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

Report Nos.: 50-259/84-34, 50-260/84-34, and 50-296/84-34

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

Docket Nos.: 50-259, 50-260 and 50-296 License Nos.: DPR-33, DPR-52, and DPR-68

Facility Name: Browns Ferry Nuclear Plant

Inspection Conducted: August 14 - September 6, 1984

Inspectors: C.W. Hehl for 1/24/85
G. L. Paulk, Senior Resident Inspector Date Signed

C.W. Hehl for 1/24/85
C. A. Patterson, Resident Inspector Date Signed

Approved by: 1/24/85
F. S. Cantrell, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope: This routine, unannounced inspection involved 123 inspector-hours in the areas of the core spray system over pressurization event of August 14, 1984.

Results: VIOLATIONS -
1) 10 CFR 50 Appendix B, Criterion IV, Inadequate quality control on solenoid valve rebuild parts.
2) Technical Specification 3.7.D.1, Inoperable primary containment valve FCV-75-26
3) 10 CFR 50 Appendix B, Criterion XI and 10 CFR 50.55 a(g), Omission of required ASME Code Testing of valves FCV 75-26 and 75-543 B. (two examples)
4) Technical Specification 6.3.A, Failure to follow procedure and inadequate procedure. (two examples)
5) 10 CFR 50, Appendix B, Criterion VII, Inadequate receipt inspection of valve parts.
6) 10 CFR 50, Appendix B, Criterion V, Activities affecting quality not in accordance with drawings or procedures (three examples).
7) TVA Topical Report TVA-TR75-1, Table 17.2-5, Independent verification not accomplished.

REPORT DETAILS

1. Licensee Employees Contacted

   J. A. Coffey, Site Director
   G. T. Jones, Plant Manager
   J. E. Swindell, Superintendent - Operations/Engineering
   J. R. Pittman, Superintendent - Maintenance
   J. H. Rinne, Modifications Manager
   J. D. Carlson, Quality Engineering Supervisor
   D. C. Mims, Engineering Group Supervisor
   R. Hunkapillar, Operations Group Supervisor
   C. G. Wages, Mechanical Maintenance Supervisor
   T. D. Cosby, Electrical Maintenance Supervisor
   R. E. Burns, Instrument Maintenance Supervisor
   A. W. Sorrell, Health Physics Supervisor
   R. E. Jackson, Chief Public Safety
   R. Cole, QA Site Representative
   T. L. Chinn, Technical Services Manager
   T. F. Ziegler, Site Services Manager
   J. R. Clark, Chemical Unit Supervisor
   B. C. Morris, Plant Compliance Supervisor
   A. L. Burnette, Assistant Operations Group Supervisor
   R. R. Smallwood, Assistant Operations Group Supervisor
   T. W. Jordan, Assistant Operations Group Supervisor
   S. R. Maehr, Planning/Scheduling Supervisor
   G. R. Hall, Design Services Manager
   W. C. Thomison, Engineering Section Supervisor
   A. L. Clement, Radwaste Group Controller

   Other licensee employees contacted included licensed reactor operators, senior reactor operators, auxiliary operators, craftsmen, technicians, public safety officers, quality assurance, quality control and engineering personnel.

2. Exit Interview (30703)

   The inspection scope and findings were summarized on September 6, 1984 with the Plant Manager and/or Assistant Plant Managers and other members of his staff.

   The licensee acknowledged the findings and took no exceptions.

3. Summary: Core Spray Event of August 14, 1984

   On August 14, 1984, while conducting a core spray system logic surveillance test the Unit 1 core spray system loop 1 was overpressurized. The unit was operating at 100 percent power. The condition was found after TVA workers responded to a fire alarm in the Unit 1 reactor building and ascertained

that the alarm had been triggered by smoking paint on the loop 1 core spray piping. Further checking revealed that hot primary coolant had backflowed into the loop when the outboard isolation valve (FCV 75-25) was inadvertently opened during the surveillance logic test because the inboard isolation check valve was partially open due to maintenance errors. The check valve had been open since unit startup on December 29, 1983. The core spray discharge pressure instrumentation in the control room pegged offscale high (greater than 500 psig). The 500 psi relief valve in the core spray discharge piping functioned to relieve pressure; however, it is unknown what the actual pressure in the discharge piping was during the event. The outboard isolation valve was open for 12-15 minutes until isolated from the control room. Thirteen workers became contaminated during the event. All workers were satisfactorily decontaminated immediately after the event.

This event is judged to be significant in terms of reactor safety because the open check valve and inadvertently opened outboard isolation valve substantially degraded the high-pressure/low-pressure isolation arrangements provided between the reactor coolant system and the low-pressure core spray system. The open isolation valves thereby significantly increased the likelihood of an interfacing loss-of-coolant accident.

The cause of the event was traced to multiple operator and maintenance personnel errors. Primary cause factors include: Inadequate maintenance and post-maintenance testing of the check valve actuator air solenoid valve, failure to follow procedures by the operator during the logic surveillance test and by the electricians during check valve position indicating light wiring changes, and lack of adequate quality assurance control of solenoid valve rebuild kits and valves.

4. Details of Core Spray Event

a. Core Spray System Description:

Portions of the following material is from the NRC System Manual Boiling Water Reactor; - Two independent loops are provided as a part of the core spray system. Each loop consists of two fifty percent centrifugal pumps driven by electric motors; a spray sparger in the reactor vessel above the core; piping and valves to convey water from the suppression pool to the sparger; and the associated controls and instrumentation.

In the case of an initiation signal (low water level in the reactor vessel or high pressure in the drywell plus low reactor vessel pressure), the core spray system starts and automatically sprays water on the fuel assemblies at a sufficient flow rate to cool the core and limit fuel cladding temperature.

The core spray system provides protection to the core for the large break in the nuclear system when the control rod drive water pumps, the Reactor Core Isolation Cooling System (RCICS), and the High Pressure

Coolant Injection Cooling System (HPCI) are unable to maintain reactor vessel water level.

The protection provided by the core spray system also extends to a small break in which the control rod drive water pumps, RCICS, and HPCI are all unable to maintain the reactor vessel water level and the automatic depressurization system has operated to lower the reactor vessel pressure so the low pressure coolant injection system (LPCI) and the core spray system can provide core cooling.

The core spray pumps and all automatic valves can be operated individually by manual switches in the control room. Operating information is provided in the control room with pressure indicators, flow meters and indicator lights.

The major equipment for one loop is described in the following paragraphs.

When the system is actuated, water is drawn from the suppression pool through a normally open motor-operated valve in the suction line to each pump. Each valve can be closed by a remote-manual switch from the control room to isolate the system from the suppression pool in the case of a leak from the core spray system. This valve, which is normally open, is located in the core spray pump suction line as close to the suppression pool as practical.

A local pressure gauge by each pump indicates the presence of a suction head for the pump. The core spray pumps are located in the reactor building below the water level in the suppression pool to assure positive pump suction pressure. The pumps, piping, controls, and instrumentation of each loop are separated and protected so that any single physical event, or missiles generated by rupture of any pipe in any system within the containment drywell, cannot make both core spray loops inoperable.

A shaft seal drain line is provided from the pump casings, which drains to the radwaste system. Leakage from the drain line is measured during primary containment leakage tests.

A low flow bypass-line is provided from the pump discharge to below the surface of the suppression pool. The bypass flow is required to prevent the pump from overheating when pumping against a closed discharge valve. Two orifices in series limit the bypass flow. A manual valve normally locked open is used to close the bypass line for maintenance.

A relief valve, set for 500 psig, protects the low pressure core spray system upstream of the outboard shutoff valve from reactor pressure. The relief valve discharges to the equipment drain sump and thence to the radwaste system.

A full-flow test line permits circulating water to the suppression pool for testing the system during normal plant operations. A normally closed, motor-operated valve (75-22) in the line is controlled by a remote-manual switch in the control room. Orifices and partial opening of the valve in the test line provide rated core spray flow at a pressure drop equivalent to discharging into the reactor vessel. A restricting orifice is provided to minimize vibration in the pump test lines. A flow indicator in the control room signals that water is or is not flowing to the core spray sparger or test line.

Two motor-operated valves are provided to isolate the Core Spray System from the nuclear system when the core spray pumps are not running. These valves admit core spray water to the reactor when signaled to open. Both valves are installed outside the drywell to facilitate operation and maintenance, and are interlocked to prevent simultaneous opening of both valves unless reactor pressure is less than 500 psi. The valve nearest the containment (75-25) is considered the outboard isolation valve and is normally closed to back up the inside check valve (75-26) which is the inboard isolation valve. The outboard motor operated valve (75-23) is normally open, to limit the equipment needed to operate in an accident condition. By closing valve 75-23, valve 75-25 can be operated for test with the reactor vessel pressurized. A drain line is provided between the two motor operated valves to measure leakage through the inside check valve or the inboard shutoff valve. The drain line has two normally closed valves.

Testable check valve 75-26 is provided in the core spray pipeline just inside the primary containment, to prevent loss of reactor coolant outside containment in case the core spray line breaks.

A normal locked-open manual valve is provided downstream of the inside check valve to shut off the core spray system from the reactor during shutdown conditions for maintenance of the upstream valves. The two core spray system pipes enter the reactor vessel through nozzles 120 degrees apart. Each internal pipe then divides into a semicircular header with a downcomer at each end, which turns and passes through the reactor vessel internal shroud. A semicircular sparger is attached to each of the four downcomers to make two practically complete circles, one above the other. Short elbow nozzles are spaced around the spargers to spray the water radially onto the tops of the fuel assemblies.

Core spray piping upstream of the outboard shutoff valve is designed for the lower pressure and temperature of the core spray pump discharge and is fabricated from carbon steel. The outboard valve and piping downstream are designed for reactor vessel pressure and temperature. Material for the portion of the system piping inside the drywell to the second isolation valve (75-23) is high toughness (A333 Grade 6) carbon steel.

Upon signals of reactor low-water level or drywell high pressure plus low reactor vessel pressure, the automatic controls turn on the core spray pumps and position valves to the spray mode. When reactor pressure decreases, the core spray shutoff valves are signalled to open. Flow to the sparger begins when the pressure differential opens the inside check valve.

Upon receipt of an initiation signal, the test bypass valve is interlocked shut. The core spray pump discharge valves are automatically opened when nuclear system pressure drops to a preselected value; the setting is selected low enough so that the low-pressure portions of the core spray system are not overpressurized, yet high enough to open the valves in time to provide adequate cooling for the fuel. Four pressure switches are used to monitor nuclear system pressure. Two switches must be tripped to initiate opening of the discharge valves. Two of the low-pressure instruments are bourdon tube and the other two are bellows-type pressure detectors. Core spray system pressure between the two pump discharge valves is monitored by a pressure switch to permit detection of leakage from the nuclear system into the core spray system outside the primary containment.

B. Details of Core Spray Event

On August 14, 1984, Surveillance Instruction 4.2.B-39A (Core Spray Six Month Logic Test) was being performed on core spray loop 1 of Unit 1. During the preliminary section of the procedure, the operator is required to open the circuit breaker for the motor-operated outboard isolation valve (FCV 75-25): (step 4.1.h of S.I. 4.2.B-39A). No independent verification was required to be performed to verify completion of this step. The operator failed to adequately perform this procedural step and the circuit breaker remained energized. Review of the procedure also indicated that step 4.1.h. referred to an incorrect 480 v. reactor MOV Board. The procedure required the operator to open the breaker on MOV Bd. 2A when in actuality the breaker is on MOV Bd. 1A. The surveillance procedure was in conflict with the data sheets filled out by the operator. (The data sheets did note the correct Board 1A). As the surveillance steps were completed, a point was reached that initiated the core spray logic circuit to open the isolation valve 75-25. Typically, the valve would not open since its circuit breaker would be opened, but in this case the outboard isolation valve did open. The time sequences are included in the below attached table.

Core Spray Overpressurization Event of August 14, 1984

Sequence of Events

Time	Occurrence
0939	Core spray logic test was in progress (S.I. 4.2.B.39A). An offshift Assistant Shift Engineer (ASE) performing the surveillance, incorrectly performed step 4.1.h, which requires opening the breaker to FCV 75-25. The breaker was not opened, and remained energized. As the test proceeded the ASE initiated a test signal to activate the core spray logic circuits. These circuits opened FCV 75-25 since its breaker had not previously been opened. The ASE proceeded to a 4 KV shutdown board room to continue the test.
0945	Fire Alarm; a firewatch on the 621 foot elevation of Unit 1 reactor building noted a bluish smoke near the core spray piping.
0946	The Unit 1 ASE (on duty) left the Unit 1 control room for the Unit 1 reactor building to assess the situation and assist the fire brigade.
0947	The Unit 2 ASE (ASE, U-2) left the Unit 1 control room and was heading toward the Unit 1 reactor building. (The Unit 2 ASE normally serves as the fire brigade leader.)
0949	Both the ASE, U-1 and the ASE, U-2 entered the Unit 1 reactor building. The ASE, U-1 headed to the first elevation while the ASE, U-2 headed to the higher elevations. The ASE, U-1 discovered that smoke was coming from the northwest corner of the reactor building below elevation 565. He then phoned the Unit 1 operator and asked him to radio the fire brigade leader (ASE, U-2) and let him know about the smoke coming out the core spray room. The fire brigade leader saw smoke coming from the 519 level and directed his crew to start down the stairwell. The ASE, U-2 reported that he saw a great deal of smoke on elevation 593 of the reactor building and that a lot of smoke was rising through the floor at the location where the core spray pipe penetrates the floor. He did not notice any smoke coming from the horizontal run of core spray piping on this elevation. Both ASEs, arriving at the scene, reported that the smoke that they saw was blue in color and that its odor made them think that the fire was electrical in nature. (ASE, U-1) upon arriving at the core spray pumps 'A' and 'C' heard the flow and had felt the heat from the core spray piping. After evaluating the situation he returned to

elevation 565 and called the Unit 1 operator. He told him the situation and told him to close FCV 75-25 valve.

0952 Unit 1 operator closes valve FCV 1-75-25.

The Unit 3 SE reported that when he arrived on elevation 519 of the reactor building that the 'A' CS pump had water or steam spraying out of the pump seal area. When he first arrived at the pump area the water on the floor had not crossed the C-Zone boundaries which had been previously established by Health Physics personnel. The water crossed these boundaries while he was there. He also reported that before he left the area the seal had stopped leaking.

0953 The assistant operations supervisor called the SE, U-1 and informed him that it sounded like water was going through the CS piping and asked him what the pressure was reading on PI 75-20. SE, U-1 then informed him that the CS pressure was greater than 500 psig. Seeing that valve 1-75-25 was closed, he suspected leakage past the 25 valve and the inboard check (75-26). He then informed the Unit 1 operator that he was going to close valve 1-75-23 and he did so.

0954 Because pressure seemed to be maintained in the CS line, the SE instructed the Unit 1 operator to vent the CS system piping by opening valve 1-75-22 which is the test return line to the torus. As soon as the valve position of 75-22 started opening, (both red and green lights on), pressure in the CS system immediately dropped and returned to normal. Valve 1-75-22 was closed immediately following the drop in pressure. The operator reported that during the time that the CS piping was being vented to the torus he noticed that the torus level indication was slightly bouncing; however, there was no change in either torus level or temperature. A review of the strip charts in the control room also indicated that there were no change in any reactor parameters or steam flow from the vessel.

After the event the licensee conducted system inspections, testing and a mechanical and structural integrity evaluation. The safety evaluation for return to service of the system addressed piping, support and component integrity from the core spray pump discharge valves to the motor operated valve (FCV-75-25).

Visual indications of the system piping for heat effects revealed that the pump discharge check valves (75-037 A&C) had seated during the event. Valve 75-25 was closed at the event termination and valve 75-22 was opened to relieve pressure to the torus. The licensee safety evaluation concluded that the core spray loop 1 was acceptable for future operation.

C. Event Followup

Event followup after Unit 1 shutdown on August 21, 1984, revealed that the air-operated testable check valve (75-26) was being held partially open by its air actuator. The air-solenoid actuator (8344 ASCO series) had been improperly reassembled during maintenance activities sometime in the past and an incorrect pilot valve insert had been used. The incorrect insert caused misoperation of the valve such that it operated the reverse direction from expected. No approved maintenance procedures were available for the mechanic to use in reassembling the valve. The licensee has been using rebuild kits from power stores to rebuild the 8344 ASCO series valves. Records indicated that the rebuild kits were ordered and stocked under No QA requirements although the original valves were designated QA Level II by the licensee. Record traceability for valve maintenance was difficult at best in that no exact time could be determined by inspection of records to indicate when the insert was replaced and what exact type of maintenance was conducted. Maintenance requests 326930 of March 1982, 231888 of March 1982, 172251 of November 1983, and 165736 of December 1983, were reviewed. All MRs were inconclusive. The air solenoid was worked in March 1982 and December 1983. No testing or insufficient operability testing was conducted after the December 1983 maintenance activities to prove operability. Also, the licensee determined that some time in the past, probably the same time the insert was incorrectly installed, the electrical position indication circuits for the check valve were changed such that when the 75-26 check valve was open, it indicated shut in the control room and vice versa. Both the magnetic limit switch and the actuator limit switch were found to have reversed wiring. This was not in accordance with Drawing 730E930. No records were available to indicate when this wiring change was made.

Review of Mechanical Maintenance Instruction (MMI) 51 (Maintenance of CSSC Valves) indicated it was inadequate in that no maintenance or post-maintenance checks of the solenoid valves were included. The solenoid valves had routinely been worked by the skill-of-the-craft method with no check valve operability test conducted.

Inspection of power stores records by the inspector revealed that the ASCO 8334 solenoid valves being ordered and used as replacement valves in the plant, specifically in the core spray system, were not in accordance with design drawing PD-420870. The required valve is a WPHTX834472. (Current design from ASCO is a WPHTX8344A72); however, the plant has been using a HTX8344A73 valve. The table below delineates the design differences:

WPHTX 834472	(Req'd by Drawings) (New ASCO Design WPHTX 8344A72)
Prefix WP:	Optional watertight (submersible) solenoid enclosure (Meets NEMA Type 6 enclosure requirements, also meets watertightness test of UL Std 921)

Prefix HT:	Optional Class H high temperature coil (suitable for ambient temp. as high as 176° F).
Prefix X:	Special Construction (has monel solenoid base as opposed to a standard stainless steel)
"8344":	High flow 4-way solenoid valve
"72":	Identifies specific valve by pipe size (3/8" NPT) and which standard solenoid enclosure is installed (General purpose or explosion-proof-watertight, "72" is GP enclosure)
'A':	Recent updated model (Solenoid is larger)

HTX 8344A73	(Obtained from power stores)
Prefix HT:	Same as above
Prefix X:	Same as above
"8344":	Same as above
:A73":	3/8" NPT with explosion-proof-watertight enclosure (Meets NEMA 7C, 7D and 4 requirements)

No safety evaluation was available to indicate that a design change had been made. Discussions with plant personnel indicated the valve mixup was due to poor communications. A receipt inspection conducted on September 11, 1982, for four ASCO solenoid valves was not completed in accordance with Standard Practice 16.4/16.5. The standard practice requires the component model/part number to be verified on the receipt inspection. The licensee placed an order on requisition #82-PK1-332421 for four ASCO WPHTX 8344A73 solenoid valves. These model number valves do not exist. When the valves were received the receipt inspection accepted the valves although the received model number on the valve was HTX 8344A73, different from the valve ordered.

5. Findings:

During the review of this event seven violations were identified as noted below:

(a) Violation of 10 CFR 50 Appendix B, Criterion IV in that there was no quality control of solenoid valve rebuild parts. (259/84-34-01)

(b) Violation of Technical Specification 3.7.D.1 which requires that primary containment valves be operable. Core spray valve FCV-75-26 was held open during periods of power operation. (259/84-34-02)

(c) Violation of 10 CFR 50, Appendix B, Criterion XI and 10 CFR 50.55 a(g) in that the required testing of valves per the ASME Code was missed for two examples. Relief valve 75-543B was not tested and check valve 75-26 was not tested after maintenance. (259/84-34-03)

(d) Violation of Technical Specification 6.3.A in that required procedures were not followed or were inadequate for two examples: (259/84-34-04)

(1) S.I. 4.2.B-39 referenced the incorrect electrical distribution board for the breaker location of motor operated valve 75-25.

(2) S.I. 4.2.B-39 was not followed and the circuit breaker for valve 75-25 was not opened when required in the procedure.

(e) Violation of 10 CFR 50 Appendix B, Criterion VII, in that there was inadequate receipt inspection of solenoid valve parts and part number differences were not resolved. (259/84-34-05)

(f) Violation of 10 CFR 50 Appendix B, Criterion V in that activities affecting quality were not in accordance with plant drawings or procedures for three examples: (259/84-34-06)

(1) MMI 51 contained inadequate post maintenance testing instructions.

(2) Electrical leads were reversed for the magnetic and actuator limit switches.

(3) Four-way solenoid valve part number was not as required by plant drawings.

(g) Violation of TVA Topical Report TVA-TR-75-1, Table 17.2-5 in that no independent verification was required or performed for the position of core spray valve 75-25 during testing. (259/84-34-07)

During the event the following items were noted to function not in accordance with design and/or system description:

(1) The 450 psi pressure annunciator upstream of FCV 75-25 did not alarm. Instrument checks after the event revealed no apparent problem.

(2) The discharge line for the 500 psi relief valve, FCV 75-543A, on the core spray discharge pipe had noticable backflow from the radwaste system termination point to the core spray pump gland seal leakoff drain. This caused contaminated coolant to spray into the core spray corner room via the core spray pump gland seal leakoff area, and contributed to thirteen persons becoming contaminated.

(3) The 500 psi discharge pipe relief valve was not under any test program requirements. The valve was bench tested by the licensee and found to relieve at 400 psi.

(4) During the steps taken to stop the surveillance and return the systems to normal, the plant received an unexpected inadvertent start of the eight diesel generators.

(5) During the fire water system piping pressurization that occurred during the event, significant mud was observed discharged from the high point Nash float valves on the piping.

The above listed items will be identified as an open item (259/84-34-08). The licensee was notified of these items at the exit on September 6, 1984.