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

Report Nos.: 50-259/89-35, 50-260/89-35, and 50-296/89-35

Licensee: Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

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

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

Facility Name: Browns Ferry Units 1, 2, and 3

Inspection at Browns Ferry Site near Decatur, Alabama

Inspection Conducted: July 15 - August 16, 1989

Inspectors: A. H. Johnson, Jr. for 8/29/89
D. R. Carpenter, NRC Site Manager — Date Signed

A. H. Johnson, Jr. for 8/29/89
C. A. Patterson, NRC Restart Coordinator — Date Signed

Accompanied by: E. Christnot, Resident Inspector
W. Bearden, Resident Inspector
K. Ivey, Resident Inspector
A. Johnson, Project Engineer

Approved by: 8/29/89
W. S. Little, Section Chief, — Date Signed
Inspection Programs,
TVA Projects Division

SUMMARY

Scope: This routine resident inspection included reportable occurrences and action on previous inspection findings.

Results: Fourteen LERS were reviewed and closed. Fourteen IFI's were reviewed and eleven were closed. Five violations were reviewed and one remains open. Seven URIs were closed with one being upgraded to a NOV and two upgraded to NCVs.

The NOV involved operator response to an off-normal condition, paragraph 3.t. The two NCVs concerned design control to prevent single failure and a missed SI, paragraph 3.u. and s.

REPORT DETAILS

1. Persons Contacted

Licensee Employees:

*O. Zeringue, Site Director
*G. Campbell, Plant Manager
*R. Smith, Project Engineer
J. Hutton, Operations Superintendent
A. Sorrell, Maintenance Superintendent
D. Mims, Technical Services Supervisor
G. Turner, Site Quality Assurance Manager
P. Carier, Site Licensing Manager
*P. Salas, Acting Compliance Supervisor
J. Corey, Site Radiological Control Superintendent
R. Tuttle, Site Security Manager

Other licensee employees or contractors contacted included licensed reactor operators, auxiliary operators, craftsmen, technicians, and public safety officers; and quality assurance, design, and engineering personnel.

NRC Attendees

*W. Little, Section Chief
*D. Carpenter, Site Manager
*C. Patterson, Restart Coordinator
*E. Christnot, Resident Inspector
*W. Bearden, Resident Inspector
*K. Ivey, Resident Inspector
A. Johnson, Project Engineer

*Attended exit interview

Acronyms used throughout this report are listed in the last paragraph.

2. Reportable Occurrences (92700)

The LERs listed below were reviewed to determine if the information provided met NRC requirements. The determination included the verification of compliance with TS and regulatory requirements, and addressed the adequacy of the event description, the corrective action taken, the existence of potential generic problems, compliance with reporting requirements, and the relative safety significance of each event. Additional in-plant reviews and discussions with plant personnel, as appropriate, were conducted.

a. (CLOSED) LER 259, 260, 296/85-12, Revision 1, Design Error in Standby Gas Treatment Cable Routing.

A TVA modifications engineer found divisional cables that had been routed and installed through cable tray fire stop pressure seals using cables designated for nondivisional application. In the BFNP

Fire Recovery Plan, cabling was added for use as spare cabling. This cabling was intended for use only in non-safety-related circuits. Due to a design drawing error, SBGT divisional cables were routed using spare cables. A junction box was also determined to be seismically unqualified. The cables in question could not be qualified as IEEE class IE and a design change was processed to correct this. The junction box was acceptably remounted. The NRC inspector reviewed the closure package for ECN P3208 and determined the item had been satisfactorily resolved. This item is closed.

b. (CLOSED) LER 259/85-18, Improper Modification of Secondary Containment Relief Panels.

On May 17, 1985, during a routine licensee inspection, the shift engineer observed that explosive bolts on the reactor zone to the refuel floor relief panels had been replaced with standard bolts and nuts. These relief panels serve to prevent excessive pressure differential between the reactor zones and the refuel floor during design basis tornadoes and during steamline breaks inside the reactor building.

The NRC inspector reviewed the LER, dated June 14, 1985, and the LER closure package and verified that it met the requirements of timeliness, content, and corrective action. The root cause was determined to be lack of administrative requirements to ensure that the proper explosive bolts were used. The licensee replaced the substituted bolts with the proper designed explosive bolts and established procedure controls to ensure adherence to the special requirements. Procedure MMI-14, "Inspection of Secondary Containment Relief Panels," was revised to include all secondary containment relief panels in a scheduled inspection to ensure that these relief panels are being properly maintained. Based on the in-office and field review of the LER and closure package, this item is closed.

c. (CLOSED) LER 260/87-07, Drywell Control Air Isolation Valves Outside Design Basis Because of Design Modification Error.

This item is identical to IFI 260/87-33-02. The IFI is being closed and is discussed in detail in Section 3 of this report. This LER is closed.

d. (CLOSED) LER 260/88-05, Revision 1, Unplanned ESF Actuations Due To Inadequate Procedures.

This item occurred on August 3, 1988, when Unit 2 received spurious low reactor water level signals, causing several ESF actuations. The spurious signals were received from level transmitters 2-LT-3-203A and 2-LT-3-203B when clearances were released on transmitters 2-LT-3-206 and 2PPT-3-207 following modifications which relocated the two instruments. All four of these transmitters are served by common sensing lines, which were drained, cut, and rerouted during the modification. The workplan for the modification did not include backfilling of the lines following their

reinstallation. The SOS was aware of probable ESF actuations upon removal of the clearance and therefore, directed IM assistance in returning the valves to their normal positions. The IM did not understand this to be a request to actually return the instruments to service and did not backfill the lines prior to repositioning the valves, thereby initiating the low level signals.

The licensee determined the root cause to be procedural inadequacies which allowed the clearance to be lifted without an adequately coordinated plan of action. The following procedural enhancements have been instituted:

- Procedure SDSP 8.4, "Modification Workplans," now requires that workplans be reviewed for potential ESF actuations in accordance with SDSP 7.9, "Integrated Schedule and Work Control."

- Procedure SDSP 14.9, "Equipment Clearance Procedure," now requires the SOS to specify the proper sequence for equipment restoration when removing clearances.

The above procedural enhancements were reviewed by the NRC inspector and appear adequate to preclude future events of this nature. This item is closed.

e. (CLOSED) LER 259/88-12, Unit 2 only, Battery Failure Concurrent With LOP/LOCA Automatic Start of RHR Pump.

On March 3, 1988, the licensee discovered during a review of the 250V DC system a condition that involved a single failure of a logic system power supply. The failure of a battery supplying logic power for division I of RHR would prevent one of the two pumps in that division from starting. The other RHR pump in the division that lost logic power receives a start signal from RHR division II logic. The battery failure also causes the start logic in division II to sense diesel generator power is available for the RHR pump that lost division logic when, due to the LOP, AC power would not be available on the AC electrical board until the diesel generator output breaker closes after the time delay for the diesel generator to obtain the rated voltage. This energizes the start relay for the RHR pump and causes the breaker to try and close onto a deenergized electrical AC board. When this occurs, the pump breaker will trip. The pump must then be manually started from the electrical distribution board.

The licensee's corrective action was to initiate three CAQR's (one for each unit) and to implement DCR 3549 through ECN E-2-P7136. This ECN required the installation of wiring and relanding of wiring on specific relays and terminal points in the shutdown boards as indicated in Work Package 2182-88.

The NRC inspector reviewed the documented corrective actions, observed the LOP/LOCA series of tests and considered the action appropriate. This LER is closed for Unit 2 only. The NRC inspector noted that this item is considered part of the BFN overall single failure issue and this issue is discussed further in paragraph 3 of this report.

f. (CLOSED) LER 260/88-13, Unplanned ESF Actuation Caused By Radiation Monitor Power Supply Failure.

On October 12, 1988, while troubleshooting using a maintenance request, instrument mechanics pulled the power supply for the Unit 2 reactor zone exhaust radiation monitor, refuel zone exhaust radiation monitor, and offgas system carbon bed vault radiation monitor. They observed arcing from a hole that had been burned in the high voltage transformer power supply. This arcing caused a high radiation signal, which resulted in an ESF actuation resulting in a SBGT train B and CREV train B auto start.

The NRC inspector reviewed the LER and the LER closure package, and verified that it met the requirements of timeliness, content, and corrective action. The root cause was determined to be a failed power supply. This was determined by the General Electric failure analysis which stated that the failure was due to component degradation through aging, aggravated by the relative lack of reliability of aluminum electrolytic capacitors. The licensee replaced the power supply and the new power supply operation was verified by performance of the applicable surveillance instruction. Based on the in-office review of the LER and closure package, this item is closed.

g. (CLOSED) LER 259/88-18, Unplanned ESF Actuations Due To Circuit Protector Trip Caused By Unstable Undervoltage Relay Failure.

This item involves two identical events which occurred 10 minutes apart on June 5, 1988. The 1A1 RPS circuit protector tripped, deenergizing the Unit 1 RPS Bus 1A and initiated several engineered safety features. Following the second occurrence, an investigation was initiated which determined the cause to be an unstable undervoltage relay in the circuit protector, which tripped when subjected to minor vibration. The defective relay was replaced and returned to the manufacturer for evaluation of potential generic problems. The manufacturer determined the probable cause of failure to be relay contact degradation due to airborne contamination creating corrosion and pitting of the contacts. The licensee also consulted NPRDs to determine whether similar failures of this type relay had occurred elsewhere within the industry. No similar failures were reported. Therefore, it was determined that this relay failure was an isolated case with no generic implications at this time. The inspector reviewed the above actions and evaluations and determined them to be appropriate. This item is closed.

h. (CLOSED) LER 260/88-19, Revision 1, Operation Over Spent Fuel Pools Without the Minimum Number of Standby Gas Treatment Trains Operable.

This event involved a SBGT train becoming inoperable when the surveillance period required by the Technical Specifications had expired.

The SBGT train was subsequently relied upon to meet a limiting condition for operation action statement. This constituted an operation prohibited by the technical specifications and resulted in the reportable event.

Issues related to this event are discussed elsewhere in this report in LER 259/88-48 (paragraph 2.j), and URI 260/88-35-02 (paragraph 3.s).

The NRC inspector reviewed the event description, cause and corrective actions and found that the LER met reporting standards. This LER is closed.

i. (Closed) LERs 259/88-22 and 259/88-43, Unplanned ESF Actuations Caused By Radiation Monitor Upscale Relay Failure

Both LERs identify separate unplanned ESF actuations resulting from failures of upscale relay coils in the reactor and refuel zones exhaust radiation monitoring circuity. Both events resulted in isolations of the Unit 1 primary containment, refuel zone, and control room ventilation systems and initiated SBGT and CREV systems.

The NRC inspector reviewed LERs 259/88-22, 259/88-22 Revision 1, 259/88-43, and other documentation provided by the licensee. The relay failures were due to undersized relays in the original design. In both cases, corrective maintenance was initiated to replace the failed relay. Following repair, the radiation monitor was functionally tested and returned to service. The original 24 volt Potter Brumfield KH4690 electromagnetic relays were determined by the licensee to be undersized when continuously operated at 24 volts. These relays are continuously energized when in use and fail due to deterioration resulting from excessive heat. Industry experience has shown that continuously operated relays should be rated at approximately 150% of their planned operating voltage. This condition had been previously identified in GE SIL 189, dated July 30, 1976. In the above LERs the licensee states that the failure to implement the recommendations of the GE SIL in a timely matter had been a contributing factor. The failure to implement this SIL was identified in July of 1987. The licensee had initiated procurement of new 36 volt relays but the new relays had not arrived at the time of the two events. Procurement delays resulted in the replacement relays not being available for installation until November 1988. The new upgraded relays are now installed in the refuel zone and reactor zone exhaust radiation monitors. As part of the corrective action to the LERs the licensee stated that vendor information is now reviewed and incorporated as required in accordance with Volume I of the Nuclear Performance Plan. These LERs are closed.

j. (CLOSED) LER 259/88-48, Unplanned ESF Actuation.

This event involves two items, the first was the discovery that the "C" SBGT Train was in operation on December 13, 1988 for no apparent reason. The train was secured and increased security surveillance of the area was provided. There was on going work in the area. The second item was the discovery, through a surveillance functional test of the existence of a undocumented inlet damper in the suction ductwork of the "C" train. The "C" train was declared inoperable, the damper was locked open, the system was flow tested and returned to service. A CAQR BFP 881087 was initiated to investigate and correct appropriate drawings and procedures.

This event is associated with LER 260/88-19, Revision 1 (paragraph 2.h) and Unresolved Item 260/88-35-02 (paragraph 3.s) and are discussed further in this report. A violation was issued in NRC IR 89-33.

The NRC inspector reviewed the event in association with the above related issues and the CAQR corrective action document. The actions proposed were adequate and were implemented in a timely manner. This item is closed.

k. (CLOSED) LER 259/89-02, Design Error On EECW Discharge Causes Plant To Be In An Unanalyzed Condition.

This LER was associated with the presence of seismically unqualified vitrified clay piping in certain portions of the EECW discharge flowpaths for various safety related components. Contrary to the requirements of FSAR section 10.10.2.2, EECW piping was found to discharge into non-qualified 24 inch RCW discharge headers. These 24 inch RCW headers were routed from the Reactor Building through the RHRSW pipe tunnels where they eventually became buried pipe and tied into 30 inch headers constructed of vitrified clay. During a seismic event, these vitrified clay headers could collapse and block the discharge flow paths to the affected components.

The licensee reported the condition to the NRC on February 8, 1989. The circumstances and events associated with this issue are discussed in greater detail in NRC Inspection Report 259, 260, 296/89-10. Any regulatory concerns associated with this issue will be followed as part of the open items identified in that report. This section will only address the technical resolution of the non-seismically qualified discharge piping.

Separate plant modifications are intended to correct the problem by rerouting the three affected EECW discharge paths to qualified discharge paths. The licensee has planned and/or performed work for the following DCN's:

- H5120A reroutes piping associated with both Unit 1/2 Control Bay Chillers to qualified Unit 1 EECW discharge piping.

- H5121A reroutes piping associated with Unit 2 Shutdown Board Room Coolers to qualified Unit 2 EECW discharge path.

- H5122A reroutes piping associated with Unit 3 Control Bay Chiller 3A to qualified Unit 3 EECW discharge path.

H5120A and H5122A were field complete as of July 14, and work on H5121A commenced on July 18. The licensee has an outstanding commitment (NC08900920021) to complete all three modifications prior to tensioning the Unit 2 reactor vessel head. The NRC inspector considers that any concerns associated with the technical resolution of this issue are satisfied. This item is closed.

l. (CLOSED) LER 296/89-02, Missed Compensatory Sampling While Conductivity Monitor Was Out Of Service.

This item involves the failure to perform compensatory reactor coolant water conductivity sampling in Unit 3 at eight hour intervals while local conductivity monitor 3-CIT-43-011 was inoperable, as required by TS 4.6.B.1.c. This compensatory sampling was required when the local monitor was removed from service for repair and calibration. Procedure SDSP 7.9, "Integrated Schedule and Work Control," did not require an IE to be performed on this type of instrument prior to allowing work to begin. The ASOS was not aware that the monitor would be rendered inoperable during troubleshooting and recalibration, thus, the required eight hour sampling was not performed for approximately 23 hours. The licensee determined the root cause to be the inadequacy in SDSP 7.9 which did not require the performance of the IE. Had the IE been performed, all appropriate personnel would have been aware of the requirement for compensatory sampling. Procedure SDSP 7.9 has been revised to require specifically IEs to be performed for chemical instrumentation equipment covered by technical specifications. The inspector reviewed the above actions and procedure revision and determined them to be appropriate. Therefore, this item is closed.

m. (CLOSED) LER 296/89-03, Unplanned Engineered Safety Features Actuations Caused By Voltage Transient On Electrical Distribution System.

This issue is also addressed in IFI 260/88-28-03. The IFI is discussed in detail and closed in paragraph 3.m of this report. This LER is closed.

3. Action on Previous Inspection Findings (92701, 92702)

a. (CLOSED) IFI 260/86-40-03, Unit 2 only, IRM Power Supply and Procedure Changes Per SIL-445.

This item involves procedural enhancements and equipment modifications suggested by General Electric SIL-445. During an outage at an operating GE/BWR, all positive and negative IRM 3/4 amp fuses connected to the 24 vdc bus B were blown because of a power

surge caused by a switching transient on the 480V power supply. After the positive 3/4 amp fuses were replaced, all IRM channels were operating normally. However, because of continued loss of the negative power supply because of the blown fuses, the IRM channels remained inoperable and unable to process flux signals. The blown negative fuses were only detected during surveillance testing performed later because there was no blown fuse indication on the control room panels. In view of the above, SIL-445 made the following recommendations:

- Procedural enhancements to require functional testing of SRM and IRM channels to ensure channel operability.

- Replacement of the 3/4 amp IRM chassis fuses with 1.5 amp fuses, and

- Modification to provide a reactor protection system INOP trip in response to a loss of negative 24V power supply to the IRMs.

As a result of the above recommendations, TVA has performed the following actions:

- Procedure 0-OI-57D, "DC Electrical System Operating Instruction," Rev. 3, Section 5.7.12 requires functional testing of IRMs and SRMs.

- The above referenced fuse replacement and system modifications have been completed, for Unit 2 only, per DCN-H1706A and Work Plan 2583-88.

The inspector reviewed TVA's actions and determined them to be adequate to close this item for Unit 2 only. Units 1 and 3 will remain open pending completion of their respective modifications.

b. (CLOSED) IFI 259, 260, 296/86-40-12, Potential for Overpressurization of Residual Heat Removal System Piping.

This item concerns a modification installed to reduce excessive pressure drop across a throttling valve in the RHR system. This item was reviewed in NRC Inspection Report 259, 260, 296/88-32, paragraph 9.e which concluded that the engineering analysis did not consider the design basis LOCA, FSAR Section 14.6.3.3.2 where torus pressure could be as high as 27 psig. The basic issue is that the portion of the RHR system in question is rated at 150 psig, which may be exceeded. With the modification, which installed an eight inch orifice plate downstream of throttling valve FCV-74-73 to reduce excessive pressure drop across that valve, the pressure between the orifice and the valve could be as high as 143 psig under normal conditions with the torus at atmospheric. During a LOCA event, the torus could be pressurized to 27 psig which would mean that the piping section could exceed 170 psig, in fact by worse case calculation 173.8 psig.

The licensee's reanalysis confirms this data. The Code of record for this system is USAS B31.1.0 - 1967. Under Section 102.2.4, Ratings: Allowance for Variations from Normal Operation, the following allowances are provided:

(1) Up to 15 percent increase above the S-value during ten percent of the operating period.

(2) Up to 20 percent increase above the S-value during one percent of the operating period.

For the LOCA condition, section (2) above would apply, which would allow a maximum pressure of 180 psig during one percent of the operating time. Since maximum analysis pressure would be 173.8 psig, the system would be within code allowable. This consideration is consistent with the NRC's staff position on similar issues resolved at SQN. This item is closed.

c. (CLOSED) IFI 259, 260, 296/87-FRP-01, Closeout TMI Item II.F.1.(3) For Containment High Range Radiation Monitors (CHRRM).

The item was opened in IR 87-33 as a violation for an inadequate design modification package. The NRC inspectors concerns about the modification at that time was `that it lacked engineering documentation and calculations to support evaluation of the adequacy of the design. TVA responded to the item in a letter to the NRC dated January 11, 1988 in which they maintained that design and justifications met regulatory requirements. The NRC concurred with the TVA position and withdrew the item in a letter to TVA dated October 31, 1988. The October 31 letter also stated that the issue would be identified as IFI 259, 260, 296/87-FRP-01, because of several deficiencies described with the violation. The IFI was opened in NRC inspection report 88-14. In that report the NRC inspector discussed the system design and installation as well as the fact that the system was not operable at that time and would be subsequently followed up in later inspections.

The NRC inspectors recent review of this issue revealed that the licensee discussions of system design in its violation response provided reasonable support in the that system design met guidelines specified in NUREG 0737, met industry standards, and is therefore considered satisfactory.

The licensee provided documentation and explanation of concerns raised by the NRC about the implementation of the modification that were described in IR 87-33. These documents and discussions, along with NRC inspectors field observations of the modification, indicate that the deficiencies involving detector orientation, electrical power supply capabilities, post modification retest requirements, and functional testing of the completed circuits have been adequately resolved. Several licensee programs exist that are routinely monitored by the NRC that will ensure the return to service of this system. These programs include: LCO Tracking (this system is required by TS 3.2.F); commitments to complete TMI items discussed in a letter from TVA to the NRC dated June 16, 1989; the licensee return to service program, and System Pre-operability Checklist (SPOC). The NRC inspector considers this item closed.

d. (OPEN) IFI 259, 260, 296/87-02-06, Baseline Walkdown Problems.

This item identified that the diesel generator starting air motors were not shown as part of the starting air system in FSAR figure 8.5-2. The licensee prepared a proposed change to the FSAR to correct this item. The inspector reviewed the proposed change to the FSAR. TVA has requested and been granted a temporary exemption from 10 CFR Part 50.71(e) for an annual update of the FSAR. The FSAR change will be made in the July, 1990 update. TVA is maintaining a "living" FSAR containing the proposed changes until the FSAR is updated. The inspector reviewed a controlled copy of the "living" FSAR in Document Control and found the proposed changes in place. Therefore, this item is considered acceptable for restart based on the temporary exemption. The item remains open since the original concern has not been corrected in the FSAR.

e. (CLOSED) IFI 259, 260, 296/87-20-02, IE Notice Closeout

This item concerns the process and adequacy of nuclear experience review activities. Specifically, after an action was identified as being required at BFNP based on the nuclear experience review, the item was being closed when the responsible supervisor stated that a particular action was committed to be done. There was no followup after work was committed to be done. The licensee committed to revise the governing procedures, BFNP Standard Practice BF-21.17, Review, Reporting, and Feedback of Operating Experience Items, and to perform a QA audit of the experience review process.

Since NRC IR 87-20, the licensee has replaced BF-21.17 with SDSP 15.9, Nuclear Experience Review Program, and performed the committed QA audit. The audit results were used in formulating SDSP 15.9 and in strengthening the experience review program. There were 22 IE Notices identified in the QA audit that were reopened based on incomplete committments.

The inspector reviewed SDSP 15.9, Revision 5 and determined that it provides adequate guidelines and checks to ensure that nuclear experience review action items are tracked until completion and acceptance of the required action. Section 6.3 of SDSP 15.9 requires the responsible supervisor to retain Attachment G, "Closure of NER Item," until the action items are fully implemented. When completed and returned to the BFN Site Licensing Manager, Attachment G tracking on the NER and/or TROI data base will be closed. The licensee's action on this item were responsive and acceptable. This item is closed.

f. (CLOSED) IFI 260/87-33-02, Failure of Drywell Control Air Isolation Valves to Fail Closed Upon Loss of Air.

During performance of Restart Test Procedure -032, the drywell control air suction valves (FCV-32-62 and -63) failed "as is" upon loss of control air instead of failing closed, as was intended. The licensee determined the cause of this malfunction to be the improper implementation of an equipment modification intended to upgrade the solenoid valves for environmental qualification. In addition, the following related problems were also noted:

- Drawings 1-47E610-32-2 (Units 1 & 3) and 2-47E610-32-2 (Unit 2) incorrectly depicted these valves as diaphragm valves.

- Drawing 1-47E610-32-2 (Units 1 & 3) erroneously indicated, in Note 8, that the air supply for these valves in both units was drywell control air.

- These valves were found to be missing from FSAR Table 7.3-1, "Pipelines Penetrating Primary Containment."

To correct these problems, the licensee has completed the following actions:

- The Unit 2 valves have been replaced per ECN W0690 and work plan 2353-88, and were successfully retested on May 22, 1989 per 2-BFN-RTP-032, CN-08.

- Drawing 2-47E610-32-2 has been revised to identify accurately the valves as air operated-vane drive motor plug valves.

- Amendment 6 to the FSAR incorporated these valves into Table 7.3-1.

The inspector reviewed the above completed actions and determined them to address adequately the identified problems as they pertain to Unit 2. Therefore, this item is closed for Unit 2, but will remain open for Units 1 and 3 pending completion of the necessary hardware modifications and drawing deficiencies. In addition, the licensee had also reported the valve malfunction in LER 260/87-07. This LER is closed in paragraph 2.c of this report.

g. (CLOSED) IFI 260/87-37-03, Reactor Water Level Sensing Lines.

This item involves questions pertaining to licensee's resolution of the February 13, 1985 reactor water level mismatch event. General Electric had performed a review of the event and submitted a report to TVA containing conclusions, determination of probable cause, and recommendations. This report was an attachment to GE letter G-ER-6-333, dated August 21, 1986. The cause of the above event was determined to be the rigid instrument piping system which would not

permit adequate movement upon thermal growth of the reactor vessel. To correct the problem, the rigid instrument piping system in Unit 2 has been replaced with a modified flexible system in accordance with ECN E-2-P7131, which completed all modifications to Reactor Water Level Instrumentation necessary to support Unit 2 restart. Therefore, this item is closed for Unit 2 only.

During the review of documentation associated with the installation of the flexible instrument piping system, additional concerns were observed. The new installation is comprised of 1 inch diameter stainless steel piping, with spring-can hangers utilized to provide the desired flexibility. As Unit 2 is currently in a cold shutdown condition, post-modification testing could verify only the "cold" settings on the spring-cans. The "hot" settings can only be verified when the reactor reaches or nears operating temperature. The review of the documentation provided, and conversations with licensee personnel indicate that, due to the cold position of the spring-cans and the actual length of the springs, the expected hot position setting will be adequate when the reactor achieves its expected thermal growth. However, at present, there are no plans to verify physically these hot settings and engineering calculations have not been provided to support the above conclusion.

A second concern involves preventive maintenance. The previously referenced GE report contains a statement which reads as follows: "The drywell instrument line piping in all 3 units appeared to have been designed for flexibility, but was a rigid system. The rigid system appears to have evolved from the years of operation and absence of plant maintenance of small diameter piping." When the inspector questioned licensee personnel as to what actions had been or would be taken to address GE's assessment, no evidence was provided to indicate that the question of maintenance and/or periodic inspections had been considered nor were any such actions anticipated in the future. It should be noted that ISI programs cannot be relied upon for these lines, as ASME Section XI specifically excludes ISI requirements for 1" diameter and under piping systems. Resolution of these concerns involving preventive maintenance and verificaton of hot spring-can hanger settings will be tracked as inspector followup item IFI 259, 260, 296/89-35-01. This is not a Unit 2 restart item but programmatically it should be addressed during the power ascention testing, prior to full power operation.

h. (CLOSED) IFI 260/87-42-03, Core Spray ECN L2003 Closeout.

This item concerned work performed under ECN L2003 which involved licensee action in response to Generic Letter 84-11, Inspection of BWR Stainless Steel Piping. This ECN was to replace 304 series stainless steel piping in system 75, Core Spray, with carbon steel to reduce the potential for intergranular stress corrosion cracking. The IFI noted that several 3/4 inch drain and test lines were omitted from the ECN and would thus remain stainless steel.

The NRC inspector reviewed Safety Evaluation L2003, Revision 3 dated October 28, 1988, which addressed IGSCC concerns for small bore piping in the Core Spray system. Generic Letter 88-01, NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping, states that this GL superseded the requirements of GL 84-11. It further states that the requirements of GL 88-01 do not apply to piping less than four inches nominal diameter regardless of code classification.

Based on the revision of the SE and guidance of GL 88-01, the licensee does not intend to replace the 3/4 inch drain and test lines of system 75. This is consistent with the NRC's staff interpretation of the IGSCC requirements. This item is closed.

i. (OPEN) IFI 259, 260, 296/88-04-04, Single Failure Criteria Involving Emergency Core Cooling Systems Identified as Part of the Restart Test Program.

This inspector followup item involved a licensee identified condition where single failure design criteria was not applied to the design of subsystem 280, Battery Boards, and subsystem 231, and the 480 Volt AC SDBD. The finding was documented on CAQR BFP 880067, Revision 1. These two issues represent significant examples of design program deficiencies. These and other examples of single failure violations are discussed elsewhere in the report along with the corrective actions to improve the design control program. This IFI involves only equipment modifications associated with CAQR BFP 880067.

CAQR BFP 88067 discusses two DC power systems: the 250V DC battery supply that the TS refer to as station unit batteries, and the 250V shutdown board batteries. The unit batteries supply certain safety related loads such as HPCI-valves and containment isolation valves. The SD battery board supply provides control power for the load shed logic of the 4160V AC shutdown boards.

The CAQR stated the loss of the 250V DC Unit Battery Board 1 would result in the loss of the DC control power for the load shed logic features to the 480V AC SDBDs

SDBD 1A (Unit 1, Div 1)
SDBD 1B (Unit 1, Div 2)

and in the loss of core spray logic for Unit 1, Division 2. This violated single failure because two divisions were affected by one failure.

Loss of this load shed feature during certain accident conditions will result in overloading the associated diesel generator. The 480V AC shutdown boards are supplied from the associated 4160V AC SDBDs.

The resolution of this problem was the reassignment of the 250V DC control logic power supplies of the 480V AC shutdown boards 1A, 2A, 1B and 2B from the unit batteries to the 4160V AC SDBDs, 250V DC SDBD batteries (SB-A, SB-B, SB-C, and SB-D).

Now with the failure of a single DC control power source such as SB-D, only the associated 4160V AC board, its diesel generator, and the 480V AC boards fed from them would be affected, thereby preserving single failure design criteria.

TVA implemented the resolution of this problem by performing work associated with ECN's E-2-P7117 and E-2-P7124 which reassigned the source of normal 480V SDBD control power feeds.

The inspector reviewed the documents provided by the TVA licensing section for closure of this issue. The implementing work instructions appeared satisfactory. The NRC inspector observed that the safety evaluation associated with this modification specified no TS changes would be necessary. The 10 CFR 50.59 evaluation did state that the bases for auxiliary electrical equipment, section 3.9 of the TS would need to be revised. A review by the inspector of the complete TS revealed that this plant modification caused a confusing relationship between two different limiting conditions.

TS 3.9.B.4 entitled "Operation with Inoperable Equipment", requires initiation of and orderly shutdown within 24 hours if a 4160V AC shutdown board and any 480V AC emergency power shutdown board are inoperable at the same time. This condition, loss of a 4160 and a 480V AC board, will occur anytime a 250V DC shutdown board is made inoperable. This is the result of the recent modification that aligns the 250V DC shutdown boards to supply control power to both of the associated AC shutdown boards.

TS 3.9.B.8 addresses the loss of a 250V shutdown battery or its associated battery board, and permits continued reactor operation for up to five days if a 250V shutdown battery or battery board is inoperable.

Since the loss of any 250V DC shutdown board now always results in the loss of control power, to both a 4160V AC and a 480V AC board, TS 3.9.B.4, and 3.9.B.8 are conflicting with one another.

The NRC inspector considers this IFI as open until the conflict between TS 3.9.B.8 and 3.9.B.4V is resolved. This is a Unit 2 restart issue.

j. (OPEN) IFI 259, 260, 296/88-05-06, Potential Single Failure - Two Sets of Two Dampers From Two Trains are Actuated Thru One Relay.

This issue represents a design deficiency that pertains to the secondary containment isolation system. Four ventilation dampers located in the equipment bay (Drawing 47E865 Damper# 1-FCO-64-65A, B, C and D), between the inner and outer equipment doors, were found to close on an initiation signal from either of two trains of the SBGT. The two signals actuate the same single relay which closes all four dampers. Failure of this relay would prevent proper operation of all four dampers.

This item was identified as a result of the Restart Test Program (RTP-65-SBGT) and documented by TVA on CAQR BFT 880186. The resolution of the CAQR was to rework the system design and return the system to one that meets design requirements pertaining to single failure. The work will not be performed until the next refueling cycle. Therefore the system hardware is to remain and certain compensatory actions will be implemented to provide assurance that the system functions will be met. This action is effectively a use-as-is disposition to a non-conforming condition, at least for the interim period before the hardware is reworked. The use-as-is disposition is a design output; a modification of design criteria and therefore requires a 10 CFR 50.59 review. The NRC inspector could find no 50.59 review associated with the CAQR. TVA was notified of the inspectors concern on August 8, 1989.

This IFI will remain open until a documented 10 CFR 50.59 evaluation is provided or performed.

Review of other CAQR's despositioned as rework or use-as-is should be performed to ensure that a trend of such oversights did not exist in previous CAQR resolutions. The current CAQR program is documented in SDSP 3.13, Revision 2, "Corrective Actions," and in concert with NEP 6.6, Revision 1, "10 CFR 50.59 Evaluations," provide reasonable assurance that recently dispositioned CAQR's are not vulnerable to problems that existed in the earlier program.

k. (CLOSED) IFI 259, 260, 296/88-21-04, Deficiencies Identified During Retest of LOP/LOCA C.

This item involves the failure of the RHR Pump 2A breaker to close automatically during performance of LOP/LOCA test C in July 1988, as required by procedure 2-BFN-RTP-L/L-C, Revision 2. Upon discovery, an unsuccessful attempt was made to manually close the breaker from the control room. The following actions were then taken:

- Voltage measurements taken in the breaker control compartment revealed that the positive side of the 250V DC close signal was present up to the breaker position switch.

- The breaker was removed from its compartment for troubleshooting. All components associated with the charging, closing, and tripping circuits were checked, revealing nothing that would indicate a lack of continuity in the positive 250V DC closing circuit.

- Secondary disconnect pin MG2 was observed to stick slightly and showed signs of arcing. The pin was cleaned with contact cleaner and exercised several times to eliminate the sticking.

- Inspection of the breaker compartment revealed the guide rail in the bottom center of the compartment to be bent. The rail was straightened and proper alignment verified.

- The breaker was racked back into its compartment, tested several times, and observed to be functioning correctly.

Subsequent licensee evaluation of the above actions and findings determined the probable cause of the failure of the breaker to be slight misalignment (due to the bent guide rail) in conjunction with the sticking MG2 pin. These actions and evaluations were documented on TE -05, Maintenance Request 908524, and CAQR-BFP880518.

As previously reported in NRC IR 88-24, at the request of NRC, the PM records for this breaker were reviewed. It was revealed that PM had not been performed on this breaker in three years. The failure to perform the required PM on this breaker and on safety-related 4.16 KV breakers, in general, resulted in the issuance of Violation 259, 260, 296/88-24-08. Therefore, as this specific breaker has been adequately addressed, and as programmatic corrective actions regarding PM on 4.16 KV breakers is being tracked by the above violation, this item is closed.

l. (CLOSED) IFI 259, 260, 296/88-21-05, Vaulting of Completed and Approved Test Results.

This concern was originally identified by the inspector during the continuous observation of the RTP. The inspector reviewed SDSP 2.5, Quality Assurance Records, and noted that completed QA records may be stored up to 30 days in fire resistant metal file cabinets. The NRC inspector observed that RTP QA records were being maintained in fire resistant metal file cabinets with restricted access. This item is closed.

During the above review, an additional concern was observed. SDSP 2.5, Revision 9, page nine contains a note specifying requirements for temporary storage of QA records. There is one set of requirements for records being temporarily stored for 60 days or less, and a second set of requirements for records beings temporarily stored for more than 60 days. TVA Topical Report TVA-TR75-1A, Revision 10, Table 17D-2, Sheet 7 makes no allowance for temporarily storing QA records for periods in excess of 60 days. This concern is identified as IFI 259, 260, 296/89-35-02, pending resolution of this potential conflict between the FSAR commitment and its implementing procedure and should be addressed prior to restart of Unit 2.

m. (CLOSED) IFI 260/88-28-03, Spurious RPS Trips Associated With RPS Alternate Power Supply and Circuit Protectors.

NRC IR 88-28 identified the concern that many spurious RPS trips were being actuated by the RPS circuit protectors. This issue was also discussed in IR 89-11.

The RPS for each of the three BFN units is divided into two trip systems (A and B) and both systems are provided with a MG set. The MG sets are powered from the 480 V auxiliary power system. Each unit also has a single maintenance power supply (alternative transformer) that can be aligned to either RPS distribution system A or B, but not at the same time. Circuit protectors are provided between the output of the MG sets and the breakers for the associated RPS distribution bus, and between the output of the regulating transformer and the connection switches to the RPS distribution. The circuit protectors will open to disconnect the RPS distribution on under voltage, over voltage, or under frequency conditions.

After the Unit 3 RPS loss of power event on March 7, 1989 (see IR 89-11, paragraphs 4 and 8) the licensee's system engineers issued a report on RPS circuit protector performance and made recommendations for minimizing or eliminating circuit protector problems.

The NRC inspector reviewed the licensee's report and discussed the status of the recommended actions with the cognizant system engineer. The NRC inspector verified that the licensee performed the following actions:

- Operating and PM instructions were revised to minimize the time that RPS buses are left on the alternate supply transformer,
- Testing of the MG set voltage regulator potentiometers was performed and PM instructions for their inspection and cleaning were enhanced.
- Modifications to improve circuit protector reliability were initiated.

The cognizant system engineer stated that these actions were taken to provide better performance of the current circuit protector design and to minimize the chances for spurious trips. System engineering also requested that DNE reevaluate the basis for the circuit protector relaying setpoints, reevaluate the current use of time delays in the circuit protectors, and perform a safety evaluation to determine if the Unit 1 and Unit 3 circuit protectors could be bypassed until unit refueling. These actions were not complete at the end of this reporting period. The system engineer stated that these evaluations could be used to enhance the current systems performance but are not necessary for it to provide its intended function.

The NRC inspector concluded that the licensee had adequately addressed the concerns raised by this item, had taken actions to preclude recurrence of spurious trips, and were actively pursuing actions to enhance the current design. This item is closed. In addition, the licensee reported the Unit 3 RPS loss of power in LERs 296/89-03 and 259/88-18. These LERs are closed in paragraph 2 of this report.

n. (CLOSED) IFI 259, 260, 296/88-32-02, Diesel Generator Overspeed RTP Test.

This item was originally identified by the licensee and involved a review of the system 82, DGs, RTP test procedure results. This review indicated that the section of the procedure involving the overspeed test of the 3A DG was either not performed or was inadequately documented. A decision to perform the overspeed test on 3A DG was made and section 5.7, Load Run, Load Acceptance Test, and Miscellaneous Tests, Data Sheet 7.21 of RTP-082 was performed on October 27, 1988. The NRC inspector reviewed data sheet 7.21 and noted that the overspeed test of the 3A DG was successful. This item is closed.

o. (CLOSED) URI 259, 260, 296/87-02-05, Ambiguous Surveillance Intervals.

This item involved the fact that certain surveillance tests required by plant technical specifications to be performed at a frequency of once per operating cycle had performance dates as much as four years old. This applied to some systems that were required to be operable at all times.

This condition was identified because of the duration of this shutdown period for the Brown's Ferry Units started May 1985 and the wording of the plant's custom TS. Standardized TS generally specify 18 months as a refueling and operating cycle. This permits the application of period extensions as well as a bounded period of time.

TVA evaluated its surveillance testing program in its response to this URI with the stated intent of identifying tests scheduled on a once per operating cycle frequency that would more prudently be on an 18 month frequency. The investigation, completed in June 1987 indicated 44 surveillances should have their frequency upgraded. The tests were primarily on secondary containment systems and control room emergency ventilation. The review also determined that the tests identified for upgrade had been performed in the prior 18 months.

TVA also revised SDSP 12.7, "Systems Pre-operability Checklists," to require review of once-per operating cycle surveillances to evaluate whether the SI needs to be reperformed prior to declaring a system operable.

The inspector reviewed the issue and TVA's corrective actions, and found that the program for scheduling of the once per operating cycle surveillances had been effective as evidenced by the successful reperformance of previously identified SI's as they approached an 18 month period since their last performance. SDSP 12.7 was reviewed with no comment. The NRC inspector also reviewed the plant's book of TS interpretations for the purpose of determining if an official TVA position had been documented on the once per operating cycle issue. The inspector found an interpretation discussion on TS wording of surveillance frequencies but the discussion did not include the "once per cycle" issue. This point was brought to the attention of licensing personnel. Licensing responded with assurance that an expansion of the interpretations would be considered to ensure long term and consistent understanding of the frequency issue until appropriate TS changes were approved.

The inspector determined that no violation of NRC requirements occurred as a result of long surveillance periods. However, the actions taken by TVA are expected to remain in place to ensure a high level of confidence will be maintained in systems required to be operable. This confidence will be obtained by the successful performance of regularly scheduled surveillance tests. This item is closed.

p. (CLOSED) URI 259, 260, 296/87-26-02, Adequacy of Sampling Program for Resolution of IEB 79-14, Phase I Deficiencies.

This item involves the question as to whether TVA's proposed sampling program would be adequate to resolve concerns regarding pre-1985 walkdowns of piping supports pertaining to IEB 79-14. TVA proposed to perform walkdowns of 60 supports to determine as-built configurations, and then perform evaluations of these configurations to determine support adequacy. The results of this sampling program were intended to provide an acceptable level of confidence in the Phase I walkdowns performed prior to 1985. Subsequent reviews by NRC staff have determined that the proposed sampling program would not provide adequate assurance as to the accuracy of pre-1985 walkdowns. Therefore, TVA has been directed to perform 100% of the Unit 2 Phase II walkdowns and subsequent engineering evaluations prior to restart. These directions are contained in NRC letters to TVA dated March 25, 1988 and June 19, 1989.

Because it has been determined that TVA's proposed sampling program cannot be utilized in conjunction with the overall IEB 79-14 program, and as future NRC reviews of the results of the 100% walkdown and evaluation will be performed as part of the overall assessment of TVA's 79-14 program, this item is closed.

q. (Closed) URI 259, 260, 296/88-28-05, Failure to Report Loss of Cooling Water to Diesel Generators.

During an overheating event associated with the 3C DG that occurred on September 29, 1988, the licensee operated the DG for surveillance testing with no cooling water available. The north and south EECW headers had been unintentionally isolated at an earlier date from the four Unit 3 DGs due to a valve alignment problem resulting from a known drawing discrepancy. The condition which would have resulted in both divisions of safety-related electrical equipment failing went undetected for three days. A contributing factor to this problem was the absence of either local or remote EECW flow instrumentation. Violation 259, 260, 296/88-28-01 was issued to document the licensee's failure to maintain configuration control.

During the licensee's subsequent evaluation of the event it was determined that no report to the NRC was required per 10 CFR 50.72 or 10 CFR 50.73. The licensee's basis for this conclusion was that no Technical Specifications were violated and that the hydrostatic testing was not normally performed during power operations. However, 10 CFR 50.73 (a)(2)(v) requires that the licensee report any event or condition that alone could have prevented the fulfilment of a safety function needed to mitigate the consequences of an accident. In this case all four DGs would have overheated when called on to perform their function. During discussions held with the licensee the inspector was informed that an LER would be submitted. The licensee subsequently submitted LER 296/88-007 dated December 30, 1988, to cover the event. This LER was classified by the licensee as an voluntary informational report and submitted approximately three months after the event occurred.

The NRC inspector determined that a violation did occur, i.e. the licensee failed to report the event to the NRC within 30 days as required by 10 CFR 50.73.

Violation 259, 260, 296/89-27-03 was subsequently issued by the NRC for three separate examples of the licensee's failure to submit a required LER in a timely matter. Since this failure constitutes another similar failure, it will be included as a fourth example of Violation 89-27-03. The unresolved item is therefore closed and any corrective actions will be followed as part of the violation. This item is closed.

r. (CLOSED) URI 259, 260, 296/88-33-03, Unauthorized, Undocumented and Inadequate Maintenance Activity.

This item involved a field observation by a NRC inspector of fasteners that displayed improper thread engagement on a bolted flange connection . The work on the connection was later found to have been performed by an improper expansion of scope of an existing maintenance request that was being worked in the same area. When notified by the NRC, the licensee initiated CAQR BFP 881020 which documented the conditions. TVA corrected the fasteners by replacing them with bolts of the proper length. Work was performed by maintenance request A803275.

TVA investigated the occurrence to consider if this type of improper work control was a trend. This effort involved use of the CAQR trending program. The results indicated no trend existed because no similar occurrences were documented within the previous 6 months. Corrective actions involved training of maintenance craft personnel in the importance of work scope and control. The apparent poor communication between the field craft and the work supervisor (who was contacted prior to performing the additional work) was stressed as the root cause. The corrective actions are considered appropriate.

The NRC inspector, after review of CAQR Trending and previous NRC documented findings, considers no violation of NRC requirements occurred and that this occurrence was isolated and not the result of a programmatic deficiency. This item is closed.

s. (CLOSED) URI 260/88-35-02, Missed SBGT Surveillances.

This item involved a SBGT train becoming inoperable when the surveillance period required by the technical specifications had expired. The item was considered unresolved at the time the inspection report was issued because the operability status of redundant trains of the SBGT was not readily available. Subsequent investigations by TVA revealed that a second train of SBGT was inoperable because its onsite power supply was not available (DG B was inoperable).

With two of the three SBGT trains inoperable, the requirements for secondary containment stated in TS 3.7.B.1.b were not met. Secondary containment was required for ongoing fuel pool activities (Ref. TS 3.7.C.4).

This event was reported as an operation prohibited by TS in LER 260/88-19, Revision 1. The information provided in the LER is the basis for resolving URI 260/88-35-02 and concluding that a violation of NRC requirement did occur. This URI is considered closed and upgraded to a licensee identified violation.

The violation, NCV 259, 260, 296/89-35-03, is considered a licensee-identified violation and is not being cited because criteria specified in section V.G.1 of the NRC enforcement policy were satisfied.

TVA investigation of the events indicated that when SI 4.2.A-12 was performed on November 29, 1988, the SI steps associated with train "C" were marked N/A as allowed. The SOS acknowledged, by his signature on the SI that acceptance criteria were incomplete. The SI scheduling section identified the need for train "C" testing and placed the item on a schedule requiring performance prior to fuel load. The impact on the operability of the SBGT was not effectively passed on to the operating shifts. TVA attributed this to a lack of a formalized process for tracking the inoperability status of equipment for which SI's had been partially completed.

The corrective actions taken by TVA was to develop a formalized process for documenting and tracking the operability status of TS required equipment.

This program process consisted of revising the procedure describing the conduct of testing (PMI-17.1) to provide directions for documenting the conditions that prevent the completion of a SI and ensuring that the SOS and the STA are notified. The STA would then be required to enter the condition in the LCO tracking system.

PMI-15.10, "Tracking of Limiting Conditions for Operations," was developed and implemented to assist the SOS with the tracking of inoperable components.

The NRC inspector reviewed the corrective actions and determined that the concepts and program were adequate and implemented in a timely manner and should prevent a recurrence of the event.

t. (CLOSED) URI 259, 260, 296/89-08-03, Loss of Approximately 200,000 Gallons From the Condensate Water Storage Tank.

This item involved an event which occurred on February 10, 1989, during which the level of the Unit 1 condensate storage tank dropped from an indication of 26.7 feet at 4:00 a.m. to an indication of 10.1 feet at 7:30 a.m.. This corresponded to a loss of approximately 200,000 gallons of water in 3.5 hours. Although the release of water was not monitored, the activity was within 10 CFR 20 limits. This loss of water was not acted upon until the evening of the same day, indicating a lack of awareness of the status of plant systems by the control room personnel. The failure to adequately maintain an awareness of plant systems status is considered a violation of plant procedure PMI-12.12, Conduct of Operations, section 4.3, "Shift Personnel Conduct," which states:

> The operator at the controls and the immediate supervisor must be continuously alert to plant conditions and activities affecting plant operations, including conditions external to the plant such as grid stability, meteorological conditions, and change in support equipment status; operational occurrences should be anticipated; alarms and off-normal conditions should be promptly responded to; and problems affecting reactor operations should be corrected in a timely fashion.

This failure to follow procedures is identified as Violation 259, 260, 296/89-35-04, Failure to Respond in a Timely Manner to Off-Normal Conditions. An NRC NOV will be issued rather than classifying it as "licensee identified" with no NOV since the violation was self disclosing. The unresolved item URI 259, 260, 296/89-08-03 is closed and upgraded to a violation.

u. (CLOSED) URI 259, 260, 296/89-11-02, Single Failure Criteria.

This item involved the TVA design control process. The NRC

inspectors concern arose over the fact that several single failure criteria violations in areas of mechanical, electrical, civil and I&C design had been identified by Restart Testing and other means in recent months. TVA investigated each issue and several CAQRs resulted. Corrective actions or suitable compensatory measures were proposed for each case.

CAQR BFT 880186 provided a discussion of the root cause and the recurrence control program for identification of single failure deficiencies.

In the root cause analysis of the CAQR, several conditions adverse to quality had been identified concerning failure to meet single failure criteria as identified below:

- SCRBFNNEB8604 - Loss of 250VDC Battery BD 2 Causes Loss of Three U-3 Core Spray Pumps.
- SCRBFNNEB8607 - Loss of 250VDC Battery Concurrent with Recirc Discharge Break Results in Only One Core Spray Loop.
- SCRBFNEEG8654 - Loss of Paralleled Diesels 1/2 D & 3D Causes Loss of SBGT Trains B & C (Common to all units).
- SCRBFNNEB8612 - Loss of ECCS Division I Inverter Power to ATU Causes Loss of Automatic Vacuum Relief of Torus.
- NCRBFNMEB8403 - Loss of Offsite Power Concurrent with an Accident Signal Causes Loss of Control Bay HVAC.
- CAQR BFP880067 - Loss of 250VDC Battery BD 1 Causes Loss of 480VAC Load-Shed Signals to Both U-1 480V Shut Down Boards and Loss of Core Spray Loop II.
- CAQR BFP880154D01, D02, D03 - Certain Battery Failures Concurrent With a LOP/LOCA Results in Inadequate Combination of RHR Pumps.

These deficiencies were identified as a failure to have an appropriate design control to ensure compliance with the single failure criteria specified in the FSAR.

The TVA Design Control Program was a basic issue that lead to the development of the design baseline verification program. The applicability of single failure criteria was addressed by the DBVP and is discussed for each system in the system's Design Basis Document. This increased visibility of such a fundamental consideration currently provides an acceptable action to prevent recurrence.

The items identified thus far were early designs performed before enhanced programs were developed.

TVA provided the NRC a copy of its Single Failure Design Criteria Document (BFN-50-729) used in the analysis of the design of fluid and mechanical systems and subsequent design changes. This document was developed to promote a general understanding of single failure requirements and was issued in June 1987 as part of the DBVP. The inspector considers the actions taken by the licensee to be appropriate. A violation for failure to implement adequate design controls required by 10 CFR 50 Appendix B, Criterion III is not being cited because the criteria specified in section V.G.1 of the NRC Enforcement Policy were satisfied.

This NCV (NCV 259, 260, 296/89-35-05) requires no response. The inspectors will continue to monitor TVA activities in this area. URI 259, 260, 296/89-11-02 is closed.

v. (CLOSED) VIO 260/84-34-03, Core Spray Relief Valves.

This violation involved the failure to test the Core Spray System suction relief valves per ASME code subsection IWV-3510 requirements. The violation was previously discussed in NRC inspection report 89-19 where corrective actions by the licensee regarding ASME Testing were found satisfactory and documented therein. The issue was not closed at that time because a CAQR was open that identified concerns related to improper relief valve sizing.

TVA Site Licensing provided the NRC with documentation that the relief valve sizing issue had been satisfactorily resolved. The resolution involved calculations of actual system flow requirements to ensure existing relief capacity was adequate. The system vendor, GE, concurred with the licensee. Calculations, GE correspondence, and other documentation were reviewed in the CAQR 88-07-69 closure package and were found complete. This violation is closed.

w. (Open) VIO 296/85-13-01 Failure to Shut Down With Two Reactor Protection System Water Level Instruments Inoperable.

Following a NRC inspection conducted to determine the circumstances surrounding the inoperability of two Unit 3 RPS RPV water level instruments (LIS-3-203 A, B) during a reactor startup on February 13, 1985, it was determined that the responsible licensee personnel failed to commence a reactor shutdown in accordance with required actions stated in Technical Specification 3.1 (Table 3.1.A). T.S. 3.1 states there shall be two operable or tripped systems for each trip function. If the minimum number of operable channels per trip system cannot be met for both trip systems, the licensee shall initiate insertion of operable control rods and complete insertion of all operable rods within four hours. Even though there existed sufficient redundant information which should have alerted operators that two required water level switches were inoperable, the licensee did not shut down and continued power escalation. The reactor was

eventually shutdown on March 9, 1985 to conduct further investigations required by TVA management following review of the circumstances associated with the event. This resulted in the NRC issuing a severity level II violation with a civil penalty (EA-85-13).

The inspector reviewed the licensee's responses to the violation and civil penalty dated August 21, 1985 and August 30, 1985. In that response the licensee has stated their inability to determine explicitly the root cause for the observed level mismatch which led to the event. It is suggested that the level mismatch was most likely caused by a loss of water in the "A" instrument reference leg. Two possible causes are as follows:

Reference level leakage via identified transgranular stress corrosion cracking in the line that existed adjacent to the X-28 drywell penetration. This cracking was found during post-event investigation and repairs have been made to the affected line.

Potential for the presence of air bubbles in the "A" reference leg. This possibility is supported by licensee engineering calculations and may have been enhanced by the above listed cause. The presence of high points in horizontal runs and the number and character of restrictions gives credibility to this possibility. Additionally, various activities affecting vessel water level and negative pressures maintained on the vessel for several days prior to the startup could have contributed to the introduction of air in the horizontal runs of the reference leg lines.

The licensee's analysis of operator actions pointed out the need for additional training in diagnosing water level instrumentation problems at off-rated conditions. A similar condition had existed during an earlier startup that occurred on November 20, 1984 when operators also failed to diagnose correctly and fully appreciate the condition.

The inspector reviewed the licensee's corrective actions for this violation. Specifically, the following corrective actions were noted:

Both of the above potential hardware problems should have been corrected by completion of ECN E-2-P7131. This ECN is related to NUREG-0737, Item II.F.2, and relocated the vertical runs of reference legs outside of the drywell so as to minimize the potential of erroneous level indications resulting from the post-accident environment in the drywell. This modification was completed by TVA during the second half of 1988 and is covered in more detail in NRC IR 88-32. The inspector noted that during the ongoing work QC inspection was included to verify instrument line slope criteria were satisfied and that the presence of high points in any horizontal runs should not be a problem.

The inspector reviewed documentation including memoranda and training department lesson plans associated with classroom and simulator training. Lesson plans included training on the types and design of the available level instruments, and their expected response during normal, off-normal and accident conditions. This completed licensee training was provided to operators, management, and STAs and was intended to enable them to more rapidly diagnose water level indication problems. Additional training as part of the planned start-up training program will cover the same subjects and is scheduled to be completed by December 22, 1989.

The inspector examined copies of training records; a manager of licensing memorandum dated March 21, 1989 (R08 890321 878); and BFN Site Quality Surveillance Monitoring Report dated April 14, 1989, (R22 890414 973). The monitoring report was conducted by the licensee to independently verify closure of the commitment to provide the training.

The inspector reviewed the licensee's Unit 2 Operational Readiness Review Interim Report dated June 9, 1989. This review performed by licensee corporate management was the first of a two phase assessment of the readiness of Browns Ferry for restart. Section VI.d covered reactor vessel water level and included various identified deficiencies some of which are as follows:

Interviews with operators and STAs indicated an inconsistent understanding of what was entailed in the reference leg modification.

Documented training to operators, STAs, and management to enable them to more rapidly diagnose level indication problems did not adequately cover the new water level reference leg installation.

Post modification testing did not verify proper function of the modified system.

The acceptance criteria band specified in the post modification test equated to approximately 27 inches of water. Significant indication errors such as trapped air would not be cause for rejection.

Site licensee management has not yet responded to this review. Due to the significance of the above licensee identified deficiencies and the apparent inconsistencies between these comments and the documentation provided by site licensee personnel, this item will remain open pending further review.

x. (CLOSED) VIO 296/86-06-06, RHR/RHRSW/Diesel Generator Inoperability.

This violation resulted from a personnel error of failure to recognize the inoperability of redundant safety systems. One system

was inoperable due to an ongoing surveillance test and the second system was made inoperable when its associated diesel generator was removed from service for scheduled maintenance. The combined affect of these out of service systems was a reduction of RHRSW systems to less than that required by the TS for the then current plant conditions.

TVA identified this violation and reported it in LER 296/86-04. The NRC considered this occurrence as unnecessary if corrective action from a previous violation (84-26-02) had been properly implemented. The NRC therefore issued a NOV.

TVA responded to the NOV in a letter to the NRC dated May 1, 1986 and detailed the corrective actions for the violation.

The inspector reviewed the TVA compliance section documentation of the followup and closeout of the plant's activities for this violation. The package was thorough and complete. All corrective action commitments were found to have been completed. Corrective actions included clarification of TS 3.0.5 as it applies to cold shutdown conditions and development of shift turnover checklists. This violation is considered closed.

y. (CLOSED) VIO 259, 260, 296/87-14-02, CREV Train B Inoperable.

This violation involved the CREV system and the fact that the system was determined to be inoperable because air flow rates were inadvertently set below the minimum allowed by the TSs. This violation was discussed previously in inspection report 89-19. In that report the licensee's corrective actions were reviewed and several aspects of the violation were closed. The following items were not closed at that time:

1) The results of special test ST 8726 designed to analyze systems flows in various damper line-ups indicated that the CREV system flowrates could exceed the TS maximum allowable with certain damper alignments.

2) The CREV system did not meet the design content of the FSAR because significant unfiltered inleakage of outside air into the control room habitability zone bypassed the system.

3) The acceptance criteria of the TS Surveillance Instructions could be met satisfactorily even though the system would not perform its intended function because of unmonitored inleakage.

4) The related issue of the effect of toxic chemical releases from accidents on transportation routes near the site did not appear to meet requirements of R.G. 1.78.

These complex issues addressing the ability of the CREV system to maintain the control room habitability during accident conditions has been under consideration by the NRC and TVA for a long period of time.

Recent actions on the remaining items are as follows:

1) After a review of special test ST 8726, flowrates were determined by the NRC inspector to be within TS limits. This determination was reached after discussion with ventilation system engineers and analysis of test results summary. This item is considered closed.

2) The fact that the CREV system does not meet its intended function is the specific topic of an amendment request submitted by TVA to the NRC in a letter dated February 14, 1989. The change request number 265T discusses in detail the deficiencies of the system design and operation. Approval of the change request or other action will be required before Unit 2 startup to resolve the CREV system operability issue.

3) The fact that a TS surveillance instruction assumes that system integrity had once been established is an acceptable practice. The SI then verifies that major components of that system continue to function as designed and other administrative programs will preserve system structural configurations. These programs will ensure that system performance will not degrade unknowingly. The NRC inspector has determined after review of the CREVs system surveillance program that it meets TS requirements as well as industry standards and is considered satisfactory for this item to be closed.

4) The issue of toxic gas releases near the site was discussed in a letter from TVA to the NRC dated June 27, 1989. In summary, that letter stated that TVA concluded that the plant meets Reg Guide 1.78 as it applies to Browns Ferry Toxic Gas Analysis. The discussion within the letter directly and clearly addresses the NRC concerns. Resolution of this issue now rests with the NRC. Since the scope of this issue exceeds the scope of the original violation and is clearly documented in the June 27, 1989 letter, the NRC inspector considers the violation regarding CREVs testing techniques as closed. This is not an acceptance of the control room toxic gas analysis or habitability issue.

This violation is closed.

z. (CLOSED) VIO 259, 260, 296/88-18-02, Failure to Initiate a CAQR for the Overloaded 1/2 D DG.

This item involved a personnel error leading to an overload of the Units 1/2 D DG which occurred during a conduct of a special test ST 88-09 in June, 1988. The craftsman was taking a reading to verify a parameter being monitored on a recorder. The actual overload condition lasted for approximately 30 seconds. However, as a result of the event, no CAQR was initiated. The inspector reviewed SDSP 3.7, "Correction Action," and SDSP 3.13, "Corrective

Actions" and noted that both procedures outline activities required when CAQRs are initiated, reviewed, and closed out. SDSP 3.7 states the following in subsection 6.1.1:

> Confirmed degradation, damage, failure, malfunction, or loss of plant structures, systems, and components that could adversely affect the performance of a safety-related function (i.e., nonconformance). This would include but not be limited to material failure, abnormal or unexpected wear, manufacturer defects, failure to function as intended, and repetitive failures.

SDSP 3.13 states the following in subsection 6.2.1.F:

> Items which have been subjected to conditions for which they have not been designed, unless done intentionally by an approved and properly authorized procedure such as overpressure, overvoltage, overheating, overstressing, or environmental conditions hazardous to their function.

This appears to be an inconsistency in that SDSP 3.7 indicates Confirmed Damage, whereas SDSP 3.13 indicates Condition for Which They Have Not Been Designed. Both SDSPs are in effect as of the end of this reporting period.

The NRC inspector further noted that SDSP 3.7, section 2.1 states the following: CAQR's initiated on or after August 16, 1988, shall be processed in accordance with SDSP 3.13, Corrective Actions. The NRC inspector also noted that SDSP 3.13, initial revision was dated August 5, 1988 and that any future occurrences of a system or component being subjected to conditions for which they have not been designed will be initiated and processed in accordance with SDSP 3.13. This item is closed.

4. Exit Interview (30703)

The inspection scope and findings were summarized on August 16, 1989 with those persons indicated in paragraph 1 above. The inspectors described the areas inspected and discussed in detail the inspection findings listed below. The licensee did not identify as proprietary any of the material provided to or reviewed by the inspectors during this inspection. Dissenting comments were not received from the licensee.

Item	Description
259, 260, 296/89-35-01	IFI, Flexibility of Reactor Water Level Sensing Lines, paragraph 3.g.
259, 260, 296/89-35-02	IFI, Storage of QA Records, paragraph 3.1.
259, 260, 296/89-35-03	NCV, Missed SI Results in a TS Violation, paragraph 3.s.

259, 260, 296/89-35-04	Violation. Failure to Respond in a Timely Manner to Off-Normal Conditions, paragraph 3.t.
259, 260, 296/89-35-05	NCV, Design Control of Single Failure, paragraph 3.u.

5. Acronyms

ASME	American Society of Mechanical Engineers
ASOS	Assistant Shift Operations Supervisor
ATU	Analog Trip Units
BF	Browns Ferry
BFNP	Browns Ferry Nuclear Power Plant
BWR	Boiling Water Reactor
CAQR	Condition Adverse to Quality Report
CHRRM	Containment High Range Radiation Monitors
CREVS	Control Room Emergency Ventilation System
DBVP	Design Baseline and Verification Program
DCN	Design Change Notice
DCR	Design Change Request
DG	Diesel Generator
EA	Engineering Assurance
ECCS	Emergency Core Cooling Systems
ECN	Engineering Change Notice
EECW	Emergency Equipment Cooling Water
ESF	Engineered Safety Feature
FSAR	Final Safety Analysis Report
GE	General Electric
GL	Generic Letter
HPCI	High Pressure Coolant Inspection
HVAC	Heating, Ventilation, & Air Conditioning
IE	Impact Evaluation
IEB	Inspection and Enforcement Bulletin
IEEE	Institute of Electrical and Electronics Engineers
IER	Inspection and Enforcement Report
IFI	Inspector Followup Item
IGSCC	Intergranular Stress Corrosion Cracking
IM	Instrument Maintenance
IR	Inspection Report
IRM	Intermediate Range Monitor
ISI	In Service Inspection
KV	Kilovolt
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LIV	Licensee Identified Violation
LOP/LOCA	Loss of Power/Loss of Coolant Accident
MG	Motor Generator
MMI	Mechanical Maintenance Instruction
NCV	Non-cited Violation
NOV	Notice of Violation
NPRD	Nuclear Plant Reliability Data System
NRC	Nuclear Regulatory Commission
PM	Preventive Maintenance

PMI	Plant Manager Instruction
QA	Quality Assurance
QC	Quality Control
RCW	Raw Cooling Water
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RTP	Restart Test Program
SBGT	Standby Gas Treatment System
SDBD	Shutdown Board
SDSP	Site Director Standard Practice
SE	Safety Evaluation
SI	Surveillance Instruction
SIL	Service Information Letter
SOS	Shift Operations Supervisor
SPOC	System Pre-Operation Checklist
SQN	Sequoyah Nuclear Plant
SRM	Source Range Monitor
STA	Shift Technical Advisor
TE	Test Exception
TMI	Three Mile Island
TS	Technical Specifications
TVA	Tennessee Valley Authority
URI	Unresolved Item
VIO	Violation