, which indicated a bad preamp. A new preamp was installed along with a capacitor connected between the preamp case and ground (temporary alteration control form No. 2-89-042-092), and it appears the noise problem has been corrected. SI-93.2, "Reactor Trip Instrumentation Functional Tests - SR and IR Nuclear Instrumentation," was successfully performed, and the source range monitors returned to service.

CAUSE OF EVENT

The cause of the first reactor trip has been postulated to be a dropped rod. The cause of the second reactor trip signal was a spurious signal from an inoperable source range monitor because of a failure to follow procedure.

ANALYSIS OF EVENT

Based on the following discussions of plant response during and after the trip, plant systems and parameters responded consistent with responses described in the Final Safety Analysis Report (FSAR). Subsequent evaluation of trip data indicates that a dropped RCCA or RCCAs resulted in the high negative rate reactor trip. This is supported by comparison of trip data with data from previous reactor trips and simulated reactor trips caused by dropped RCCAs.

Investigation into the cause of the dropped RCCA, as described in this investigation, did not identify any hardware failures, associated system problems, or anomalies that would have resulted in dropping RCCAs. All control rod banks have been successfully exercised, with all RCCAs indicating movement on demand. Therefore, it is concluded that this trip is a result of a spurious electrical fault.

Furthermore, none of the plant data indicates evidence of physical separation of RCCAs from jackshafts. This is supported by the lead operator's observation that two rod bottom indicating lights were illuminated prior to the tripped RCCAs reaching full insertion. The two rod bottom lights, which could be indicative of the dropped rods that caused the reactor trip, provide support that the RCCAs remained physically attached to the jackshafts. This is based on the fact that the signal for the RPIs and rod bottom lights is derived from the position of the control rod jackshafts in the coil stack. Since the rod bottom lights illuminated on the suspected dropped RCCAs, evidence exists that the RCCAs remain physically coupled to the jackshafts.

Occurrence of dropped RCCAs is a FSAR Condition II event. Reactor protection systems are designed to trip the reactor in the event of a group of RCCAs inadvertently dropping into the core.

A single dropped RCCA may or may not result in a reactor trip, depending upon the reactivity worth of the RCCA. In order to bound a situation where a single dropped RCCA does not result in a reactor trip, the FSAR Chapter 15 analysis assumes a dropped RCCA occurring at 100 percent power without tripping the reactor. The FSAR analysis shows that, in all cases of dropped single assemblies, the departure from nucleate boiling remains greater than 1.30 at power, and consequently, dropped single assemblies do not cause core damage.

RCS Temperature

Pretrip Tavg was at or about 578 degrees F. Posttrip Tavg declined to 547 degrees F following the reactor trip. The operator worked within the guidelines of ES-0.1. When Tavg continued to decline below 547 degrees F, the operator took manual control of the AFW system and emergency borated in accordance with ES-0.1. The decline in Tavg was stopped at approximately 543 degrees F and was stabilized at approximately 547 degrees F. RCS temperature response as a result of this trip was within the bounds of the accident analysis.

ANALYSIS OF EVENT (Continued)

Heatup/Cooldown Limits

Technical Specifications require a cooldown limit of 100 degrees F in any 1-hour time period. These limits were not exceeded.

Pressurizer Level

Pressurizer level was varying around 59 percent pretrip. Response of the pressurizer level to the transient closely paralleled that of RCS pressure and temperature. The level dropped because of cooling of the RCS. At approximately 10 minutes into this event, the lowest level (21 percent) was attained. The level increased and stabilized in accordance with the program. Pressurizer level response was within the bounds of the accident analysis.

Feedwater Flow

Feed flow was steady for 100 percent power. All four main regulator valves and main feed pumps A and B were in automatic before the trip. AFW started as designed following the trip, and flow to the steam generator from AFW continued at greater than 440 gal/min per steam generator as expected while steam generator levels remained below 33 percent. Manual control of AFW was taken by the operators in accordance with ES-0.1. No manual boration was required. Technical specification and FSAR requirements and analysis were not challenged.

Steam Pressure

Pretrip steam generator pressure varied from 850 to 857. Posttrip steam generator pressure increased to approximately 990 because of the turbine trip. Steam pressure returned to no-load pressure, and Tavg returned to 547 degrees F. Technical specification and FSAR requirements and analysis were not challenged.

Containment Pressure/Temperature/Radiation

No perturbations were observed in containment pressure, temperature, or radiation. Technical specification and FSAR requirements and analysis were not challenged.

Forced/Manual Circulation

All reactor coolant pumps continued to maintain flow post-reactor trip; therefore, forced flow was not lost, and no FSAR assumptions were challenged.

ANALYSIS OF EVENT (Continued)

Reactor Power

The reactor was operating at approximately 100 percent rated thermal power. Upon receipt of the trip signal, the shutdown and control banks dropped into the core, and reactor power rapidly decreased as expected.

Prior to the trip, the NIS power (in percent) and core delta flux were being recorded on NR45. These trends showed that the unit was at approximately 100 percent power and a delta flux of approximately -0.5 percent. At the time of the event, the delta flux trace made a negative spike, returned to approximately -0.5 percent, then spiked a second time in the negative direction. As the delta flux was returning to mid-scale, the selected indication for NR45 was changed to a source range and an intermediate range detector. Realigning the indicated parameters on NR45 is consistent with instructions of ES-0.1.

The delta flux spikes in the negative direction are consistent with expected NIS response to a dropped rod. The first negative spike is a result of the dropped rod(s). As the rods pass the midplane of the core, the delta flux will swing back to approximately the initial value. The second negative spike is caused by the rods falling after the reactor trip. The indications on the recorder were then realigned in accordance with ES-0.1. No Technical specification or FSAR limit was challenged.

Shutdown Margin

Pretrip, the reactor was operating above the minimum insertion limits, and by definition, adequate shutdown margin was available.

Following the trip, expected cooldown occurred as had been previously discussed. Adequate shutdown margin was maintained in accordance with ES-0.1 and SI-38, "Shutdown Margin." Technical specifications and the accident analysis were not violated.

RCS Pressure

Prior to the event, RCS pressure varied at or near 2,230 psig. When the reactor trip occurred, the pressurizer pressure dropped to approximately 2,010 psig within one minute. Pressure recovered to 2,245 psig within the following 20 minutes and stabilized at or near 2,230 psig within approximately one hour of the trip. The decrease in pressure can be attributed to the cooldown. As discussed in the FSAR, Section 15.2.8, the AFW system was capable of removing enough residual heat to prevent overpressurization of the RCS. Therefore, the accident analysis was not challenged.

CORRECTIVE ACTIONS

The following corrective actions were taken to ensure the plant was ready for restart and 100 percent power operation.

1. An investigation was performed to verify the correct operation of the rod control system. An action plan was established to systematically troubleshoot the system and determined the system was operational.

2. A condition adverse to quality report was generated to document this event.

3. The AFW LCV on loop 2 (2-LCV-3-148) and loop 3 (2-LCV-3-156) were repaired and returned to service.

4. A walkdown of Unit 2 feedwater/condensate systems was performed to verify no damage occurred as a result of the transient. No damage was identified.

5. The condensate booster pump B suction valve (2-FCV-2-87) was repaired and returned to service.

6. The NIS source range N31 channel noise problem was corrected, and the source range N31 channel was returned to service.

7. The generator hydrogen temperature control valve's (2-TIC-24-48) failure to control was repaired, and it was returned to service.

8. Intermediate range N36 channel power supply was repaired and returned to service.

9. Rod position indicators C11 and E3 on shutdown bank C were repaired and returned to service.

10. A training letter was issued to all licensed personnel and shift technical advisors on July 21, 1989, addressing the failure to comply with AOI-4 when the N31 source range monitor was declared inoperable.

11. A real-time computer has been connected to selected rod cluster control assembly stationary gripper coils. This would provide clear information should Unit 2 experience another rod drop incident. This computer should remain in service until the Unit 2 ice outage tentatively scheduled to start March 5, 1990.

12. The plant will monitor selected rod bottom lights using a video camera or equivalent instrumentation. This will provide clear information should Unit 2 experience another rod drop incident. It is planned for this equipment to remain in service until the Unit 2 ice outage.

Additional Information

There has been one other unplanned reactor trip occurring from a dropped rod. Reactor trip No. 39 occurred on March 17, 1981, and resulted from an electrical connection failure.

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