UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA ST., N.W., SUITE 3100
ATLANTA, GEORGIA 30303

Report Nos. 50-259/79-45, 50-260/79-45 and 50-296/79-45

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

Facility: Browns Ferry Nuclear Plant

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

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

Inspection at: Browns Ferry site near Decatur, Alabama

Inspectors: H C Dance — 1/4/80
H. C. Dance — Date Signed

D S Price — 1/4/80
D. S. Price — Date Signed

H C Dance /for — 1/4/80
R. F. Sullivan — Date Signed

Approved by: R. C. Lewis — 1/4/80
R. C. Lewis, Acting Branch Chief, RONS Branch — Date Signed

SUMMARY

Inspection on December 10-14, 1979

Areas Inspected

This special, unannounced inspection involved 78 inspector-hours onsite in the areas of Unit 3 containment integrity following the December 6, 1979, startup, plant tours and Plant Operations Review Committee (PORC) meeting.

Results

Of the 3 areas inspected, no apparent items of noncompliance or deviations were identified in 2 areas; 3 apparent items of noncompliance were found in 1 area [Violation-failure to control activities that involve safety as evidenced by loss of primary containment integrity, paragraph 5; Violation-failure to shut down the reactor after not meeting a limiting condition for operation (LCO), paragraph 5; Infraction-failure to make timely notification to the NRC of an event requiring prompt notification, paragraph 5].

DETAILS

1. Persons Contacted

Licensee Employees

***H. L. Abercrombe, Power Plant Superintendent
***J. L. Harness, Assistant Power Plant Superintendent
*G. T. Jones, Outage Director
M. T. Dover, Outage Engineer
L. A. Franklin, Assistant Health Physics Supervisor
J. V. Purvis, Health Physics Technician
M. J. Hazel, Health Physics Technician
J. B. Studdard, Operations Supervisor
**R. Hunkapillar, Assistant Operations Supervisor
R. G. Metke, Results Section Supervisor
B. F. Miller, Shift Engineer (SRO)
E. G. Thornton, Shift Engineer (SRO)
H. L. Harrow, Assistant Shift Engineer (SRO)
J. H. Bratcher, Assistant Shift Engineer (SRO)
J. C. Cox, Unit Operator (RO)
L. E. Johnson, Mechanical Results Mechanic
R. McPherson, Mechanical Results Engineer
W. C. Thomison, Assistant Results Supervisor
J. D. Daniels, Boilermaker Foreman, Outage
J. W. Jackson, Boilermaker, Outage
J. Cummings, Machinist, Outage
J. E. Swindell, Assistant Outage Director
J. R. Pittman, Instrument Engineer
**T. Jordan, Shift Engineer (SRO)
*C. Rozear, Quality Assurance Engineer

NRC Resident Inspector

*J. W. Chase, Resident Inspector (In Training)

*Attended exit interview on December 12, 1979
**Attended exit interview on December 14, 1979
***Attended exit interviews on December 12 and 14, 1979

2. Exit Interview

The inspectors met with those indicated in paragraph 1 on December 12 and 14, 1979 and discussed the items of noncompliance. The licensee stated that in their view, containment integrity had existed until it was demonstrated that the drywell to torus differential pressure could not be maintained at which time the appropriate Technical Specification (TS) action was taken. Regarding the reporting requirement, the licensee stated that they were operating within the action statements of the Surveillance Requirements section of the Technical Specifications and therefore they considered

no requirement for prompt reporting of the item until after their evaluation was completed. Regarding the shutdown requirements which were identified during the December 14 meeting, the licensee stated their interpretation of Technical Specifications was that the surveillance requirements section (4.7.A.2) allowed continued operation for 48 hours following detection of the leak prior to reactor shutdown.

Additionally, the licensee reconfirmed his intentions following the containment hatch leakage that action was being initiated to place identification and seal mechanisms on each containment closure prior to the January 2, 1980 Unit 1 refueling outage; by the end of the refueling outage to issue specific procedures pertaining to hatches; and to continue the investigation of the cause of the equipment hatch bolts being loose.

3. Licensee Action on Previous Inspection Findings

Not inspected.

4. Unresolved Items

Unresolved items are matters about which more information is required to determine whether they are acceptable or may involve noncompliance or deviations. One new unresolved item identified during this inspection is discussed in paragraph 5.

5. Loss of Primary Containment Integrity

a. Discussion

The reactor was taken critical at 0645 hours on December 6, 1979, following a scheduled refueling outage. Rated temperature and pressure were achieved at 1215 hours on December 7 and the reactor mode switch was placed in the RUN position shortly thereafter at 1235 hours.

Nitrogen inerting of the torus and drywell on December 8, 1979 was commenced in accordance with System Operating Instruction No. 76, Containment Inerting System. Drywell oxygen concentration had decreased to less than the four percent required by TS 3.7.A.5.b at about 1100 hours on December 8. The drywell was then rapidly pressurized through the six inch purge line to about 1.35 psig at 1235 hours. The normal makeup nitrogen system was placed in service but the recorder chart indicated makeup flow was not achieved. Containment pressure decreased to atmospheric in about two hours (1430 hours). Plant personnel recognized that a problem existed and proceeded along several work paths which included:

(1) The drywell to torus differential pressure, required by TS 3.7.A.6.a.(1) to be greater than 1.3 psig, could not be maintained. TS 3.7.A.6.b required that this differential pressure be restored within six hours or the reactor placed in cold shutdown within the next 24

hours. The torus to drywell vacuum breakers were suspected of not being seated so each valve was cycled while observing valve position indication. One or more valves leaking would have permitted drywell pressure to leak into the torus.

(2) Check of the nitrogen inventory determined the amount on hand to be marginal for a second drywell pressurization. Additional nitrogen was ordered and arrived onsite at about 0001 hours on December 9.

(3) Check of the nitrogen system revealed the normal (two-inch line) nitrogen makeup valve's solenoid was defective and not permitting nitrogen flow. This valve was repaired and returned to service at about 0230 on December 9.

(4) In the meantime, continuation of reactor startup and surveillance test activities took place, such as placing the turbine on line and performing associated tests, cycling the main steam isolation valves, and nuclear instrumentation checks.

The inspector concluded that at 1430 hours on December 8, information was available for the licensee to recognize that significant leakage from the containment was probable. The drywell pressure of 1.35 psig had decreased to atmospheric pressure in less than two hours. The inspector calculated from the slope of the drywell pressure decay that this amounted to a containment leakage of approximately 8000 CFH. This is significantly above the maximum allowable containment leakage rate of 516 SCFH at 25 psig. The drywell to torus differential pressure is not recorded but should have had a similar pattern as the drywell pressure decay. Had there been internal containment leakage from the drywell to the torus, it would have resulted in an equalized pressure of approximately 0.7 psig for the two volumes. Therefore, with the apparent containment leak rate noted above and without knowing the source of leakage, a breach in containment integrity was apparent. Since TS 3.7.A.2 (which requires that containment integrity be maintained) was not met and an action statement is not provided, 10 CFR 50.36(c)(2) requiring the reactor to be shut down, is applicable. The licensee did not recognize or consider these two requirements and continued operation while pursuing the operational matters listed previously.

Three items of noncompliance were identified in the above sequence. The licensee failed to control activities involving safety components as evidenced by the lack of containment integrity experienced between December 6-9, 1979. Paragraph 5.c. discusses this item further. The second item is the failure to shut down the reactor as required by 10 CFR 50.36(c)(2), on December 8, 1979, when information was available that indicated containment integrity was not maintained. The third item is the failure to report to the NRC within 24 hours that the Limiting Condition of Operation of TS 3.7.A.2 had not been met.

The containment was repressurized at 0250 hours on December 9 and pressure cycled between 1.5 psig and 0.5 psig for about three hours and then between 1.7 psig and 1.3 psig for another two and one-half hours. This pressure cycling occurred because containment pressure was observed to rapidly decrease when nitrogen flow was halted, thus requiring nitrogen makeup via the 6" purge line to be admitted periodically. At approximately 0430 hours, inspections in the plant identified the Southeast drywell equipment hatch to be leaking. The licensee recognized this leak as applying to TS 4.7.A.2.h which permitted 48 hours to correct the problem. Two or three of the twelve securing bolts in the top right section of the hatch were found torqued to less than the required 500 foot pounds. By 0930 hours the bolts had been retorqued and a successful local leak rate test completed. The containment excessive leak rate was terminated as evidenced by the successful local leak rate test, normal drywell pressure readings, and the nitrogen makeup consumption.

b. Nitrogen Consumption Monitoring

Nitrogen consumption is monitored by several means. Surveillance Instruction-2, Instrument Checks and Observations, requires a daily recording of nitrogen makeup requirements (from FR/PR-76-14). It also requires the daily recording of liquid nitrogen tank level (in percent and inches-from LI-84-2A/B and LI-84-13A/B). These instruments are available for monitoring by operations personnel. The nitrogen makeup requirements are also continuously recorded on a strip chart which is required by the licensee to be reviewed daily by Results Section personnel. A formal review system that monitored for short-term and long-term trends had not been established. Technical Specification 4.7.A.2.j. requires that when the primary containment is inerted, the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. The plant superintendent stated that a formal review system would be established.

It was recognized by the licensee during this event that the nitrogen makeup system did detect abnormal nitrogen consumption. It was also noted on subsequent pressurization that it was not possible to maintain the 1.5 psig containment pressure using the normal two inch nitrogen makeup line. Therefore, the inspector concludes that a long-term existence of this event without detection was not possible. The procedures involved in monitoring nitrogen consumption were very general. The following observations were noted by the inspector:

SI-2, Instrument Check and Observations, required checking daily makeup requirements but did not indicate what amount was abnormal. No formal system existed to review the data to satisfy the TS 4.7.A.2.j intent.

SOI-76, Containment Inerting System, did not give adequate guidance on what was excessive leakage. The procedure made no reference to Containment Integrity TS 3.7.A.2. Thus, it was not clear how operating

personnel could relate excessive consumption to loss of containment integrity.

SOI-64, Primary Containment, provides general guidance (i.e., check for leaks) on low drywell pressure.

The issue of whether plant procedures properly address expected indications of, or operator actions for a loss of primary containment integrity, is designated as an Unresolved Item.

c. Hatch Removal and Installation Procedures

The inspectors established through interviews and procedure review that no procedure existed for removal and installation of the drywell equipment hatches and other containment closures, excluding the drywell head. These jobs had been considered within the skill of the craft by TVA with a specific formal local leak rate test being required at the conclusion of the installation. Assignment of work and specific instructions were provided verbally. Discussions with three personnel involved in hatch closures resulted in three techniques for performing the work. Thus, the prerequisites, storage, sequencing, torque values, and testing requirements were communicated verbally to the foreman conducting the work. The closure of containment boundaries cannot be left to the chance that the proper instructions are correctly communicated and understood by those performing the work. The absence of these procedures is an apparent item of noncompliance with Criterion V of 10 CFR Appendix B and TS 6.3.A.1 which require detailed written procedures for components involving nuclear safety. The licensee stated that procedures would be developed before the end of the Unit 1 refueling outage scheduled to begin January 2, 1980.

d. Procedural Coordination

System Operating Instruction (SOI) No. 64, Primary Containment, Step III B requires the shift engineer to verify that the drywell access hatches are replaced and sealed. This was initialed as being performed on data sheet 64-1, page 2, for Unit 3. The data sheet was dated 12/6/79 at 1400 hours and is required by step I.B.1 of General Operating Instruction (GOI) 100-1, Integrated Plant Operations. Thus, it is noted that any subsequent opening of the access hatch would not cause GOI-100-1 to require the re-performing of SOI-64. The plant superintendent stated that the procedures would be revised to address this matter. In this instance, it was also noted that the Unit 3 shift engineer's log had entered that GOI-100-1 was completed at 0345 hours on 12/6/79 even though, as stated above, SOI-64, data sheet 64-1, page 2, required to be completed by GOI-100-1 was dated several hours later.

e. Documents Reviewed

Documents and records listed below were among those reviewed during the investigation.

(1) Unit 3 Shift Engineer's Journal from 12/5/79 through 12/9/79.

(2) Unit 3 Operator's Journal from 12/5/79 through 12/9/79.

(3) Unit 3 Assistant Shift Engineer's Journal from 12/5/79 through 12/9/79.

(4) Unit 3 Surveillance Instruction 2, Instrument Checks and Observations, for the period December 2 through December 13, 1979.

(5) General Operating Instruction 100-1, Integrated Plant Operations, dated 12/6/79 for Unit 3.

(6) System Operating Instruction No. 64, Primary Containment, data sheets dated 12/5 and 12/6/79 for Unit 3.

(7) System Operating Instruction No. 76, Containment Inerting System, dated 12/6/79 with enclosed check list dated 11/27/79 for Unit 3.

(8) Strip Chart of primary containment nitrogen makeup flow and pressure; FR/PR-76-14 for the period 12/7/79 through 12/9/79 for Unit 3.

(9) Strip chart of oxygen concentration in the drywell, Unit 3. for the period 12/8/79 through 12/9/79.

(10) Strip chart of power range power level for Unit 3 for the period 12/8/79 through 12/9/79.

(11) Data sheet from performance of SI 4.7.A.1. g-2, Test Bolted Double-Gasketed Seals performed on equipment access hatch X-1B-EQUIP (Southeast) on 11/15/79 and 12/9/79 and hatch X-1A-EQUIP (Northwest) on 11/20/79 and 12/5/79.

(12) Valve Group Journal from 12/3/79 to 12/7/79.

(13) Drywell Health Physics Journal, 11/23/79 to 12/6/79.

(14) Fabrication Drawings of the Drywell Equipment Hatches

f. Test Results

Following completion of refueling activities, local leak rate testing was verified to have been performed on both drywell equipment hatches on November 15 and 20, 1979. Additionally, the integrated containment leak rate test was completed successfully on November 25, 1979. This

test had been witnessed by Region II inspectors. Subsequently, the Northwest equipment hatch was reopened on December 4 and reclosed on December 5. Test data results after the closure were confirmed to be satisfactory. Discussions with three personnel preparing for and performing this work on the Northwest equipment hatch did not indicate that any work had been performed on the Southeast equipment hatch. Discussions with one individual directly involved in correcting the leaking Southeast equipment hatch on December 9 established that two or three bolts in the top right sector were tight but torqued to less than 500 foot pounds. Following retorquing, test results indicated a satisfactory local leak rate test was performed.

g. Hatch Identification

Discussion with personnel indicated that identification of the two equipment hatches of each reactor is by building orientation such as Northwest, North, or West and Southeast, South, or East. Identification of the equipment hatches are by X-1A and X-1B in technical specifications and test procedures. No identification was present on the hatches themselves as observed by the inspectors. To minimize any potential confusion, the plant superintendent has committed to providing identification for these hatches by January 2, 1980.

h. Sequence of Events

Time	Date	Event
0645	12/6/79	Unit 3 critical following refueling .
1215	12/7/79	Reactor Coolant System at normal operating temperature and pressure.
1235	12/7/79	Reactor mode switch in RUN position.
0620	12/8/79	Started purging nitrogen into containment.
1100	12/8/79	Drywell atmosphere less than 4% oxygen (TS 3.7.A.5.b satisfied)
1235	12/8/79	Drywell to torus differential pressure initially established [TS 3.7.A.6.a.(1) satisfied] Drywell pressure at 1.35 psig
1430	12/8/79	Drywell pressure 0 psig.
1650	12/8/79	Brought turbine on the line (power at 20%)

Time	Date	Event
2040	12/8/79	Completed cycling torus to drywell vacuum breakers.
2100	12/8/79	Increased power to 25%.
0023	12/9/79	Commenced inserting control rods (power at 25%).
0250	12/9/79	Containment repressurized and controlled shutdown stopped (power at 20%).
0430	12/9/79	Drywell leak discovered on equipment hatch.
0830	12/9/79	Equipment hatch bolts fully torqued.
0930	12/9/79	Equipment hatch passed leak test.
0745	12/10/79	NRC resident inspector informed of leak on equipment hatch. Plant personnel evaluating event.
1430	12/10/79	Licensee decides to issue prompt report.
1625	12/10/79	NRC inspectors formally informed prompt report being issued.
	12/11/79	Prompt report fascimile, issued by licensee, received in NRC Region II (date stamped 0841 hours EST 12/12/79).

Investigations by the licensee and the Region II inspectors were unable to establish the specific cause of the access hatch bolts not being fully torqued. This report does identify several items which may have contributed to the event. On December 12, Region II issued a letter to TVA confirming certain actions that included a formal investigation of events which resulted in a loss of primary containment; verifying present nitrogen consumption rates on all three Browns Ferry containments; reviewing the adequacy of management controls of plant maintenance activities that impact on plant safety; and reviewing procedures for prompt reporting of plant events to NRC. The results of the investigation and the details relating to the other actions will be discussed in a meeting with NRC in the near future.

6. Plant tour.

A tour of the unit 2 reactor building was made on December 14, 1979. A personnel entry and exit of the unit 2 drywell was monitored.

No items of noncompliance or deviations were identified.

7. Plant Operations Review Committee (PORC) Meeting

The inspector observed a PORC meeting held on December 10, 1979 to ascertain whether provisions of Technical Specification 6.2.B dealing with membership, review process and quorm were met. In addition, the minutes of this meeting were subsequently reviewed to confirm that they accurately reflected the content of the meeting.

No items of noncompliance or deviations were identified.