GNS '800814 101

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : J. R. Calhoun, Director of Nuclear Power, 1750 CST2-C

FROM : H. N. Culver, Chief, Nuclear Safety Review Staff, 249A HBB-K

DATE : August 14, 1980

8008150002 (19)

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NSRS REVIEW REPORT NO. R-80-12-BFN

Attached is the NSRS report of a routine review of selected activities at BFN. Categories of items reviewed included open items from previous reviews and potential new safety concerns as identified by NSRS.

Our recommendations, as stated in section III of the report, show one open item requiring action by NUC PR for resolution. By copy of this memorandum, we are informing EN DES of one other recommendation requiring Design attention. We have not established a resolution date for the open items; however, we feel that these items should be resolved in a timely manner. You are requested to inform NSRS of your plans and schedule for implementing our recommendations by September 5, 1980.

If you have any questions regarding this matter, we will be pleased to discuss them.

H N Culver

H. N. Culver

LFB:KRW
Attachment
CC (Attachment):
~~MEDS, E4B37 C-K~~
M. N. Sprouse, W11A9 C-K
F. A. Szczepanski, Jr., 417 UBB-K

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

REVIEW

NSRS REPORT NO. R-80-12-BFN

Subject: Tennessee Valley Authority

Browns Ferry Nuclear Plant - Units 1, 2, and 3

Routine Review

(1) Old Items

(2) Potential New Concerns

Dates of Onsite Review: June 9-13, 1980

Reviewers: L. F. Blankner — Date: 8-13-80

R. W. Travis — Date: 8-13-80

Approved by: K. W. Whitt — Date: 8/13/80

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REPORT R-80-12-BFN

I. Scope

This was a routine review of selected activities at Browns Ferry Nuclear Plant and involved items from previous reviews and potential new concerns as identified by NSRS. Each item is discussed in detail in section IV. The review entailed two man-weeks on-site during the week of June 9-13, 1980.

II. Status of Items Reviewed

During this review of BFN, items from the Operational Event Report, items 1-28-80-2, 3-11-80-3, 3-21-80-1, 3-29-80-1, 3-31-80-1, 5-13-80-3, 5-15-80-1, and 5-16-80-1; OER 5-13-80; OIE Bulletin 80-07; LER 260/8020; Employee Concern 79-10-01; an NSRS staff report dated April 30, 1980; and steam tunnel modifications were reviewed, and the plant actions were considered appropriate with a few exceptions.

A. Old Items

1. BFN-1, Main Steam Relief Valve Tailpipe Thermocouple Bypass (OER 3-21-80-1)

   Item closed. (Reference section IV.A.1 for details.)

2. Review of JIWR Logs and Administrative Controls (Employee Concern 79-10-01)

   Item closed. (Reference section IV.A.2 for details.)

3. R-80-12-BFN-01, Main Steam Line Insulation and Ventilation in the Steam Tunnels

   Item remains open pending completion of unit 2 steam tunnel insulation modification and unit 1 tunnel door closure. (Reference section IV.A.3 for details.)

4. Scram Reports 92 through 95 for BFN-2

   Item closed. (Reference section IV.A.4 for details.)

B. Potential New Concerns

1. R-80-06-BFN-02, Surveillance Instructions not Performed/Management Notification (OER 5-16-80-1)

   This item is closed. (Reference section IV.B.1 for details.)

2. R-80-12-BFN-03, BFN-3 LIS-3-56A and B, Reactor Water Level (OER 5-13-80-3)

   Item remains open pending switch replacement. (Reference section IV.B.2 for details.)

3. R-80-12-BFN-04, BFN-2, PS-3-22D, Reactor Pressure B, Setpoint Drift (LER 260/8020)

   Item remains open pending NUC PR resolution of access control to PS-3-22A, B, C, and D. (Reference section IV.B.3 for details.)

4. R-80-12-BFN-05, BFN-3 Unit Scram (OER 5-15-80-1)

   Item remains open for NSRS tracking purposes. (Reference section IV.B.4 for details.)

5. R-80-12-BFN-06, Performance of Core Spray Pump 1A (OER's 1-28-80-2 and 3-11-80-3)

   Item remains open for NSRS tracking purposes. (Reference section IV.B.5 for details.)

6. R-80-12-BFN-07, BWR Jet Pump Assembly Failure

   Item remains open for NSRS tracking purposes. (Reference section IV.B.6 for details.)

7. R-80-12-BFN-08, EECW Flow Deficiencies (OER 5-13-80)

   Item remains open for NSRS tracking purposes. (Reference section IV.B.7 for details.)

8. R-80-12-BFN-09, BFN-1 High Worth Control Rod (OER Items 03-29-80-1 and 03-31-80-1)

   Item remains open pending review of documentation from NUC PR management to BFN concurring in rod sequence modifications. (Reference section IV.B.8 for details.)

9. BFN-3 Functional Test of Control Rod 18-55

   Item closed. (Reference section IV.B.9 for details.)

III. Recommendations on Open Items

A. Items for which NUC PR Actions are Required

1. Item II.A.3, R-80-12-BFN-01, Main Steam Line Insulation and Ventilation in the Steam Tunnel

NUC PR should determine and implement appropriate action to assure the proper operational configuration of the unit 1 steam tunnel. The steam tunnel door should be closed and special ventilation should not be required under normal operating conditions. (Reference section IV.A.3 for details.)

B. Items for which EN DES Action Required

1. Item II.A.3, R-80-12-BFN-01, Main Steam Line Insulation and Ventilation in the Steam Tunnels

EN DES should coordinate with NUC PR to resolve the problem of high temperature in unit 2 main steam tunnel. (Reference section IV.A.3 for details.)

IV. Details

A. Old Items

1. BFN-1, Main Steam Relief Valve Tailpipe Thermocouple Bypass (OER 3-21-80-1)

The Operational Event Report item 3-21-80-1 read as follows:

"Unit 1 - While testing main steam relief valves (MSRV's) at 250 pounds pressure, MSRV 1-23 gave no indication of the valve opening as recorded on the thermocouple located on the tailpipe of the MSRV. Further investigation revealed jumpers had disabled the thermocouples on MSRV's 1-23 and 1-34. Removing the jumper on 1-23 restored the thermocouple operability and the valve tested satisfactorily. Valve 1-34, however, gave no tailpipe temperature changes as recorded by the thermocouple when the valve was actuated. Entry into the drywell revealed that on 1-23 and 1-34, thermocouple walls were approximately 1/4-inch diameter as opposed to the nominal 1/2-inch diameter. The thermocouple probe on 1-23 had been changed to fit the wall. However, 1-34 was also changed but was broken, and there was no operable thermocouple. MSRV 1-34 thermocouple probe is being machined to fit the existing wall and will be installed by mid-day today."

The original safety concern of the NSRS relative to this item was why the jumpers were not identified on the prestartup committee report for unit 1, cycle 3. The report was dated March 15, 1980, and the Temporary Alteration Control Form (TACF) was dated March 15, 1980. Therefore, the answer to the concern seems to be that it was not installed when the report was drafted. The plant actions appear to be in accordance with the guidance provided by the plant instructions. However, while reviewing this item, it was noticed that "white-out" was used on the original form in the control room TACF book on the points where the jumper was installed. While the use of "white-out" is not prohibited by the plant Standard Practices, it is a procedure which is open to abuse. Attention will be directed in future NSRS reviews toward abuse of the use of "white-out" on official documents.

2. Review of JIWR Logs and Administrative Controls (Employee Concern 79-10-01)

This was a general followup review to a previous employee concern. A review of the TACF log book was conducted with no major deficiencies noted.

In a conversation with a member of plant management, it was learned that almost all temporary alterations were now reviewed by PORC even though only the ones on CSSC equipment are required. An instrument maintenance representative stated that a thorough technical review was made for all proposed temporary alterations on instrumentation.

A review of the interface between the Outage Section and the remainder of the plant staff concerning temporary alterations was begun. The tracking of temporary alterations when used for Outage work will be reviewed in greater depth during a future visit.

3. R-80-12-BFN-01, Main Steam Line Insulation and Ventilation in the Steam Tunnels

   In a memorandum from H. G. Parris to H. N. Culver dated April 3, 1980, it was stated that:

   > "The modification to the duct work and the piping insulation has been completed on units 1 and 3 and will be completed during the next refueling outage on unit 2. It appears at this time that this has solved the high ambient temperature problem. The doors on units 1 and 3 will be closed, but the unit 2 door will remain open until the modification is complete."

   On June 10 the doors to the steam tunnels on units 1 and 2 were still open. A fan was blowing into the unit 1 tunnel, and an air conditioner was blowing into unit 2. The unit 3 door was closed. A management representative was asked about this on June 11. He said that the doors on units 1 and 3 should be closed but that the unit 2 modifications would not be completed until an outage planned for June 19.

   Subsequent to the plant site review, item 06-20-80-1 appeared in the Operational Event Report reporting an attempt to close the unit 1 main steam tunnel doors. The temperature rose to 176 degrees Fahrenheit before the door was reopened. There were no steam leaks, and the ventilation system was properly balanced. An investigation was continuing.

   It appears that the high temperature problem has not been resolved. This should be further evaluated and a satisfactory resolution reached to assure that the steam tunnel operational configuration intended by design can be maintained.

4. BFN-2 Scram Reports 92 through 95 (NSRS Staff Report Dated April 30, 1980)

   This review was performed to determine the status of scram reports 92, 93, 94, and 95 for BFN-2. At the time (March 12, 13, and 14, 1980) of an earlier review, these reports were in the form of an incomplete draft pending ongoing tests and investigations.

The scram reports have been completed and approved. From discussions with site personnel, the NSRS reviewer was satisfied that no unusual or unexpected results had been disclosed by the testing and monitoring subsequent to the four scrams. This item is closed.

B. Potential New Concerns

1. R-80-12-BFN-02, Surveillance Instructions (SI) not Performed/ Management Notification (OER 5-16-80-1)

   On April 4, 1980, it was determined by plant personnel that a portion of SI 4.11.C.3 and .4, "Fire Protection System," had not been conducted for three out of the last four months. This failure to perform required surveillance testing was not brought to plant management's attention until May 6, 1980, when a corrective action report was prepared by the responsible plant section. The cause of the failure to inform plant management was not conclusively determined. It appears to have been a random failure. Conditions that NSRS becomes aware of which indicate a failure by plant management to keep informed of plant operational deficiencies will be reviewed during future visits.

   On June 2, 1980, a licensee event report (LER), BFRO-50-296/8017, was issued. SI 4.11.C.3 and .4 was conducted in three out of four previous months with the last page, page 181, missing. This left one fire zone untested. There was no indication that page 180 was not the last page of the instruction. Problems also exist in instructions where revisions have been made which caused a page to be added in the middle of the instruction SI 4.11.D.4 is an example. Page 79 is followed by page 79A. Page 79A can be omitted and there is no indication that there should be a page between 79 and 80. According to BFRO-50-296/8017, the procedure data sheets are being revised to ensure no data sheets are eliminated in the future.

2. R-80-12-03, BFN-3, LIS-3-56A and B, Reactor Water Level (OER 5-13-80-3)

   On May 13, 1980, the following appeared in the Operational Event Report:

   > "Unit 3 - Level switches LIS-3-56A and B were found to have sticking contacts on low-level trips for the recirculation MG set. These switches are located on the Yarway. This requires a 30-day report to NRC."

The NSRS reviewed this item for safety significance. Several LER's have been written on the subject over the last few months. The Yarway instrumentation will be replaced with analog alarm and trip units.

Operation personnel should be made aware that until the present reactor level instrumentation is replaced, there exists the possibility that the recirculation pumps will not automatically trip on low reactor water level.

There is also a problem with certification of the replacement instrumentation. Nothing presently available meets IEEE Standard 323-1974. Correspondence between NUC PR and EN DES indicates that a concentrated effort has been and is being made to purchase suitable replacement instrumentation.

Replacement of reactor water level instrumentation will be a problem for an undefined period. If the equipment ordered is installed, it may not meet the IEEE Standard 323-1974. EN DES and NUC PR should work together in an effort to reach an expedited resolution. NSRS will follow the progress of this effort.

3. R-80-12-BFN-04, BFN-2, PS-3-22D, Reactor Pressure B, Setpoint Drift (LER 260/8020)

This event occurred on April 21, 1980, and was reported to the NRC on May 20, 1980. During normal operation the reactor high pressure switch PS-3-22D setpoint was found to exceed the required setpoint during performance of SI 4.1.A-5. The cause of the occurrence was attributed to setpoint drift. The switch is a Barksdale model B2T-A12SS and is calibrated monthly. PS-3-22A, B, C, and D may have been involved in the spurious trips of February 1980. No connection could be made between those trips and this setpoint drift. It was thought possible that the switches had been damaged causing the setpoint drift, but this could not be shown.

The four switches are painted yellow to identify them as critical instrumentation. There is administrative control for access to the switches, but there are no effective physical barriers. A rope is draped across the panels with a sign attached that requires notification of the shift engineer before crossing. There are plans for a wire mesh screen around the instruments. This item will remain open until the physical barriers are installed or an alternate fix is implemented.

4. R-80-12-BFN-05, BFN-3 Unit Scram (OER 5-15-80-1)

This was scram no. 82 on BFN-3. The cause was a generator field ground detector. The scram report stated that it was probably caused by an electrician performing EMI-6, "Motors, Generators, and Controls - General," when relay 86C was energized. Bridge and megger tests were performed on the field in accordance with DPM N73013 and no grounds were indicated. EMI-6 has been changed, as of May 20, 1980, to ensure that the trip cutout (TCO) block between the generator field group detector and relay 86C is pulled while performing portions of EMI-6 related to the generator field.

On June 7, 1980, unit 3 was again tripped when the generator field ground relay was being returned to service following routine testing of the generator shaft voltage. The OER stated that the trip was caused by the sensitivity of these relays. NSRS will review this during a future visit.

The scram report was adequate as defined by BF 12.8, "Unit Trip and Reactor Transient Analysis." BF 12.8 requires that "pertinent charts and printouts" be incorporated in the report. Several charts and printouts were included along with the GOI 100-1 data sheets. The narrative section of the report should contain more details concerning the analysis of the charts and printouts thereby showing how the cause of the scram was determined. This would aid plant personnel in investigating subsequent scrams.

5. R-80-12-BFN-06, BFN-1, Performance of Core Spray Pump 1A (OER's 1-28-80-2 and 3-11-80-3)

This review was performed to determine whether satisfactory post-maintenance and surveillance test results were obtained subsequent to the removal of a wood block from the impeller of core spray pump 1A. NSRS received notification of the wood block via OER items 1-28-80-2 and 3-11-80-3. The chief sources of information for this review were BF STEAR 80-03, "Analysis of Wood Block in '1A' Core Spray Pump," and the surveillance data file for SI 3.1.1, "Core Spray Pump Performance."

a. BF STEAR 80-03 was prepared to authorize a special post-maintenance test of core spray pump 1A following removal of a wood block found wedged between two blades of the pump impeller. Plant personnel had initiated a request for inspection of the pump's internals based on performance test data. The wood block, which was removed during the winter 1980 refueling outage of BFN-1, was recovered with a fragment (2.25" by 1.5" x 1") missing. The unreviewed safety question determination (USQD) for BF STEAR 80-03 concluded that safe operation of core spray loop I, containing pumps 1A and 1C, could be assured by performing a special test to determine that the missing wood fragment was not situated so as to block minimum flow bypass protection of pump 1A. Such a test had been performed and was documented in the BF STEAR file. Upon review, it was concluded that BF STEAR 80-03 had been prepared and performed adequately with respect to safety considerations. It was noted that plant management was processing corrections for several administrative deficiencies in the STEAR documentation. This subitem is closed.

b. Review of core spray pump performance data (SI 3.1.1) indicated that post-maintenance performance of core spray pump 1A, while acceptable under surveillance test criteria, may indicate degraded performance. Summaries of recent core spray performance results are provided in tables 1 and 2. In comparison with its companion 1C pump and with other core spray pumps, 1A pump exhibits more fluctuation and less developed head than appears warranted (note especially total head developed by other BFN-1 pumps). The review included an examination of preoperational test data which indicated that pump 1A may have been fouled by the wood block even before initial site testing. Review of the vendor's test data gives no evidence of significant differences between 1A and other core spray pumps. There is a possibility that the fluctuation in pump data results from an instrumentation deficiency. However, this data was not examined by the reviewer due to time constraints. It was commented by mechanical maintenance personnel that, following removal of the wood block, a careful inspection of the pump internals disclosed no sign of deterioration or unusual wear. This subitem is closed.

c. The NSRS reviewer noted that on two occasions, when performance of core spray pump 1A fell in the "Acceptable Range" (marginal performance), plant personnel determined that instrument recalibration should be performed and the SI test rerun. In both cases, delay in recalibration occurred such that the next scheduled (monthly) flow test served as the retest. This does not satisfy the implied urgency of IWP-3230 (as incorporated in SI 3.1) which requires doubling the frequency of pump tests in the event data falls in the "Alert Range." It would appear that the same interval (i.e., two weeks for IWP performance tests) should provide a sufficient interval for recalibration and retest.

It was also noted that there was no reference to a trouble report to document a request for recalibration of instrumentation. However, plant personnel assured the reviewer that recalibration had been effected. Maintenance records for the affected instruments were not revised at this time due to time constraints. Performance test results for pump 1A will be examined during a future site review.

d. The reviewer noted that consistent differences can be noted in indicated total head of some pumps having identical system configurations. For example, core spray pumps 2A, 2C, and 2D appear to develop 15-20 psi less than pumps 3A, 3C, and 3D. However, pumps 2B and 3B develop equal total heads.

The reviewer questioned whether different reference data could be in use for instrument calibrations as had been discovered in RHR system pressure sensors at an earlier time. A representative of plant management stated that a review of this question would be conducted. This matter will be examined during a future site review.

6. R-80-12-BFN-07, BWR Jet Pump Assembly Failure (OIE Bulletin 80-07)

This review was performed to determine whether an adequate in-vessel inspection of jet pump hold-down beams had been performed during the winter 1980 refueling outage of BFN-1 and whether the requirements of IE Bulletin 80-07, "BWR Jet Pump Assembly Failure," had been implemented. The chief sources of information for this review were a videotaped recording of the beam inspection and test instructions SI 4.6.E, "Jet Pumps," and TI 52, "Jet Pump Failure Detection." These instructions received PORC approval on June 10, 1980.

a. Review of portions of a videotape produced during inspection of jet pump assemblies on BFN-1 revealed that surface physical characteristics of the beam and its keeper mechanism were observed with considerable clarity during the inspection. The NSRS reviewer was informed that an NRC inspector was present to watch the video display during underwater inspection of the BFN-1 jet pumps. At the time of the BFN-1 outage, only a visual inspection of hold-down beams was feasible since an ultrasonic test jig had not then been manufactured for BWR-4 applications.

b. Review of SI 4.6.E and TI-52 indicated that the requirements of IE Bulletin 80-07 would be fully implemented once these procedures were placed in effect.

NSRS will examine results from the SI 4.6.E surveillance tests at a future site review.

7. R-80-12-BFN-08, EECW System Flow Deficiencies (OER 5-13-80)

This review was performed to determine whether failure by the EECW system to deliver adequate cooling flow to vital auxiliary components is a recurring problem and to determine what plant actions were being taken to monitor and improve system performance. NSRS received notification of EECW flow deficiencies via OER items 5-13-80-2 (3B RHR seal head exchanger) and 5-13-80-5 (core spray system I room cooler and 3D diesel generator jacket cooler). The chief sources of information for this review were the data file for MRI 303, "EECW Quarterly Flow Verification," discussions with site personnel, and the data file for SI 3.2.4, "EECW Check Valve Test."

a. The surveillance and correspondence files concerning MRI 303 data disclosed that deficient EECW flows had been detected during three of the last five flow tests over a period of approximately two years. It was noted that "As Left" rather than "As Found" data was being recorded for the official test record. Nevertheless, test personnel maintained that acceptable data was being recorded based on the fact that since deficient EECW flow rates were always totally inadequate, a record identifying which components had required correction was adequate for the data file. Plant management stated that an assignment had been made to evaluate recent test results and initiate action as required to remedy performance deficiencies. At the present time, double the required flow is being maintained ("As Left") on those components having minimal requirements.

It was determined that following the most recent flow tests about one-half gallon of clams were removed from the 3D diesel generator jacket cooler. A small quantity of mud had been found in the 3B RHR pump seal heat exchanger.

b. Discussion of SI 3.2.4 results with test personnel indicated that the lack of EECW flow to the BFN-1 core spray system I room cooler had probably resulted from siltation. Normal flow was restored by rapidly cycling the manual isolation valve in the EECW supply line.

c. It was noted in review of SI 3.2.4 data that there was on occasion neither a record of results nor a cross-reference to maintenance records (such as a trouble report number) by which results of corrective maintenance could be traced to deficient SI data. Where needed, results of corrective maintenance were obtained by discussion with plant personnel.

d. Following the site visit, a review of associated LER's (see below) has been performed. The LER's fail to describe the causes of the deficiencies. Each LER commits to a system flushing evaluation. It was noted on site that the frequency of performing MRI 303 had been increased from once per six months to once per quarter.

<u>Associated LER's</u>

BFRO-50-259/8042	Core Spray System I Room Cooler
BFRO-50-296/8015	RHR Pump Seal Heat Exchanger 3B
BFRO-50-296/8016	Diesel Generator Heat Exchanger 3D

There appears to be no policy statement or program document to control this type of testing. If generalized, failure to record "As Found" data or to cross-reference findings of degraded conditions to followup actions (i.e., submission of TR's, DCR's, etc.) may indicate need for a general standard for performing tests and acting on results. Nevertheless, plant management in this case is aware of and taking action concerning the EECW deficiencies. The timeliness and results of these actions will be examined by NSRS during a future site review.

8. R-80-12-BFN-09, BFN-1 High Worth Control Rod (OER Items 03-29-80-1 and 03-31-80-1)

This review was performed to determine whether adequate corrective action had been taken subsequent to discovery of a high worth control rod which caused a short period event during startup of BFN-1 following the winter 1980 refueling outage. NSRS was notified of the event by OER items 03-29-80-1 and 03-31-80-1. The sources of information for this review were discussions with site personnel and data sheets for SI 4.3.B.1.a, "Control Rod Coupling Integrity Check." The NSRS reviewer was shown a revised A-2 rod sequence for startup of BFN units 1, 2, and 3. This sequence, which was approved on April 11, 1980, contains notation that CRD 46-19 should be withdrawn one rod notch at a time for unit 1 only. The revision was made to reverse the sequence, by group, of the rods withdrawn during startup. It was explained to the NSRS reviewer that the sequence change imposes criticality during startup at a point where individual rod worths are much less than rod 46 in the former sequence. Plant management also stated that correspondence from NUC PR in Chattanooga was being prepared to document concurrence in the sequence change by the Reactor Engineering Branch and General Electric, both of which had been consulted prior to revising the rod sequence. Verification of this commitment will be made during a future site review.

9. BFN-3 Functional Test of Control Rod 18-55 (OER's 1-26-80 and 1-27-80)

This review was performed to determine whether a complete functional test of CRD 18-55 had been completed at the next available opportunity following a partial rod stroke limited by operating conditions. By OER's dated January 26, 1980, and January 27, 1980, NSRS was notified that corrective maintenance followed by a half-stroke functional test had been performed on January 26, 1980, while BFN-3 was operating at a high-power level. Full-stroke testing of the rod was prevented by fuel thermal limits in the vicinity of the control rod blade. Site personnel informed the NSRS reviewer that verbal concurrence had been obtained from NUC PR in Chattanooga for the partial stroke test. The nuclear engineers' log book confirmed that a full stroke of rod 18-55 had been completed on February 22, 1980, when power was reduced for a rod sequence exchange. This had been the earliest opportunity for the test. This item is closed.

V. Personnel Contacted

H. L. Abercrombie
A. L. Burnett
M. W. Haney
J. L. Harness
J. R. Pittman
R. E. Smith
J. E. Swindell

VI. Documents Reviewed (References)

A. Memorandum from R. T. Smith to H. L. Abercrombie dated March 15, 1980, "Prestartup Committee Report for Unit 1 Cycle 3"

B. Standard Practice BF 8.2, "Temporary Alterations," dated January 8, 1980

C. Standard Practice BF 12.8, "Unit Trip and Reactor Transient Analysis," dated September 13, 1978

D. Scram Report No. 82, Unit 3

E. EMI-6, "Motors, Generators, and Controls - General," dated May 30, 1980

F. SI 4.11.C.3 and .4, "Fire Protection System," dated January 6, 1978

G. Memorandum from H. G. Parris to H. N. Culver dated April 3, 1980, "Browns Ferry Nuclear Plant Employee Concern 79-10-01 - Operating Practices where Protective Systems are Bypassed"

H. Standard Practice BF 3.4, "Unit Prestartup Review," dated June 23, 1978

I. Memorandum from Roy H. Dunham to H. S. Fox dated January 24, 1980, "Browns Ferry Nuclear Plant - ECN P-0126 - DCR No. 1398 - Trip Calibration System"

J. Memorandum from J. R. Calhoun to R. H. Dunham dated March 6, 1980, "Browns Ferry Nuclear Plant - ECN P-0126 - DCR No. 1398 - Trip Calibration"

K. Memorandum from Roy H. Dunham to J. R. Calhoun dated April 1, 1980, "Browns Ferry Nuclear Plant - ECN P-0126 - DCR No. 1398 - Trip Calibration"

L. Memorandum dated January 31, 1980, from H. L. Abercrombie to J. G. Dewease," STEAR 80-03 - Analysis of Wood Block in "1A" Core Spray Pump - Browns Ferry Nuclear Plant" (L52 800131 838)

M. Memorandum dated March 10, 1980, from J. G. Dewease to H. L. Abercrombie, "STEAR 80-03 - Analysis of Wood Block in "1A" Core Spray Pump - Browns Ferry Nuclear Plant" (L33 800207 801)

N. Preoperational Test GE-12, "Core Spray System," Official Data for Browns Ferry Nuclear Plant Units 1, 2, and 3

O. Surveillance Instruction 4.3.8.1.a-A

P. Scram Reports 92, 93, 94, and 95 for Browns Ferry Nuclear Plant Unit 2

Q. Technical Instruction 32, "Jet Pump Failure Detection, Units 1, 2, and 3" (PORC reviewed June 10, 1980)

R. Revisions to portions of Surveillance Instruction 4.6.E, Units 1, 2, and 3 (PORC reviewed June 10, 1980)

S. MMI 303, "Emergency Equipment Cooling Water System Quarterly Flow Verification" and data file

T. Surveillance Instruction 3.1, "Inservice Pump Testing Required by ASME Section XI"

U. Surveillance Instruction 3.1.1, "Core Spray Pump Performance"

TABLE 1

CORE SPRAY PUMP PERFORMANCE DATA FOR BFN-1

Date of Surveillance Test	Core Spray Pump Total Head (psia)[a]			
	1A	1C	1B	1D
2/2/80	248.5	248.0	258[b]	-
3/5/80	228.0	249.0	249	254
4/5/80	240.0	245.0	259	254
5/7/80	228.0	244.0	258	259
6/4/80	224.5	243.8	259	233

TABLE 2

CORE SPRAY PUMP PERFORMANCE DATA FOR BFN-2 AND BFN-3[c]

	Core Spray Pump Total Head (psia)[a]			
Core Spray Pump	2A	2C	2B	2D
	229	227	238	222
	237	237	239	222
	224[b]	220[b]	234	224
Core Spray Pump	3A	3C	3B	3D
	246	247	235	244
	255	245	238	243
	246	245	238	256

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[a] Data taken at nominal flow rate of 3200 gpm.

[b] Data taken at recorded flow rate of 3300 gpm.

[c] Data represents most recent three sets of SI results.

OPTIONAL FORM NO. 10
UNITED STATES GOVERNMENT

Memorandum

GNS '80 0821 052

TENNESSEE VALLEY AUTHORITY

800822C0301

(20)

TO : J. R. Calhoun, Director of Nuclear Power, 1750 CST2-C

FROM : H. N. Culver, Director of Nuclear Safety Review Staff, 249A HBB-K

DATE : August 20, 1980

SUBJECT: BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3 - CONTROL ROD DRIVE SYSTEM MALFUNCTIONS AT UNITS 1 AND 3 - NUCLEAR SAFETY REVIEW STAFF REVIEW REPORT NO. R-80-13-BFN

Attached is the NSRS report of a special review of station activities and documentation regarding recent incidents and evaluations concerning control rod drive (CRD) system performance.

Our recommendations that are set forth in section III of the report are specific to the conclusions and findings set forth in section V of the report. Because of the nature of the problems identified in the findings and our concern that specific actions are required to cope with these problems, the recommendations are set forth in very specific terms. These recommendations we believe will provide solutions to the problems that are identified. Your response should therefore either be directed toward your actions to implement these specific recommendation, or your proposal to implement changes that will accomplish the objectives of our recommendations if alternate methods of accomplishing such objectives exist. Special attention is essential to assure completion during the forthcoming major outages of the BFN units. Please provide your plans and schedule for implementing the recommendations by September 22, 1980.

If you have any questions regarding this matter, we will be pleased to discuss them.

Original Signed By
H. N. Culver

H. N. Culver

LFB:LML
Attachment
cc (Attachment):
~~MEDS, EAB17 C-K~~
H. N. Sprouse, W11A9 C-K
W. F. Willis, E12B16 C-K

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

REVIEW

NSRS REPORT NO. R-80-13-BFN

SUBJECT: TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT - UNITS 1, 2, & 3

SPECIAL REVIEW OF INCIDENTS AND ACTIVITIES CONDUCTED TO RESOLVE DEFICIENCIES IN THE CONTROL ROD DRIVE SYSTEM PERFORMANCE

DATES OF ONSITE REVIEW: July 3-7, 1980
July 23-25, 1980

REVIEWER: Terry D. Taylor for H. L. Jones — 8-20-80 Date

REVIEWER: L. F. Blankner — 8-20-80 Date

APPROVED BY: K. W. Whitt — 8/20/80 Date

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I. Scope

This special review of incidents and activities at the Browns Ferry Nuclear Plant was made to what action was being taken by the line organization to prevent recurrence of two incidents involving deficient performance of the control rod drive (CRD) system on Browns Ferry units 1 and 3. The review was completed in two segments and required a total of approximately 88 man-hours of onsite time in the examination of documentation, observation of special performance testing, physical characteristics of sytems, and discussions with site personnel.

II. Background

While being shutdown for maintenance on June 28, 1980, the Browns Ferry unit 3 reactor failed to shut down properly when 76 of 185 control rods did not insert fully upon actuation of a manual shut-down signal (scram) of the reactor protection system. Within the next 15 minutes, the reactor was fully and safely shut down by action of station operations personnel.

This incident represents a very significant safety concern in that backup protection against a severe incident of this type is not included in the basic plant design. Instead, an extremely high factor of control rod system reliability is maintained by emphasizing proper design, operation, and maintenance of the control rod system.

Following the shutdown on June 28, a comprehensive series of inspections, tests, and analyses were begun by NUC PR. Analysis of event and test data indicated that the partial failure to scram was apparently due to the presence of a substantial quantity of water in the east side scram discharge header (SDH) [that is, the SDH that services the control rod hydraulic control mechanisms on the east side of the reactor building]. A detailed summary of the analysis that led to this conclusion was reported in reference B, which also provided a safety analysis justifying startup of the unit. An NSRS reviewer was sent to Browns Ferry on July 3, 1980, to monitor tests, evaluate data, and review preparation of the "Report and Safety Evaluation of the Browns Ferry unit 3 Partial Scram Failure of June 28, 1980," issued July 6, 1980. When completed, this analysis was concurred in by NSRS.

While unit 3 was being maintained in a safe shutdown condition, the NRC on June 30, 1980, issued a "Confirmation of Action" letter to TVA confirming a verbal agreement that the unit would remain out of service until NRC concerns were satisfied. On July 3, 1980, the NRC issued IE Bulletin 80-17, "Failure of 76 of 185 Control Rods to Fully Insert During Scram at a BWR." This bulletin required that all owners of domestic BWR plants conduct a comprehensive performance test to determine whether the control rod system was functioning adequately. There were also requirements in this bulletin for action to enhance the capability to mitigate the results of an ATWS (anticipated transient without scram) event and for periodic monitoring of the scram discharge piping to detect accumulation of water.

Startup of Browns Ferry unit 3 commenced on July 7, 1980. Numerous tests including those required by IE Bulletin 80-17 were conducted satisfactorily before and during this startup and the unit was returned to service on July 11, 1980.

By July 13, TVA had added a positive vent and an ultrasonic test (UT) system to monitor the east bank SDH continuously in unit 3.

The NRC issued supplement No. 1 to IE Bulletin 80-17 on July 18, 1980. This supplement directed design reviews and required additional measures, including installation of a permanent UT monitoring system, to further reduce the risk of an ATWS event at an operating BWR.

On July 19, while performing control rod scram tests, station personnel detected the presence of a vacuum in the scram discharge volume. This vacuum prevented proper drainage of the scram discharge instrument volume (SDIV) until operations personnel took action to eliminate the vacuum.

On July 22, 1980, the NRC issued supplement No. 2 to IE Bulletin 80-17, requiring installation of positive vents on the scram discharge headers at all operating BWR's.

At this point in time, the NSRS commenced the second portion of an onsite review with findings as reported below.

III. Recommendations

On July 25, 1980, an exit meeting was conducted by the NSRS representatives with those persons indicated in paragraph VI to summarize the purpose, scope, and findings of the special review. Except as noted below, the following findings were discussed and acknowledged by plant personnel as preliminary conclusions and recommendations at that time. NSRS' recommended modifications will provide corrective action to eliminate identified possible fundamental deficiencies in the present scram discharge headers (SDH)/scram discharge instrument volume (SDIV) arrangements. Figures 1 and 2 illustrate original and recommended configurations, respectively.

A. Ultrasonic Monitoring System

1. Site management and personnel should continue a conscientious effort to maintain and monitor the ultrasonic monitoring system. This method is the sole means of determining how much water is in the 6-inch scram discharge headers. (See section V.A.1 for details.)

2. Plant procedures should be prepared and approved to formalize the operating and maintenance (calibration and functional tests) requirements which have been implemented for the ultrasonic monitoring system. Such procedures should also address recordkeeping requirements and recognition of normal and abnormal monitor displays. (See section V.A.2 for details.)

3. NUC PR should determine and take appropriate action as required to address potential for degradation of UT equipment, especially the probe and couplant, from thermal, radiation, or acceleration damage. NUC PR should continue a very conscientious effort to monitor for signs of deterioration of the sensor and signal lead combination during daily calibrations. (See section V.A.3 for details.)

4. *A periodic calibration test for the inplace UT probes should be devised and implemented.

5. *NUC PR should evaluate possible deficiencies caused by use of a single UT monitor with dual inputs on units 1 and 2.

*Not discussed at exit interview.

B. Clean Radwaste (CRW) System Vents

NUC PR should prepare a safety-related DCR or FCR to document the modifications of CRW system (addition of positive vents) required by supplement No. 2 to IE Bulletin 80-17. Completion of a site safety evaluation or USQD as appropriate should be included in this documentation.

NSRS understands that formal review and associated documentation is in the process of being prepared.

C. CRW Vent Path Verification

Until physical modifications are accomplished which eliminate the need for the CRW vent connections, periodic functional testing of the vent paths by procedure should be completed and documented.

It is NSRS's understanding that the need for periodic testing has been eliminated by a vent line modification installed per FCR 185.

D. Modifications to Clean Radwaste and Control Rod Drive System

NSRS recommends the following modifications to the vent/drain portions of the scram discharge volume (see figure 2):

1. Disconnect each SDH vent line from the CRW system and provide a positive vent to atmosphere.

2. Cross-connect the SDH vent lines inboard of the vent valves.

3. Provide an atmospheric sump tank for collection of SDIV drains.

4. Disconnect the SDIV drain from the CRW system and provide a positive drain path (downstream of the drain valve) to the atmospheric sump tank.

These modifications should be accomplished during the next refueling outage of units 2 and 3, presently scheduled for the fall of 1980. For unit 1, it is NSRS' understanding that a NUREG-0576 outage is planned for the fall of 1980. The modification should be accomplished on unit 1 at that time.

E. Modification to Scram Discharge Instrument Volume

Modifications should be made to the control rod system such that drainage from the 6-inch SDH into the SDIV is assured regardless of a functional vent path. Two modifications are essential to accomplish this (see figure 2):

1. An extension should be added to the east side SDH end. The extension would turn downward to form a new east side SDIV. The new tank drain line would be configured as described in item D.

2. The exisiting SDIV should be modified to be identical to the new east side SDIV.

In addition, NUC PR should consider a means to monitor SDIV drainage flow rate, such as by periodially closing the SDIV drain valve or monitoring sump fill rate in an independent tank such as the atmospheric sump tank recommended in III.D.3.

Besides ensuring scram discharge volume drainage independent of a functional vent path, these modifications will increase the capacity to handle and detect leakages into the SDH and decrease the post-scram drain time. These modifications should be accomplished in the time frame specified in item D.

F. SDIV Level Detectors

1. NUC PR should continue the present flush commitment for the SDIV level switches;

2. Qualified differential pressure transmitters should be substituted for the present level switches;

3. The sensing arrangment for SDIV level detectors should be disconnected from the drain lines and routed instead to the SDIV tank.

4. NUC PR should provide a diverse, highly reliable and repeatable means to monitor for accumulation of water in the SDIV. Because of adverse effects of corrosion fines on displacement sensors such as float switches or differential pressure transmitters, strong consideration should be given for providing a UT monitoring system as one of the monitoring means. (Note this item was not discussed at the site exit interview.)

These modifications should be accomplished in the time frame specified in item D.

IV Status of Open Items

This review encompassed numerous tests, evaluations, modifications, and operational and maintenance procedures and data involved in evaluating and providing additional protection for operation of the Browns Ferry Nuclear units. Plant actions were considered appropriate with the following exceptions:

Open Items

A. R-80-13-BFN-01, Ultrasonic Monitoring System

This item remains open for NSRS tracking purposes. (See sections III.A.1 and V.A for details.)

B. R-80-13-BFN-02, Procedures for Ultrasonic Monitoring System

This item is open pending action by NUC PR on recommendation III.A.2. (See section V.A for details.)

C. R-80-13-BFN-03, Potential for Degradation of UT Equipment

This item is open pending action by NUC PR on recommendation III.A.3. (See section V.A for details.)

D. R-80-13-BFN-04, Calibration of UT Sensors

This item is open pending action by NUC PR on recommendation III.A.4. (See section V.A for details.)

E. R-80-13-BFN-05, Dual Input UT Monitor

This item is open pending action by NUC PR on recommendation III.A.5. (See section V.A for details.)

F. R-80-13-BFN-06, Clean Radwaste (CRW) System Vents

This item is open pending action by NUC PR on recommendation III.B. (See section V.B for details.)

G. R-80-13-BFN-07, CRW Vent Path Verification

This item is open pending action by NUC PR on recommendation III.C. (See section V.C for details.)

H. R-80-13-BFN-08, Modifications to Clean Radwaste and Control Rod Drive Systems

This item is open pending action by NUC PR on NSRS recommendations for modifications described in section III.D. (See section V.D. for details.)

I. R-80-13-BFN-09, Modifications to Scram Discharge Instrument Volume

This item is open pending resolution by NUC PR of NSRS recommendations as listed in section III.E. (See section V.E for details.)

J. R-80-13-BFN-10, SDIV Level Switches

This item is open pending action by NUC PR on recommendation III.F. (See section V.F for details.)

V. Details

Onsite portions of this review were conducted during the intervals of July 3-7 and 23-25, 1980. Listings of documents reviewed and personnel contacted are provided in sections VI and VII respectively. In the initial review, a NSRS reviewer reviewed the operating documents provided in reference A, observed portions of special testing performed under STI's 176 and 177, and monitored preparation of the "Report and Safety Evaluation of the Browns Ferry Unit 3 Partial Scram Failure of June 28, 1980," (reference B). This safety evaluation, which provided justification for restart of BFN-3 following the incident on June 28, was concurred in by NSRS upon completion. Discussion of specific details observed during the second review follows below.

A. Ultrasonic Monitoring Methods and Results

Compliance with IE Bulletin No. 80-17 was met initially by monitoring the scram discharge headers (SDH) and scram discharge instrument volume (SDIV) using a manually manipulated ultrasonic probe operated in the pulse-echo mode. During performance testing per STI 176, as well as by verification testing performed on a calibration standard, it was observed that the ultrasonic test (UT) monitoring system provided a distinctive, accurate response for determining the level of water in a 6-inch SDH. This method was therefore acceptable for daily monitoring on a temporary basis. However, this technique also had four notable drawbacks: (a) since the UT system was sensitive only to an air-water interface, indications of water might be overlooked (or be undetectable, as with a full header), (b) the system could not be used continuously, (c) personnel operating the system were receiving substantial doses of radiation while monitoring the radioactively hot SDH piping daily, and (d) only experienced UT personnel could operate the monitor.

During the second onsite review, the NSRS observed that an inplace, continuour monitor had been substituted for the manual system. Observation of the UT video and strip chart presentation were being made at least once per 30 minutes. According to station personnel, functional tests were made once per 8 hour shift and calibration checks, once per day. The UT procedure, attendant certification packages, system monitors, and operating procedures were observed and discussed with operations and UT personnel. The implementation of UT monitoring and maintenance seemed to be adequate with some reservations. It was observed that the maintenance/operating procedures for the UT devices consisted of a NUC PR division procedure (reference D) lacking specific plant approval and an internal memorandum (reference C). Operations personnel were instructed to review the continuous UT chart on each unit at least once per 30 minutes. It was determined that operations personnel had been provided with specific training concerning UT displays and that operators should be able to recognize displays that would indicate an accumulation of water, despite the transducer echoes and RF interferences that were in evidence on some of the video screens and strip charts. The monitoring by operations personnel, in general, appeared to be effective and timely.

Certification of UT equipment and the frequency of functional tests and calibrations seemed very good, as should be expected considering the importance of the UT system to plant safety. The reviewers observed that two UT monitors were in use to monitor opposing (east and west) SDH piping on unit 3, whereas a single monitor was in use in each of the other units. The use of a strip chart to provide a continuous record of UT data was a very significant improvement in UT monitoring technique. Concerning units 1 and 2, a UT examiner stated that although both SDH's were being monitored by the single UT video scope through two concurrent inputs, the scope and connected strip chart would respond to indications of water in either SDH. However, a demonstration also showed that failure of either input would result in no change on either the video or strip chart displays. Backup verification of this deficiency may have been evident in the scram data for a unit 1 scram which occurred at 2108 on July 22, 1980 (reference E). This data appeared to match the dynamic response expected of SDH drainage characteristics. However, it appeared that after the west header drained, the instrumentation indicated total system drainage for about 15-20 minutes, until someone connected the east header lead, at which time the instrument showed that the east header was half drained. It was further determined that since each UT sensor is embedded permanently in a small quantity of couplant applied to the underside of the SDH, the "system" functional tests conducted once per shift do not include the sensor and signal cable. Instead, the functional test demonstrates that the UT scope and strip chart respond properly to an input made via a separate test probe and signal cable. The once-per-day system calibration is performed by demonstrating that a spare UT scope and the UT monitor both respond similarly to the same input (i.e., the associated SDH sensor) and that the signal gain of the UT system has not changed appreciably since initial calibration.

In reviewing UT calibration and operating data, the reviewers noted that data was retrievable, although it was not being filed in secure storage as QA records would be maintained. It was noted that there is one constant indication of water in the SDH/SDIV system: on unit 3, a UT probe indicates daily that there is water to an indicated level about 1.6 inch in the horizontal portion of the 2-inch SDIV drain line. The NSRS reviewers were not able to review the UT strip chart resultant from the manual scram conducted July 23 per IE Bulletin No. 80-17 on unit 1. However, station personnel stated that the chart showed very good agreement with expected conditions in the SDH during and after the scram.

The reviewers requested data on the qualification of UT probes and couplant to withstand the high radiation and thermal loading received from the SDH piping. No hard data had been obtained through initial plant inquiries to the vendor. The reviewers also inquired about potential damage or disruption of the probe-couplant-pipe interfaces from severe acceleration caused by scram discharge events. The station had no data on this matter. However, a UT examiner stated that loss of couplant interface from any of the three stated causes would result in a prominent change in the UT video display for that UT transducer. Thermal or radiation-induced deterioration of the sensor media might be seen as a change in system gain during the daily calibration tests.

The following was concluded from this review:

1. The continuous UT monitoring technique in present use at BFN provides a considerably better means to monitor the SDH piping than the once-per-day check originally used. This method is acceptable in the short term--that is, until design changes to the scram discharge headers have been completed. However, since the causes of the two incidents at BFN may not have been found and corrected, site management and personnel must continue a very conscientious effort to maintain and monitor the UT system properly.

2. Plant procedures have not been prepared and approved to formalize requirements for all aspects of this program, including operating and maintenance (calibration and functional testing) requirements. Such procedures should address recordkeeping requirements and provide examples for recognition of normal and abnormal displays.

3. The UT sensors may not be adequately qualified for thermal, radiation, and acceleration effects.

   (Note: At the exit interview, a senior plant management representative stated that assignments had been made to prepare applicable plant procedures, even though use of a division procedure and internal memorandum are permissible. He also committed to inquire into the qualifications of the UT sensors through NUC PR division management.)

4. The UT sensor and signal lead cannot be given a valid functional test except when a scram discharges water into the SDH's. No calibration can be performed in place.

5. (This finding was not discussed at the exit interview.) On units 1 and 2 the single UT monitors appear to be deficient in one aspect. Loss of either of the dual inputs does not cause a significant change in the display. Also, the functional and calibration tests as performed do not verify that a UT input lead "makes up" properly when installed on the video monitor.

B. Clean Radwaste System (CRW) Vents

The "positive" vents installed in the CRW system in all three units were observed by the inspectors. Physically, two different types of devices were installed by the station. One device was an internal filter-liner similar in shape to an OBA (oxygen breathing apparatus) canister; the other was a filter container of a kind used on particulate air samplers. The positive vent paths were connected with 1/4-inch stainless steel tubing to a 4-inch CRW standpipe on each CRD bank. There were two (east and west) positive CRW vents on each BFN unit except for unit 3, where a second vent had been installed on the east side CRW headers. Both unit 1 positive vents were of the OBA canister type. There was water and some steam evident in the opening of the west header vent. Both east and west header vents appeared to be stained from release of water from the CRW system. All the canister-type vents contained a rubber diaphragm flap which would prevent a discharge from the CRW to the reactor zone. In response to inquiries, the NSRS was told that the size and capacity of the positive vents had been determined during conversations with the NRC (no EN DES participation) and that the vents were installed under trouble reports (TR's) with field change request (FCR) followup. The use of TR's was said to be acceptable because the CRW was a non-CSSC system. The FCR followup was being used because of unusually high interest in the incidents on BFN 1 and 3. No unreviewed safety question determination (USQD) was available on site. Site management stated that EN DES had concurred in the FCR for this alteration, thereby ensuring that a USQD had been completed in accordance with EN DES procedures. On July 25, 1980, a work plan containing the FCR for this design change was reviewed briefly. The safety evaluation required by BFA-28 was not found in this work plan. At the time of this observation, plant personnel stated that the rubber diaphragms had been removed from OBA canister-type vents following discussion with NSRS. The diaphragms did not meet the requirements of a positive (clear of <u>any</u> obstruction) vent path.

It was concluded from this review that use of trouble reports to effect a plant alteration (i.e., add positive vents in the CRW system to satisfy requirements of IE Bulletin No. 80-17) was unacceptable. This conclusion is based on the fact that the alteration was intended to affect a CSSC--namely, the CRD scram discharge system. Further, the use of a TR circumvented the preparation of a USQD by EN DES or a safety evaluation by the station (for a DCR or FCR, respectively) prior to performing the modification.

C. <u>CRW Vent Path Verification</u>

The reviewers traced out and felt all exposed portions of the reactor zone CRW system in units 1, 2, and 3 that were not included in contaminated zones. This search was made because of the vacuum condition that occurred in the unit 1 SDIV and because of the steam-water effluent noticed on a unit 1 positive vent device. Steam in the CRW system was evident in all three reactor zones, but especially in unit 1, where a strong venting of steam was noted at the 565' elevation where two 4-inch CRW standpipes had faulty seams on the cap welds.

Except for a slight venting at a chemical sample sink on the 621' elevation, the CRW system is designed to vent/drain into a submerged termination in the reactor building equipment drain tank (RBEDT). No evidence of strong release of steam was seen in units 2 and 3, although small portions of the system piping were warm to hot to the touch. However, in unit 1 the RBEDT was steaming constantly and the major portion of the system at the 565' elevation was extremely hot to the touch, indicating a pressurized steam blanketing in the CRW headers into which the SDH/SDIV vents and drain exhaust. A senior member of the plant management stated that the source of the unit 1 steam leaks had been localized to one of four locations by an intensive investigation and that corrective maintenance has been planned to correct this condition.

Following installation of the positive vent paths discussed in item B above, testing of the path of communication of the vent was attempted by forcing air from an open section of the SDH vent line into the CRW system and also into the SDH. A reverse leak check of each SDH vent valve was performed during this testing. No blockage was found between the SDH and CRW headers. Based on the known facts that a vacuum has occurred in the unit 1 SDIV and that water is known to stand in a horizontal portion of the unit 3 SDIV drain, this test was not a satisfactory functional test of the SDH vents. Instead of injecting pressurized air, the test should have made an attempt to pull a vacuum from the direction of the SDIV. Inability to draw a vacuum would have proved that a clear air path existed between the vent and drain.

D. Modifications to Clean Radwaste and Control Rod Drive Systems

The CRW system exerts a complicated influence on the 6-inch SDH/SDIV subsystem as they function to accumulate and discharge water received from a unit scram. In the present configuration, the vents and drain for the subsystem are piped to the CRW system, which is closed except for its submerged discharge, and the "positive vents" added per IE Bulletin No. 80-17. A considerable and most important portion of the scram discharge volume subsystem is composed of each 6-inch SDH volume and its 2-inch drain line to the SDIV. In all three BFN units, the east-side SDH's and their drains have a very gradual slope and extensive, constricted length of flow path, which results in a lengthy drainage time (20-30 minutes) under the most favorable of conditions. The information that a quantity of water may reside in a horizontal portion of SDH/SDIV vent or drain line (see section A above) and that differential pressures can exist between local points in the same CRW header (BFN-1 incident of July 19) gives rise to a serious concern regarding the need to disconnect the CRW system altogether from the scram discharge portions of the CRD system. For example, in case of a pressurized CRW system, as in unit 1, addition of a positive vent to the SDH does not remove the possibility of some form of liquid lock on the SDIV drain line. Also, the failure of a non-redundant SDH vent valve, as occurred at Dresden unit 3, could result in accumulation of water in the SDH piping.

From a review of the system physical configuration, NSRS concluded that several modifications are required to promote prompt, reliable drainage of the SDH's. (See section III.D for details.)

E. Modifications to Scram Discharge Instrument Volume

Results of performance tests and the Dresden unit 3 incident reported in IE Bulletin No. 80-17, supplement No. 2, have demonstrated an unacceptably long interval to drain the east-side SDH via its narrow bore, gently sloped, and extremely long drain connection to the SDIV. The BFN-3 tests also revealed that even under the best of conditions, east header drainage rate cannot result in an accumulation of water in the SDIV. With the east vent or drain line blocked, or for other causes (such as interactions of the CRW system), it is quite possible to have a nearly full east-side SDH with no indication of abnormality on the SDIV level monitoring system. Because of the near prohibition on shutting the SDIV drain valve for intervals in excess of one hour required by IE Bulletin No. 80-14, monitoring of scram discharge system leakage rate is not presently possible.

It has been concluded from this review that there should be an SDIV for each SDH, that the interconnection between the SDH and its SDIV should be of minimum length, and that the interconnection should be of the same size as the SDH so as to eliminate potential line restriction. With proper design, such a configuration would ensure that leakage will drain to the SDIV regardless of the availability of an external vent path.

It is recommended that NUC PR consider a means to monitor SDIV drainage flow rate, such as by providing a monitored sump tank or by periodically closing the SDIV drain valve. (This suggestion was not discussed at the site exit meeting.)

F. <u>SDIV Level Switches</u>

Recently, SDIV level switches have failed to operate properly on several occasions at BFN. In discussions with site personnel, it was learned that problems have been due to deposition of corrosion fines in the float-type switches currently in use. An analysis of fines recovered from the BFN-3 SDIV level switches during a flush shows presence of no unusual constituents in the fines. Site personnel have accelerated the frequency of calibration of these switches to once per month and following each scram. Replacement of these switches with a differential pressure transmitter is overdue, except that no replacement transmitter presently available is qualified to meet applicable standards. Based on difficulties associated with corrosion fines, use of such transmitters alone may not be an ideal solution to the present problem.

Scram data have shown that SDIV level switches A and B, which receive a low side input from the SDIV drain line provide invalid indication during drainage of the SDIV through the drain line. There is a possibility, although very remote, that this bias, coupled with failure of either a C or D level switch, could fail to induce a scram should high level occur in the SDIV during normal operation.

For these reasons, NSRS reached the following conclusions.

1. The present flush commitment for the SDIV level switches is necessary to continued plant safety.

2. The SDIV level switches are very unreliable.

3. The sensing arrangement for SDIV level instruments A and B could inhibit the safety function of these switches under certain circumstances.

4. The SDIV lacks a diverse, highly reliable and repeatable means to monitor for accumulation of water in the SDIV. Because of adverse effects of corrosion fines on displacement sensors such as float switches or differential pressure transmitters, strong consideration should be given for providing a UT monitoring system as one of the monitoring means.

G. **Emergency Measures**

Following the initial scram of unit 3 on June 28, 1980, the operator reset the Reactor Protection System (RPS) logic three times to allow sufficient drainage of the SDV so that repeated scrams completed insertion of all rods. Had a persistent scram signal (such as low reactor water level or high drywell pressure) occurred, reset would have been prevented, thereby preventing reopening of the SDV drain and vents. In such an event rods could still be inserted by manual controls one at a time. Provision for this alternate mode of insertion is made in EOI-47. The NSRS considered whether by procedure a means should be identified to override a scram signal in order to allow emergency reset of RPS logic for the sole purpose of repeating control rod scrams. It was concluded that such a procedure is not required. This is based on the following considerations:

1. Reset capability is not a design basis function of the RPS logic and

2. Recurrence of the type of event occurring on unit 3 is unlikely due to the use of UT monitoring devices and forthcoming system modifications.

VI. **Personnel Contacted**

H. L. Abercrombie, Plant Superintendent
*A. L. Burnette, Shift Engineer
*T. L. Chinn, Mechanical Engineer, NUC PR Operations Section
R. E. Edmondson, Lead Electrical Maintenance Engineer
*J. L. Harness, Assistant Superintendent
*G. T. Jones, Assistant Branch Chief, Outage Management Branch
*R. G. Metke, Results Supervisor
J. W. Miller, Assistant Outage Director
K. L. Montgomery, Instrument Maintenance Foreman
*E. D. Nave, Reactor Engineer (Acting)
L. Parvin, Engineering Aide (NDE Examiner, UT-Level II)
J. R. Pittman, Instrument Maintenance Supervisor
*R. T. Smith, Quality Assurance Supervisor
*J. E. Swindell, Outage Director
*J. A. Teague, Electrical Maintenance Supervisor
W. C. Thomison, Assistant Results Supervisor

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*J. W. Chase, NRC Resident Inspector

*Present at exit interview.

VII. **Documents Reviewed**

A. Data package of BFN-3 charts and recorder outputs (assembled by BFN management) pertinent to the BFN-3 shutdown on June 28, 1980.

B. "Report and Safety Evaluation of the Browns Ferry Unit 3 Partial Scram Failure of June 28, 1980," issued July 6, 1980.

C. Memorandum dated July 18, 1980, from N. L. Abercrombie to R. T. Smith and J. B. Studdard, "Ultrasonic Monitoring on the Scram Full Discharge Headers."

D. NUC PR Division Procedure N-UT-11, "Ultrasonic Examination for Detecting and Measuring Fluid Levels in Austenitic and Ferritic Systems" (Procedure, Equipment Certifications, and Data File).

E. Chart recording of UT data for scram occurring on unit 1 at 2108 on July 22, 1980. UT data obtained during routine operation of BFN-1, -2, and -3 on July 23-24, 1980.

F. Special Test Instruction (STI) 174, Unit 3 only, "Special Testing for CRD System."

G. Summary of Test Results for STI's 176 and 177, prepared by F. D. Kelly.

H. Special Test Instruction (STI) 176, Unit 3, "Control Rod Drive and Scram Discharge Volume System Test."

I. Special Test Instruction (STI) 177, Unit 3, "Scram Discharge Volume System Test.

J. "Report of Results, Special Test 182, Implementation of NRC IE Bulletin 80-17, Browns Ferry 1."

K. Mechanical Maintenance Instruction (MMI) 14.3.13-C, Units 1 and 2, "Inspection of Control Rod Drive 1-Inch Vent Line from Scram Discharge Volume Headers."

L. Memorandum from M. N. Sprouse to J. R. Calhoun dated July 15, 1980, "Browns Ferry Nuclear Plant - Preliminary Identification of Design Changes - Scram Discharge Subsystems," (NEB 800715 273).

M. Special Maintenance Instruction (SMI) 150, "Scram Solenoid Relay Response Times" (Test Document and Data).

N. Special Electrical Maintenance Instruction (SEMI) 19, Unit 3, "Scram Pilot Valve Electrical Independence Test" (Data Only).

O. Special Electrical Maintenance Instruction (SEMI) 19.2, Units 1, 2, and 3," Operation and Calibration of RPS Reset Timers" (Test Document and Data).

P. NRC IE Bulletin 80-17, "Failure of 76 of 185 Control Rods to Fully Insert During a Scram at a BWR."

Q. NRC IE Supplement Nos. 1 and 2 to Bulletin 80-17, "Failure of 76 of 185 Control Rods to Fully Insert During a Scram at a BWR."

R. Letter from James P. O'Reilly (NRC) to H. G. Parris dated June 30, 1980, "Confirmation of Action" (GNS 800707 157).

S. Letter from James P. O'Reilly (NRC) to H. G. Parris dated July 8, 1980 (GNS 800714 100).

T. Letter from Jimmy L. Cross (TVA) to James P. O'Reilly (NRC dated July 11, 1980, "Office of Inspection and Enforcement Bulletin 80-17 - RII:JPO 50-259, -260, -261 - Browns Ferry Nuclear Plant" (A27 800711 012).

U. Letter from L. M. Mills (TVA) to Harold R. Denton (NRC) dated July 16, 1980 (A27 800716 009).

V. Letter from Darrell G. Eisenhut (NRC) to Hugh G. Parris dated July 18, 1980 (GNS 800724 103).

W. TVA 45D dated July 23, 1980, from W. C. Thomison to BFN Management, "CRD Crud Metal Analysis."

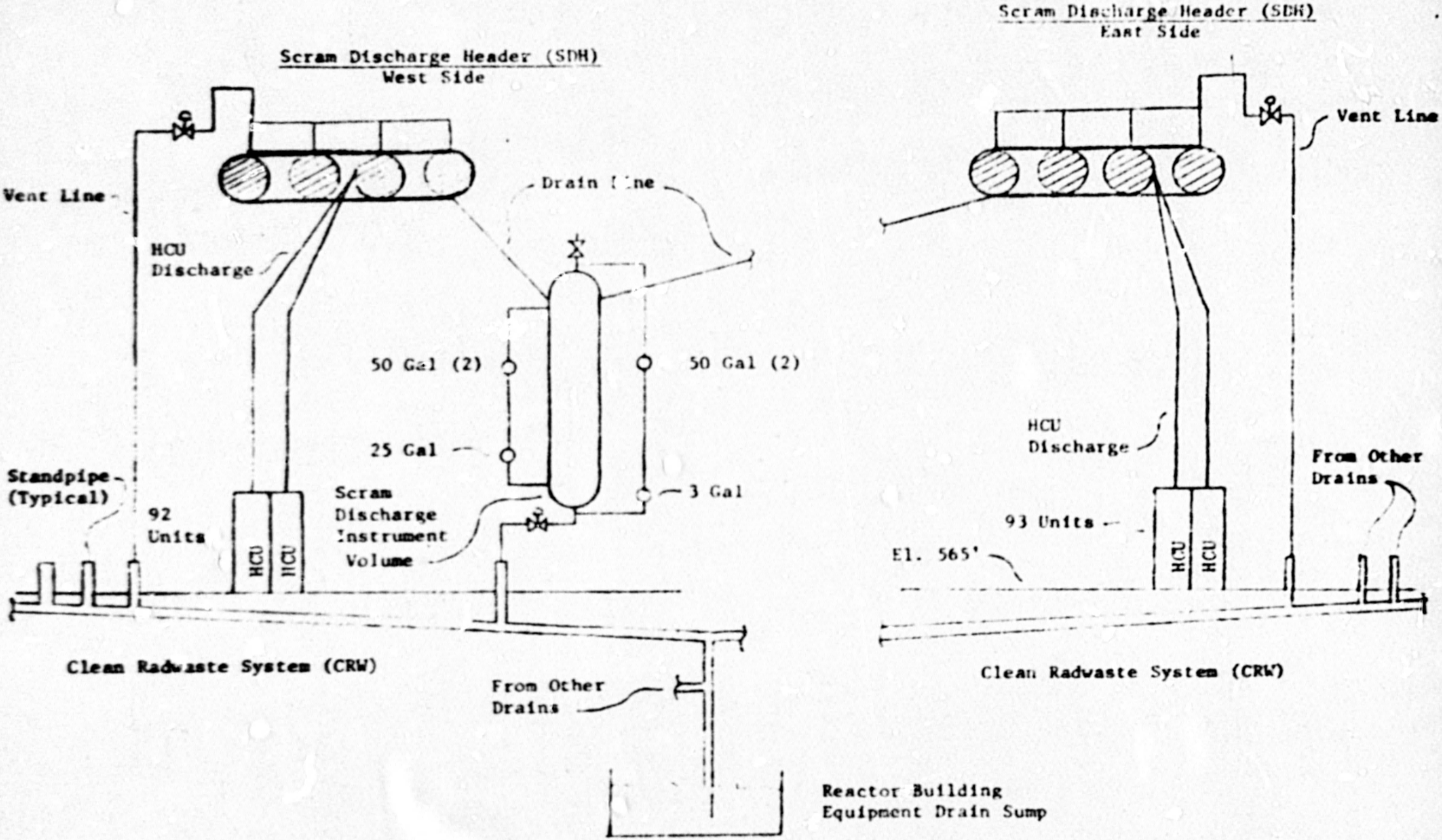
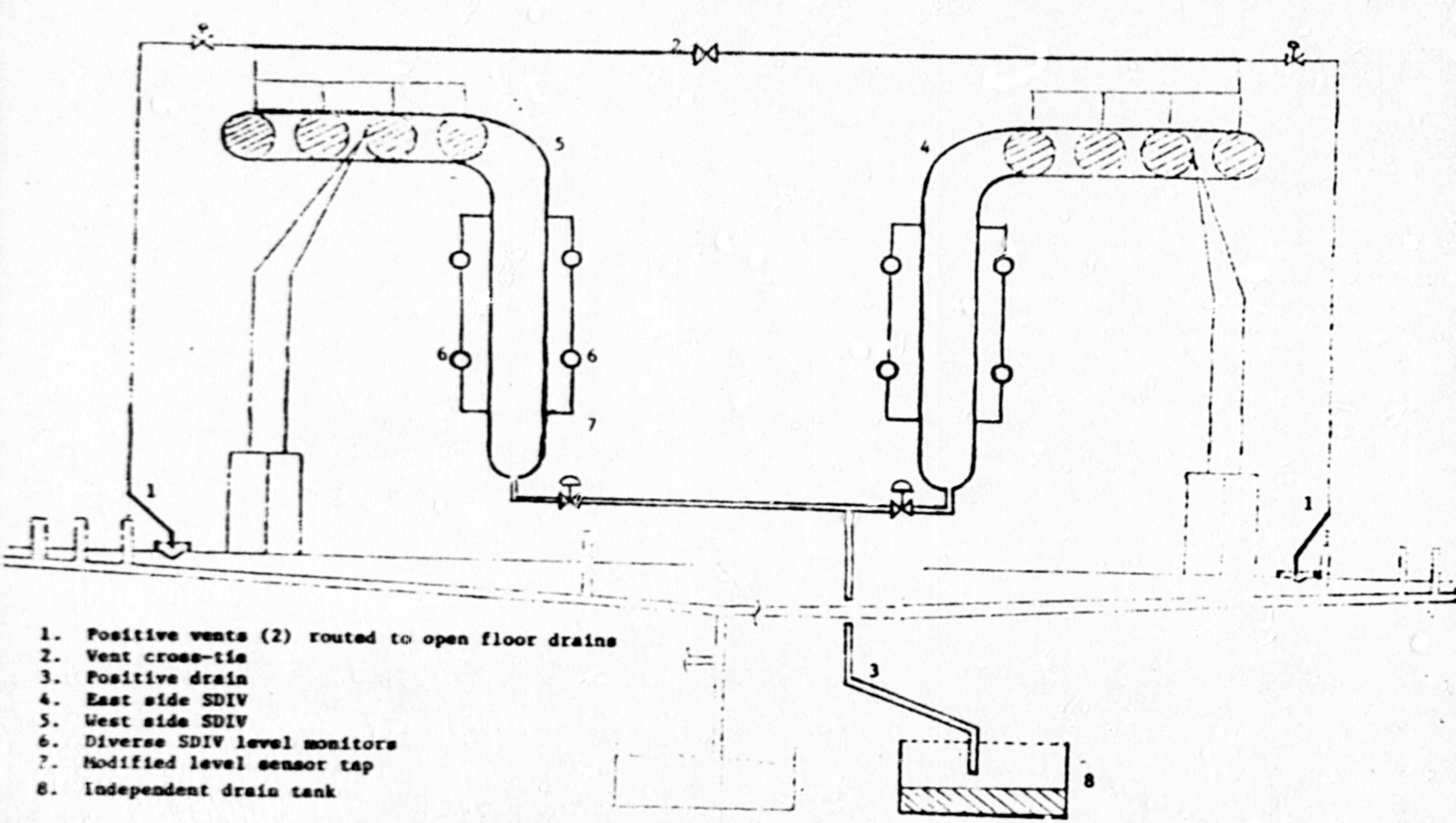


FIGURE 1- CONTROL ROD DRIVE SYSTEM (Not to Scale)



1. Positive vents (2) routed to open floor drains
2. Vent cross-tie
3. Positive drain
4. East side SDIV
5. West side SDIV
6. Diverse SDIV level monitors
7. Modified level sensor tap
8. Independent drain tank

FIGURE 2 - MODIFIED CONTROL ROD DRIVE SYSTEM (NOT TO SCALE)

TVA 64 (OS-9-69)
UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

GNS '81 1007 050

TO : H. J. Green, Director of Nuclear Power, 1750 CST2-C

FROM : H. N. Culver, Director of Nuclear Safety Review Staff, 249A HBB-K

DATE : October 7, 1981

SUBJECT: WATTS BAR NUCLEAR PLANT UNIT 1 - NUCLEAR SAFETY REVIEW STAFF REPORT NO. R-81-20-WBN

Attached is the NSRS report of a routine review conducted on selected dates at WBN during the period August 31 - September 18, 1981 in the area of preoperational testing. This review was described in my memorandum to you dated August 26, 1981 (GNS 810826 052).

Two previous recommendations were closed, and no new items were identified during this review.

Cooperation at the plant was excellent. This consideration is appreciated. If you have any questions regarding this report, please contact H. Randall Fair at extension 6590 in Knoxville.

H N Culver

H. N. Culver

HRF:LML
Attachment
cc (Attachment):
C. C. Mason, Watts Bar Nuclear NUC PR
MEDS, 100 UB-K

NR 8

NSRS FILE

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

REVIEW

NSRS REPORT NO. R-81-20-WBN

Subject: TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT - UNIT 1
ROUTINE REVIEW

Dates of
Onsite Review: August 31-September 18, 1981

Reviewer: ______________________ ______________
H. Randall Fair Date

Approved: ______________________ 10/7/81
Kermit W. Whitt Date

TABLE OF CONTENTS

I. SCOPE

This was a routine review of selected activities at Watts Bar Nuclear Plant, Preoperational Test Section. The review included preoperational test conduct and preoperational test instruction preparation.

II. CONCLUSIONS

Preoperational testing witnessed was performed with no observed deficiency. Personnel experience and confidence in systems integrity gained during "dry run" tests contributed significantly to this successful testing effort. (Reference V.B for details.)

III. RECOMMENDATIONS

None

IV. STATUS OF PREVIOUSLY IDENTIFIED ITEMS

A. R-81-16-WBN-01, Conduct of Preoperational Tests

This item remains open until test personnel have been reinstructed in the area of change sheet control. (Reference section V.A.1 for details.)

B. R-81-16-WBN-02, Documentation of Test Deficiencies

Test personnel were reinstructed in the necessity of timely completion of test deficiency documentation. This item is considered closed. (Reference section V.A.2 for details.)

C. R-81-16-WBN-03, Test Record Information

Further examination of the established interface between NUC PR, CONST, and EN DES in the area of drawing control revealed that the actions of NUC PR were appropriate. This item is considered closed. (Reference section V.A.3 for details.)

V. DETAILS

A. Review of Previously Identified Items

1. R-81-16-WBN-01, Conduct of Preoperational Tests

The Preoperational Test Section supervision plans to retrain test personnel on the necessity of continuity between test procedures and change sheets to test procedures in establishing the required prerequisites and initial conditions for test conduct. This item will remain open until the retraining is complete.

2. R-81-16-WBN-02, Documentation of Test Deficiencies

Retraining of test personnel was performed to stress the necessity of timely completion of documentation describing deficiencies discovered during test conduct. This item is closed.

3. R-81-16-WBN-03, Test Record Information

Upon further investigation into the control of test record drawings, it was determined that this recommendation was misdirected to NUC PR. This area will be the subject of future reviews of CONST. This item is considered closed.

B. Conduct of Preoperational Tests

Major portions of Preoperational Test Instruction 13B, "Onsite AC Distribution System (Diesel Generator Loading Logic)," were observed. The test was an integrated test involving automatic functions of systems required to operate during an accident and/or a station blackout. The inherent nature of such a test would dictate that problems in coordination and communication as well as with equipment would be experienced. This was not the case. The test director stated, as was readily apparent, that dry runs of the test had been performed prior to the official run. Consequently, the testing progressed smoothly and was accomplished in less time than had been estimated without any significant deficiencies.

Areas of Preoperational Test Section activity that were observed during conduct of this test included the following:

- Transfer of portions of systems required for the test.
- Control of test record drawings and the approved official copy of the test procedure.
- Adequacy and completion of prerequisites.
- Control established by the test director of the areas in the plant where the test was to be performed and the systems being tested.
- Coordination with shift operations supervision with regard to system availability and operation.
- Control of procedure change sheets.
- Identification and documentation of test deficiencies.
- Resolution of acceptance criteria discrepancies.
- Procedure adherence.

•Operation of plant equipment.

•Reducing data and comparison to acceptance criteria.

•Planning and orientation sessions.

C. Review of Preoperational Test Instructions

A general review of the following Preoperational Test Instructions was performed prior to the site visit:

TVA 2B, "Containment Vessel Pressure and Leak Test (Penetrations)"

TVA 2C, "Containment Vessel Pressure and Leak Test (Isolation Valves)"

TVA 10, "Control Building Air Conditioning"

TVA 41, "Containment Isolation System"

Comments were given to the Preoperational Test Section supervisor and are included as an appendix to this report.

VI. PERSONNEL CONTACTED

M. K. Jones, Preoperational Test Section, Supervisor
R. H. Smith, Preoperational Test Section, Assistant Supervisor
J. A. Holmes, Test Director
W. G. Bogly, Test Director
W. Byrd, Shift Technical Advisor

VII. DOCUMENTS REVIEWED

A. Preoperational Test Instructions

1. TVA 2B, "Containment Vessel Pressure and Leak Test (Penetrations)"

2. TVA 2C, "Containment Vessel Pressure and Leak Test (Isolation Valves)"

3. TVA 10, "Control Building Air Conditioning"

4. TVA 41, "Containment Isolation System"

5. TVA 13B, "Onsite AC Distribution System (Diesel Generator Loading Logic)"

B. TVA scoping documents for procedures listed in item VII.A.

C. Preoperational Test Section files for procedures listed in item VII.A.1-4.

D. Various test record drawings.

APPENDIX

TVA 2B

- Procedure does not provide for CONST verifying as-built drawings.

- Procedure gives test pressure to be used as:

  1. 12.8 psi greater than containment pressure for the absolute method and mass flow method.

  2. At least 15.4 psig for the rotameter method.

  The scoping document says, "Type B and C tests shall be conducted at a pressure greater than 14.5 and less than 15.5 psig."

- There is no precaution for maintaining ambient tempreature 30° F as required in the scoping document.

TVA 2C (draft)

- There are no prerequisites for flushing or cleaning lines per scoping document.

- There are no instructions given on determination of leakage for piping penetrations with more than two valves (see scoping document section 7.5).

- Appendix H references step 2.2.1--should be 2.1.1.

- Step 2.3.2 has no signoff.

- No environmental condition precaution for maintaining temperature 30° F.

TVA-10

- Prerequisites do not include control room habitability zone penetration seals being in place.

TVA-41

No comment.

TVA 64 (OS-9-69)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

GNS '81 1204 051

TO : H. G. Parris, Manager of Power, 500A CST2-C

FROM : H. N. Culver, Director of Nuclear Safety Review Staff, 249A HBB-K

DATE : December 4, 1981

SUBJECT: SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2 - NUCLEAR SAFETY REVIEW STAFF REVIEW REPORT NO. R-81-24-SQN

Attached is the NSRS report of a routine review conducted at SQN during the period October 13 through November 6, 1981 regarding followup of previously identified NSRS items and review of activities related to the unit 2 startup test program. The report is the result of site visits described in my memorandums to H. J. Green dated August 5, 1981 (GNS 810806 051) and October 5, 1981 (GNS 811005 050).

Our review resulted in closure of two previously identified items, R-80-05-SQN-09 and R-81-01-SQN-02, and identification of two new concerns, R-81-24-SQN-01 and -02, requiring NUC PR resolution. No specific response is required from NUC PR on the two new recommendations; however, you are requested to schedule corrective action consistent with your prioritizing system. NSRS will follow the scheduling and resolution of these items during subsequent reviews.

In regard to item R-80-05-SQN-09, we are closing this item out since the potential safety problem associated with landing a helicopter in the vicinity of the diesel and ERCW lines has been reduced by the addition of a new landing pad. NSRS considers NUC PR's reaction to this concern has been one of nonresponsiveness and in so doing allowed a potentially hazardous and unsafe condition to exist for over a year. The storage of flammable materials in this area is still a concern NSRS will be following up on. As part of some of the recent changes POWER is making to improve safety, actions should be taken to assure timely response and resolution of such safety concerns. The details of this item as well as all other items raised or closed out are provided in section III of the attached report and correspond to applicable recommendations in section II.

If you have any questions regarding this report, contact R. C. Sauer at extension 4815 in Knoxville.

K. W. Whitt
for H. N. Culver

RCS:LML
Attachment
cc (Attachment):
A. W. Crevasse, 401 UBB-C
H. J. Green, 1750 CST2-C
MEDS, 100 UB-K
F. A. Szczepanski, 417 UBB-C

NSRS FILE

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

REVIEW

NSRS REPORT NO. R-81-24-SQN

SUBJECT: TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT - UNITS 1 AND 2
ROUTINE REVIEW

DATES OF
ONSITE REVIEW: *OCTOBER 13-16, 1981
+NOVEMBER 3-6, 1981

REVIEWERS: Robert C. Sauer — 1 Dec 81
+ROBERT C. SAUER — DATE

Robert C. Sauer for — 2 Dec 81
+*RONALD W. TRAVIS — DATE

APPROVED BY: K. W. Whitt — 12/3/81
KERMIT W. WHITT — DATE

TABLE OF CONTENTS

I. SCOPE

This was a routine review of site activities leading to and including initial criticality of unit 2 at Sequoyah Nuclear Plant (SQN). In addition, followup reviews of two previously identified NSRS open items were conducted for closeout.

II. CONCLUSIONS AND RECOMMENDATIONS

The following paragraphs contain the conclusions followed by recommendations if applicable. An E or R in brackets has been placed at the end of each recommendation. The [R] indicates that NSRS has concluded the recommendation is based on a regulatory requirement or a TVA commitment. The [E] indicates NSRS has determined that the recommendation has no regulatory basis. It is considered an enhancement and is based on subjective judgment.

A. R-81-24-SQN-01, Inadequate Procedural Controls in Installing the Unit 2 On-line Reactivity Computer

The bistables of the power range channel used to input the reactor flux level to the reactivity computer for unit 2 startup testing activities were not tripped when the channel was removed from service. The cause of this deficiency was either failure to follow the procedure (TI-67) being used to trip the inoperable power range channel or having an inaccurate procedural step NOTE in the reactivity computer setup procedure (TI-25).

Recommendation

To remove the possibility of ever reusing the incorrect guidance given in Technical Instruction TI-25 during the remaining unit 2 low-power physics startup test program or to mislead WBN should they use SQN's procedure in developing their own startup test instructions, the SQN plant staff should revise TI-25 by deleting steps 2.2, 2.3, 2.4, 7.4, 7.5, and 7.6 of paragraph 4.B. Deletion of the these steps would also make TI-25 compatible with TI-67, referenced for performance in TI-25, which completes the same required action for the affected power range channel. (See section III.C.2.a for details.) [E]

B. R-81-24-SQN-02, Inadequate Trip Switch Identification Utilized in TI-67

The trip switch identification nomenclature in TI-67 is not consistent with current plant switch identification practices. NSRS found cases where the plant unique identification for trip switches was not being utilized necessitating review of instrument tabs to correlate vendor supplied component identification to the corresponding plant switch designation.

Recommendation

The SQN plant staff should evaluate TI-67 and other applicable procedures for switch/component identification consistency and usefulness to the plant employee utilizing this information. (See paragraph III.C.2.b for details.) [E]

C. The SQN plant staff appears to be taking necessary precautions to ensure data, graphs, recordings, and identification of unit 2 startup test interruptions are being properly marked and identified. This had been an administrative concern previously found deficient by NSRS involving unit 1 startup test activities and was addressed in NSRS Report No. R-80-20-SQN.

III. STATUS OF SELECTED PREVIOUSLY IDENTIFIED ITEMS

The following paragraphs contain summarized statements of action taken by the Office of Power (POWER) in resolving previously identified NSRS items. The items presented do not represent the total of our concerns but only those reviewed during this report period.

A. (Closed) R-80-05-SQN-09, <u>Temporary Helicopter Pad Located Between the Diesel Generator and Auxiliary Buildings</u>

The lack of corrective action taken by NUC PR to resolve this item was considered nonresponsive; however, the recommendation and the item are being closed out because the hazard and safety conditions initially noted by NSRS no longer exists. (See section IV.A.1 for details.)

B. (Closed) R-81-01-SQN-02, <u>Failure of OPQAA to Perform Audit Function</u>

Corrective action taken by OPQAA to resolve this item was considered sufficiently responsive to close this item out. (See section IV.A.2 for details.)

IV. DETAILS

A. <u>Previously Identified Open Items</u>

1. R-80-05-SQN-09, <u>Temporary Helicopter Pad Located Between the Diesel Generator and Auxiliary Buildings</u>

This item involved requesting the Division of Nuclear Power (NUC PR) to take measures to assure that a temporary helicopter landing facility was not used until an analysis could be made to determine the effects of a helicopter crashing into this area. In addition, NUC PR was requested to perform an analysis and to implement any corrective actions needed associated with the storing of petroleum distillates in this area. The details of this item were provided in paragraph IV.C.5 of NSRS Report No. R-80-05-SQN dated June 27, 1980 with recommended corrective action provided in recommendation paragraph III.A.7.

This was the third time NSRS had reviewed this item since it was first identified. The previous two followup reviews were performed as NSRS review reports R-80-11-SQN and R-81-12-SQN, references V and Y respectively. This review indicated that nothing had been done to modify the landing pad to disallow its use. Further, the SQN plant staff had not conducted or documented a review of the possible safety affects involving loss of diesels and other ERCW serviced safety systems or components should a helicopter impact this area or petroleum distillates explode. NSRS therefore concludes that NUC PR's reaction to this item has been one of nonresponsiveness.

To determine what actions, if any, NUC PR had taken indirectly in resolving this issue, NSRS held discussions with Public Safety, Compliance, and with personnel in Transportation Services. These discussions resulted in our concluding that <u>presently</u> a low probability exists that the old pad will ever be mistaken for the proper landing area. This conclusion was based on the new landing area being constructed outside the protected area, quite distirct, and with markings quite dissimilar from the pad within the protected area. There are also landing lights and a wind sock at the new facility. In addition, the helicopter pilots will be given specific preflight briefings to ensure that they are fully aware of the markings on the new landing pad. On this basis, NSRS considers the item closed. NSRS will review plant flammable storage practices during our fire prevention/protection review planned for early 1982.

2. R-81-01-SQN-02, <u>Failure of OPQAA to Perform Audit Function</u>

This item involved failure of OPQAA to comply with commitments made to review the conduct of startup test activities associated with the unit 1 startup test program through formal audits as specified in section 14.1.1.2.1 of the Sequoyah FSAR. The details of this item were provided in paragraph V.B.2.a of NSRS Report No. R-81-01-SQN dated February 18, 1981 with recommended corrective action provided in recommendation paragraph IV.B.

Subsequent to the issuance of the R-81-01-SQN review report, NSRS and OPQAA held a meeting in the NSRS offices on April 2, 1981 (reference BB) to discuss the OPQAA audit role during startup testing activities at SQN.

Included in this discussion were two memorandums involving NSRS clarification of its concern (reference Z) and the OPQAA response (reference AA). The outcome of the meeting was that OPQAA agreed to either witness some of the remaining startup tests or to monitor the plant QA Staff's program for monitoring test conduct adequacy. In addition, consideration of this discussion would be evaluated for incorporation into the unit 2 startup testing audit program.

NSRS review of OPQAA activities subsequent to the meeting revealed that OPQAA had conducted an audit of the activities and documentation associated with startup testing for both units 1 and 2 during the period July 21-24, 1981. Though no startup test witnessing was accomplished on unit 1, an audit team comprised of OPQAA and NSRB specialists was assembled prior to the startup of unit 2 to monitor the plant's conduct of activities associated with achieving initial criticality. The OPQAA reviewers indicated that additional audits would be conducted to follow startup activities for this unit.

NSRS considers this effort to be very positive and finds the action taken appropriate for resolution of this item. NSRS will continue to monitor OPQAA activities and resultant reports associated with the startup testing of unit 2. This item is closed.

B. Unit 2 Precriticality Review

NSRS evaluated the activities of NUC PR in preparation for initial criticality in unit 2. The requirement/commitments with which these plant activities were compared came from the Sequoyah FSAR, section 14.1.4.2, tables 14.1-2 and 14.1-3 (reference R); NRC Regulatory Guide 1.68, section 2 (reference T); and the unit 2 license (reference H). There were several items yet to be completed prior to initial criticality, but through a combination of Field Services scheduling, the Compliance Staff COMTRAK Program (reference S), S.U.-7.1 (reference A), GOI-1 (reference F), GOI-2 (reference G), and CO-SQNP-SAP, Appendix C (reference E), it appeared that all requirements were being tracked.

C. Unit 2 Startup Activities Review

1. Pre-Startup Functional Test Results Review

The results of three completed pre-startup functional test surveillances were reviewed to ensure that:

a. Measured functional test results were within the established acceptance limits set by the surveillance instruction.

b. Those personnel charged with the responsibility for review and acceptance of the functional test data have documented their review and determination that the equipment was operable.

c. Performance of the surveillance was within the time requirements set by the unit 2 Technical Specifications.

Those completed surveillance instructions and associated functional tests reviewed were:

SI-93 Reactor Trip Instrumentation Functional Tests (prior to startup), R4, dated March 30, 1980, with associated functional test completions:

- IMI-92-SRM-FT Functional test of SRM channels completed November 2, 1981

- IMI-92-IRM-FT Functional test of IRM channels completed November 2, 1981

- FT-17 Functional test of turbine stop valve closure completed November 2, 1981

- FT-18 Manual reactor trip functional test completed November 4, 1981

SI-95 Functional Test of Intermediate and Power Range Neutron Flux Channels Prior to Low Temperature Physics Tests, R2, dated April 10, 1980 with associated functional test completions:

- IMI-92-IRM-FT Functional test of IRM channels completed November 4, 1981

- IMI-92-PRM-FT Functional test of PRM channels completed November 4, 1981

SI-279 Reactor Trip Instrumentation Functional Tests (with Rod Control System Operable and Reactor Trip Breakers Closed), R1, dated November 4, 1980 with associated functional/surveillance test completions:

- FT-19 Functional test of reactor trip breaker channels (automatic) completed October 30, 1981

- SI-268 Verification of P-4 interlock completed October 30, 1981

No major problems or deficiencies were noted.

2. Initial Criticality Performance Review

NSRS observed SQN's preparations and subsequent test performance conduct of startup test SU-7.2, "Initial Criticality," in achieving the initial nuclear reaction in the unit 2 reactor at 1025 CST on November 5, 1981. Initial criticality was achieved through dilution of the boron-treated primary coolant with all control rod banks withdrawn. The evolution was conducted in a controlled and efficient manner.

During the performance of the test, the NSRS observer verified that the proper revision of the procedure was in use, that all temporary changes had been complied with, all prerequisites had been signed accomplished, shift manning was proper and in accordance with the procedure and license conditions, that procedural steps were being properly signed off as they were accomplished, and that special test equipment required by the procedure was in use and had been calibrated. In addition, NSRS observed test data acquisition, the processing of this data for inverse multiplication plots to trend and to project the point at which initial criticality would be achieved, and test personnel performance in reviewing the final just critical data to determine if the test procedure's acceptance criteria had been met.

NSRS concerns associated with the performance of the initial criticality startup test involved the following:

a. R-81-24-SQN-01, Inadequate Procedural Controls in Installing the Unit 2 On-line Reactivity Computer

Subsequent to the startup, it was discovered by SQN plant operations personnel that the NIS power range channel (N-43) used to input the flux level to the reactivity computer did not have its bistables placed in the tripped condition when the channel was removed from service as required by Technical Specification table 3.3-1, action item No. 2. After review of the installing procedure, Technical Instruction TI-25, "Setup and Operation of the Reactivity Computer," the SQN plant staff concluded that steps necessary to complete the setup procedure were missing. These steps included using an external high voltage power supply to power the N-43 detector and removing its instrument fuses to trip its associated bistables. Temporary change 81-1847 to TI-25 was issued to make the additions.

NSRS review of this matter revealed that one of two possible causes brought about this event and both are contrary to the previously identified cause of missing procedural steps. The first involves the failure to remove the affected power range channel's control power fuses prior to connecting the detector's output to the reactivity computer as required by TI-67, "Procedures for Removing an Inoperable Protection Channel from Service," and referenced to be performed per paragraph 4.B, step 2.1 of TI-25. Had the control power fuses been removed as required by TI-67 they, in effect, would have tripped the bistables similar to that which the instrument fuses had previously accomplished and therefore could have averted this event.

The second possible cause, and the most plausible, involves a misleading NOTE in the power range channel removal from service sequence step detailed in paragraph 4.B, step 2.4 of TI-25. This NOTE specifies not to remove the control power fuses while the drawer assemblies are in the bypass mode as this will remove the bypass function. This NOTE is incorrect and may have resulted in the reinstallation of the control power fuses previously removed during performance of TI-67, thus restoring the previously tripped bistables. Since TI-67 performs action steps similar to the tripping/bypassing steps detailed in TI-25, paragraph 4.B, steps 2.2, 2.3, 2.4, 7.4, 7.5, and 7.6, these TI-25 steps should be deleted to prevent recurrence of a similar problem and to ensure compatibility between the two technical instructions.

Both potential causes were discussed to some degree with the reactor engineer who is investigating this event. This item will remain open until corrective action has been taken to resolve this issue and the resolution has been evaluated by NSRS.

b. R-81-24-SQN-02, Inadequate Trip Switch Identification Utilized in TI-67

TI-67, "Procedures for Removing an Inoperable Protection Channel from Service," identifies, in cases, the wrong component prefix identification for tripping bistables in the protection racks. For example, in placing the bistables associated with power range channel N-43 into a tripped condition, TI-67, Note (B1) step 1, page 194, requires tripping TS/431C and TS/431D of protection rack 10. Review of Mechanical Instrument Tabulation drawing 47B601-68, R16, sheet 16, indicates the vendor-supplied component number prefix for this bistable to be TB versus TS. The actual transfer switch required to defeat this bistable in protection rack 10 is TS 68-44 D/E as specified on the instrument tab and on the Mechanical Control Diagram for the Reactor Coolant System No. 47W610-68-3, R9. Similar discrepancies exist throughout TI-67 and possibly other station procedures.

The SQN plant staff needs to evaluate TI-67 and other applicable station procedures to determine the adequacy, consistency, and usefulness of the component identification nomenclature being utilized. NSRS considers the approach used in several portions of TI-67 of first identifying the appropriate plant designated identification number and then followed by the affected vendor component identification as the most desirable approach in resolving this issue. To translate this approach to the case identified above, the proper instruction should have stated: "Trip TS 68-44 D/E (TB-431 C/D) in rack 10."

3. Boron Analysis Review During Dilution to Criticality

The NSRS reviewer observed the drawing of two reactor coolant samples and the performance of the subsequent boron analysis on these samples in accordance with TI-11, "Chemical Analytical Methods," Method B.5. The observation was made to ensure:

- Sample lines were being continuously purged to ensure meaningful samples could be taken readily

- The pH meter had been buffer standardized

- Titrants had been standardized

- Reagents used had not expired

- Consistency in the determination of boron recovery from the sample was achieved

NSRS concerns in this area were administrative and corrective action has been taken or is anticipated in a timely manner. These concerns include:

a. A copy of TI-16, "Sample Points and Sampling Methods," and TI-11 were not available in the unit 2 hot sample room as required by 10CFR50, Appendix B, Criterion VI, and implemented by TVA-TR75-1, paragraph 17.2.6. These requirements specify that approved documents are to be made available at the location where an activity is to be performed before the work is started. A similar item involving unit 1 had previously been found by NSRS during its startup testing program. The details of that item were discussed in NSRS Report No. R-81-01-SQN dated February 18, 1981.

   When informed, the results supervisor immediately requested that a copy of the procedures be placed in the hot sample room. This item will be evaluated further during a subsequent review of unit 2 startup testing.

b. The chemical technicians on duty during the midnight shift on November 5, 1981 had failed to complete several of the columns of data required by chemical worksheet No. 11-5, including identification as to who the technician was that performed the boron analysis. The responsible chem techs will be instructed to complete the forms properly.

V. PERSONNEL CONTACTED

J. M. Anthony, Assistant Operations Supervisor
D. R. Bucci, Preoperational Test Engineer
G. N. Buchanan, Field Services Scheduling
F. D. Cuzzort, Preoperational Test Engineer
R. W. Fortenberry, Reactor Engineer

*W. J. Glasser, OPQA - Sequoyah QA Coordinator
W. M. Halley, Supervisor, Preoperational Test Section
*R. L. Hamilton, Supervisor, Quality Assurance Section
W. H. Kinsey, Supervisor, Power Plant Results Section
*G. B. Kirk, Sequoyah Compliance Staff
+A. M. Carver, Sequoyah Compliance Staff
°+* J. M. McGriff, Assistant Plant Superintendent, H&S Groups
+R. L. Moore, Team Leader of OPQAA Unit 2 Startup Activities Audit Group
B. T. Springer, Transportation Services - Muscle Shoals
G. W. Smith, Member of OPQAA Unit 2 Startup Activities Audit Group
+C. H. Whittemore, OPQA - Watts Bar QA Coordinator

*Present at exit meeting October 16, 1981
+Present at exit meeting November 6, 1981
°Senior station representative at exit meeting

VI. DOCUMENTS REVIEWED (REFERENCES)

A. Sequoyah Nuclear Plant - Startup Test SU-7.1, "NSSS Startup Sequence,"

B. Sequoyah Nuclear plant - Startup Test SU-7.2 "Initial Criticality,"

C. Sequoyah Nuclear Plant - Preoperational Test W-9.3, "Rod Drop Time Measurement"

D. Sequoyah Nuclear Plant - Preoperational Test W-11.3, "Incore Thermocouple and RTD Cross Calibration"

E. Central Office - Sequoyah Nuclear Plant - Startup Approval Procedure - (CO-SQNP-SAP) Appendix C, "Initial Criticality and Zero Power"

F. Sequoyah Nuclear Plant - General Operating Instruction GOI-1, "Plant Startup from Cold Shutdown to Hot Standby"

G. Sequoyah Nuclear Plant - General Operating Instruction GOI-2, "Plant Startup from Hot Standby to Minimum Load"

H. Facility Operating License DPR-79, Sequoyah Nuclear Plant

I. Sequoyah Nuclear Plant - Standard Practice SQA-44, "Plant Startup Test Program"

J. Sequoyah Nuclear Plant - Standard Practice SQA-93, "Nuclear Plant Management Services Section"

K. Sequoyah Nuclear Plant - Standard Practice SQA-97, "Management Action Tracking System"

L. Sequoyah Nuclear Plant - Standard Practice SQA-117, "Compliance Staff"

M. Sequoyah Nuclear Plant - Standard Practice SQA-128, "Method of Operation"

N. Compliance Section Instruction Letters

O. Operational Section Instruction Letters

P. Sequoyah Nuclear Plant - Technical Instruction TI-26, "Inverse Count Rate Ratio (ICRR) Plotting"

Q. Sequoyah Nuclear Plant, "Operational Quality Assurance Manual, Part II, section 4.2

R. Sequoyah Nuclear Plant, "Final Safety Analysis Report"

S. COMTRAK - Computer Program Printout

T. US NRC Regulatory Guide 1.68, "Preoperational and Initial Start-up Test Programs for Water-Cooled Power Reactors," November 1973

U. Memorandum from H. N. Culver to J. R. Calhoun dated June 27, 1980, "Sequoyah Nuclear Plant Unit 2 - NSRS Review Report No. R-80-05-SQN," (GNS 800627 002)

V. Memorandum from H. N. Culver to J. R. Calhoun dated August 25, 1980, "Sequoyah Nuclear Plant Unit 2 - NSRS Review Report No. R-80-11-SQN," (GNS 800826 002)

W. Memorandum from H. N. Culver to H. J. Green dated January 14, 1981, "Sequoyah Nuclear Plant Units 1 and 2 - Nuclear Safety Review Staff Review Report R-80-20-SQN," (GNS 810115 154)

X. Memorandum from H. N. Culver to A. W. Crevasse and H. J. Green dated February 18, 1981, "Sequoyah Nuclear Plant Units 1 and 2 - Nuclear Safety Review Staff Review Report No. R-81-01-SQN (GNS 810218 002)

Y. Memorandum from H. N. Culver to Those listed, "Sequoyah Nuclear Plant - Nuclear Safety Review Staff Review Report No. R-81-12-SQN," (GNS 810717 051)

Z. Memorandum from H. N. Culver to A. W. Crevasse dated March 2, 1981, "Clarification of a Nuclear Safety Review Staff Report Item," (GNS 810302 002)

AA. Memorandum from A. W. Crevasse to H. N. Culver dated March 23, 1981, "Sequoyah Nuclear Plant Units 1 and 2 - Nuclear Safety Review Staff Review Report No. R-81-01-SQN" (GNS 810313 101)

BB. Memorandum from A. W. Crevasse to H. N. Culver dated April 7, 1981, "Sequoyah Nuclear Plant Units 1 and 2 - Nuclear Safety Review Staff Review Report No. R-81-01-SQN" (GNS 810409 103)

CC. Office of Power Quality Assurance and Audit Staff, Quality Program Audit Report No. OPQAA-SQ-81-SP-03, "Startup Testing - Units 1 and 2, " July 21-24, 1981

DD. Sequoyah Nuclear Plant - Administrative Instruction AI-4, "Plant Instructions Document Control"

EE. Sequoyah Nuclear Plant - Administrative Instruction AI-5, "Shift and Relief Turnover"

FF. Sequoyah Nuclear Plant - Administrative Instruction AI-9, "Control of Temporary Alterations and Use of the Temporary Alterations Order"

GG. Sequoyah Nuclear Plant - System Operating Instruction SOI-92.1, "Nuclear Instrumentation System"

TVA 64 (OS-9-65)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

GNS '81 1013 050

TO : W. F. Willis, General Manager, E12B16 C-K

FROM : H. N. Culver, Director of Nuclear Safety Review Staff, 249A HBB-K

DATE : October 13, 1981

SUBJECT: INTERFACE BETWEEN POWER AND OEDC ON ITEMS INVOLVING THE NRC OFFICE OF INSPECTION AND ENFORCEMENT - NUCLEAR SAFETY REVIEW STAFF REPORT NO. R-81-23-NPS

In response to your recent request, NSRS has examined the problems that presently exist between OEDC and POWER on items involving the NRC Office of Inspection and Enforcement. We have reviewed the OEDC proposal for modifying the existing system and have had benefit of discussions with POWER and OEDC staff involved with this problem.

NSRS has concluded that changes to the present system should result in improvement in the timeliness and accuracy of responses to NRC. Reassignment of responsibility to OEDC to interface with NRC would aggrevate problems that TVA is presently attempting to improve. The attached report provides our findings and recommendations to POWER and OEDC.

H. N. Culver

H. N. Culver

HNC:LML
Attachment
cc (Attachment):
G. H. Kimmons, W12A9 C-K) For action on recommendations,
H. G. Parris, 500A CST2-C) please respond in 30 days.
MEDS, 100 UB-K

--HNC

NSRS FILE

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

REVIEW

NSRS REPORT NO. R-81-23-NPS

SUBJECT: INTERFACE BETWEEN POWER AND OEDC ON ITEMS INVOLVING THE NRC OFFICE OF INSPECTION AND ENFORCEMENT

DATE OF REVIEW: OCTOBER 6-7, 1981

REVIEWERS: James A. Crittenden
JAMES A. CRITTENDEN — 10-13-81 DATE

James A. Crittenden for
MARVIN V. SINKULE — 10-13-81 DATE

APPROVED BY: James A. Crittenden for
MARVIN V. SINKULE — 10-13-81 DATE

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I. BACKGROUND

The Office of Power (POWER) Nuclear Regulatory Staff is responsible for interfacing with the NRC. The Nuclear Regulatory Staff is responsible for coordinating TVA's efforts in responding to NRC, obtaining information from various divisions within POWER, and obtaining information from various other offices, including the Office of Engineering Design and Construction (OEDC).

A number of problems have been identified by OEDC relating to the preparation and response to the Nuclear Regualtory Commission-Office of Inspection and Enforcement (NRC-OIE) on nonconformance reports, IE Bulletins, and NRC inspections reports during the design and construction of nuclear plants. In a 45D from M. N. Sprouse to G. H. Kimmons dated September 23, 1981, EN DES indicates that the existing arrangement has made it difficult and inefficient for TVA to resolve promptly matters with NRC relating to design and construction.

OEDC has proposed that POWER delegate to OEDC the responsibility for communicating directly with NRC-OIE on matters relating directly to design and construction.

II. FINDINGS

The NSRS investigation has been made by discussing the issues involved with representatives from OEDC and POWER who have responsibility for preparing and reviewing responses to OIE on nonconformance reports, IE Bulletins, and NRC inspection reports. The Nuclear Safety Review Staff (NSRS) reviewers also examined examples of responses which had been sent to NRC that contained errors or that were not transmitted within the proper time allocated for responses. NSRS review was directed toward determining the following:

1. What are the nature of the problems that exist with the present system?

2. Would the OEDC proposal solve the problems that presently exist?

3. Would a change from the existing system introduce new problems?

4. What, if any, changes should be made?

The two problems identified in the 45D from Sprouse to Kimmons relates to timeliness and the accuracy of responses that go back to NRC. These two problems are interrelated and must be considered together. The factors within the existing system that have led to problems in both timeliness and accuracy are as follows:

1. There is no clear definition of duties and responsibilities for the organizational units involved with preparation or review of responses. This has led to duplication of effort in the review process resulting in additional time required to complete an

approved response. For example, a response prepared by the line organization is reviewed by the Division of Engineering Design (EN DES) Nuclear Engineering Branch (NEB) Nuclear Licensing Section and by EN DES Quality Assurance Branch (QAB) and by OEDC Quality Assurance (QA). The review by the QA organizations is not deemed necessary and contributes to time delay. Following these reviews there is further review by POWER.

2. There has in the past been inadequate understanding on the part of the line organization regarding the type of information that should be presented in responses. Lack of understanding has led to the need to rewrite many responses causing timeliness problems. Recognizing this problem, EN DES management has written a number of memorandum during the past six-eight months to correct this problem. Lack of a procedure has prevented specific training of the line organization.

3. The slowness in the preparation of final responses has resulted in some cases in the transmittal of draft responses by OEDC to POWER. In some cases the final OEDC position is not received until after a response has been transmitted to NRC-OIE. This has led to problems in response content.

4. Clear definition of what POWER can change without going back to OEDC has not been established. Additionally, there is not a good understanding of who in OEDC can approve changes made by POWER. With the existing system, POWER Nuclear Regulatory Staff may contact a staff representative in EN DES and obtain approval for a change which EN DES management may not support.

5. Only a small fraction of the responses going to NRC appear to have timeliness and accuracy problems. The NRC has indicated there has been improvement in timeliness and accuracy on responses during the past few months.

The above problems appear to result from a lack of good understanding between OEDC and POWER as well as inadequate control within OEDC. Most of these problems can be solved by establishment of improvements within the existing system.

The only advantage of the OEDC proposed modification to the system appears to result from decreasing the time required by the POWER Nuclear Regulatory Staff to review submittals. It is not clear if this time improvement (five days) would really be realized since the function presently assigned to POWER would then have to be accomplished by some group in OEDC. Thus the timeliness of responses would not be greatly changed by changing the system.

One factor that NSRS examined as a part of the evaluation was the introduction of new problems by reassignment of responsibility. The only problem in this area results from the fragmentation of the responsibility for interfacing with the NRC. The advantages of the single point of contact with TVA are:

1. A generic review would be performed to determine if the response from one TVA organization would affect the functions of other TVA organizations.

2. Consistent responses would be prepared regarding TVA policies or positions on questions raised by NRC.

3. Since TVA is a large and complex organization, a single point of contact provides NRC with a communications channel to ask questions on any part of our nuclear program and receive coordinated reponses.

NSRS concludes that the advantages of the single point of contact should be retained and, in fact, strengthened. Where informal contact is required, such contact can be maintained within agreed upon guidelines.

III. CONCLUSION

The existing system for dealing with the NRC-OIE should be retained. The problems that have been identified with the existing system are not extensive and can best be coped with by the various participants. Recommendations are made to improve the existing system

IV. RECOMMENDATIONS

A. R-81-23-NPS-01, Issue POWER-OEDC Interdivisional Procedure

An interdivisional procedure should be issued which clearly defines the responsibilities of POWER Nuclear Regulatory Staff and OEDC NEB Nuclear Licensing Section in preparing responses to NRC-OIE. The procedure should also address the time allowed for each organization to review and prepare a response to 50.55(e) items, IE inspection reports, and IE Bulletins.

B. R-81-23-NPS-02, Issue POWER Nuclear Regulatory Staff Procedure

POWER Nuclear Regulatory Staff should issue a procedure which defines the responsibilities of members of the staff for: (a) receipt of IE inspection reports, 50.55(e) responses, and IE Bulletins, (b) transmittal of the reports and bulletins to OEDC NEB Nuclear Licensing Section or other affected organizations, (c) review of responses for IE inspection reports, IE Bulletins, and 50.55(e) items, and (d) requirements for coordination with the organization that prepared the response when Nuclear Regulatory Staff members change any part of the response. This coordination should be documented in case future questions arise pertaining to authorization for change.

C. R-81-23-NPS-03, Issue OEDC NEB Nuclear Licensing Section Procedure

OEDC NEB Nuclear Licensing Section should issue procedures to define the responsibilities of members of the section for:

(a) receipt of IE inspection reports, IE Bulletins, and nonconformance reports, (b) transmittal of inspection reports and bulletins to EN DES branches or Division of Construction (CONST) projects, (c) review of responses to inspection reports, bulletins, and nonconformances, (d) notifying NRC of 50.55(e) items, and (e) approval of Nuclear Regulatory Staff suggested changes to Nuclear Licensing Section responses.

D. R-81-23-NPS-04, Issue EN DES-CONST Procedures

EN DES and CONST should issue procedures which define the responsibilities of persons preparing nonconformance reports and responses to inspection reports and bulletins. Specific details should be included in the procedure for: (a) accurately identifying nonconformances, (b) completely describing the apparent cause, (c) giving specific and complete details on the actions to be taken to prevent recurrence, and (d) consideration for generic implication of the nonconformance. Training sessions should be conducted, as necessary, to inform persons responsible for preparing responses regarding the developed procedures and of the importance of accuracy and timeliness.