UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

Report Nos: 50-259/85-57, 50-260/85-57, and 50-296/85-57

Licensee: Tennessee Valley Authority
500A Chestnut Street Tower II
Chattanooga, Tennessee 37401

Docket Nos.: 50-259, 50-260, and 50-296

License Nos.: DPR-33, DPR-52, and DPR-68

Facility Name: Browns Ferry Nuclear Plant

Inspection Conducted: November 21 - December 31, 1985

Inspectors: AWJohnson Jr for — 2/5/86
G. L. Paulk, Senior Resident Inspector — Date Signed

AWJohnson Jr for — 2/5/86
C. A. Patterson, Resident Inspector — Date Signed

AWJohnson Jr for — 2/5/86
C. R. Brooks, Resident Inspector — Date Signed

Approved by: FCantrell — 2/5/86
F. S. Cantrell, Section Chief, — Date Signed
Division of Reactor Projects

SUMMARY

Scope: This routine inspection involved 280 resident inspector-hours in the areas of operational safety; maintenance observation; surveillance testing; reportable occurrences; receipt, storage and handling of equipment program; and emergency drill.

Results: Four Violations summarized below:

1. 10 CFR 50.59 for failure to have prior NRC approval for changing a secondary containment isolation damper timing.

2. Technical Specification 6.3.A.6 for failure to conduct sixteen surveillance instructions during shutdown and refueling.

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3. 10 CFR 50, Appendix B, Criterion V for four examples:

   a) Failure to have 4160 volt shutdown board A control power connected per plant drawing.

   b) Failure to have diesel generator oil pressure switch functional as per plant drawing.

   c) Failure to have an adequate procedure to conduct LPRM changeout.

   d) Failure to conduct charcoal bed iodine removal analysis in accordance with the test requirements.

4. 10 CFR 50, Appendix B, Criterion II for failure to carry out the Quality Assurance Program in accordance with written procedures after the discovery of nonconforming fuel channels.

REPORT DETAILS

1. Licensee Employees Contacted

   W. C. Bibb, Site Director
   T. F. Ziegler, Assistant to the Site Director
   R. L. Lewis, Plant Manager
   J. E. Swindell, Superintendent - Operations/Engineering
   T. D. Cosby, Superintendent - Maintenance
   J. H. Rinne, Modifications Manager
   J. D. Carlson, Quality Engineering Supervisor
   D. C. Mims, Engineering Group Supervisor
   R. M. McKeon, Operations Group Supervisor
   C. G. Wages, Mechanical Maintenance Supervisor
   J. C. Crowell, Electrical Maintenance Supervisor
   R. E. Burns, Instrument Maintenance Supervisor
   A. W. Sorrell, Health Physics Supervisor
   R. E. Jackson, Chief Public Safety
   T. L. Chinn, Senior Shift Manager
   J. R. Clark, Chemical Unit Supervisor
   B. C. Morris, Plant Compliance Supervisor
   A. L. Burnette, Assistant Operations Group Supervisor
   R. R. Smallwood, Assistant Operations Group Supervisor
   S. R. Maehr, Planning/Scheduling Supervisor
   G. R. Hall, Design Services Manager
   W. C. Thomison, Engineering Section Supervisor
   C. E. Burke, Radwaste Group Controller

   Other licensee employees contacted included licensed reactor operators, auxiliary operators, craftsmen, technicians, public safety officers, Quality Assurance, Design and engineering personnel.

   Mr. W. C. (Bill) Bibb was named Site Director at Browns Ferry effective November 25, 1985. Mr. Bibb comes under contract to the plant from his position as Vice President of Operations Services for Management Analysis Company. He will be Site Director under a management contract for up to two years until either a permanent site director is employed or an assistant site director is adequately trained. Mr. Bibb has thirty years of experience in nuclear power plant operations and project management of nuclear plant construction.

2. Exit Interview (30703)

   The inspection scope and findings were summarized on January 6, 1985 with the Plant Manager and/or Assistant Plant Managers and other members of his staff.

   The licensee acknowledged the findings and took no exceptions. The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection.

3. Licensee Action on Previous Enforcement Matters (92702)

(Closed) Unresolved Item (259/260/296/85-49-01) The missed surveillances resulted in a violation discussed in this report.

(Closed) Unresolved Item (259/260/296/85-39-03) The charcoal bed heaters have been determined not to be needed and have been deenergized. All TVA site directors now periodically meet to discuss related plant problems. This item is closed.

(Closed) Unresolved Item (259/260/296/85-49-03) The damper timing problem is addressed as a violation in this report.

(Closed) Unresolved Item (259/260/296/85-49-04) Recent reviews of problems occurring during surveillance testing have been indicated on Surveillance Procedure cover sheets. This item is closed.

(Closed) Unresolved Item (259/260/296/85-49-02) The licensee initiated Discrepancy Report 85-0567 for failure to properly control nonconforming fuel channels in accordance with Part III, Section 7.1 of the Nuclear Quality Assurance Manual (NQAM) and Standard Practice 16.5, Nonconforming Material, Components, and Spare Parts. This item is upgraded to a violation (259/260/296/85-57-11) for failure to implement the Quality Assurance Manual. An additional example of this violation is discussed in paragraph 9.

(Closed) Inspector Followup Item (259/85-25-05) Regarding the issue on individual cell voltages greater than +0.04 volts, the licensee received a letter from the battery manufacturer which stated that values of up to +0.1 volt are acceptable provided that the critical voltage of 2.13 volts is maintained. The manufacturer also stated that battery float voltage may be allowed to go as high as 2.33 volts per cell for up to one year. Regarding the minimum battery temperature for Shutdown Board 3EB, the licensee has completed its Design Study Request (DSR) on this subject. The DSR concluded that battery operability was not impaired by temperatures as low as 0 degrees F. Timely closeout of this open item was hampered by the lack of a Battery Load Profile in the design basis documents for the battery. A load profile was generated and the battery capacity requirements were calculated per IEEE-485-1976, "IEEE Recommended Practice for Sizing Large Load Storage Batteries for Generating Stations. This item is closed.

4. Unresolved Items* (92701)

a. HPCI Welding Problems

On November 20, 1985, the licensee identified that the unit two high pressure coolant injection (HPCI) system had a damaged pipe anchor support. The damage was at penetration XII for the HPCI steam supply

*An Unresolved Item is a matter about which more information is required to determine whether it is acceptable or may involve a violation or deviation.

where it attaches to the building steel. The damage apparently was due to the method of welding in that the welding has warped and distorted the steel. The licensee is evaluating the configuration and the question of HPCI operability. This will remain an unresolved item for further review (260/85-57-01). The licensee has determined this item not reportable pending further evaluation.

b. Cable Separation Problem

On November 23, 1985, the licensee identified a concern that a violation of divisional electrical separation related to the residual heat removal system may have occurred. During preparation to perform a work plan involving cable 2R1746, it was thought that the division I cable may have been run in a division II conduit. Further information was discovered indicating that cable 2R1746 ran from panel 9-9 cabinet 2 breaker 228 to panel 25-62. This is from a division I panel to a division II panel.

This was determined not to be reportable by the licensee pending an office of engineering evaluation. If the cable is determined to be installed as shown on the drawing, then the generic implications will be examined. This will be an unresolved item pending the office of engineering evaluation. (250/260/296/85-57-02).

c. HPCI Control System Low Voltage

The licensee reported on October 30, 1985, that reliable operation of the HPCI control system cannot be assured at the minimum voltage (200 VDC) required by Section 8.6.2.3 of the FSAR. Information from Woodward and General Electric indicates that for reliable operation of the HPCI control system the input to the electro-mechanical governor control box should be no less than 42 VDC. During actual testing on Unit 2, the voltage to the control box dropped to less than 40 volts when the 250 volt DC system voltage was dropped to 200 volt DC.

This will continue to be tracked under Open Item (259/85-32-01).

d. Reactor Protection System (RPS) Instrument Racks Not Seismically Qualified

The licensee reported by the Emergency Notification System on December 20, 1985 that the Unit 2 RPS instrument racks containing the scram Barton level instruments (LIS 203 A-D) plus other safety instruments were not seismically qualified.

A seismic event could prevent or cause a reactor scram, prevent automatic depressurization system blowdown, prevent core spray initiation, prevent ATWS recirculation pump trip and other problems. This will remain unresolved pending further review (259/260/296/85-57-03).

e. Additional Comments

The licensee is presently evaluating some 40 items identified as potential licensee event reports. The above items (a and b) are examples of these. The licensee tracks these items with a safety issues list. Examination of the list shows that half of the items are past the 30-day due date for reporting. This will be an unresolved item pending further review of the licensee compliance with the Commission Order (EN 85-47) dated June 17, 1985 given to insure that potentially significant safety conditions are promptly evaluated and reported (259/260/296/85-57-04).

5. Operational Safety (71707, 71710)

The inspectors were kept informed on a daily basis of the overall plant status and any significant safety matters related to plant operations. Daily discussions were held each morning with plant management and various members of the plant operating staff.

The inspectors made frequent visits to the control rooms such that each was visited at least daily when an inspector was on site. Observations included instrument readings, setpoints and recordings; status of operating systems; status and alignments of emergency standby systems; onsite and offsite emergency power sources available for automatic operation; purpose of temporary tags on equipment controls and switches; annunciator alarm status; adherence to procedures; adherence to limiting conditions for operations; nuclear instruments operable; temporary alterations in effect; daily journals and logs; stack monitor recorder traces; and control room manning. This inspection activity also included numerous informal discussions with operators and their supervisors.

General plant tours were conducted on at least a weekly basis. Portions of the turbine building, each reactor building and outside areas were visited. Observations included valve positions and system alignment; snubber and hanger conditions; containment isolation alignments; instrument readings; housekeeping; proper power supply and breaker; alignments; radiation area controls; tag controls on equipment; work activities in progress; radiation protection controls adequate; vital area controls; personnel search and escort; and vehicle search and escort. Informal discussions were held with selected plant personnel in their functional areas during these tours. Weekly verifications of system status which included major flow path valve alignment, instrument alignment, and switch position alignments were performed on the residual heat removal systems.

A complete walkdown of the accessible portions of the Spent Fuel Pool and Fuel Pool Cooling system was conducted to verify system operability. Typical of the items checked during the walkdown were: lineup procedures match plant drawings and the as-built configuration, hangars and supports operable, housekeeping adequate, electrical panel interior conditions, calibration dates appropriate, system instrumentation on-line, valve position alignment correct, valves locked as appropriate and system indicators functioning properly.

a. Secondary Containment Isolation Dampers

While following up on the licensee's action regarding stroke times on Secondary Containment Isolation Dampers (Unresolved Item 85-49-03), the inspector became aware of a potential Unreviewed Safety Question as defined in 10 CFR 50.59. A November 25, 1985 memorandum from Plant Engineering to Operations summarized the results of a new analysis of the design basis fuel handling accident. The memo concluded that secondary containment isolation damper closure times shorter than 10 seconds are sufficient to mitigate the consequences of a Design Basis Accident. Discussion with licensee representatives confirmed that this memorandum was being used to consider the dampers operable with up to a 10 second stroke time as opposed to the two second criteria contained in Section 5.3.4.2 of the FSAR until a change could be made to the FSAR. Justification for this was contained in the design report, Radiological Impact of Ventilation Damper Closing Time During a Design Basis Fuel Handling Accident, dated October 31, 1985. This report concluded that radiation levels calculated in the analysis are significantly increased beyond the fuel handling design basis accident reported in the Final Safety Analysis Report for Browns Ferry. Since these results remained a small part of 10 CFR 100 limits and were below EPA protective action guides, increasing the damper closure time to more than two seconds was considered acceptable.

The inspector questioned whether this was an Unreviewed Safety Question per 10 CFR 50.59 since this would increase the consequences of an accident previously evaluated in the Safety Analysis Report. As of December 9, 1985, (14 days after implementation of the new 10 second closure time criteria) no evaluation to determine whether the change constituted an Unreviewed Safety Question had been performed. Licensee representatives informed the inspector that an Unreviewed Safety Question Determination (USQD) per Standard Practice 17.18 should have been prepared and promptly initiated one. On December 10, 1985, the USQD was reviewed by the Plant Operations Review Committee (PORC) and approved by the Plant Manager. The conclusion of the USQD was that this was not an Unreviewed Safety Question. The reason for this conclusion was that the consequences of a fuel handling accident are not significantly increased because the off-site dose for a 10 second closure time remains only a small part of 10 CFR 100 criteria and that dose rates within NRC guidance are previously evaluated. This appears to conflict with the design report which concluded that a significant increase of radiation levels would occur. Resolution of this conflict depends on the definition of "significant". The postulated increase in off-site dose was actually a factor of 20 greater than that reported in the FSAR. This question is moot, however, since 10 CFR 50.59 does not require a judgment on whether the increase is significant or not.

The licensee was informed on December 13, 1985, that NRR and Region II had reached a consensus that the change constituted an Unreviewed Safety Question and that prior Commission approval was required per 10 CFR 50.59.

Further support that prior NRC approval was required came from a review of Section 5.4.B of the Browns Ferry Technical Specifications. This Section requires that the secondary containment shall be as described in Section 5.3 of the FSAR. Section 5.3.4.2 of the FSAR states that reactor zone isolation dampers are closed in two seconds upon receipt of signals from the radiation monitor. A change to this section of the FSAR would constitute a change to the technical specification and would therefore require prior Commission approval. This failure to obtain prior Commission approval for a change to the technical specification and a change which involved an Unreviewed Safety Question is a Violation of 10 CFR 50.59 (259/260/296/85-57-05).

b. High Radiation from LPRM Changeout

During a local power range monitor (LPRM) manipulation on November 20, 1985, the set point (100 mr/hr) for area radiation monitor (ARM) 2-RM-90-141 on the refuel floor was exceeded. This resulted in a secondary containment isolation and standby gas treatment initiation.

The monitor alarmed when an LPRM was inadvertently raised to the surface of the unit. two spent fuel pool resulting in the radiation increase. Each LPRM is forty feet in length consisting of a 15 foot "hot" end located in the core and a 25 foot "cold" end beneath the core. The whole assembly is very flexible. Each LPRM is held in the core top guide plate by a spring loaded plunger. By use of a special LPRM tool the LPRM is removed from the core top guide plate and raised to five feet below the water surface level. A safety hook is attached to a collar on the LPRM. The LPRM is transferred to the spent fuel pool by moving the refueling platform toward the spent fuel pool and the remaining length of the LPRM in the core dragged across the vessel flange. While the LPRM was dragged across the edge of the vessel flange, the tip of the hot end of the LPRM was pulled off but remained in the LPRM tool. This allowed the LPRM to slide through the tool preventing a cinch hold. Attempts to position the LPRM into the storage position in spent fuel pool were hampered by the LPRM sliding through the tool. During this movement, the LPRM became lodged behind a source pin rack along the side of the spent fuel pool. This whole process of positioning the LPRM is complicated as the spent fuel pool is 32 feet deep and the LPRM is 40 feet long and very flexible. To store the LPRM the hot end is positioned in one corner of the pool and the cold end bent upward at the other end of the pool and tied to the side of the pool. Although the procedure did not address the abnormal conditions of a broken tip and the LPRM being stuck, operations continued resulting in an unnecessary radiation hazard to personnel.

First, an attempt was made to remove the LPRM from behind the source pin rack by over-riding the limit switch on the monorail hoist which allowed the LPRM to come closer to the pool surface than five feet. This did not work. Next J-hooks and the safety hook were used to try to dislodge the LPRM. During this time, the LPRM sprang upward and to

the surface of the water. This resulted in exceeding the 10 mr/hr surface radiation level allowed in procedure step 5.c.2.1. The measured radiation on an area radiation monitor 53 feet away was 450 mr/hr. The health physics technicians covering the job immediately detected the increase in radiation and operations personnel lowered the LPRM. All personnel involved had their thermoluminescent dosimeters processed. The highest dose was calculated at 22 mrem. Radiation readings were taken from the refuel floor, reactor building, elevation 664, area radiation monitors and are listed below:

Unit 2	During Event	After Event (Background)
2-90-140	85 mr/hr	12 mr/hr
2-90-141	450 mr/hr	17 mr/hr
2-90-142	15 mr/hr	0.8 mr/hr
2-90-143	6 mr/hr	0.9 mr/hr

Failure to have an adequate procedure appropriate to the circumstances was given as an example of a violation against 10 CFR 50, Appendix B, Criterion V (259/260/296/85-57-06). Additionally, the licensee identified other procedural deficiencies after the event had occurred concerning radiological cautions, pre-job briefing, and physical movement of the LPRM. These were reported in Licensee Event Report 260/85017.

Additional discussion with plant personnel involved in the LPRM changeout revealed that LPRM tips had been broken several times in the past without correction of the cause. The broken tips were attributed to moving the LPRM sideways (horizontal) with the LPRM tool weight tending to keep the tool upright (vertical) causing an undue force on the LPRM tip. The procedure was changed to require removal of the tool once the safety hook was attached and prior to moving the LPRM toward the spent fuel pool. Besides the personnel involved in the LPRM changeout no futher management involvement took place until after the event occurred. The corrective actions after the event included a procedure revision and retraining of personnel. A report written describing the event was very thorough.

6. Maintenance Observation (62703)

Plant maintenance activities of selected safety-related systems and components were observed/reviewed to ascertain that they were conducted in accordance with requirements. The following items were considered during this review: the limiting conditions for operations were met; activities were accomplished using approved procedures; functional testing and/or calibrations were performed prior to returning components or system to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; proper tagout clearance procedures were adhered to; Technical Specification adherence; and radiological controls were implemented as required.

Maintenance requests were reviewed to determine status of outstanding jobs and to assure that priority was assigned to safety-related equipment maintenance which might affect plant safety. The inspectors observed the below listed maintenance activities during this report period:

a. Diesel Failure to Start on November 19, 1985

On November 19, 1985 during performance of the monthly Operability Surveillance Instruction, SI-4.9.A.1.a, the "B" diesel generators failed to start. A similar failure occurred on August 27, 1985 and was documented in Inspection Report 85-45. The diesel governor was suspected as the cause of the failure to start and was replaced with a new one. An evaluation of the old governor is planned. However, the exact cause of the failure to start has not been determined. The diesel generator was considered operable after performance of other maintenance items. The cause of the failure to start will remain an inspector followup item for further evaluation (259/260/296/85-57-07).

b. Diesel Failure to Start on December 16, 1985

On December 16, 1985, the "B" diesel generator failed to start during performance of Surveillance Instruction SI 4.9.A.1.a (diesel generator monthly operability). The cause of the failure was a defective cell switch located in the diesel generator output breaker (1822) compartment. The same diesel failed to start on 8-27-85 and 11-19-85. The cause of the previous failures have not been determined. The licensee is conducting a failure evaluation of the switch failure. This will remain an inspector followup item pending review of this evaluation (259/260/296/85-57-08).

c. Diesel Oil Pressure Switch Not Functional

The licensee found after completion of maintenance on the 1B diesel generator that a connection block for four oil tubing lines had not been drilled for the bottom line. The oil line connected from the connection block to the backup oil pressure switches PS-82-29 A, B, C, D (MB-3), which signals the diesel air start motors to disengage as the engine oil pressure increases during startup. The primary signal to disengage the air start motor is normally taken from the engine speed sensor. This condition had not been detected in the past because the pressure switch was periodically tested by disconnecting the oil line at the pressure switch. The condition was found this time by pressurizing the oil line upstream of connection block with no observed response from the pressure switch. This condition was found to be common to all four unit one and two diesels and has existed since original plant installation. The unit three diesels were not affected as the oil lines were connected by a series of T connections instead of a connection block. The licensee is conducting a 10 CFR 21 evaluation of the connection blocks. There are six connection blocks in each diesel generator local control cabinet. All of the blocks on the A diesel generator were checked and no additional problems found. Inspection of the other diesel generators is planned.

Additional questioning of why this condition was not indicated in the control room was conducted. A low low oil pressure light should have been received when the engine was running with no oil pressure to the pressure switch. All four diesel generator lights were found not functioning. The low low oil pressure light is adjacent to the emergency stop button on the diesel generator control board in the control room. The lights were replaced and this caused a continuous illumination of the lights when the associated diesel was running due to the pressure switch sensing no oil pressure. The licensee removed the lights for the units one and two diesels. The inspector reviewed the Plant Operating Instruction OI-82 for the diesel generator and found the indicating light was not addressed in the procedure. Plant Drawing 45N767-4 shows the pressure switch circuitry and associated indicating light. Failure to have the pressure switches installed per plant design drawings, failure to adequately monitor and maintain operational readiness of the low low oil pressure indicators, and failure to have an adequate operating instruction to address the alarm pressure sensing condition are collectively given as another example of the violation against 10 CFR 50, Appendix B, Criterion V (259/260/296/85-57-06).

7. Surveillance Testing Observation (61726)

The inspectors observed and/or reviewed the below listed surveillance procedures. The inspection consisted of a review of the procedures for technical adequacy, conformance to technical specifications, verification of test instrument calibration, observation on the conduct of the test, removal from service and return to service of the system, a review of test data, limiting condition for operation met, testing accomplished by qualified personnel, and that the surveillance was completed at the required frequency.

a. Shutdown Board Control Power Wired Incorrectly

During performance of Surveillance Instruction SI-4.9.A.2.c for the battery discharge test of the shutdown board battery A, all 250 volt DC control power to the 4160 volt shutdown board A was lost. Troubleshooting by the licensee found that the electrical power cables for the normal and alternate sources of 250 volt D.C. control power were reversed. The normal control power is from a small 250 volt DC battery (shutdown board battery A) which is rated for three hours; and the alternate control power is a large plant 250 volt D.C. battery (unit two battery connected to number two battery board) which is only required to maintain voltage for 30 minutes as discussed in the Final Safety Analysis Report (FSAR) section 8.6. The licensee reported this error on November 19, 1985, by the emergency notification system.

The consequence of the error was that control power to the 4160 volt shutdown board would be supplied from a source of power other than as stated in the FSAR. Since the control power would have been supplied from a source of power only required to maintain voltage for 30 minutes, the shutdown board would not have been functional for the required amount of time following an accident. Additionally, the

additional load on battery board two supplying the 4160 volt shutdown board control power may have affected its ability to supply its post accident loads for the time required. The wiring error would have confused the operators in an accident, complicating the accident scenario.

The time of the error was traced to original construction since to reverse the leads one set of leads had to be spliced six inches to make the proper connection. The leads should have been correctly installed under engineering change notice (ECN) E-19 in 1973. No work plans could be found that were associated with the ECN. It could only be conjectured why the error had not previously been detected. One possible explanation was that the identification label for the control power transfer switch for the normal and alternate positions was missing and was found inside the 4160 volt shutdown board. Past surveillance battery discharge tests (done every two years) of the shutdown board battery and the unit battery should have identified this discrepancy if done correctly. It can only be conjectured that the surveillance tests were not adequately done or if the error was identified no corrective action was taken. Failure to have the control power connected as per plant Drawing 010608860 is another example of the violation against 10 CFR 50 Appendix B, Criterion V. This was discussed with plant management in an exit meeting on January 6, 1985 (259/260/296/85-57-06).

b. Missed Surveillances

A violation of Technical Specification 6.3.A.6 was identified after review of unresolved item 259/260/296/85-49-01 and Licensee Event Report 259/85-50 (259/260/296/85-57-09). Sixteen surveillances required by technical specifications were missed due to inadequately scheduling for the plant conditions of shutdown and refueling. Surveillance Instruction SI-1, Surveillance Program, used by plant personnel to schedule surveillances contained errors and was generally not in accordance with technical specifications. This violation was discussed in an exit meeting on January 6, 1986. Previous to this event the licensee utilized an outside contractor to evaluate technical specifications. The outside contractor identified many errors and discrepancies during their review of SI-1 as stated in paragraph six of the contractor report. This was discussed in Inspection Report IE 85-45, paragraph nine. The errors discussed in LER 259/85-50 were not previously identified. This area was also a long-term Regulatory Performance Improvement Program Item 9.7. During this report period the plant manager stated that an ~~additional~~ review is being considered to review all surveillance instructions.

c. Standby Gas Treatment System Iodine Removal Efficiency

Surveillance Instruction SI-4.7.B-6, Standby Gas Treatment System (SGTS), Iodine Removal Efficiency, is intended to satisfy Surveillance Requirement 4.7.B.2.a. This requirement calls for periodic laboratory analysis of carbon samples from the SGTS in accordance with ANSI N510-1975. ANSI N510-1975 "Testing of Nuclear Air-Cleaning Systems" in turn requires that laboratory testing of adsorbent samples be performed in accordance with RDT M 16-1T (1973) "Gas-Phase Adsorbents for Trapping Radioactive Iodine and Iodine Compounds". The licensee ships SGTS carbon samples under contract for testing to SAIC (Science Applications International Corporation) of Rockville, Maryland.

A review of the test results submitted to the licensee and telephone conversation with an SAIC representative indicate that two requirements of RDT M 16-IT are not being satisfied during the laboratory testing. Section 4.5.g of RDT M 16-IT requires that the test assembly be pre-equilibrated for 5 hours or until the differential temperature across the test bed stabilizes at less than 1 degree C. whichever is greater. Test results for SI 4.7.B-6 performed on July 13, 1985 for SGTS Train A state that the equilibration period was "0" minutes. Section 4.5.h of RDT M 16-IT requires that a complete test shall consist of three determinations which may be made simultaneously or sequentially. Individual test values are to be recorded for each sample and an average iodine removal efficiency is to be calculated. Test results for SI 4.7.B-6 performed on July 13, 1985 for SGTS Train A indicate that only one test was performed. These deficiencies are another example of the violation against 10 CFR 50, Appendix B, Criterion V (259/260/296/85-57-06).

SI 4.7.B-6, Attachment B, Test Parameter Sheet was also found to be in error. Step 5 of the data sheet specified the concentration of methyl iodine to be 0.05 to 0.15 mg/m3. This is in error since RDT M 16-IT requires the concentration to be 1.5 to 2.0 mg/m3. The correct concentration was used during the surveillance test, however. This error will be tracked as an Inspector Follow-up Item (259/260/296/85-57-10).

d. Reactor Protection System Surveillance

During a detailed systems review by General Electric Engineers under contract by the licensee, Surveillance Instruction 4.1.A.2, Reactor Protection System, Manual Scram, was found to contain deficiencies which indicated that various functions were not being adequately tested by the surveillance procedure. The licensee revised the procedure and conducted a walk-through of the revision as part of the procedure review and approval process. During this walk-through, the licensee discovered a wiring error which was previously undetected due to the inadequate procedure. Two relays in the manual scram channel on Unit 3 (relays 5A-K15A and 5A-K15C on Drawing 730E915-11) were wired in series as opposed to parallel. The only effect of this wiring error was that

misleading information could be provided to the operator. The backlight on the manual scram pushbutton which indicates whether the manual scram signal is cleared or not could be extinguished prior to the scram channel being fully reset. This problem exemplifies the value of the detailed systems review and corrective action being undertaken by the licensee and contractors. Per the enforcement policy contained in 10 CFR 2, Appendix C, the licensee is given credit for self-identification and correction of this problem and no violation will be issued.

8. Reportable Occurrences (90712, 92700)

The below listed licensee events reports (LERs) were reviewed to determine if the information provided met NRC requirements. The determination included: adequacy of event description, verification of compliance with technical specifications and regulatory requirements, corrective action taken, existence of potential generic problems, reporting requirements satisfied, and the relative safety significance of each event. Additional in-plant reviews and discussion with plant personnel, as appropriate, were conducted for those reports indicated by an asterisk. The following licensee event reports are closed:

LER No.	Date	Event
*296/85-23	10-23-85	Inadvertent Containment Isolation
*296/85-22	8-23-85	Inadvertent Containment Isolation
*296/85-19	7-25-85	Ventilation Exhaust Radiation Monitor Surveillance Testing revealed less than minimum monitors operable
*259/85-34	6-29-85	Late completion of Surveillance Test Requirement
*259/85-46	9-12-85	Unmonitored Turbine Building Effluent Release

No violations or deviations were identified in this area.

9. Receipt, Storage and Handling of Equipment Program (38702)

In October, 1985, six fuel channels were found outside the reactor building with no control over storage and handling (refer to Inspection Report 85-49, paragraph 5.B). Since no control over these fuel channels had existed since the fall of 1984, the inspector followed up on the corrective action. Mechanical Maintenance had responsibility over the fuel channels. The channels were moved inside, uncrated, cleaned, inspected and wrapped with polyethylene. Mechanical maintenance then returned the fuel channels to power stores in accordance with Standard Practice 16.4, paragraph 5.3, Turn In of CSSC Items to the Site Power Stores Section. Per this Standard Practice, receipt inspections at this point must verify that the items' quality has not been downgraded by physical damage or by apparent inadequate storage conditions. Fuel channels are defined as Fuel Related Components in Part II, Section 7.1 of the Nuclear Quality Assurance Manual (NQAM). Paragraph 2.3 of this section of the NQAM contains the receipt inspection requirements for Fuel Related Components. On November 19, 1985, Browns Ferry Power Stores receipt inspected the fuel channels; however, the following requirements of Part II, Section 7.1 of the NQAM were not satisfied during the inspection:

a. Inspection personnel were not certified fuel receipt inspectors per paragraph 2.3.3 and 2.1.4.

b. The inspection was not documented on Attachment 4, Fuel and Component Receipt Inspection Master Checkoff Log as required per paragraph 2.3.4.3.

c. An anomalous condition which could cause the fuel related component to be unacceptable to TVA was found but was not documented on a Site Fuel Discrepancy Report as required by paragraph 2.3.4.4. The anomalous condition was excessive chloride contamination on Fuel Channel number 2239 (1010 ug/dm^2 vs. Limit 80 ug/dm^2). The channel was subsequently cleaned to within acceptable chloride concentration.

d. Resolution of the excessive chloride contamination on Fuel Channel number 2239 was not reviewed and approved by the following as required by paragraph 2.3.5.2: Site Reactor Engineer, Chief Reactor Engineering Branch, Plant Operations Review Committee (PORC), and the Plant Superintendent.

Power Stores personnel were unaware that these requirements were imposed on fuel channels. Quality Engineering personnel who would normally be involved in receipt inspection of fuel related componen~~ts~~ were not involved in the receipt inspection or chloride contamination clean-up activity. This is an additional example of a violation for failure to implement the Quality Assurance Manual (259/260/296/85-57-11).

10. Emergency Preparedness Drill of November 13, 1985 (92705)

The resident inspector followed up on the personal contamination event that occurred during the annual emergency drill as discussed in I.E. Report 85-53.

The initial estimate of the skin dose received during the November 13, 1985, Tc-99m contamination incident was described in I.E. Report 85-53, paragraph 10. Through additional investigation into the event the licensee has concluded that the initial frisker readings of 500,000 c.p.m. were in error by a factor of 10.

The correct reading should have been 50,000 c.p.m. The licensee additionally conducted an investigation into the frisker efficiency using a finger phantom cover with chamois cloth to simulate human skin. The phantom was spiked with a known quantity of Tc-99m. The frisker efficiency was determined to be 0.97 percent (103.5 dpm/cpm) for the finger geometry. The determination of frisker dead time yielded 60 micro-seconds, making it possible to perform dead time corrections for all survey measurements. Using the new figures the total radiation received in the most severe case was 833 mrem.

The Health Physics Supervisor was questioned as to how an error could occur by the technician conducting the personnel contamination survey. No conclusive answers were obtained, however, the supervisor committed to perform re-training on personnel contamination survey techniques and counting methods for all personnel responsible for such functions. The item will be an inspector follow-up item (259/85-57-12).