TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS INVESTIGATION REPORT NO. I-85-735-SQN

EMPLOYEE CONCERN: XX-85-102-011

SUBJECT: NDE INSPECTORS CANNOT WRITE NOTICE OF INDICATION FOR PRESERVICE-RELATED DEFECTS

DATES OF INVESTIGATION: OCTOBER 25 - NOVEMBER 5, 1985

INVESTIGATOR: E. F. Harwell 12/6/85
E. F. HARWELL DATE

REVIEWED BY: R. C. Sauer 12/6/85
R. C. SAUER DATE

APPROVED BY: [signature] 12/11/85
M. S. KIDD DATE

I. BACKGROUND

A Nuclear Safety Review Staff (NSRS) investigation was conducted to determine the validity of an expressed employee concern as received by Quality Technology Company (QTC)/Employee Response Team (ERT). The concern of record, as summarized on the Employee Concern Assignment Request Form from QTC and identified as XX-85-102-001, stated:

> "Sequoyah: NDE inspectors can only write a Notice of Inspection (correction: Indication) on in-service related defects. Preservice related defects can only be identified by a Maintenance Request. Nuclear Power Dept. concern."

II. SCOPE

A. The scope of the investigation was determined from the stated concern of record to be two specific issues requiring investigation:

1. NDE inspectors report service-related defects only on Notices of Indication (NOI).

2. Preservice defects are reported only on a Maintenance Request (MR).

B. In conducting this investigation NSRS reviewed the requirements of the Nuclear Quality Assurance Manual (NQAM), plant surveillance instructions, and plant instructions which govern defect reporting. Nuclear Central Office ISI group managers and level III's, plant QC section supervisors, and Power Operations Training Center (POTC) NDE trainers were interviewed concerning the training, instructions, and practices of NDE inspectors on reporting defects. NSRS also reviewed random samples of NOI's generated during the present (Environmental Qualification) outage.

III. SUMMARY OF FINDINGS

A. Requirements and Commitments

1. ASME Section XI, "Rules for Inservic Inspection of Nuclear Power Plant Components."

2. Title 10 Code of Federal Regulations Part 50, "Domestic Licensing of Production and Utilization Facilities."

3. NQAM, Part II, Section 5.1, "Inservice Inspection - Nuclear Power Plant Components."

4. NQAM, Part II, Section 5.3, "Maintenance and Modification Inspection Program."

B. Findings

1. Part II, Section 5.1, of the NQAM (ref. 1) requires an NOI be written if a defect is found in the examination area for both preservice and inservice examinations.

2. Individuals A and C stated in their interview that ISI inspectors are instructed to prepare NOIs for either preservice or inservice inspection detected defects that are found within the scope of the examination area. However, if an inspector finds the examination area is not ready for inspection (i.e., needs polishing or grinding) he does not perform the inspection, but prepares an MR to have the area properly prepared for subsequent inspection. If a defect (i.e., arc strike) is found which is outside the scope of the examination area, is obviously not a service-related flaw and can be readily corrected, the inspectors are instructed to prepare an MR. The inspectors are instructed to notify their supervisor if significant items are found outside the examination area and the reporting is done via other nonconforming condition reporting methods.

3. Individual D stated in his interview that plant QC inspectors are instructed to prepare an NOI for defects found while performing an ASME Section XI preservice or inservice examination. However, defects found during examinations conducted after repairs or modifications for initial acceptance are recorded on the workplan data sheet, or the weld record data sheet, or on an MR, depending on the type work control document. This is in accordance with NQAM Part II, Section 5.3 (ref. 2) requirements and plant instructions.

4. Fifty-eight NOI's (ref. 8) were reviewed and determined that they appropriately referenced an MR to cover any corrective actions required.

5. Two hundred and twenty-seven MRs (ref. 8) were reviewed associated with inservice examinations to determine if any noted defects for which NOIs were not written should have been written. No inadequacies were identified. The MRs reviewed were determined to be properly written for corrections found during examination and coupled to an NOI, and as deficient items discovered outside the official inspection area but considered necessary for correction by the inspector.

IV. CONCLUSIONS AND RECOMMENDATIONS

The concern of record could not be substantiated because this investigation revealed that NOIs are prepared for both preservice and inservice defects found within the area of scope for ASME Section XI examinations.

DOCUMENTS REVIEWED IN INVESTIGATION I-85-735-SQN AND REFERENCES

1. NQAM, Part II, Section 5.1, Revision dated March 28, 1984, "Inservice Inspection - Nuclear Power Plant Components"

2. NQAM, Part II, Section 5.3, Revision dated July 30, 1984, "Maintenance and Modification Inspection Program"

3. SQNP Surveillance Instruction SI-114.1, Revision 5, dated September 14, 1984, ASME Section XI, "In-service Inspection Program"

4. SQNP Technical Instruction TI-51, Revision 29, dated September 5 1985, "Assignment of Detailed Test Methods and Responsibility for Nondestructive Testing"

5. SQNP Administrative Instruction AI-12, Revision 20, dated August 2, 1985, "Adverse Conditions and Corrective Actions"

6. SQNP Modification and Additions Instruction M&AI-1, Revision 9, dated August 5, 1985 "Control of Weld Documentation and Heat Treatment"

7. SQNP Quality Assurance Section Instruction Letter No. 10.4, Revision 7, dated August 16, 1985, "Inspection - QC Inspections"

8. Sequoyah Notices of Indication (NOI) SQ-0139 through SQ-0202 (58 total) and 227 MRs involving ISI work beginning U1C3 up to November 2, 1985

NRC

TVA 64 (OS-9-65) (OP-WP-5-85)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : H. L. Abercrombie, Site Director, Sequoyah Nuclear Plant

FROM : K. W. Whitt, Director of Nuclear Safety Review Staff, E3A8 C-K

DATE : DEC 10 1985

SUBJECT: NUCLEAR SAFETY REVIEW STAFF INVESTIGATION REPORT TRANSMITTAL

Transmitted herein is NSRS Report No. I-85-616-SQN

Subject NUCLEAR PLANT SECURITY

Concern No. XX-85-099-001

No response or corrective action is required for this report. It is being transmitted to you for information purposes only. Should you have any questions, please contact R. C. Sauer at telephone 2277 .

Recommend Reportability Determination: Yes ____ No X

Director, NSRS/Designee

RCS:JTH
Attachment
cc (Attachment):
R. P. Denise, LP6N35A-C
R. J. Griffin, SQN E-18
G. B. Kirk, SQN
D. R. Nichols, E10A14 C-K
QTC/ERT, Watts Bar Nuclear Plant
Eric Sliger, LP6N48A-C
J. H. Sullivan, SQN
W. F. Willis, E12B16 C-K (4)

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TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

INVESTIGATION REPORT NO. I-85-616-SQN

EMPLOYEE CONCERN: XX-85-099-001

SUBJECT:	**NUCLEAR PLANT SECURITY**
DATES OF INVESTIGATION	**OCTOBER 9-12, 1985**

INVESTIGATOR: MW Alexander — M. W. ALEXANDER — 12/6/85 DATE

REVIEWED BY: R. C. Sauer — R. C. SAUER — 12/6/85 DATE

APPROVED BY: for M. S. KIDD — 12/9/85 DATE

I. BACKGROUND

A Nuclear Safety Review Staff (NSRS) investigation was conducted to determine the validity of an expressed employee concern as received by Quality Technology Company (QTC)/Employee Response Team (ERT). The concern of record, as summarized on the Employee Concern Assignment Request Form from QTC and identified as XX-85-099-001, stated:

> "Sequoyah--the plant needs to be more secure. The protected area needs to be larger."

The ERT follow-up group was requested to obtain additional clarification from the concerned individual (CI) as to the specifics of his issue. ERT made several attempts to contact the CI but he/she had moved and left no forwarding address.

II. SCOPE

A. The scope of the investigation was defined by the concern of record and entailed investigating two issues in order to either validate or refute the concern.

1. The plant needs to have greater protection provided.

2. The present protected area (PA) needs to be larger.

B. The Sequoyah Nuclear Plant Physical Security Plan was reviewed to determine the basic features of the power block security concept. In addition, cognizant nuclear central office and plant personnel who participated in the development of the power block concept, as well as plant personnel who are subject to security plan requirements on a day-to-day basis, were interviewed to assess past and current program control adequacy.

III. SUMMARY OF FINDINGS

A. <u>Requirements and Commitments</u>

1. U.S. Code of Federal Regulations, Title 10, Part 73.55, "Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage."

2. Sequoyah Nuclear Plant Physical Security/Contingency Plan.

B. <u>Findings</u>

1. Several years ago the decision was made to implement a new concept in plant security at Sequoyah--the Power Block Concept (PBC). The basic objectives of PBC were to reduce the size of the physical areas and number of personnel that must be protected and focus more concentrated security efforts in smaller safety-related "vital" areas. The design of PBC was

carried out by TVA engineers and included a thorough review of the proposed concept and information from security plans at other utilities and nuclear facilities. PBC was submitted to NRC approximately two years ago. Based on discussions with cognizant personnel it appears NRC conducted an extensive review of PBC and issued their approval June 11, 1985 (ref. 6). PBC was immediately initiated by TVA on June 22, 1985. Even though the change from the existing security system to the PBC concept was considered drastic due to the significant amount of work involved and personnel reorientation required, it was implemented without any significant problems.

2. From the review of pertinent information and discussions with TVA personnel involved in the security program, this investigation revealed no objective evidence that the level of security has been decreased as a result of PBC or that there are any significant weaknessns in the program. On the contrary, all of the personnel interviewed felt that plant security was at least as good under PBC and most felt there had been an overall improvement in the level of effectiveness of the security program especially in the reduction of the size of the areas to be protected. Any negative comments dealt with implementation problems which are being worked out during the initial break-in phase.

3. The plant Quality Assurance Staff conducted a QA survey of plant security activities July 15-19, 1985 (ref. 3). Seven deficiencies were identified. All were related to incomplete implementation of PBC. Four of these deficiencies have been resolved and the remaining three will be corrected by January 1, 1986.

4. During the week of September 9, 1985, the Nuclear Security Program Section in the Division of Nuclear Services conducted a survey (ref. 2) at the request of the Site Director to determine the impact of PBC on plant operations. It covered the period June 22 through September 9, 1985, and included interviews with numerous craftsmen, engineers, Public Safety officers, and general employees. It identified sixteen problems or concerns raised by plant personnel during the interviews. These issues generally related to difficulties in implementing PBC (i.e., delays in obtaining security approvals to open vital areas, possible relaxation of some administrative controls, too many vehicles on the designated vehicle list, etc.) and indicated no overall security program weaknesses.

5. NRC has conducted two inspections since PBC was instituted July 8-11 (ref. 4) and September 3-6 (ref. 5), 1985. Reports of these inspections indicated one Level V violation of regulations (lowest level) related to reduced security lighting levels which was corrected on September 5, 1985. Other plant security activities (package searching, maintenance of protected area barriers, and maintenance of vital area barriers) were found to be acceptable.

IV. CONCLUSIONS AND RECOMMENDATIONS

The employee concern appears to be unsubstantiated for the following reasons:

- The results of this investigation revealed no objective evidence that the Sequoyah security program under the PBC provides a lesser degree of security than previous programs. On the contrary, it appears that plant security is at least as effective under PBC, and the feeling of most personnel interviewed is that it is better.

- The information reviewed indicated that the concept of reducing the size of the PA was thoroughly reviewed by TVA and NRC, and this investigation revealed no reasons to increase its size.

DOCUMENTS REVIEWED IN INVESTIGATION I-85-616-SQN AND REFERENCES

1. Sequoyah Nuclear Plant Physical Security/Contingency Plan (Safeguards Information)

2. TVA memorandum from John Hutton to H. L. Abercrombie dated October 7, 1985, "Survey for Power Block Concept (PBC) Impact on Plant Operations" (L46 851001 863)

3. Sequoyah Nuclear Plant QA Staff Survey No. 22-85-5-001, dated July 31, 1985, "Physical Security Program Survey" (Safeguards Information)

4. U.S. NRC Inspection Report No. 50-327/85-25 and 50-328/85-25 dated August 9, 1985 (Safeguards Information)

5. U.S. NRC Inspection Report No. 50-327/85-30 and 50-328/85-30 dated October 1, 1985 (Safeguards Information)

6. Letter from E. G. Adensam (NRC) to H. G. Parris (TVA) dated June 11, 1985, "Issuance of Amendment No. 38 to Facility Operating License DPR-77 and Amendment No. 30 to Facility Operating License DPR-77 - Sequoyah Nuclear Plant, Units 1 and 2" (A02 850614 001)

NRC

TVA 64 (OS-9-65) (OP-WP-5-85)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : H. L. Abercrombie, Site Director, Sequoyah Nuclear Plant

FROM : K. W. Whitt, Director of Nuclear Safety Review Staff, E3A8 C-K

DATE : DEC 10 1985

SUBJECT: NUCLEAR SAFETY REVIEW STAFF INVESTIGATION REPORT TRANSMITTAL

Transmitted herein is NSRS Report No. I-85-615-SQN

Subject FREQUENCY OF RADIATION SURVEYS

Concern No. XX-85-098-002

No response or corrective action is required for this report. It is being transmitted to you for information purposes only. Should you have any questions, please contact R. C. Sauer at telephone 2277 .

Recommend Reportability Determination: Yes ____ No X

Director, NSRS/Designee

RCS:JTH
Attachment
cc (Attachment):
R. P. Denise, LP6N35A-C
R. J. Griffin, SQN E-10
G. B. Kirk, SQN
D. R. Nichols, E10A14 C-K
QTC/ERT, Watts Bar Nuclear Plant
Eric Sliger, LP6N48A-C
J. H. Sullivan, SQN
W. F. Willis, E12B16 C-K (4)

0154U

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS INVESTIGATION REPORT NO. I-85-615-SQN

EMPLOYEE CONCERN XX-85-098-002

SUBJECT: FREQUENCY OF RADIATION SURVEYS

DATES OF INVESTIGATION: OCTOBER 21-23, 1985

INVESTIGATOR: R.C. Sauer for 12/9/85
D. J. HORNSTRA DATE

REVIEWED BY: R.C. Sauer 12/9/85
R. C. SAUER DATE

APPROVED BY: for M. S. KIDD 12/9/85
DATE

I. BACKGROUND

A Nuclear Safety Review Staff (NSRS) investigation was conducted to determine the validity of an employee concern received by Quality Technology Company (QTC)/Employee Response Team (ERT). The concern of record, as summarized on the Employee Concern Assignment Request form from QTC and identified as XX-85-098-002, stated:

> "Sequoyah - Radiation areas are not monitored often enough. Nuclear Power concern. The Concerned Individual has no additional information."

Further information was requested from the ERT followup group. Based upon this followup, the concern of record was classified to be a concern that the frequency of radiation surveys at Sequoyah Nuclear Plant (SQN) is not adequate.

II. SCOPE

A. The scope of this investigation was to determine the frequency of radiation surveys required in radiation areas and to determine the implementation adequacy of these requirements.

B. This investigation included the review of requirement documents, interviews of Health Physics personnel, and review of radiation survey documents.

III. SUMMARY OF FINDINGS

A. Requirements and Commitments

1. 10CFR20.201 (ref. 1) defines a survey as "an evaluation of the radiation hazards . . . . When appropriate, such evaluation includes a physical survey of the location of materials and equipment, and measurements of levels of radiation or concentrations of radioactive material present." Furthermore, it requires that surveys be conducted as "reasonable under the circumstances to evaluate the extent of radiation hazards that may be present."

2. SQN Technical Specification 6.11 (ref. 2) requires that: "Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure."

3. SQN Final Safety Analysis Report (FSAR), Section 13.5.7 (ref. 3), identifies Radiation Control Instructions (RCIs) which include an RCI for the "Radiological Hygiene Program."

4. SQN FSAR, Section 12.3 (ref. 4), commits SQN to implement a health physics program in conformance with the TVA Office of

Power Radiation Protection Program (RPP) Manual established by the Radiological Health Staff.

5. The RPP, Section A.3.2 (ref. 5), requires that routine and special radiological surveys shall be conducted and documented in accordance with 10CFR20.

6. The Radiation Protection Manual (RPM), Area Plan 3, Procedure 0301.03 (ref. 6), Section 3.6, requires that radiation surveys be performed on both a routine and an unscheduled basis. The RPM requires the performance of surveys "to ensure compliance with regulatory requirements." Furthermore, the RPM states that dose rates should be checked frequently if work was being done in an area where the dose rate may vary. It also states that a person should not unnecessarily expose himself to radiation while performing surveys.

B. Findings

1. The frequency of surveys required by Radiological Control Instruction RCI-1, Section X (ref. 7), was found to satisfy the requirements and commitments. RCI-1 states:

   > Surveys shall be performed on a routine basis to assess radiation exposure rates, contamination, and airborne radioactivity levels. Additional surveys shall be performed whenever required by plant conditions or work requirements to assure the protection of personnel and to monitor plant conditions.

2. The specific frequency of radiological surveys required in areas with an active Radiation Work Permit (RWP) is established in RCI-14 (ref. 8) and was found to meet the requirements of RCI-1.

   RCI-14, Section III, requires that:

   > Periodic radiological surveys will be performed in all areas covered by an active RWP. The survey period will vary, depending upon radiological conditions, but will not exceed seven days . . . .

   Provisions are made for more frequent surveys if system changes occur to change the radiation dose rate. RCI, Section V, requires that:

   > If the job location is in an area where significant changes in dose rate are likely to occur, a radiological survey should be performed just prior to the start of work.

3. The RPM requirement that a person should not unnecessarily expose himself to radiation while performing radiation surveys

[i.e., maintain exposure of HP technicians as low as reasonably achievable (ALARA)] has been satisfied by an exception in RCI-14 that:

> At the discretion of the plant health physicist or his assistant, the survey period may be extended for ALARA purposes, in increments of 7 days, by making the extension in writing to the responsible shift supervisors.

Additionally, according to HPSIL-7 (ref. 9), routine surveys (a survey once every seven days) may be deleted for an individual area if an RWP is not in effect in the particular area or if radiation levels exceed 1000 millirem per hour and no work is scheduled in that area. Thus, radiation exposure of health physics personnel will be maintained ALARA if no surveys are required to support ongoing work.

4. For many areas of the plant which are routinely accessible, surveys are documented on preprinted survey sheets which establish the weekly survey routine to ensure that a survey is conducted once every seven days.

5. Surveys are scheduled on these preprinted sheets for specific shifts throughout the week. A review of these preprinted sheets found that numerous areas outside the regulated area (i.e., the cafeteria and hallway by the electrical shop) were surveyed more frequently than once a week to check for the presence of transferable contamination.

6. Routine surveys of the containment building and various rooms in the auxiliary building are scheduled based upon work planned during operation or for a particular outage. A survey status list and/or a monthly schedule of routine surveys are maintained at the health physics lab/control point to ensure that the frequency of surveys meet the requirements of RCI-14. A review of the monthly schedule at unit 1 containment control point (marked-up calendar) indicated that containment surveys were currently being conducted on a five-day schedule.

7. Surveys for the auxiliary and containment buildings were reviewed for the period of July through September 1985. The frequency of radiation surveys of 15 locations for the duration of this period indicated that these locations had received a routine survey on a seven-day schedule.

8. RWP Timesheets (ref. 10, 11, 12) from 1984 demonstrated that surveys had been conducted on at least a seven-day schedule in accordance with RCI-14. Due to the nature of the work, one of the timesheets (ref. 10) had radioactivity/contamination surveys performed on five days in an eight-day period.

9. Based upon interviews with Individuals C and D, few personnel (less than 25 percent) review the survey sheets at this time in

the outage (two to three months into the outage) prior to entry into containment on an RWP. Personnel were observed at the control points for unit 1 for a period during which approximately 20 individuals processed through the control point, with none reviewing surveys. A check of the associated RWP timesheets showed that these individuals had previously worked in containment on those timesheets. Individual D stated that when an RWP timesheet is first opened, all radiation hazards are discussed by the HP with the associated foreman, using the survey map. The HP at the control point reiterates this information when the work crew enters the RWP for the first time. Additional instructions to workers on subsequent entries are provided to the workers only on a case-by-case basis. A Control Point HP Technician (Individual C) was observed giving instructions to workers on special dosimetry requirements on a reentry on one job due to the nature of the work on reactor coolant pumps. Radiation levels were not reiterated to these individuals since it was unchanged from their last entry.

IV. CONCLUSIONS AND RECOMMENDATIONS

The concern of record is not substantiated. The frequency of radiation surveys, with the flexibility to have more surveys when changes in radiation levels are anticipated, was judged to adequately meet the requirements.

DOCUMENTS AND REFERENCES REVIEWED IN INVESTIGATION I-85-615-SQN

1. 10CFR20.201, "Surveys"

2. SQN Technical Specification 6.11, "Radiation Protection Program"

3. FSAR Section 13.5.7, "Radiation Control Instruction"

4. TVA FSAR Section 12.3, "Health Physics Program"

5. Radiation Protection Plan, Section A3.2, "Radiological Control Zones and Standards," dated November 2, 1983

6. Radiation Protection Manual, Area Plan 3, Procedure No. 0301.03, "External Exposure Limits and Controls," dated December 29, 1983

7. SQN Radiological Control Instruction, RCI-1, R27, "Radiological Hygiene Control," Section X, "Radiological Surveys and Records," dated September 2, 1985

8. SQN Radiological Control Instruction, RCI-14, R4, "Radiation Work Permit Program," dated July 10, 1985

9. SQN Health Physics Section Instruction Letter, HPSIL-7, "Routines," dated August 13, 1984

10 Radiation Work Permit (RWP) 02-2-84214 Timesheet 0002

11. RWP 02-2-84247 Timesheet 0034

12. RWP 02-2-84250 Timesheet 0030

13. NRC NUREG-75/087, September 1975, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition"

14. NRC Regulatory Guide 8.8, February 2, 1973, "Guide for Administrative Practices in Radiation Monitoring"

15. ANSI N13.2-1969, "Guide for Administrative Practices in Radiation Monitoring"

16. SQN HPSIL-1 Survey Sheets for July-October 1985

17. Radiation Work Permits and Survey Sheets at Containment Control Point Stations on November 6, 1985, and November 22, 1985

NRC

TVA 64 (OS-9-65) (OP-WP-5-85)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : H. L. Abercrombie, Site Director, Sequoyah Nuclear Plant

FROM : K. W. Whitt, Director of Nuclear Safety Review Staff, E3A8 C-K

DATE : DEC 10 1985

SUBJECT: NUCLEAR SAFETY REVIEW STAFF INVESTIGATION REPORT TRANSMITTAL

Transmitted herein is NSRS Report No. I-85-619-SQN

Subject TRAINING OF SEQUOYAH SHIFT ENGINEERS AND ASSISTANT SHIFT ENGINEERS ON ELECTRICAL STATION OPERATION

Concern No. XX-85-093-001

The attached report contains one Priority 3 [P3] recommendation which requires you to take some form of investigative or corrective action within the next four months (April 1, 1986). No formal response is required for this report unless you disagree with the proposed action. Please notify us if actions taken have been completed sooner. Should you have any questions, please contact R. C. Sauer at telephone 2277.

Recommend Reportability Determination: Yes ____ No X

Director, NSRS/Designee

RCS:JTH
Attachment
cc (Attachment):
R. P. Denise, LP6N35A-C
R. J. Griffin, SQN E-18
G. B. Kirk, SQN
B. C. Morris, BFN
D. R. Nichols, E10A14 C-K
QTC/ERT, Watts Bar Nuclear Plant
Eric Sliger, LP6N48A-C
J. H. Sullivan, SQN
W. F. Willis, E12B16 C-K (4)

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TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS INVESTIGATION REPORT NO. I-85-619-SQN

EMPLOYEE CONCERN: XX-85-093-001

SUBJECT: TRAINING OF SEQUOYAH SHIFT ENGINEERS AND ASSISTANT SHIFT ENGINEERS ON ELECTRICAL STATION OPERATION

DATES OF INVESTIGATION: OCTOBER 2 - NOVEMBER 22, 1985

LEAD INVESTIGATOR: C. L. Breeding — C. L. BREEDING — 12/6/85 DATE

INVESTIGATOR: N. T. Henrich — N. T. HENRICH — 12/6/85 DATE

REVIEWED BY: R. C. Sauer — R. C. SAUER — 12/6/85 DATE

APPROVED BY: for M. S. KIDD — 12/9/85 DATE

I. BACKGROUND

A Nuclear Safety Review Staff (NSRS) investigation was conducted to determine the validity of an expressed employee concern received by Quality Technology Company (QTC)/Employee Response Team (ERT). The concern of record, as summarized on the Employee Concern Assignment Request Form from QTC and identified as XX-85-093-001, stated:

> "Sequoyah: Shift engineers (SE) and assistant shift engineers (ASE) are inadequately trained in electrical station operation (switchyard, off-site power feed, etc.) such that there could be an excessive delay in restoring off-site power feed to the plant in the event of an emergency. C/I feels that SE/ASE personnel should receive better training in this area. The C/I has no further information."

II. SCOPE

The scope of this investigation as determined from the concern of record entailed four specific issues requiring investigation:

A. Shift engineers (SE) are inadequately trained in electrical station operation.

B. Assistant shift engineers (ASE) are inadequately trained in electrical station operation.

C. In the event of an emergency, excessive delays in restoring offsite power feed to the plant could result.

D. Shift engineers and assistant shift engineers should receive better training in this area.

NSRS reviewed documentation which delineates shift engineer (SE) and assistant shift engineer (ASE) training requirements. Typical duties of the SE and ASE in switchyard operation were reviewed along with applicable operating procedures. A review of the type, scope, and quantity of electrical training provided the SE and ASE was conducted. The investigation used Institute of Nuclear Power Operations (INPO) guidelines and evidence of current problems with switchyward operation to determine the adequacy of this training.

III. SUMMARY OF FINDINGS

A. <u>Requirements and Commitments</u>

1. 10CFR55 is the basic implementing regulation for licensing reactor operators and senior reactor operators. Appendix A to 10CFR55, "Requalification Programs for Licensed Operators of Production and Utilization Facilities," establishes the basic requirements and the regulatory basis for licensing operators.

2. Regulatory Guide 1.8, "Personnel Selection and Training," dated May 1977, describes an NRC acceptable method of implementing the regulations with regard to personnel qualifications.

3. TVA-TR75-1A, "TVA Topical Report," Revision 8, in Table 17D-3 gives regulatory guidance for quality assurance during station operation. This document commits TVA to Regulatory Guide 1.8 and to 10CFR55 with no exceptions.

4. ANSI/ANS 3.1 - 1981, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants," establishes the criterion for the selection, qualification, and training of personnel for stationary nuclear power plants.

5. NUREG - 1021, dated February 1985, Rev. 1, "Operator Licensing Examiner Standards," provides guidance and establishes procedures and practices for the examining and licensing of applicants for NRC operator licenses. This document endorses ANSI/ANS 3.1 - 1981.

6. Nuclear Power Area Plan Program Procedure 0202.05, "Nuclear Plant Operator Training Program," March 15, 1985, summarizes and consolidates training requirements for all nuclear operating personnel.

7. Sequoyah Final Safety Analysis Report, chapter 13, stipulates that procedure 0202.05 will be followed for the training of nuclear plant operating personnel.

B. <u>Findings</u>

1. 10CFR55 (ref. 1) establishes the procedures and criteria for issuance of reactor operating licenses to operators of nuclear facilities including senior reactor operators (shift engineers and assistant shift engineers). In order to obtain a license as a reactor operator or senior reactor operator, the candidate must demonstrate an understanding of the design and operation of the Sequoyah facility including auxiliary systems (switchyard and offsite power supplies) which affect it.

2. ANSI/ANS standard 3.1 (ref. 4) - 1981 has been adopted by the NRC and identifies training requirements for reactor operators and senior reactor operators to be licensed by the NRC. Section 5.2 of this standard requires plant specific system instruction on power plant systems including electrical systems. In addition, it also specifies the content of required nuclear power plant fundamentals training which includes fundamentals of electrical theory.

3. NUREG-1021 (ref. 5) provides guidance to NRC examiners in determining the qualifications of an applicant for reactor operator and senior reactor operator licenses. Section ES-402, category 6, specifies that the candidate be able to reproduce from memory sketches and descriptions of various plant systems including electrical distribution systems and their mechanical components (in plant and switchyard). The candidate must also be able to discuss the design intent, construction, operation, and interrelationships of those systems on nuclear power plant operation and reactor safety. NUREG-1021, section ES-502, specifies control manipulations and plant evolutions for which an applicant for an SRO license must demonstrate proficiency. Control manipulations not performed at the plant may be performed on a simulator. One of the specified plant evolutions is a response to loss of electrical power and/or degraded power sources. A candidate's performance can be evaluated using the Sequoyah plant simulator.

4. A comprehensive operator training program has been developed and implemented to ensure that Sequoyah reactor operators and senior reactor operators meet the qualifications and training requirements established or endorsed by the NRC. This training program is described in Nuclear Power Area Plan Procedure 0202.05 (ref. 6) entitled "Nuclear Plant Operator Training Program."

5. Training of Sequoyah operators in electrical operation of plant and switchyard systems is conducted from the initial auxiliary unit operator training through the assistant shift engineer training. This training is comprehensive and covers details of electrical theory and the actual operation of switchyard equipment. The operators are required to pass tests to demonstrate their knowledge. The operation of electrical switchgear is a normal and routine part of the unit operator job. The electrical training program for nuclear plant operators is presented in four steps in Nuclear Power Area Plan Procedure 0202.05.

   a. Step 1 is a 13-week program on basic electrical theory and equipment. It is presented during the Nuclear Plant Operator Training Program (NOTP) during the student level II phase (prior to training for reactor operator or senior reactor operator). All ASEs and SEs must have successfully completed this training or its equivalent.

   b. Step 2A is a 2-week, inplant electrical training program on plant electrical systems (onsite and offsite) presented during the student level III phase. All ASEs and SEs must have successfully completed this training or its equivalent.

c. Step 2B is defined as unit operator upgrade electrical training and is a 4-week program of inplant training on plant electrical systems and station service. All ASEs and SEs must have successfully completed this training or its equivalent.

d. Step 3 is a 6-week ASE upgrade electrical training program required prior to taking the accrediting examination for ASE. All ASEs and SEs must have successfully completed this training or its equivalent. This training addresses both offsite and onsite electrical systems.

6. The Sequoyah Nuclear Plant Operator Training Program, which includes the electrical training, was one of the first in the nation to receive accreditation from INPO (ref. 11). This accreditation required a complete review of the training program and approval by an independent INPO Accreditation Board. INPO continues to review accredited programs on a regular basis to ensure the training meets their standards. Accreditation was received in January 1984.

7. Although no evidence was found of any poor operation of the switchyard at Sequoyah, there does appear to be poor relations between the operators at the plant and some Power System Operations (PSO) personnel. Normal operation of the switchyard is accomplished when the PSO dispatcher at the Chickamauga Dam Control Center calls the ASE at Sequoyah and gives instructions for any new configuration of the switchyard. The instructions are written down by the ASE and repeated verbatim to the dispatcher so that there will be no question as to what is to be done. Some PSO individuals that were interviewed felt that the nuclear plant operators did not react quickly enough to their requests for switchyard changes. They felt that this could endanger the reliability of the power system grid. PSO was also critical of the short notice, or, in some cases no notice, that they were given before one of the nuclear units was taken off line.

8. The "emergency" referred to in the concern is related to power system emergencies. No documented evidence was found in this investigation to substantiate the complaint of PSO personnel that Sequoyah switchyard operations were not carried out on a timely basis. Sequoyah shift engineers that were interviewed stated, however, that switchyard operations did not take first priority if the nuclear units were in an abnormal status.

IV. CONCLUSIONS AND RECOMMENDATIONS

A. Conclusions

This employee concern was not substantiated by this investigation because:

1. The Sequoyah shift engineers and assistant shift engineers are given extensive training in the operation of the switchyard (both classroom and on-the-job). The training meets NRC requirements.

2. No examples of poor switchyard operation or operation of this equipment in a manner that endangered the nuclear equipment at Sequoyah was found.

3. The shift engineers and assistant shift engineers receive training in electrical station operation that meets the NRC requirements and the Sequoyah training program has received INPO accreditation.

B. Recommendations

I-85-619-SQN-01 - Relations Between Plant Operator and PSO

There does appear to be some poor relations between PSO and the Sequoyah Nuclear Power organizations. This is of no nuclear safety significance, but in the interest of TVA power production and system reliability this issue should be addressed by Sequoyah and PSO management. This is an NSRS tracking item only. [P3]

TVA 64 (OS-9-65) (OP-WP-5-85)

NRC

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : H. L. Abercrombie, Site Director, Sequoyah Nuclear Plant

FROM : K. W. Whitt, Director of Nuclear Safety Review Staff, E3A8 C-K

DATE : DEC 10 1985

SUBJECT: NUCLEAR SAFETY REVIEW STAFF INVESTIGATION REPORT TRANSMITTAL

Transmitted herein is NSRS Report No. I-85-636-SQN

Subject TVA MANUFACTURE OF A DRAVO ASME-CLASS SPOOL PIECE

Concern No. XX-85-068-007

No response or corrective action is required for this report. It is being transmitted to you for information purposes only. Should you have any questions, please contact R. C. Sauer at telephone 2277.

Recommend Reportability Determination: Yes ____ No X

Director, NSRS/Designee

RCS:JTH
Attachment
cc (Attachment):
- R. P. Denise, LP6N35A-C
- R. J. Griffin, SQN E-18
- G. B. Kirk, SQN
- D. R. Nichols, E10A14 C-K
- QTC/ERT, Watts Bar Nuclear Plant
- Eric Sliger, LP6N48A-C
- J. H. Sullivan, SQN
- W. F. Willis, E12B16 C-K (4)

0150U

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS INVESTIGATION REPORT NO. I-85-636-SQN

EMPLOYEE CONCERN: XX-85-068-007

SUBJECT: TVA MANUFACTURE OF A DRAVO ASME-CLASS SPOOL PIECE

DATES OF INVESTIGATION: OCTOBER 8 - NOVEMBER 21, 1985

LEAD INVESTIGATOR: C. L. Breeding 12/6/85
C. L. BREEDING DATE

REVIEWED BY: R. C. Sauer 12/6/85
R. C. SAUER DATE

APPROVED BY: for M. S. KIDD 12/9/85
M. S. KIDD DATE

I. BACKGROUND

A Nuclear Safety Review Staff (NSRS) investigation was conducted to determine the validity of an expressed employee concern received by Quality Technology Company (QTC)/Employee Response Team (ERT). The concern of record, as summarized on the Employee Concern Assignment Request Form from QTC and identified as XX-85-068-007, stated:

> "Sequoyah - TVA may have manufactured a spool piece to replace, under ASME Section XI, a DRAVO ASME-class spool piece. When the spool piece was replaced, the Code nameplate from the DRAVO spool piece was removed, and affixed to the TVA manufactured spool. This may have been noted by a cognizant inspection individual (position unknown), and not reported due to the individual not wanting to get involved."

The ERT followup group was contacted for further details and information. None was available because they received this concern from an anonymous telephone call.

II. SCOPE

A. The scope of this investigation as determined from the stated concern to be that of four issues requiring investigation.

1. TVA may have manufactured an ASME Section XI spool piece.

2. TVA replaced a DRAVO spool piece with TVA manufactured spool piece.

3. The code nameplate was moved from the DRAVO piece to the TVA piece.

4. TVA inspector may have been aware of switch but did not report it.

B. The concern did not specify the location or equipment or piping that is of concern; therefore, a search was made for all DRAVO pipe supplied to Sequoyah. Also, the concern mentioned the ASME code used for the inservice inspection of nuclear power plant components; therefore, the requirements of this code for the Sequoyah plant were researched. The ability of TVA to manufacture spool pieces was also investigated.

III. SUMMARY OF FINDINGS

A. Requirements and Commitments

1. 10CFR50, Appendix A, General Design Criterion 1, "Quality Standards and Records," requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of safety functions to be performed.

2. 10CFR50.55a, "Codes and Standards," requires that components of the reactor coolant pressure boundary be designed, fabricated, erected, and tested in accordance with the requirements for class 1 components of section III of the ASME Boiler and Pressure Vessel Code or equivalent quality standards. This requirement went into effect for nuclear plants with construction permits submitted after January 1, 1975. Sequoyah had already received a construction permit.

3. The original design of Sequoyah was in accordance with ANSI B31.1 code for power piping with installation and inspection to ANSI B31.7. Additions and modifications at Sequoyah were made in accordance with the ASME code after April 1973 as stated in Sequoyah Nuclear Plant Design Criteria Manual, SQN-DC-V-3.0, "General Design Criteria for the Classification of Piping, Pumps, Valves, and Vessels," table 3.1-2. This table lists the code requirements for the plant design and their related TVA safety class.

4. The TVA Nuclear Quality Assurance Manual (NQAM), Part II, Section 2.3, revision 8/20/84, establishes controls to assure that repairs and replacements of ASME Section XI components are performed in accordance with ASME Section XI, IWA-4000 and IWA-7000, requirements.

5. TVA Construction Specification N2G-877, "Identification of Structures, Systems, and Components Covered by the Sequoyah Nuclear Plant Quality Assurance Program," requires certified material test reports (CMTRs), material traceability, and inspection documentation for Quality Level I materials.

6. The TVA Division of Nuclear Power Procedure DPM-N76A10, revised September 28, 1984, "Purchase Specifications for CSSC Metallic, Wire, and Cable Used Inside Primary Containment, Welding, and Brazing Materials, Valve Parts, and Pump Parts," specifies the "Code of Record," for Sequoyah. The code listed is ANSI B31.7-1971 Addenda.

7. Sequoyah Nuclear Plant Standard Practice SQA162, "Purchase Specifications for CSSC Materials," also lists ANSI B31.7 as the "Code of Record."

B. <u>Findings</u>

1. TVA has manufactured spool pieces at Sequoyah. Spool pieces are generally pipes, but sometimes the term "spool piece" is used for a pipe with flanges on each end that can be bolted into place. To manufacture one requires that a longer pipe be cut (thereby yielding two spool pieces) and flanges be welded on to make it a bolt-in "spool piece." TVA does this type of work

in compliance with ANSI B31.1 and B31.7 the "Codes of Record" for the plant. Fabricating pipe pieces to replace worn out, damaged, or otherwise unsuitable pipe is a normal part of operating and maintaining a power plant. A search of the maintenance records for spool pieces fabricated at Sequoyah was made. Several have been produced since the plant went into operation. A detailed review of the maintenance requests and inspection reports for four examples of this work was made. Maintenance requests for the fabrication of spool pieces in the component cooling system (MR 0654546), water treatment system (MR A049809), Auxiliary Feedwater System (MR A237954), and Reactor Coolant Pump Motor oil cooler (MR A299465) were reviewed along with the QA inspection reports of this work (refs. 14-17). No deviations from TVA procedures or code requirements were found.

2. No record of any DRAVO spool pieces having been delivered can be found at Sequoyah. DRAVO was contacted, and they have no record of supplying any spool pieces to Sequoyah. TVA records at the site and in the Chattanooga central offices show no contracts with DRAVO for pipe or spool pieces.

3. Spool pieces do not normally have ASME nameplates affixed. Nameplates are used on pressure vessels and other pressure containing devices but not on pieces of pipe. Most of the pressure vessels, piping, and other equipment at Sequoyah was designed and procured under the ANSI B31.1 code that did not require nameplates. Therefore, it is unlikely that a nameplate could have been moved since almost none exist at Sequoyah and there is no requirement for a nameplate on a new piece of pipe. Some of the spool pieces fabricated at Sequoyah were for temporary service such as flood mode crossties or nitrogen filling of steam generators. These temporary spool pieces are often reused and are labeled when built so they can be identified when needed.

4. It is permissible, even required in some cases, for the pipe identification number (the heat number) to be transferred from the original pipe to any spool piece cut from that pipe. An inspector is required to witness this activity. It is possible that an observer of this activity could have misconstrued the transfer of heat numbers to be the moving of a nameplate.

IV. CONCLUSIONS AND RECOMMENDATIONS

A. This employee concern is not substantiated for the following reasons:

1. No evidence of DRAVO spool pieces could be found at Sequoyah, and no record of their purchase was found.

2. Even though TVA does manufacture spool pieces for repair, replacement, or modification of plant piping systems, there could have been no exchange with DRAVO.

3. Code nameplates are not required at Sequoyah; therefore, the concern about any removal or attachment is not valid. No evidence of such activity was found in this investigation.

4. Inspection personnel at Sequoyah are familiar with the requirements for spool piece manufacture and know that nameplates are not required. There would, therefore, be no reason for an inspector to report an activity that did not violate a requirement or procedure.

B. This concern appears to have resulted from a misconception or misunderstanding of the requirements for producing spool pieces at Sequoyah. It is possible that an observer misconstrued the transfer of piping heat numbers to be the transfer of a nameplate. No action at Sequoyah is required.

**DOCUMENTS REVIEWED IN INVESTIGATION I-85-636-SQN
AND REFERENCES**

1. ASME Boiler and Pressure Vessel Code, Section XI, Article IWA-7000, IWA-4000, and Section III

2. 10CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records"

3. ANSI B31.1 - 1971 Addenda, "Power Piping"

4. ANSI B31.7 - 1971 Addenda, "Nuclear Power Piping"

5. NQAM Part II, Section 2.3, Revision 8/20/84, "Repairs and Replacement of ASME Section XI Components"

6. TVA Construction Specification N2G-877, "Identification of Structures, Systems, and Components Covered by the Sequoyah Nuclear Plant Quality Assurance Program," revised May 3, 1985

7. TVA Construction Specification N2M-865, "Field Fabrication, Assembly Examination, and Tests for Pipe and Duct Systems"

8. Division of Nuclear Power, Division Procedure DPM-N76A10, Revised September 28, 1984, "Purchase Specifications for CSSC Metallic Wire and Cable Used Inside Primary Containment, Welding and Brazing Materials, Valve Parts, and Pump Parts"

9. Sequoyah Nuclear Plant Standard Practice SQA162, "Purchase Specifications for CSSC Material," dated October 9, 1985, Rev. 0

10. U.S. NRC Regulatory Guide 1.26, Rev. 3, dated February 1976, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plants"

11. Sequoyah Nuclear Plant Design Criteria Manual, SQN-DC-V-3.0, "General Design Criteria for the Classification of Piping, Pumps, Valves, and Vessels"

12. SNP Construction Procedure No. P-34, "Heat Number Validation," revised December 12, 1978

13. Sequoyah FSAR, Chapter 3

14. MR 064546, dated January 8, 1981, "Fabricate Spool Piece for the Component Cooling System"

15. MR A049809, dated January 13, 1982, "Fabricate and Install Spool Piece for Train A DI," for Water Treatment System

16. MR A0237954, dated August 15, 1984, "1-PIPG-003, Rework to Venturi Spool Section to Match Length of Cavitating Venturi Sections," for the Auxiliary Feedwater System

17. MR A299465, dated September 20, 1985, "1-068, Fabricate Spool Piece for No. 1 RCPM 0.1 Cooler," for the Reactor Coolant Pump Motor Oil Cooler

TVA 64 (OS-9-65) (OP-WP-5-85)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : H. L. Abercrombie, Site Director, Sequoyah Nuclear Plant

FROM : K. W. Whitt, Director of Nuclear Safety Review Staff, E3A8 C-K

DATE : DEC 10 1985

SUBJECT: NUCLEAR SAFETY REVIEW STAFF INVESTIGATION REPORT TRANSMITTAL

Transmitted herein is NSRS Report No. I-85-750-SQN

Subject PERFORMANCE OF REMOTE VISUAL INSPECTIONS/RIGID PIPE SUPPORT

Concern No. XX-85-065-001

No response or corrective action is required for this report. It is being transmitted to you for information purposes only. Should you have any questions, please contact R. C. Sauer at telephone 2277 .

Recommend Reportability Determination: Yes _____ No X

Director, NSRS/Designee

RCS:JTM
Attachment
cc (Attachment):
- R. P. Denise, LP6N35A-C
- R. J. Griffin, SQN E-10
- G. B. Kirk, SQN
- D. R. Nichols, E10A14 C-K
- QTC/ERT, Watts Bar Nuclear Plant
- Eric Sliger, LP6N48A-C
- J. H. Sullivan, SQN
- W. F. Willis, E12B16 C-K (4)

0155U

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

INVESTIGATION REPORT NO. I-85-750-SQN

EMPLOYEE CONCERN NO. XX-85-065-001

SUBJECT: PERFORMANCE OF REMOTE VISUAL INSPECTIONS OF RIGID PIPE SUPPORT

DATES OF INVESTIGATION: OCTOBER 11-23, 1985

INVESTIGATOR: R.C. Sauer for

E. F. HARWELL

12/9/85

DATE

REVIEWED BY: R.C. Sauer

R. C. SAUER

12/9/85

DATE

APPROVED BY:

M. S. KIDD

12/9/85

DATE

I. BACKGROUND

A Nuclear Safety Review Staff (NSRS) investigation was conducted to determine the validity of an expressed employee concern received by Quality Technology Company (QTC)/Employee Response Team (ERT). The concern of record, as summarized on the Employee Concern Assignment Request Form from QTC and identified as XX-85-065-001, stated:

> "During Spring outage (February or March 1984) at Sequoyah, CI witnessed 2 ISI inspectors (names known) from baseline group performing "Remote Visual Inspections" on ERCW system rigid pipe supports in auxiliary building elevation 669' on horizontal pipe runs off the ceiling. CI defines "Remote Visual Inspections" as perfunctory, poorly performed visual inspections made from remote distances without actually verifying the mandatory inspection attributes on the inspection checklist."

The QTC/ERT followup group was contacted to obtain the names of the two inspectors in order to narrow the scope of the investigation.

II. SCOPE

A. The scope of the investigation was determined from the concern of record to be that of two specific issues requiring investigation:

1. Inspectors made inadequate visual inspections of suspended, rigid ERCW pipe supports in the auxiliary building at the 669' elevation during the February/March 1984 time frame.

2. Visual inspections must be performed at close proximity to verify specific mandatory inspection attributes (particulars) on the inspection checklist.

B. To accomplish the investigation, NSRS reviewed a computer printout of hanger examinations performed during the Sequoyah unit 1 cycle 2 (U1C2) outage (ref. 3). A determination was made as to which ERCW hangers on the 669' elevation could have been examined by the inspectors named by the CI. These inspection reports were then reviewed. Interviews were conducted with three ISI inspectors, the inspection supervisor in charge during the outage, a plant Quality Engineering and Control Group supervisor, and the onsite Authorized Nuclear Inservice Inspector (ANII) from Hartford Steam Boiler Company. Thirty ERCW hangers from the group inspected by one of the named inspectors were reexamined under the cognizance of the NSRS investigator. The results of this reexamination were reviewed to determine if the supports had been examined properly and if the programmatic procedures used in the inspections were adequate.

III. SUMMARY OF FINDINGS

A. Requirements and Commitments

1. ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components"

2. 10CFR50.55a

3. Sequoyah Technical Specifications, Section 4.0.5 and 3.4.4.10

4. NQAM, Part II, Section 5.1, "Inservice Inspection - Nuclear Power Plant Components"

5. Area Plan Program Procedure 1502.7 (formerly DPM N80E3), "NDE Procedures Approved for Use on CSSC Items at All Nuclear Plants"

B. Findings

1. The two ISI inspectors (Individuals C and D) named by the CI did not work together on ERCW hanger inspections. Individual D worked mostly on ultrasonic examinations during the UIC2 outage.

2. Individual C performed 20 ERCW hanger visual inspections on elevation 669 on February 27, 1984, accompanied by Individual E (in training).

3. Both individuals (C and E), when interviewed, said it was impossible to perform an adequate visual inspection of a hanger without having hands-on access.

4. The onsite ANII witnessed inspections performed by this pair on several occasions but not on this particular day.

5. The inspection reports did not indicate any type examination other than direct visual was utilized.

6. Individual C submitted 31 ERCW support inspection reports for the day in question.

7. The results of reexamining all the supports during this investigation are as follows:

   a. Arc strikes and weld splatter were found on embedded steel but had been there since initial construction and were painted over.

   b. Some pipe clamps had unequal distance betwen the ears but had equal loading around the pipe.

   c. One support had been deleted, but it appeared on the weld support isometric. A support in a grouping of five was improperly tagged with the deleted support number which resulted in an extensive inspection sheet being generated.

   d. One base plate had a loose bolt, but a conduit had to be moved to determine this condition.

These discrepancies were evaluated by the cognizant Level III NDE engineer (Individual I) and determined to be acceptable (ref. 7).

IV. CONCLUSIONS AND RECOMMENDATIONS

A. The employee concern could not be substantiated for the following reasons:

1. The two inspectors named by the CI did not work together on ERCW hanger inspections.

2. The two inspectors who did work together said it was impossible to do an adequate inspection remotely and recognized that it would be a violation of procedures to do so. Both said that it was not worth jeopardizing their jobs to do a poor inspection since they were not being pressured to meet a particular quota of inspections each day.

3. The reexamination of ERCW pipe hangers conducted during this investigation did not identify any major problems.

4. A plant QA staff manager said that he had not heard of an incident such as this employee concern and would have been notified if it had been reported to a supervisor.

5. The onsite ANII said he witnessed the two individuals performing inspections and did not believe they would do anything other than a proper inspection.

B. The CI may have witnessed an ISI inspector performing a preliminary walkdown of the ERCW system, prior to inspection, where a determination is made concerning the need for metal identification tags, insulation removal, and scaffolding and misconstrued this as a remote visual inspection of hangers. The actual documented inspection takes place at a later time when the identified preliminary findings have been addressed.

DOCUMENTS REVIEWED IN INVESTIGATION I-85-750-SQN AND REFERENCES

1. SQN Surveillance Instruction SI 114.1 Rev. 5, dated September 14, 1984, Unit 1 ASME Section XI In-Service Inspection Program

2. SQN Surveillance Instruction SI 114.1 data packages for the U1C2 outage

3. Printout of ERCW Hanger inspections performed during the unit 1 cycle 2 outage prepared by the NCO ISI Group

4. Inspection Records of visual inspections performed on ERCW hangers of Elevation 669 by Individual C during outage*

5. Preservice and Inservice Visual Examination Procedure, N-VT-1, Rev. 4, dated July 1, 1983

6. N-VT-1, Rev. 7, dated June 20, 1985

7. Memorandum (45D) from M. E. Gothard to Fonda Harwell dated November 27, 1985, entitled "Unit 1, Cycle 2, In-Service Inspection Employee Concern Allegation" with results of reexamination attached**

*These records are considered confidential as they contain the name of one of the individuals named in this employee concern.

**This document is considered confidential as it contains information critical to this investigation and is in the personal possession of E. F. Harwell.

TVA 64 (OS-9-65) (OP-WP-5-85)

NRC

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : S. Schum, QTC/ERT Program Manager, Watts Bar Nuclear Plant

FROM : K. W. Whitt, Director of Nuclear Safety Review Staff, E3A8 C-K

DATE : DEC 12 1985

SUBJECT: TRANSMITTAL OF ACCEPTED FINAL REPORTS

The following final reports have been reviewed and accepted by NSRS and are transmitted to you for preparation of employee responses.

I-85-299-WBN (IN-85-272-004)
I-85-364-WBN (IN-85-955-001)
I-85-440-WBN (IN-86-200-003)
I-85-441-WBN (IN-86-221-001)
I-85-549-WBN (IN-85-278-002)
I-85-720-WBN (IN-85-964-003)
I-85-753-WBN (IN-85-001-005, IN-85-007-003)

K. W. Whitt

Please acknowledge receipt by signing below, copying and returning this form to J. T. Huffstetler, E3B37 C-K.

NAME DATE

GDM
Attachments
cc (Attachments):
R. P. Denise, LP6N35A-C
E. R. Ennis, WBN
D. R. Nichols, E10A14C-K
Eric Sliger, LP6N48A-C
W. F. Willis, E12B16 C-K (4)

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TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS INVESTIGATION REPORT NO. I-85-299-WBN

EMPLOYEE CONCERN IN-85-272-004

MILESTONE 3

SUBJECT: CABLE OVERHEATING DUE TO FIRE-RETARDANT COATING

DATE OF INVESTIGATION: December 4, 1985

INVESTIGATOR: G. R. Owens 12/10/85
Date

REVIEWED BY: J. D. Smith 12/10/85
Date

APPROVED BY: for W D Stevens M. A. Harrison 12/10/85
Date

I. BACKGROUND

Employee Concern IN-85-272-004 was received by the Quality Technology Company (QTC) Employee Response Team that stated:

> 'Valcoat,' used to fireproof electrical cable in both units, may cause cables to overheat causing degradation of the cable insulation.

Note: The name of the fire-retardant coating used to coat electrical cables was Vimasco rather than Valcoat. Discussions with cognizant Office of Construction and Office of Engineering personnel failed to reveal any one who was familiar with a cable fireproofing called Valcoat.

II. SCOPE

NSRS Investigation Report No. I-85-569-WBN was found to encompass the concern related to possible cable overheating due to Vimasco coating. The investigation findings in that report therefore apply here. In addition, NSRS Investigation Report No. I-85-708-WBN addresses this issue as well.

III. SUMMARY OF FINDINGS

None; not reinvestigated.

IV. CONCLUSIONS AND RECOMMENDATIONS

Refer to NSRS Investigation Report No. I-85-569-WBN. No additional response is necessary. This item is closed; corrective action is tracked by I-85-569-WBN.

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS INVESTIGATION REPORT NO. I-85-549-WBN

EMPLOYEE CONCERN IN-85-278-002

MILESTONE 7

SUBJECT: PLANT RECORDS - ORIGINAL AND COPIES

DATES OF INVESTIGATION: November 25-27, 1985

LEAD INVESTIGATOR: John Knightly 12/6/85
J. J. Knightly Date

INVESTIGATOR: A M Gentry 12-6-85
A. M. Gentry Date

REVIEWED BY: J. D. Smith 12-9-85
J. D. Smith Date

APPROVED BY: for W D Stevens 12-10-85
M. A. Harrison Date

I. BACKGROUND

The Nuclear Safety Review Staff (NSRS) investigated Employee Concern IN-85-278-002 which Quality Technology Company (QTC) had identified during the Watts Bar Employee Concern Program. The concern was worded:

> No distinction is made between original records (hand-written ink entries), records which are partial copies and have original entries, and records which are completely copied, with respect to the document which is retained for archival record purposes. CI would not provide additional details/specifics. Constr. Dept. concern.

II. SCOPE

NSRS reviewed applicable requirements and procedures, records files, reports of audits, and correspondence concerning original records and copies. Additionally, records personnel responsible for archival records storage and maintenance were contacted to discuss records management practices as they relate to the employee's concern.

III. SUMMARY OF FINDINGS

A. Applicable Requirements and Commitments

1. ANSI N45.2.9-1974, "Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants" - "All . . . quality assurance records shall be legible, completely filled out and adequately identified to the item involved. . . . These records may be either the original or a reproduced copy."

2. Watts Bar Nuclear Plant Quality Control Instruction QCI-1.08, "Quality Assurance Records" - "The Responsible Engineer or Inspector . . . ensures that . . . documents are the original or a legible reproduced copy suitable for microfilming."

B. Findings

1. The use of legible record copies in place of originals was not a violation of nuclear records requirements. Additionally, in TVA's microfilm records storage system a copy is not necessarily distinguishable from an original. However, the use of original records is regarded as an important administrative practice which generally promotes greater legibility and microfilmability.

2. NUC PR-CONST Joint Audit JA-8300-01 conducted at Watts Bar April 4-14, 1983 identified legibility problems in selected N-5 data packages. It was determined that these legibility problems were not with CONST-originated documents but with vendor documentation. The corrective action included efforts by EN DES to secure and distribute best-available vendor record copies.

3. NUC PR inspections of the first 149 rolls of microfilmed records transferred from WBN CONST on August 16, 1985 identified only 40 potential legibility problems in their sample of 3,146 images selected from a total of 314,615. This is a potential problem rate of 1.3 percent. The NUC PR and CONST Document Control Unit (DCU) personnel stated their agreement that the actual problem rate is even lower because many of these potential problem records have been found to be acceptable. These are records which contain fully legible signatures, identifiers, key points for indexing, and DCU-A review stamps. (See memorandum WBN 841119 151 for details.) It was stated that in every instance the best available copy will be provided. Based on their extensive review of records transferred to date and CONST DCU's corrective action on problem images, NUC PR DCU personnel indicated a high level of confidence in the legibility of the transferred records. NSRS spot checks of these records did not identify any notable problems.

4. Certain types of older records, such as pre-1982 inspection records which included several different inspection points on the same document, were more likely to include some copies and partial copies. Procedural changes since 1982 which have provided for one inspection per record have helped correct this cause. DCU personnel stated that few legibility problems are encountered with current records and that received records are the originals in "99 percent of the cases." WBN Construction Quality Assurance records audit WB-A-85-10 dated July 9, 1985 and an NSRS spot check of current records did not identify problems related to the employee's concern.

IV. CONCLUSIONS AND RECOMMENDATIONS

Conclusions

Except in selected instances, the employee's concern was not substantiated. No widespread use of copies was identified nor was illegibility found to be a major problem.

Recommendations

None.

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS INVESTIGATION REPORT NO. I-85-720-WBN

EMPLOYEE CONCERN IN-85-964-003

MILESTONE 6

SUBJECT: IMPROPER DOCUMENTATION FOR SUBSTITUTED EQUIPMENT

DATE OF INVESTIGATION: December 5, 1985

INVESTIGATOR: John Mashburn — J. W. Mashburn — Dec 10, 85 Date

REVIEWED BY: WD Stevens — W. D. Stevens — 12/10/85 Date

APPROVED BY: WD Stevens — M. A. Harrison — 12/10/85 Date

I. BACKGROUND

The Nuclear Safety Review Staff (NSRS) was assigned Employee Concern IN-85-964-003 which had been identified by Quality Technology Company (QTC) during the Watts Bar Employee Concern Program. The concern was stated as follows.

> Material/Equipment is ordered dedicated to a specific system, unit, etc., but it is frequently installed/used elsewhere and it is unknown if documentation is revised to reflect this cannibalization. The concerned individual has no further information.

This concern is essentially the same as a previously-investigated concern reported in NSRS Investigation Report No. I-85-172-WBN dated November 22, 1985.

II. SCOPE

The previous investigation in this area was reviewed for applicablity to this concern, including the time period which it addressed versus when this concern was filed. It was found to be applicable, so a separate investigation was unnecessary.

III. SUMMARY OF FINDINGS

(See NSRS Investigation Report No. I-85-172-WBN.)

IV. CONCLUSIONS AND RECOMMENDATIONS

(See NSRS Investigation Report No. I-85-172-WBN.)

There were no recommendations.

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS INVESTIGATION REPORT NO. I-85-440-WBN

EMPLOYEE CONCERN IN-86-200-003

MILESTONE 6

SUBJECT: USE OF "REDHEADS" IS NOT SAFE

DATES OF INVESTIGATION: October 17-November 26, 1985

INVESTIGATOR: Pamela K. Howard 12/6/85
P. K. Howard Date

REVIEWED BY: P R Washer 12/11/85
P. R. Washer Date

APPROVED BY: M A Harrison 12/11/85
M. A. Harrison Date

I. BACKGROUND

NSRS has investigated Employee Concern IN-86-200-003 which the Quality Technology Company (QTC) identified during the Watts Bar Employee Concern Program. The concern is worded:

> The use of "Red Heads" for support is not safe in that the concrete could be honeycombed around the "Red Head". CI has no additional information.

II. SCOPE

The scope of the investigation was determined from the stated concern to be that the use of expansion shell anchors is not safe because the surrounding concrete could be undetectably honeycombed. The activities performed by NSRS during this investigation are listed below.

A. Review of Office of Construction (OC) WBN plant procedures including:

1. WBN-QCP-1.14, R17 and Various, "Inspection and Testing of Bolt Anchors Set in Hardened Concrete and Control of Attachments to Embedded Features"
2. WBN-QCP-1.42-2, R4 and Various, "Bolt and Gap Inspection for Bolt Anchor Assemblies"
3. WBN-QCP-1.47, R6, "Concrete/Grout Preplacement Inspection"
4. WBN-QCP-2.02, R7 and Various, "Concrete Placement and Documentation"

B. Review of Office of Engineering (OE) Civil Design Standard DS-C1.7.1, R3, "General Anchorage to Concrete"

C. Review of TVA Commitments and Requirements including:

1. Final Safety Analysis Report (FSAR)-WBN, Section 3.8, "Design of Category I Structures"
2. American Concrete Institute (ACI) 304-73, "Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete"
3. TVA General Construction Specification G-2, "Plain and Reinforced Concrete"
4. TVA General Construction Specification G-32, "Bolt Anchors Set in Hardened Concrete"
5. TVA General Construction Specification G-34, "Repair of Concrete"

D. Review of documentation including:

1. WBN Unit 1 Final Report, R2, "NRC-OIE Bulletin 79-02"
2. Nonconforming Condition Reports (NCRs) Nos. 146, 983, 3474, and 3799

E. Interviews with QC personnel knowledgeable in concrete installation and inspection practices.

III. SUMMARY OF FINDINGS

Based upon the review of applicable specifications, procedures, and documents, NSRS has not substantiated the identified concern. Described below are the results of the investigation that support the basis for the NSRS determination.

A. Review of TVA Commitments and Requirements

The FSAR for WBN Section 3.8 identifies the codes, standards, and specifications for which the design and construction of applicable structures was based.

1. Through the FSAR TVA was committed to batch, place, cure, and test concrete in accordance with ACI 304-73 and TVA General Construction Specification G-2. G-2 contains the controls by which concrete was placed and compacted to prevent the occurrence of honeycombing.

2. Through the FSAR TVA was committed to install and test bolt anchors set in hardened concrete in accordance with TVA General Construction Specification G-32. G-32 contains requirements for the qualification of anchors by each site in project-placed concrete, random proofloading of anchors, and the evalua[illegible]n of test results. The concrete would have to have adequate [illegible]ength for the anchors to pass the proofload test.

B. Review of QC WBN Plant Procedures

1. The requirements of G-2 were implemented in WBN-QCP-2.02 and -1.47. These procedures contain the acceptance criteria and the required documentation to implement G-2. In conclusion, plant procedures were in place to implement G-2 requirements although it was noted that many earlier concrete pours made at WBN were made in accordance with G-2 only since WBN-QCP-2.02 was not issued until June 1975.

2. The requirements of G-32 were implemented in WBN-QCP-1.14 and -1.42-2. These procedures contain the acceptance criteria and the required documentation to implement G-32. Additionally, WBN-QCP-1.14 specified that a visual check of the concrete condition be performed upon anchor inspection to allow evaluation of any damage that would render concrete acceptability doubtful.

C. OE Civil Design Standard DS-C1.7.1 Section 3.0 provides criteria for designing concrete anchorages utilizing self-drilling expansion shell anchors. A review of the standard revealed that a safety factor of 5.0 for tensile capacity and 4.0 for shear capacity was specified for anchor selection. This means that anchors should have been designed to take four to five times the load considered for normal operating conditions to allow for anchor installation errors, undetected concrete problems, and increased loads unforseen at the time of the design.

D. Documentation Review

1. Appendices H and I of the WBN Unit 1 Final Report on NRC Bulletin 79-02 were reviewed for relevance to the subject. The bulletin addressed pipe support base plate designs using concrete expansion anchors. Appendix H provided justification for factors of safety less than 5.0 but greater than 4.0 as determined through the sampling process for WBN. The report notes that tests have shown that the properties of the aggregate in the concrete affect the anchor capacity. The most significant point made for justification was that anchor capacities in TVA concrete are higher than those reported by the manufacturers; therefore, a lower factor of safety than 5.0 could be used without an increase in the potential for anchor failures. As noted in the report, the factor of safety of 5.0 specified in the bulletin was based partially on concerns with the affect of concrete degradation and cracking during a seismic event on the capacity of the anchors. Appendix I provided the results of cyclic load testing of expansion anchors. Tests were performed in both cracked and uncracked concrete. The report concluded that self-drilling expansion anchors would maintain their load-carrying capacity during and following a seismic event.

2. NCRs 146, 983, 3474, and 3799 were reviewed to determine if the quality program onsite was properly implemented in instances where concrete honeycombing was detected and considered to be major damage. The correction action for these NCRs was to repair the concrete areas in question. No problems were evident during the documentation review for the repaired areas; therefore, the program and its implementation appeared to be effective.

E. Interviews with QC personnel were conducted to obtain information relating to practices employed for concrete installation and inspection. This information was used to determine methods of documentation retrieval.

IV. CONCLUSIONS AND RECOMMENDATIONS

Conclusions

The concern was not substantiated since no existing evidence indicates that the capacity of expansion shell anchors was inadequate, due to honeycombed concrete that was not evaluated and/or repaired, to ensure performance of the anchors as intended.

Recommendations

No action is considered necessary.

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS INVESTIGATION REPORT NO. I-85-364-WBN

EMPLOYEE CONCERN IN-85-955-001

MILESTONE 6

SUBJECT: SECURITY POWER LOSS

DATES OF INVESTIGATION: December 4-7, 1985

INVESTIGATOR:	R. C. Cutshaw	12/9/85 Date
REVIEWED BY:	W D Stevens W. D. Stevens	12/9/85 Date
APPROVED BY:	for W D Stevens M. A. Harrison	12/9/85 Date

I. BACKGROUND

A concern was received by the Quality Technology Company (QTC) Employee Response Team that stated:

> Power is frequently and consistently lost at the Central Alarm Station when operations is transferring power to support on-going activities. This renders all systems (computers, vital area access doors, communications, etc.) inoperable. PSS management continuously requests advance notice of these events to no avail. Knoxville has been made aware of this problem and has responded by saying that the central alarm station power is independent of other power sources and this event cannot occur-yet it does. CI has no additional information.

II. SCOPE

The scope of this investigation was determined by the concern of record:

A. Determine the requirements regarding uninterruptible power to the Central Alarm Station (CAS).

B. Determine if power is frequently and consistently lost at the CAS, and what equipment is affected by the loss.

C. Assess the nuclear-safety/security implication of power loss at the CAS if substantiated.

III. SUMMARY OF FINDINGS

A. Requirements

Although the need for a redundant, independent, and/or continuous power supply to the Central Alarm Station (CAS) and Secondary Alarm Station (SAS) in total is not stated per se in 10CFR73, the general performance objectives for alarms, communications equipment, barriers, and other security-related devices dictates that such will be the case for certain security systems and subsystems within the CAS and SAS.

B. Interviews with a number of cognizant Public Safety Service Officers, specifically CAS/SAS operators and supervisors, revealed the following.

1. There was no recollection of total power losses to the CAS and/or SAS that affected all security-related or required equipment as stated in the concern of record. This was due to redundant and separate power trains.

2. Prior to August 1985 there had been an infrequent problem with a fluctuation of power to a security-related computer system in the CAS. This power fluctuation did have a temporary detrimental effect on a required system.

3. The power fluctuation corresponded in time with plant power-switching operations.

4. In August of 1985 an uninterruptible power source (UPS) was added to the computer system in question, thereby correcting the fluctuation-effect problem.

5. Since August 1985 power-switching operations have affected only the printout function of a required security system. The printout function was not a requirement, and its temporary loss did not have an adverse affect on the facility's security posture. The Public Safety Service is in the process of modifying that system to correct that problem.

IV. CONCLUSIONS AND RECOMMENDATIONS

A. Conclusions

1. The concern [illegible] is partially substantiated in that prior to August 1985 power-switching surges did have a detrimental [illegible] effect on a required security system. However, the addition of a UPS to the involved required system in August 1985 mitigated the nuclear-safety/security aspect of the concern. ([illegible] concern was filed with QTC prior to August 8, 1985.)

[illegible] modifications to security systems [illegible] hardware should [illegible] potential concerns over loss of function of [illegible] functions.

B. Recommendations

None.

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS INVESTIGATION REPORT NO. I-85-753-WBN

EMPLOYEE CONCERNS IN-85-001-005 AND IN-85-007-003

MILESTONE 6

SUBJECT: VENDOR WELDING

DATES OF INVESTIGATION: November 12, 1985

INVESTIGATOR: John C. Catlin Sr 12-9-85
J. C. Catlin Date

REVIEWED BY: J. D. Smith 12-9-85
J. D. Smith Date

APPROVED BY: for W D Stevens 12-10-85
M. A. Harrison Date

I. BACKGROUND

NSRS conducted an investigation regarding two employee concerns received by Quality Technology Company (QTC). Concern IN-85-001-005 received on October 15, 1985 stated: "Vendor welds were bought off even though they exhibited 'shoddy workmanship'." The allegation was nonspecific. Concern IN-85-007-003 received June 10, 1985 stated: "General look over vendor welds should be performed. Vendor welds are not inspected at WBNP 1 or 2. They are easily distinguishable from field welds because of the bad quality of the vendor welds. Vendor welds would not pass the same acceptance. . . ." This allegation was also nonspecific. During the course of the investigation a similar concern was noted; i.e., IN-85-372-001. This concern had been investigated by the Office of Construction and closed out by QTC.

II. SCOPE

The scope of the investigation included attempts to find a more specific example of the allegation and to track the example to its conclusion. QTC could provide no additional information other than to verify that the concerns were similar to IN-85-372-001.

III. SUMMARY OF FINDINGS

A. Requirements and Commitments

The nonspecific nature of the allegations rendered all requirements and commitments indeterminate.

B. Findings

1. Employee Concern IN-85-372-001 cited manway hatch covers as a specific example of substandard vendor welds.

2. NCR 6341 was written on September 25, 1985 which defined the nonconforming condition as: "Contractor welds for stiffener plates on hatch covers appear to not meet requirements of AWS D1.1. Welds appear to be undersized in places and have undercut and overlap. Reference employee concern IN-85-372-001."

3. NCRs 6345 and 6345A were written on September 25 and 26, 1985 covering Units 1 and 2, respectively. The nonconforming condition noted on the NCRs was similar to that of NCR 6341.

4. A statement was issued on Employee Concern IN-85-372-001 which stated in part that QC agreed that these welds were not of the quality expected of TVA personnel and that the contractor welds for stiffener plates on these hatch covers did not appear to meet the requirements of AWS D1.1 and also that the welds appeared to be undersized in places and have undercut and overlap. These were structural attachment welds which were not part of the reactor primary containment; and, therefore, they did not require a leak tightness test.

5. Disposition of all th[illegible] NCRs by Engineering was to "use as is" in accordance with mem[illegible]andum B26 851018 007.

IV. CONCLUSIONS AND RECOMMENDATIONS

A. Conclusions

1. The objective evidence of a similar employee concern substantiated the observed allegation of both concerned individuals (CI).

2. A typical case of a similar problem had been identified, reported, and documented in accordance with applicable procedures. Disposition was to "use as is."

3. Specific conclusions regarding these nonspecific allegations could not be reached.

B. Recommendations

None.