TVA

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Joseph R. Bynum
Vice President, Nuclear Operations

April 10, 1991

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) 50-328/91003

The enclosed LER provides details concerning the discovery of the breaker for the Unit 2 No. 3 cold leg accumulator isolation valve being in the locked closed (power on) position. Technical Specification (TS) Surveillance Requirement 4.5.1.1.1.c requires that the valve be open with power removed. This event is being reported in accordance with 10 CFR 50.73(a)(2)(i)(B) as an operation prohibited by TSs and 10 CFR 50.73(a)(2)(ii)(B) as a condition that was outside the design basis of the plant.

Mr. Bill Little, of your NRC staff, was notified on April 2 and again on April 5, 1991, that issuance of this LER was delayed, and that the LER would be issued by April 12, 1991.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

J. R. Bynum

Enclosure
cc: See page 2

cc (Enclosure):

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NRC Form 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

Approved OMB No. 3150-0104
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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 1

TITLE (4) Power not removed from cold leg accumulator isolation valve as a result of inappropriate personnel actions.

EVENT DAY (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 3	0 1	9 1	9 1	0 0 3	0 0	0 4	1 0	9 1		

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)	__ 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)	XX 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	XX 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
Gregory S. Kniedler, 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

SUPPLEMENTAL REPORT EXPECTED (14)

__ YES (If yes, complete EXPECTED SUBMISSION DATE) | X | NO

EXPECTED SUBMISSION DATE (15) MONTH DAY YEAR

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

On March 1, 1991, at 0127 Eastern standard time (EST) with Unit 2 in Mode 1, it was discovered that the breaker for the operator for the No. 3 cold leg accumulator (CLA) isolation valve 2-FCV-63-80 was locked in the closed position. This was discovered during the performance of Surveillance Instruction (SI) 0-SI-OPS-063-013.0, "Cold Leg Accumulator Valves Power Removal Verification." The last documented manipulation of this breaker was on February 14, 1991, when an evolution was being performed in attempt to stop inleakage of reactor coolant into the CLA. This evolution was initiated at 2019 EST on February 14, 1991, and completed at 2032 EST with the components thought returned to their required conditions/positions. No independent verification of the breaker's restoration was performed. The cause of the event is attributed to inappropriate personnel actions. Immediate corrective action was to restore the breaker to its correct position. Additional corrective actions include discussions with Operations personnel to clarify requirements, disciplinary action, procedure clarifications, and further training.

DESCRIPTION OF EVENT

On March 1, 1991, at 0127 Eastern standard time (EST), with Unit 2 operating in Mode 1, (100 percent power, 2,235 pounds per square inch gauge [psig], and 578 degrees Fahrenheit [F]), it was discovered that the breaker for the operator of the Unit 2, No. 3 cold leg accumulator (CLA) (EIIS code BQ) isolation valve, 2-FCV-63-80, was locked in the closed position. This was discovered during the performance of Surveillance Instruction (SI) 0-SI-OPS-063-013.0 "Cold Leg Accumulator Valves Power Removal Verification," which is required by Technical Specification (TS) Surveillance Requirement (SR) 4.5.1.1.1.c. SR 4.5.1.1.1.c requires that: "At least once every 31 days when the RCS pressure is above 2000 psig by verifying that power to the isolation valve operator is disconnected by removal of the breaker from the circuit." The TS basis for the requirements of TS 3.5.1.1 is "The accumulator power operated isolation valves are considered to be 'operating bypasses' in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissible conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required." The shift operations supervisor (SOS) was immediately notified of this condition and TS Limiting Condition for Operation (LCO) 3.5.1.1 was entered at 0127 EST. The breaker was unlocked, opened, and locked in the open position (i.e., power removed) at 0131 EST and LCO 3.5.1.1 was exited.

On February 1, 1991, 0-SI-OPS-063-013.0 was performed and the breaker for the operator to 2-FCV-63-80 was verified in the locked open position as required by TS SR 4.5.1.1.1.c. On February 14, 1991, an evolution was initiated in an attempt to reduce the 0.21 gallons per minute (gpm) inleakage of the reactor coolant system (RCS) into the Unit 2, No. 3 CLA. The performance of this evolution was done in accordance with Administrative Instruction (AI) 30, "Nuclear Plant Conduct of Operation," which states: "Limited evolutions of short duration may be performed by an operator without a procedure provided that positive configuration control is maintained in accordance with AI-58, a procedure does not exist for the activity and the operation is not complex. The SOS and unit assistant shift operations supervisor (ASOS) will determine if any operation will be allowed without a procedure based upon the complexity, duration of the operation, TS requirements and Final Safety Analysis Report Description/Bases/Assumptions. Any evolution performed without a procedure shall be documented in the operator journal."

The evolution to stop the inleakage of RCS into the No. 3 CLA was to consist of unlocking and closing the breaker for valve 2-FCV-63-80, the repositioning of four valves (2-FCV-63-80, -78, -71, and -84), and the operation of the 2A Safety Injection System (SIS) pump. The Unit 2 ASOS was to remain at the breaker throughout the evolution to maintain positive control and the Unit 2 lead main control room (MCR) unit operator (UO) would maintain positive control over the valve manipulations in the MCR.

DESCRIPTION OF EVENT

This lineup was intended to vent pressure from the upstream side of check valve 63-624 to the holdup tank and then apply a large differential pressure to backseat the check valve by starting the SIS pump. Refer to Updated Final Safety Analysis Report (UFSAR) Figure 6.3.2-1.

Before performing this evolution, the activity was discussed with the Operations Superintendent, the off-going SOS, the onshift SOS and the onshift Unit 2 ASOS. The activity was determined to constitute a limited evolution based on the small number of manipulations, anticipated short duration, and positive controls that would be implemented. The initial review for the evolution by the onshift SOS and onshift Unit 2 ASOS consisted of a review of flow prints, technical specifications, and procedures. Electrical prints were not reviewed. The evolution was initiated at 2019 EST on February 14, 1991, with entry into LCO 3.5.1.1. These activities were logged into the operator's log as required by AI-30 for limited evolutions. The Unit 2 ASOS unlocked and closed the breaker for the 2-FCV-63-80 operator. The Unit 2 lead UO then shut 2-FCV-63-80 from the MCR. The isolation valve automatically opened. This was reported to the ASOS at the breaker. The ASOS opened and then closed the control power breaker to the 2-FCV-63-80 operator to clear any possible lock-in safety injection signals. The ASOS then requested the UO to close 2-FCV-63-80 and it again reopened automatically. The ASOS locked the breaker open, then proceeded to the MCR to review drawings to determine what was preventing the valve from remaining closed. The P-11 interlock on 2-FCV-63-80 was identified by review of electrical prints. This interlock causes the valve to open automatically when the RCS pressure is greater than or equal to 1,970 pounds per square inch gauge (psig). Discussion with the SOS and Unit 1 ASOS took place to determine a course of action to address the P-11 interlock relative to 2-FCV-63-80. The Unit 2 ASOS returned to the location of the breaker for valve 2-FCV-63-80 and unlocked and closed the breaker. Valve 2-FCV-63-80 was then closed by the UO, the ASOS opened the breaker as soon as the valve indicated closed, and the valve remained closed. The Operations Superintendent was not recontacted or consulted when the interlock was encountered.

The evolution continued with the opening of valves 2-FCV-63-78, -71, and -84. Once the pressure in the line was relieved, valves 2-FCV-63-78 and -71 were closed, and the SIS pump 2A was started. After this evolution, valve 2-FCV-63-80 was opened, the SIS pump 2A was stopped, and valve 2-FCV-63-84 was closed.

The Unit 2 ASOS who performed the manipulations of the valve's motor-operator supply breaker, remained in the vicinity of the breaker when it was not in the locked open position. The evolution was completed with valves returned to their normal positions at 2032 EST, and LCO 3.5.1.1 was exited. Following the evolution the ASOS returned to the MCR. The SOS asked the ASOS if power had been removed from the operator to valve 2-FCV-63-80. The ASOS reported that power had been removed. This was logged in the SOS and ASOS logs. However, no independent verification was performed following system restoration as required by AI-37, "Independent Verification." AI-37, Section 6.1.2,

DESCRIPTION OF EVENT

requires that breakers in the Emergency Core Cooling System (ECCS) shall be independently verified to be in the correct position and/or condition when the system or component is being returned to service or restored to a stand-by line up. The stated objective of AI-37 is to minimize the possibility of human error in the performance of designated activities by verifying that the activity conforms to specified requirements.

The normal method of documenting independent verification is to use clearance sheets, system operating instruction power availability and valve checklists or AI-58, "Maintaining Cognizance of Operation Status - Configuration Status Control," Appendix B, "Configuration File Sheets." An allowable exception to configuration control requirements, according to AI-58, was followed. AI-58, Section 2.2.2.1, allows exceptions to requirements for configuration log entries if "equipment involved is continuously monitored by operator at local site until it is returned to NORMAL status." These requirements were met and accordingly no configuration log entries were made. As a result of the allowed exception of AI-58, Operations personnel assumed that independent verification was not absolutely required, i.e., since the evolution was not documented by the above noted normal methods; therefore, an independent verification was not performed.

Following discovery of power on the valve on March 1, 1991, an investigation was conducted to evaluate potential causes of the mispositioning of the 2-FCV-63-80 motor operator breaker. Operator logs were reviewed to determine if any manipulations of the breaker occurred between February 14 and March 1, 1991. No evidence was identified of any operations of the breaker or valve other than that on February 14, 1991.

In response to a concern that a manipulation of the breaker for the operator to 2-FCV-63-80 could have occurred but not been logged, an extensive interview process was performed. SOSs, ASOSs, lead UOs, and balance of plant UOs who were assigned to operate Unit 2 between February 14 and March 1, 1991, were interviewed. No evidence of any operations other than that on February 14, 1991, was identified during the interview process.

The possibility of an unintentional operation of the breaker to 2-FCV-63-80, because of a confusion between Units 1 and 2, was evaluated. This possibility was considered because of the Unit 1 forced outage that occurred between February 18 and 26, 1991. No evidence was identified of an unintentional operation.

The Operations incident investigation team members performed a preliminary assessment and determined that the event was not reportable under 10 CFR 50.72. This determination was based upon the initial assumption that the event did not constitute a departure from the plant's design basis. As the investigation proceeded, a draft analysis of the event's safety implications was prepared by the incident investigation team. This draft analysis was sent to Nuclear Engineering (NE) for independent review.

DESCRIPTION OF EVENT

On March 22, 1991, at 1646 EST after further investigation, it was determined that Unit 2 had operated in a condition outside the design basis during the time frame of February 14, 1991, to March 1, 1991, as a result of the breaker to 2-FCV-63-80 being in the locked closed position. The basis is that a single failure in the operator breaker or control circuit could cause the valve to inadvertently close, isolating the No. 3 CLA. Should this single failure occur following a Loss of Coolant Accident (LOCA), because of a rupture in RCS cold leg loops 1, 2, or 4, only two CLAs would be available for injection. The parameters used in the LOCA analysis of Section 15.3 and 15.4 of the UFSAR, requires three CLAs to be available for injection. This determination was based on consultation with Westinghouse Electric Corporation LOCA specialists. A one-hour notification phone call to the NRC was made in accordance with 10 CFR 50.72.(b)(1)(ii)(B) as a result of this determination at 1737 EST on March 22, 1991.

Upon further review, it was determined that the evolution performed would not achieve its intended function to further backseat check valve 63-624 by starting of the SIS pump. The leakrate for the primary check valve 63-562 (0.68 gpm) was evaluated to be greater than the leakrate for the secondary check valve 63-624 (0.21 gpm), using results from the leak rate testing performed before restart from the Cycle 4 refueling outage. The measured leakrate from testing at reduced pressure is extrapolated to a leakrate at full pressure conditions. The leakage from the RCS into the CLA through this header and not by other means had been previously determined as a result of extensive troubleshooting activities and by confirmation that water samples from both the CLA and RCS were very similar. The pipe header pressure between check valves 63-562, -624, -634 (RHR pump, loop 3 cold leg injection secondary check valve), and -555 (SIS pump, Loop 3 cold leg injection secondary check valve) is therefore considered to have very likely been at approximately 2,235 psig, normal RCS pressure at 100 percent reactor power. Therefore, when the upstream side of check valve 63-624 was vented to the holdup tank the differential pressure across 63-624 was approximately 2,235 psig. Starting of the 2A SIS pump pressurized the line between the pump and check valve 63-555 to approximately 1,500 psig. As indicated, the pressure in the line downstream of check valve 63-555 was approximately 2235 psig. Since the piping between check valves 63-561, -624, -634, and -555 was already at a pressure greater than the SIS discharge pressure, starting the SIS pump would not have applied any further pressure differential across 63-624. It was not recognized at the time the evolution was planned and performed that the downstream side of valve 63-555 was at a higher pressure than what the SIS pump could achieve.

Review of this evolution also considered whether starting of the SIP could have actuated the other three SIS secondary check valves 63-551, -553, and -557 for loops 1, 2, and 4, respectively, necessitating leakrate testing in accordance with SR 4.4.6.2.2.d. As designed, the minimum pressure downstream of these check valves, assuming no primary check valve leakage, would be approximately 600 psig, the pressure of the CLAs. When the SIP was started with a 1,500 psig discharge pressure, a 900 psid could have been developed across the three SIS secondary check valves in the direction of the RCS.

DESCRIPTION OF EVENT

However, since water is incompressible and the piping between the primary and secondary check valves is water solid, the volume of water moved across the SIS check valves would be extremely small. If it is conservatively assumed that the CLA check valves (-622, -623, and -625) were leaking back to the CLAs (although there has been no indication of level or concentration changes), the most water that could have been moved across the SIS check valves would be approximately the leakrate of the CLA check valves measured during tests conducted at startup from the last refueling outage (on the order of 0.26 to 0.32 gpm). From this review it is concluded that the evolution did not result in actuation or "flow through the valve" as intended by SR 4.4.6.2.2.d and therefore testing to reverify check valve leakrate was not required.

As a result of further review of this evolution and the associated TS and design and licensing basis, it is concluded that the CLA isolation valve 2-FCV-63-80 should not have been closed. TS LCO 3.5.1.1, Action Statement "b" states: "With one cold leg injection accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 12 hours." The corresponding bases states: "If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required." The review of this TS action statement and bases by the involved Operations personnel, concluded that the isolation valve could be immediately opened and the action could be met by stationing individuals at the valve control and breaker to immediately reopen the valve in event of an accident, and by ensuring that the evolution was completed well within an hour. It was further reasoned that this evolution was less "severe" than the periodic accumulator drain and refill evolutions that were being necessitated by the check valve back-leakage. However, following further review of the technical specification action statement as written and the accident analysis, as further detailed in the analysis section, it is concluded that intentional closure of the isolation valves should not occur in modes 1, 2, 3, and with pressurizer pressure above 1000 psig.

CAUSE OF EVENT

The direct cause of this event is attributed to inappropriate personnel actions in placing the breaker in the locked closed rather than the locked open position. The cause of that incorrect action could not be determined. Discussion with the ASOS indicated his belief that the breaker was locked open. A contributor to the event is lack of independent verification. Independent verification of manipulations of ECCS components is required by AI-37, however personnel believed that independent verification was not required given the process and procedures that were being used for this evolution.

AI-37 requires independent verification for the temporary alterations of removing and returning ECCS systems from and to service. AI-30 provides information relative to "system configuration control of CSSC safety related systems" and controls for implementing limited evolutions without formal procedures. AI-30 also refers to AI-58

CAUSE OF EVENT

for a detailed description of maintaining the alignment of these systems in accordance with their appropria.e valve and power availability checklists. AI-58 lists exceptions to configuration log entries for specific activities including limited evolutions. Implementing the subject evolution in this manner eliminated the normal method of documenting independent verification and led the Operations personnel to believe that independent verification was not absolutely required. Additionally, the SOS had a high level of confidence in the performance of the involved ASOS.

The root cause of this event, however, is considered to be the judgement made that this activity constituted a limited evolution not requiring a procedure. While AI-30 provides flexibility for licensed personnel evaluation of the condition and therefore did not specifically prohibit this judgement, TVA considers that this activity should clearly have been recognized as being outside the scope of the limited evolution process. Further, when the P-11 interlock was encountered during the evolution this should have further indicated to the personnel involved that the activity was not a limited evolution and that a procedure was required. Had a procedure been prepared it is believed that the technical issues would have been appropriately identified and addressed. Additionally any evolution involving manipulation of ECCS components would have required written independent verification of return to normal. A contributing factor to the incorrect judgement is considered to be an inadequate preevolution review. The review performed consisted of review of the flow diagrams to assess the flow paths, the TSs, and peer review among several SROs. However the review did not include review of electrical, control or logic prints nor did it adequately assess TS and FSAR impact/significance. As a result of discussions concerning this evolution with Operations management and operating personnel it is concluded that inadequate training has been provided to ensure appropriate and consistent implementation of limited evolutions.

The error in the initial reportability determination is considered to have resulted from lack of engineering involvement in the assessment relative to design basis.

ANALYSIS OF EVENT

This event is being reported in accordance with 10 CFR 50.73, paragraph a.2.i, as an operation prohibited by TS 3.5.1.1 and 10 CFR 50.73, paragraph a.2.ii, as a condition that was outside the design basis of the plant.

With the breaker for the isolation valve locked closed (i.e., power to the valve), instead of locked open, a potential exists that a spurious active single failure in the control circuit could cause the valve to inadvertently close, isolating the No. 3 CLA. Locking open the operator breaker (i.e., power removed) prevents a spurious active single failure. Should this single failure occur following a large break LOCA because of a rupture in RCS cold leg Loops 1, 2, or 4; only two CLAs would be available for injection. The parameters in the LOCA analysis described in UFSAR, Section 15.4, requires three CLAs to be available for injection; assuming one of the four CLAs is lost to the sump through the break in the cold leg.

ANALYSIS OF EVENT

An example of spurious active single failure is the unintended energizing of a power-operated valve to open or close. Spurious failures may occur independently of the component's environmental surroundings. Spurious operation is the change in equipment state because of electrically induced faults. Thus with power removed, no active failure of the valve may be postulated to occur. Upon swapover to recirculation from the sump, a passive failure may be assumed; however, the borated water in the CLAs would have already been discharged to the RCS. Therefore removing power assures three CLAs will be capable of injecting into the RCS, in the event of a RCS cold leg LOCA, and thus the design basis will be satisfied.

With the breaker to the operator of 2-FCV-63-80 locked closed, the postaccident peak clad temperature (PCT) could exceed the design limit, provided a large break LOCA occurs in the RCS cold leg pressure boundary piping loops 1, 2, or 4, and a single spurious failure is applied to 2-FCV-63-80 (valve fails to remain open), thus resulting in the elimination of one of the three remaining available CLA's. An additional train of ECCS (assumed to exist since the single failure occurred in the spurious actuation of 2-FCV-63-80) would be available, but would not supply sufficient flow to substitute for the loss of a CLA. This is because of the inability of the ECCS pumps to deliver the required volume of water (equal to or greater than an accumulator discharge) in the short time interval necessary. Because of the net loss in delivered flow, the time to resubmerge the bottom of the fuel after initial core uncovery, would be extended by more than 12 seconds and PCT could exceed the design limit.

SQN's Individual Plant Evaluation for COMPONENT FAILURE RATES generically documents failure rates for various type components in the plant.. The failure rate for a motor operated valve (failure to remain in its normal position open or closed) is 1E-7/hour. An additional analysis showed that the conditional probability of a large break LOCA and one CLA motor operated isolation valve closing is negligible over a period of 14 days. The conditional probability of these two events, both occurring within a 14-day period, is 2.62E-10.

The limiting break size in terms of highest PCT for a small break LOCA is a 3-inch diameter break. The depressurization transient for this break is shown in UFSAR Figure 15.3.1-2. The extent to which the core is uncovered is shown in UFSAR Figure 15.3.1-3. For a small break LOCA and a failure of valve 2-FCV-63-80 to remain open, the PCT would not exceed 2,200 degrees F.

Beyond purely spurious failures, an evaluation was also conducted to determine whether closure of the isolation valve (which is not specifically environmentally qualified) could be expected to result from environmentally induced accident conditions. The results of this evaluation concluded that it is not expected that a harsh environment would cause spurious actuation and closure of 2-FCV-63-80 during the time period under which closure could adversely affect calculated PCT.

ANALYSIS OF EVENT

In review of the actual conditions over the subject period, it is noted that valve 2-FCV-63-80 remained open following this evolution during the time period the breaker was in the locked closed (power on) position (February 14 to March 1, 1991). The period of time that the valve was closed during the evolution was a short duration and positive controls were in place to immediately reopen the valve in event of an accident condition. There were no challenges to the ECCS during this time frame and no accumulator failures occurred. Isolation valve 2-FCV-63-80 is checked each shift to ensure that the valve is open in accordance with 2-SI-OPS-000-002.0, "Shift Log," which is required by TS SR 4.5.1.1.1. No deviations of the valve from the open position were identified during the approximate two week time frame.

Although the potential existed to challenge the design basis, there were no challenges to the ECCS or failures of 2-FCV-63-80, and therefore this event did not adversely affect the health and safety of the public.

CORRECTIVE ACTIONS

Immediate corrective action was to place the breaker for the operator for 2-FCV-63-80 in the locked open position.

The Plant Manager has discussed with the Operations personnel involved with this event the importance of performing independent verification for activities affecting nuclear safety. The Operations Superintendent has discussed with each of the Operations crews the circumstances of this event and the importance of performing independent verification in accordance with AI-37. Additionally the Operations personnel involved in this event will receive appropriate disciplinary action by April 19, 1991.

To provide interim controls until associated procedures are revised, a night order was issued by the Operations Superintendent to (1) require the Operations Superintendent's approval before performing a limited evolution (i.e., without a procedure) until further training is provided, (2) to require discussion with the Operations Superintendent if an unexpected response is encountered during a limited evolution and (3) to clarify that the independent verification requirements of AI-37 applies to component manipulations regardless of the AI-58 method that is used to control the configuration. Associated procedures will be revised to further clarify the need for independent verification by May 15, 1991.

While TVA believes that the subject activity should have been conducted with an approved procedure, TVA also believes there still remain certain simple manipulations involving deviations from normal configurations that should properly be considered operation of the facility rather than changes in the facility. For certain simple, short duration manipulations that will not require a bypass of permissives and for which direct positive control is maintained, generation of special procedures is not considered warranted and could impede reasonable facility operation. However TVA recognizes that these evolutions must be adequately and consistently controlled.

CORRECTIVE ACTIONS

Additional criteria for evaluation and conduct of limited evolutions is being developed by TVA based on Operations management and personnel input, regulatory requirements, and input from other utilities. Bypassing of interlocks will be specifically disallowed under limited evolutions. A resultant training package is being prepared which will be provided to all licensed personnel. Additional guidance will be incorporated into associated procedures as appropriate. TVA is additionally evaluating whether an additional temporary procedure process should be established to handle activities/evolutions that are beyond the scope of limited evolutions but do not warrant development of a new formal procedure.

To provide clarification and promote consistency in future interpretation of TS 3.5.1.1, Action b, Operations management will review the position that intentional closure of the CLA isolation valves in Modes 1, 2, 3, and with pressurizer pressure above 1000 psig should not occur with licensed personnel.

TVA is additionally evaluating current processes/interfaces used to support initial reportability determinations, with particular reference to nuclear engineering involvement. This evaluation will be completed by April 15, 1991, and governing procedures/processes will be revised as appropriate.

ADDITIONAL INFORMATION

Previous mispositioning events were reviewed to determine if an event resulted from similar causes. None were identified such that corrective actions taken should have reasonably been expected to prevent this event.

Inspection Report Nos. 50-327/89-15, 50-328/89-15, and Notice of Violation 89-15-05 involved making a change to the facility as described in the FSAR without performing a written evaluation to determine whether the change involved an unreviewed safety question. The change involved taking the boron injection tank (BIT) out of continuous recirculation, resulting in the low flow alarm actuating and rendoring the BIT inoperable. There was no procedure used to initially isolate BIT recirculation. Corrective action for this violation included a revision to AI-30 to define the conditions and controls under which manipulations can be performed without procedures. This revision was made with the intent to provide flexibility to address any number of unforseen simple scenarios; however, in hindsight, additional detail or training should have been provided to ensure appropriate and consistent implementation.

COMMITMENTS

1. Associated procedures will be revised to further clarify the need for independent verification by May 15, 1991.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Sequoyah Nuclear Plant Unit 2	0 5 0 0 0 3 2 8	9 1	-- 0 0 3	-- 0 0	1 1 OF 1 1

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

COMMITMENTS

2. Additional criteria for evaluation and conduct of limited evolutions is being developed by TVA based on Operations management and personnel input, regulatory requirements, and input from other utilities. Bypassing of interlocks will be specifically disallowed under limited evolutions. A resultant training package is being prepared which will be provided to all licensed personnel by April 26, 1991.

3. Additional guidance (regarding limited evolutions) will be incorporated into associated procedures as appropriate by April 26, 1991.

4. To provide clarification and promote consistency in future interpretation of TS 3.5.1.1, Action b, Operations management will review the position that intentional closure of the CLA isolation valves should not occur with licensed personnel by April 19, 1991.

5. TVA is additionally evaluating current processes/interfaces used to support initial reportability determinations, with particular reference to nuclear engineering involvement. This evaluation will be completed by April 15, 1991.

6. Governing procedures/processes will be revised as appropriate as a result of commitment Number 5.