UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

Report Nos.: 50-259/95-43, 50-260/95-43, and 50-296/95-43

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 12, 1995

Inspector: Leonard D. Wert, Jr., Senior Resident Inspector — 8/31/95 Date Signed

J. Munday, Resident Inspector
R. Musser, Resident Inspector
M. Morgan, Resident Inspector
M. Janus, Resident Inspector

Approved by: Mark S. Lesser, Chief, Reactor Projects, Section 4A, Division of Reactor Projects — 8/31/95 Date Signed

SUMMARY

Scope:

This routine resident inspection involved inspection on-site in the areas of operations, plant support, maintenance activities, and surveillance testing, Unit 3 recovery actions, and review of open items, including a Three Mile Island item. Several hours of backshift coverage were routinely worked during most work weeks. Deep backshift inspections were conducted on July 23 and 30, and August 5, 6, and 8, 1995.

Enclosure

Results:

No violations, noncited violations, unresolved items, or inspector followup items were identified.

Operations:

A review of fire brigade manning and self contained breathing apparatus controls was conducted. Brown Ferry's fire brigade is manned by dedicated personnel from a separate Fire Operations group. Control room manning is not adversely affected during fire fighting activities. Responses to fire alarms and medical emergencies have been noted to be highly professional. Although the annual self contained breathing apparatus qualifications of some onshift operations personnel had been allowed to lapse, specific regulatory requirements and commitments were met. (paragraphs 2.3 and 2.4)

Plant Support:

During the report period, the licensee completed significant portions of a major upgrade of site security equipment. New security facilities including the West personnel access building were placed into operation. The shift to the revised procedures and upgraded equipment was accomplished smoothly. (paragraph 3.0)

Unit Three Recovery:

System pre-operability walkdowns monitored during this period and the previous report period were conducted in a thorough manner and with a questioning attitude. Equipment deficiencies were identified and corrective actions initiated promptly. Inspection of the Unit 3 Q-list development indicated that the project is proceeding on schedule with acceptable overall quality. (paragraphs 5.1, 5.2 and 5.3)

REPORT DETAILS

1.0 Persons Contacted

Licensee Employees:

J. Brazell, Site Security Manager
*R. Coleman, Radiological Controls Manager
*J. Corey, Chemistry and Radiological Controls Manager
*T. Cornelius, Emergency Preparedness Manager
C. Crane, Assistant Plant Manager
*J. Johnson, Site Quality Manager
R. Jones, Unit 3 Startup Manager
*G. Little, Operations Superintendent
*R. Machon, Site Vice President, Browns Ferry
J. Maddox, Maintenance and Modification Manager
R. Moll, Plant Operations Manager
G. Pierce, Technical Support Manager
*E. Preston, Plant Manager
*S. Rudge, Site Support Manager
*J. Sabados, Chemistry Manager
*P. Salas, Licensing Manager
T. Shriver, Nuclear Assurance and Licensing Manager
D. Stinson, Recovery Manager
*S. Wetzel, Acting Compliance Licensing Manager
*H. Williams, Engineering and Materials 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 Personnel:

*L. Wert, Senior Resident Inspector
J. Munday, Resident Inspector
*R. Musser, Resident Inspector
M. Morgan, Resident Inspector
M. Janus, Resident Inspector (Brunswick)

*Attended exit interview

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

2.0 Plant Operations (71707, 92901, 40500)

2.1 Operations Status and Observations

Unit 2 operated at power during this inspection period. On July 17, power was reduced slightly to reduce the generator bus duct temperature. Perturbations on the system grid, high system demand, high outside air temperatures, and abnormal system voltage resulted in bus duct temperature increasing to the alarm setpoint. Paragraph 2.2 of this report contains additional details.

Activities within the control rooms were monitored routinely. Inspections were conducted on day and night shifts, during weekdays and on weekends. Observations included control room manning, access control, operator professionalism and attentiveness, and adherence to procedures. The inspectors noted that operators were cognizant of plant conditions and were generally attentive in their duties. Instrument readings, recorder traces, annunciator alarms, operability of nuclear instrumentation and reactor protection system channels, availability of power sources, and operability of the Safety Parameter Display System were monitored. Unit 1/2 control room observations also included emergency core cooling system lineups, primary and secondary containment integrity, reactor mode switch position, scram discharge volume valve positions, and rod movement controls. Observations in the Unit 3 control room focused on major activities in progress and operable systems.

Daily discussions were held with plant management and various members of the plant operating staff. One of the inspectors attended the daily Plan of the Day meetings. Plant tours were taken throughout the reporting period on a routine basis. Observations included valve position and system alignment, snubber and hanger conditions, containment isolation alignments, instrument readings, housekeeping, power supply and breaker alignments, radiation and contaminated area controls, tag controls on equipment, work activities in progress, and radiological protection controls. Informal discussions were held with plant personnel during these tours.

The tours in the Unit 1 areas focused on maintenance activities and systems required to be operable to ensure that appropriate attention is provided to the shutdown unit. The RHR heat exchanger 1B outlet valve disc was replaced with one of a different design in an effort to increase the amount of flow the valve could pass. Paragraph 4.1 provides additional details.

2.2 Main Generator Bus Duct Temperature Increase

On July 17, 1995, the main generator bus duct temperature increased to the annunciator setpoint of 176 degrees F. The unit operators determined that it was due to increased reactive loading caused by disturbances on the grid and reduced the reactive loading accordingly. Local temperature readings were obtained of the bus duct which indicated that the peak temperature reached was approximately 207 degrees F. Operations lowered reactor power to reduce the temperature more quickly. When the temperature returned to normal, power was slowly raised back to one hundred percent. The inspector was in the control room at the time the annunciator was received and noted that the annunciator response procedure contained no guidance for adjusting the reactive loading.

The procedure was written assuming the increased temperature was caused by a problem associated with the bus duct cooling system. The actions required by the procedure regarding investigation of the cooling system performance were completed. No problems with the duct cooling system were evident. The inspector discussed the actions of the operators with Operations management. Management stated that procedures would be revised to provide additional guidance for this type of event.

2.3 Fire Brigade

During this report period the inspectors reviewed the composition of the fire brigade and the licensee's ability to combat a fire. The fire brigade consists of five members one of whom is the brigade leader. In addition, a Senior Reactor Operator serves as the site Incident Commander. The fire brigade members are not shift operations minimum staffing personnel. Normal duties of the fire brigade include surveillance testing of fire related equipment such as detectors, pumps, and hoses. The Fire Operations Manager stated that these routine duties do not interfere with the fire brigade's ability to respond to a fire. The surveillance procedures are written such that the test can be immediately secured and abandoned. The site Incident Commander normally serves as an extra SRO on shift responsible for duties outside of the control room. Similarly, his duties do not impede his ability to respond rapidly. In all cases, no other duties would take precedence over responding to a fire alarm. In the event that a fire occurs and additional fire fighting assistance is needed, the site Incident Commander can request offsite assistance from local fire departments. The inspector reviewed the brigade members assigned during normal working hours as well as during backshift hours and noted the minimum complement was satisfied.

Site personnel are trained during General Employee Training to report all fires to the main control room regardless of the size or presence of flame. This call can be made via the emergency number, 3911, or by radio. The 3911 phone is normally monitored by the Unit One operator. Upon receipt of this call the fire brigade is dispatched. All calls made via the emergency number are also monitored in the fire house. Generally, the fire brigade is aware of the condition of the fire prior to receiving notification from the main control room. In addition, the brigade members monitor hand held radios in the event the fire is reported via this method.

Fires causing actuation of a smoke or flame detector annunciate in the Unit 1 control room. The fire alarm panel has a distinct sound which serves to distinguish it from other alarms in the control room and results in rapid response by the operators. The resident inspectors have observed control room fire alarm responses on several occasions and noted that operator actions were professional.

The fire brigade members receive recurrent training on a regular basis. One aspect of this training includes fighting fires associated with energized electrical switchgear. Site procedures allow the use of water to combat electrical fires based on the spray pattern and the voltage of the electrical source. However, it is desirable to deenergize the source if at all possible.

In addition, each shift is tested in a drill at least once every three months. Each shift performs at least one drill per year during the backshift hours.

2.4 Use of Self Contained Breathing Apparatus

The inspector reviewed the licensee's ability to operate the plant in the event that the operators had to don SCBAs. The licensee's Final Safety Analysis Report section 10.12.5.3, states that the worst case accident resulting in toxic gas entering the main control room would be from a barge accident on the Tennessee River involving chemicals. However, the chemicals listed as being of concern would be detected by smell by the operators in sufficient time to don protective equipment without experiencing any physical impairment. The inspector reviewed correspondence between the NRC and the licensee and determined that while the FSAR statement is factual, training to support the operators ability to identify the toxic gas is not a regulatory requirement. Correspondence from the NRC to the licensee dated November 20, 1990, stated that due to the low probability of a toxic gas incident, this training was not necessary to bring the licensee into compliance with regulatory requirements or to ensure adequate public health and safety.

The inspector determined during a review of Operations qualifications that a substantial number of licensed operators were unqualified for use of a SCBA. Most of the operators had received training on SCBA use in the past but actions to keep the qualifications current were not effective. The fact that some operators were unqualified had been previously identified and reviewed by the inspectors (IR 94-24). That inspection had focused on Appendix R requirements and commitments. At that time, Operations management stated that it was their intent to maintain operators performing emergency duties qualified for SCBA use. This matter was again discussed with Operations management during this report period. The licensee stated that all operators would be provided SCBA training and with the exception of medical restrictions, would become SCBA qualified.

Although TS require normal control room manning of at least two senior reactor operators, four reactor operators, and one shift technical advisor, the licensee stated that only one senior reactor operator and two reactor operators would be required (in the CR) to safely operate the plant if a toxic gas problem occurred. The inspector verified that each control room contained enough SCBAs for those required to stay and operate the plant. Five SCBAs were located in each main control room and approximately seventy-five spare bottles of air were available to replenish their supply. The other SCBAs required by Appendix R procedures were verified during the inspection in IR 94-24.

During this review, the inspector noted that the procedure for combatting the release of a hazardous chemical, 0-AOI-100-6, Release Of Hazardous Chemicals Or Gases, only contained steps for controlling ventilation to the affected areas and for identifying and isolating the source. There were no provisions for protecting other personnel on site. This was discussed with Operations and site emergency planning representatives who stated that if a hazardous chemical was released on site which affected personnel, the site would be evacuated in accordance with the normal site evacuation procedures.

The inspectors concluded that the licensee was meeting regulatory requirements and commitments regarding the number and location of SCBAs. Operations management indicated that more attention will be focused on the maintenance of SCBA qualifications by operations personnel.

No violations or deviations were identified.

3.0 Plant Support (71750, 40500)

During this report period Phase I of the site security upgrade was completed. Changes made included a new protected area access building, central alarm station, secondary alarm station, and protected area fence and alarm systems for portions of the site. In addition, the use of hand geometry palm print readers for plant access was introduced. On August 4, at 6:00 p.m., the licensee opened and manned the new East gate and West gate security access control points. On the morning of August 5, one of the inspectors observed operations within the West access point and noted that ongoing activities (including standardization and calibration of portals) were being conducted in accordance with procedural guidance. The new hand geometry systems were in operation and an adequate number of security staff was available to assist employees entering the plant.

The inspectors toured the protected area and noted that the perimeter fence was intact and not compromised by erosion or disrepair. The fence fabric was verified to be intact and secured. The inspectors observed personnel and packages entering the protected area and verified they were searched either by special purpose detectors or physical patdown. Until the site security upgrade is complete, the licensee will utilize the old central and secondary alarm stations as well as the new stations. The inspectors toured all four alarm stations paying particular attention to the coordination of each alarm station with the other three. The inspectors noted that alarms were responded to appropriately with no confusion from the security force manning the alarm stations.

No violations or deviations were identified.

4.0 Maintenance Activities and Surveillance Testing (62703, 92902, 61726, 92901, 37551, 92903)

4.1 Maintenance and Surveillance Observations

Maintenance activities were observed and/or reviewed during the reporting period to verify that work was performed by qualified personnel and that approved procedures in use adequately described work that was not within the skill of the trade. Activities, procedures, and work requests were examined to verify proper authorization to begin work, provisions for fire hazards, cleanliness, exposure control, proper return of equipment to service, and that limiting conditions for operation were met.

Surveillance tests were reviewed by the inspectors to verify procedural and performance adequacy. Testing was witnessed to ensure that approved procedures were used, test equipment was calibrated, prerequisites were met, test results were acceptable, and system restoration was completed.

The following maintenance activities were reviewed and witnessed in whole or in part:

Work Order 95-11697-00 RHR Heat Exchanger 1B Outlet Valve

During this report period the disc for the RHR heat exchanger 1B outlet valve was replaced with one of a different design. It was previously determined that the valve was unable to pass the required amount of service water flow as documented in IR 95-38. Engineering determined that the previous valve disc which was fluted could be replaced with one which was not fluted and would therefore pass a greater amount of flow. The inspector reviewed the work package and TDCN 25771. Work on the valve was observed as well as acceptance testing and no discrepancies were noted. The post maintenance test was completed satisfactorily with the valve being able to pass the required flow.

Work Order 95-13124-00 Control Rod Drive Scram Pilot Air Header Pressure Control Valves

On August 10, the inspector witnessed maintenance personnel calibrate pressure switch 2-PA-85-38B to a new setpoint. This switch provides control room annunciation on an abnormal scram pilot air header pressure. Additionally, the setpoint for the scram pilot air header pressure control valves, 2-PIC-85-66 and 2-PIC-85-67, was changed. The reason for the changes was to provide anticipatory control room annunciation for an abnormal scram pilot air header pressure. Prior to the changes, the annunciator would alarm if the standby controller could not restore air pressure in the event the primary controller failed. This modification adjusted the annunciator setpoint to coincide with the pressure at which the standby pressure control valve started controlling. In addition, the setpoint for the two pressure control valves was changed slightly. The inspector noted that the coordination during this activity was good. A maintenance foreman and Senior Reactor Operator were observed at the job site which was considered appropriate for a high risk activity such as this.

Paragraph 5.1 discusses the inspection of Unit 3 ECCS instrumentation testing.

4.2 Core Spray Testable Check Valve

The inspector observed maintenance activities on one of the Unit 3 CS testable check valves (3-CKV-75-0054). In May, it had been determined that testing of the valve was not fully opening the valve disk. A TOE (Technical Operability Evaluation) had been written to support operability of the Unit 2 valves since they were tested the same way. IRs 95-26 and 95-31 contain additional details of the issue. The 75-0054 valve cover had been disassembled and workers were to cycle the valve manually with a torque wrench to determine the torque required to open the valve. Initial attempts were unsuccessful due to workers not being well informed on the issues associated with the valve and the torque values expected. The following day, the inspector observed cycling of the valve. The torque required to break the valve off the closed seat and the running torque to continue stroking it to the open position were less than the force that CS flow would be expected to generate. The inspector cycled the disk by hand and noted that once it was moved off the seat with just a small effort, the disk moved very freely throughout the rest of travel to full open. The inspector noted that some of the torque measured when manually cycling the disk was due to packing forces associated with the actuator shaft. The inspector concluded that these observations strengthened the conclusions of the TOE on the Unit 2 CS valves. At the close of this report period, the licensee was still performing maintenance on the valve. Additional review of the maintenance will be conducted as followup to Violation 95-31-01: Core Spray Testable Check Valve Testing Not In Accordance With Requirements.

4.3 RHR Chemical Decontamination Tap Hydrostatic Test

The inspector reviewed problem evaluation reports BFPER950875 and BFPER950882 which identified that an advanced authorized change document was used to support a hydrostatic test but was not incorporated into the applicable design output documents.

Design Change Notice W20667A added chemical decontamination taps to the Unit 2 RHR shutdown cooling suction line and the A and B loop LPCI injection lines in February, 1993. At that time, minimum temperature required for the hydrostatic test, 70 degrees fahrenheit, could not be maintained. The required temperature for the test was lowered to 40 degrees fahrenheit by an advanced authorization and was subsequently performed. The documented justification for this advanced authorization was to expedite construction testing since the higher temperature could not be maintained. This advanced authorization was never incorporated into the appropriate FDCN.

The inspectors concern was with the lack of engineering justification for the advanced authorized document. The licensee was asked if the lower temperature was too low for the piping which was tested. Engineering determined that because the A and B loop LPCI injection lines were made of stainless steel the lower test temperature was acceptable. However, because the shutdown cooling suction line contained carbon steel material, it was not acceptable to hydrostatically test the pipe at less than 70 degrees fahrenheit. A review of the actual test performed indicated that while a temperature of 40 degrees fahrenheit was allowed the actual test temperature was only 69 degrees fahrenheit. A preliminary evaluation from Engineering indicated that this slight reduction in temperature would have no detrimental effect on the piping.

The inspector and the licensee reviewed the licensee's procedures for determining hydrostatic testing requirements and identified several weaknesses. Final resolution of these issues and development of corrective action will be included in the two problem evaluation reports.

No violations or deviations were identified.

5.0 Unit 3 Restart Activities (37828, 61726, 62703, 37550, 92903) (Unit 3)

5.1 Unit 3 Status and General Observations

The inspectors reviewed and observed the licensee's activities involved with the Unit 3 restart. This included reviews of procedures, post-job activities, and completed field work; observation of pre-job field work, in-progress field work, and QA/QC activities; attendance at restart progress meetings, and management meetings; and periodic discussions with both TVA and contractor personnel, skilled craftsmen, supervisors, and managers.

On August 2, an inspector observed removal of a blank flange and gasket assembly and re-installation of a piping flange spacer and gasket assembly in the 3D RHRSW piping. The blank flange had been installed to completely isolate the non-operating Unit 3 portion of the RHRSW system from Unit 2. Complete isolation of this portion of RHRSW was required during preparations for the upcoming Unit 3 restart. The work was accomplished per work order 94-06433-00 and TVA general engineering specification G-53, "ASME Section II & Non-ASME Section III Bolting Material". The WO package, the WO, and related WO clearance tagging and cleanliness controls were adequate and work on the system was satisfactory performed. The inspector noted that some basic job personnel industrial safety work practices such as wearing of safety glasses and hard hats was not performed at all times. This item was brought to the attention of the work foreman by the inspector and was corrected.

While observing the above work, the inspector noted that a small amount of Containment Atmospheric Dilution (CAD) system piping, located directly over the piping flange, had been cut out and removed to allow access to the RHRSW blanks. Since the inspector was aware that the CAD system had already undergone a SPOC I walkdown, the inspector confirmed that the piping removal was being monitored. The inspector determined that the activities were directed by FDCN F37661A and DCN W17627 and that the removal of the piping was known by both the system engineer and test engineer.

The inspector observed the performance of the following surveillances and calibrations of reactor protection system instrumentation: 3-SI-4.2.A-6(D), Primary Containment Isolation System Main Steam Line Low Pressure Instrument Channel B2 Calibration; and 3-SI-4.2.B-7(A), Core and Containment Cooling Systems Reactor Low Pressure Instrumentation Channel A Calibration. The inspector observed the conduct of these activities in the auxiliary instrument room. The lead technician performed the required calibrations in the instrument cabinets located here and directed activities of the other technicians located in the control room and the field. The inspector noted good communication between the various individuals and the control room, and that the lead technician maintained good control of the activities in the field. The inspector reviewed the procedures in use in the field and verified that they were the current revisions, properly annotated, and were being properly followed. The inspector observed that the steps which required independent verification were properly verified by an individual not associated with the specific job. The inspector observed that any discrepancies or problems were discussed, noted and resolved prior to the continuance of the work activity. The inspector questioned the technicians about the various aspects of their work, and found them to be quite knowledgeable of the work and the systems involved.

5.2 Control Rod Drive System Phase 1 SPOC Walkdown

On August 9 and 10, 1995, the inspector accompanied licensee personnel during the performance of the Phase 1 SSP-12.55, Unit 3 System Pre-Operability Checklist (SPOC) Walkdown of the Control Rod Drive (CRD) System. This SPOC process is a systematic process intended to identify and evaluate any issues which potentially affect the performance of the system. The system walkdown was led and directed by the system start-up engineer and performed by a group of personnel including an AUO, individuals from electrical and mechanical maintenance, quality assurance and the Technical Support Manager. Given the extent of the system, the walkdown was scheduled to be completed over a period of three days. The inspector attended the initial pre-job briefing and observed the completion of the walkdown in all areas of the plant with the exception of the main control room and the rod gallery under the reactor vessel.

The entire CRD system was walked down by the licensee, starting with the two CRD pumps and their associated suction and discharge piping, strainers and instrument and control stations. Next came the CRD hydraulic flow control equipment, drive and exhaust filters, stabilizing valves, and pressure and flow instrumentation. This was followed by a walkdown of the CRD scram solenoid pilot air header supply piping and instrumentation. Following the completion of this activity, the next area reviewed was the most extensive and time consuming, 185 individual CRD hydraulic control units. This included the scram discharge volume vent and drain valves and level instrumentation, accumulator level and pressure instrumentation, alarm panels, and insert and withdraw riser piping. The walkdown of both the east and west HCU racks and associated piping required the better part of 1.5 days. The remaining work involved the rod position indication system, scram discharge volume level instrumentation located in the auxiliary instrument room, all associated controls and instrumentation in the main control room, and the rod drives, Rod Position Indication cabling and CRD supports (shoot out steel) located under the reactor vessel.

During the observance of the initial day's activities, the inspector observed that while the team members were identifying a substantial number of deficiencies, they appeared to be moving rather quickly. As a check, the inspector returned to portions of the system previously reviewed, performed an independent review, and could not find any deficiencies not already identified by the team. The inspector performed this independent review of selected portions of the system several times during the course of the walkdown and noted that the team had conducted a thorough walkdown of the system.

The list of the items identified during the walkdown reflected the thoroughness of the inspection effort. Items ranged from missing condulet cover gaskets, improperly mounted flex conduit connector to nicked or cracked wire insulation, and valve packing leaks. The inspector noted that the team members paid particular attention to bolt/thread engagement, labeling, piping supports, hangers and restraints, and general housekeeping and material conditions of the system. During the walkdown, several members including the inspector questioned the lack of adequate piping supports/hanger/restraints on the insert and withdraw riser piping above the HCU racks. Based on these questions, the licensee determined that their pipe support engineering staff would be required to review this area prior to completion of the walkdown.

The inspector concluded that the system walkdown process was effective in identifying problems and issues which need to be corrected prior to system operability. The portions of the CRD system walkdown which the inspector observed were performed in accordance with the guidelines in Appendix D of SSP 12.55, and were thorough and complete, identifying a wide range of problems. Finally, the inspector noted that the required management level team member provided useful guidance and direction during the walkdown and provided the appropriate level of attention to this process.

5.3 Unit 3 Q-List Program

On January 9, 1991, TVA submitted the plans for return to service of BFN Units 1 and 3. Enclosure 2 stated that a Q-list would be developed and fully implemented prior to the restart of Units 1 and 3. The Unit 3 Q-list program was to be implemented in accordance with the Unit 2 precedent.

The inspector reviewed the following documentation to gain an understanding of the regulatory requirements, commitments, and past problems associated with the BFN Q-list:

- NUREG-1232, Volume 3, Supplement 2 Safety Evaluation Report on Browns Ferry Nuclear Performance Plan (Section 3.15).
- Section III.14.1 of the Browns Ferry Nuclear Performance Plan.
- IR 89-16 (Inspection of the Unit 2 Q-list program)
- Generic Letter 82-28, licensee's response, and SER of June 1, 1989
- TVA Nuclear Quality Assurance Plan, Revision 5
- 10 CFR 50, Appendix B

The inspector reviewed the following procedures associated with the Q-list program:

- SSP 3.3: Q-List/Critical Structures, Systems, and Component (CSSC) List Use and Control
- BFEP-PI-87-52: Development of the Q-List
- SSP 3.2: Augmented QA Program
- SSP 9.51: Equipment Management System

NRC requirements specify that all safety related structures, systems, and components be identified. At BFN, the Q-list is intended to perform this function. The Unit 3 Q-list is being loaded into the Equipment Management System (EMS) data base. It will include "as constructed" information on Unit 1,2,3, and common unit equipment required for safe shutdown of Unit 3. Equipment is evaluated against the Safe Shutdown analysis and applicable design criteria to determine if the equipment is safety related, quality related, or non quality related. Additional details of the specific quality or safety functions are also entered into the database. The Unit 3 Q-list upgrade project is scheduled for completion in September 1995.

Actual loading of data is accomplished through EMS loading DCNs, one of which has been initiated for each of the 37 systems which have significant safety related equipment. The DCNs are part of the Master Equipment List program and include addition of Unique Equipment Identification (UNIDs) to drawings and equipment labels.

The inspector met with personnel responsible for the Unit 3 Q-list development. The process for Q-list development and application of the EMS database were discussed. A copy of DCN T34380A (RHR system EMS loading) was provided to the inspector. The inspector was also briefed on EMS database access (read only).

The inspector selected approximately 40 Unit 3 and common components during routine tours of the facility for verification that the proper data was loaded into EMS and the system could be utilized by workers. Most of the selections were from the RHR, RHRSW and EDG systems. Additionally, the inspector focused on support components related to internal flood protection such as watertight doors and sump pumps since the Plant Safety Analysis states that flooding is an important contributor to risk. In all cases, the EMS information reflected the appropriate safety classification and safety function notes. Flood protection components were noted by "Q44-Special Event-Protection from external event". In a few cases, the UNIDs from the equipment labels did not match the EMS database. By using the "browse" feature, the inspector was often able to track down the UNID and verify the entry. The inspector noted that the Unit 3 equipment was often labeled better than Unit 2 or common equipment. For example, the labels for the EDG building outside watertight doors did not match the UNIDs used in EMS. At the close of the inspection report, the licensee was reviewing these observations. The inspector will meet with personnel responsible for implementation of labeling program for additional review.

The inspector also verified that some equipment designated as "design in progress" did in fact have open DCNs. The DCN process requires that the "as designed" Q-list information be validated and entered when modification work is completed.

During the Unit 2 restart effort, the CSSC list had to be relied upon often since the Q-List had not been updated. Basically, the CSSC list is referenced when equipment is not specifically addressed in the Q-list. The Unit 3 effort is incorporating most of the CSSC list into the "system notes" and thus use of the CSSC list will be less frequent. SSP 3.3 still contains requirements to reference the CSSC list if necessary.

The inspector also noted that comparison to Unit 2 data and resolution of any differences is required as part of the upgrade process. During observation of work activities, the inspectors have not identified any deficiencies involving improper safety classification of equipment.

Based on this review, the inspector concluded that the Unit 3 Q-list development is proceeding well and in accordance with regulatory requirements and commitments.

No violations or deviations were identified.

6.0 Review of Open Items (92700) (92901) (92902) (92903) (92904) (TI 2515/65)

The open items listed below were reviewed to determine if the information provided met NRC requirements. The determinations included the verification of compliance with TS and regulatory requirements, and addressed the adequacy of the event description, the corrective actions 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.

6.1 (CLOSED) VIO 260/94-24-03, Improper Maintenance Actions Involving Clearance Boundary.

This violation occurred as a result of a maintenance worker not recognizing that a drain valve attached to the drywell delta pressure heat exchanger was part of a clearance boundary. Subsequently the heat exchanger and the drain valve with the clearance tag still attached was removed from the system. Upon discovery of the event the licensee stopped the work, reviewed the requirements and expectations with the involved individuals, and then allowed work to resume. Further corrective actions taken include the issuance of a memorandum from the Maintenance and Modifications Manager to the craft personnel stating his expectations concerning work involving a clearance. A summary of this event has also been added to the annual training of maintenance personnel. This item is therefore closed.

6.2 (CLOSED) LER 260/93-10, Technical Specifications Surveillance Test Was Not Performed Due To Inadequate Procedures And Drawings.

This event occurred when a local leak rate test was not performed following maintenance on the internals of a test connection valve. The valve is part of the primary containment boundary and therefore required a leak test in accordance with TS 4.7.A.2.g. The licensee stated that the cause of the event was inadequate documentation regarding primary containment boundaries. The procedures used to plan the activity did not list this valve as a primary containment boundary valve. Corrective action to prevent recurrence included training the planners to understand the criteria for a primary containment boundary, revising the controlling procedure, SSP-8.4, Containment Leak Rate Programs, to include all valves which are part of the primary containment boundary, and issuing color coded drawings depicting the testing boundaries. The inspector reviewed the drawings and procedure revisions and noted that the primary containment boundaries on the drawings were clearly depicted but the procedure was cumbersome to use. The engineering staff stated that further revision of SSP-8.4 is planned to make it easier to use. Based on these corrective actions this item is closed.

6.3 (CLOSED) IFI 296/84-41-04, Relocation of HPCI EGM Control Boxes.

This item was open to track the relocation of the HPCI EGM control box due to the harsh environment of high temperatures and high humidity the box could be exposed to while mounted on the HPCI skid. The licensee initiated DCN W17834A which, in part, relocated the Unit 3 HPCI EGM control box to the south wall of the HPCI room, elevation 519'. The inspector verified that the control box was relocated as stated above. The functionality of the controls will be monitored during restart and operability testing. Based on this review, the item is closed for Unit 3. Previously the item was closed for Unit 2 (IR 88-21) and remains open for Unit 1.

6.4 (OPEN) TMI Action Item II.K.3.18, Automatic Depressurization System (ADS) Logic Modification.

The inspector reviewed item II.K.3.18, to determine present status of licensee efforts. DCN W17531A was reviewed, with the following upgrades noted:

Original timer was replaced with a seismically-qualified timer.

The timer setpoint of 120 seconds was reduced to 95 seconds in order to comply with BFN Unit 3 TS Section 3.2.B.

An inhibit switch was added to both ADS initiation logic trains.

A time delay relay was added to each ADS logic train. This bypasses the high drywell pressure signal after a Level 1 reactor vessel water level signal is present and the timer times out.

A new annunciator, "ADS LOGIC BUS A OR B INHIBITED", was installed. This alarm provides an operator alert that ADS has been inhibited by use of the bus "A" or bus "B" keylock switches.

The inspector performed walkdowns of various portions of the modification, reviewed the associated drawings, and confirmed proper installation of the ADS instrumentation. No problems were identified. TI 2515/65 items 3.02.a.(1), 3.02.a.(2) and 3.02.a.(3), have been met. However, per TI 2515/65 guidance, this action item remains open pending further inspections of ADS equipment calibration/operability, functional testing, and procedural upgrades.

6.5 (CLOSED) IE Bulletin 84-02, Failures of General Electric Type HFA Relays In Use In Class 1E Safety Systems.

This bulletin addressed similar failures of GE HFA relays which had been reported in several GE service reports, and requested licensees to inform the NRC of their plans, including schedules for implementing the manufacturer's recommendations in the subject GE reports. The reported relay failures were identified as GE type HFA 51 Series AC relays utilizing nylon or Lexan as the coil spool material. GE determined that the deterioration of the coil wire insulation resulted in shorted turns, causing increased coil temperatures and eventual coil failure.

The licensee previously completed these activities for Unit 2 prior to its restart. These activities are documented in NRC IRs 88-28 and 88-32.

The licensee has identified, inspected, and replaced all suspect HFA relays used on Unit 3 systems. This activity involved the inspection and replacement of approximately 308 relays. During the time of this inspection, 295 work packages had been completed. The remaining relays had been replaced, but the packages remained open pending the completion of PMTs. The inspector discussed these work activities with the cognizant engineers and determined that all activities required by the bulletin had been completed for Unit 3. The inspector notes that any relays located in systems common to Unit 1, 2 and 3 were replaced prior to the restart of Unit 2. The inspector selected various relays at random from the list of those requiring replacement and verified that the activities were completed. Based on this review of the licensee's completed activities, the inspector considers this item closed for Unit 3 and acceptable for restart.

7.0 Exit Interview (30703)

The inspection scope and findings were summarized on August 17, 1995, with those persons indicated in paragraph 1 above. The inspectors described the areas inspected and discussed in detail the inspection findings listed below. Although proprietary material was reviewed during the inspection, proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

Item Number	Status	Description and Reference
260/94-24-03	Closed	VIO-Improper Maintenance Actions Involving Clearance Boundary (paragraph 6.1)
260/93-10	Closed	LER-Technical Specifications Surveillance Test Was Not Performed Due To Inadequate Procedures And Drawings (paragraph 6.2)
296/84-41-04	Closed	IFI-Relocation of HPCI EGM Control Boxes (paragraph 6.3)
TMI II.K.3.18	Open	TMI Action Item II.K.3.18, Automatic Depressurization System (ADS) Logic Modification (paragraph 6.4)
Bulletin 84-02	Closed	BU 84-02, Failures of GE HFA Relays in 1E Safety Systems (paragraph 6.5)

8.0 Acronyms and Initialisms

ADS	Automatic Depressurization System
ASME	American Society of Mechanical Engineers
BFN	Browns Ferry Nuclear Plant
CFR	Code of Federal Regulations
CR	Control Room
CRD	Control Rod Drive
CS	Core Spray
CSSC	Critical Structues, Systems, And Components
DCA	Drawing Change Authorization
DCN	Design Change Notice
DPR	Demonstration Power Reactor
ECCS	Emergency Core Cooling Systems
EDG	Emergency Diesel Generator
EMS	Equipment Management System
F	Fahrenheit
FSAR	Final Safety Analysis Report
GE	General Electric
HCU	Hydraulic Control Unit
HPCI	High Pressure Coolant Injection
IFI	Inspector Followup Item
IR	Inspection Report
LER	Licensee Event Report
LPCI	Low Pressure Coolant Injection
NRC	Nuclear Regulatory Commission
PER	Problem Evaluation Report
QA	Quality Assurance
QC	Quality Control
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water System
SBGT	Standby Gas Treatment
SCBA	Self Contained Breathing Apparatus
SER	Safety Evaluation Report
SI	Surveillance Instruction
SPOC	System Preoperational Checklist
SRO	Senior Reactor Operator
SSP	Site Standard Practices
TI	Temporary Instruction
TMI	Three Mile Island
TOE	Technical Operability Evaluation

TS	Technical Specifications
TVA	Tennessee Valley Authority
UNID	Unique Equipment Identification
VIO	Violation
WO	Work Order