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 27 1985

SUBJECT: NUCLEAR SAFETY REVIEW STAFF INVESTIGATION REPORT TRANSMITTAL

Transmitted herein is NSRS Report No. I-85-776-SQN

Subject SOCKET WELDS NOT INSPECTED

Concern No. XX-85-108-001 and XX-85-108-002

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):
- Jim W. Coan, WBN P-104 SB-K
- 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)

0219U

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS INVESTIGATION REPORT NO. I-85-776-SQN

EMPLOYEE CONCERNS: XX-85-108-001
XX-85-108-002

SUBJECT: SOCKET WELDS NOT INSPECTED

DATES OF INVESTIGATION: NOVEMBER 18-26, 1985

LEAD INVESTIGATOR: E. F. Harwell 12/17/85
E. F. HARWELL DATE

REVIEWED BY: M W Alexander 12/17/85
M. W. ALEXANDER DATE

APPROVED BY: R.C. Sauer 12/20/85
R. C. SAUER DATE

I. **BACKGROUND**

A Nuclear Safety Review Staff (NSRS) investigation was conducted to determine the validity of two expressed employee concerns as received by the Quality Technology Company (QTC)/Employee Response Team (ERT). The concerns of record, as summarized on the Employee Concern Assignment Request Forms from QTC and identified as XX-85-108-001 and XX-85-108-002, respectively, stated:

> Sequoyah: C/I states welds in Unit #1 accumulator rooms and/or fan rooms were never inspected. Timeframe is nine or ten years ago. Welds on 2" stainless steel (socket welds) and hangers on the radius pipe in those areas. C/I has no additional info.

> Sequoyah: Programmatic breakdown on the weld inspection process. Nine or ten years ago C/I states that some welds on 2" stainless steel socket welds were not inspected as required. C/I has no additional info.

Since both concerns dealt with 2" stainless socket welds in the same timeframe, it is assumed that the two are related and will be addressed in a single investigation.

II. **SCOPE**

A. The scope of this investigation was determined from the stated concerns of record to be that of two specific issues requiring investigation.

   1. The construction weld inspection program was inadequate nine or ten years ago and did not assure that all stainless steel socket welds were inspected as required.

   2. Specifically, welds on 2" stainless steel (socket welds) and hangers in unit 1 accumulator and/or fan rooms were not inspected by the construction organization nine or ten years ago.

B. To accomplish the investigation, NSRS reviewed construction procedures and instructions related to weld inspection requirements and documentation at the time of interest of this concern (1975-76) and subsequent procedures and instructions. A review of mechanical drawings and flow diagrams for unit 1 was performed to determine what piping is present in the accumulator and fan rooms. A review of weld map isometrics of those systems in the area of interest were performed to determine weld map numbers. A review of the weld report computer printouts was then performed for these weld maps. A random review was performed of individual weld data sheets or computer data cards. Interviews were conducted with personnel having knowledge of inspection requirements during the concern timeframe (such as inspectors, mechanical engineers, welding engineer, and a QC records clerk).

SUMMARY OF FINDINGS

A. Requirements and Commitments

1. ANSI Standard B31.7 (1969) and 1970 Addendum - Nuclear Power Piping - Governed installation and inspection requirements for Sequoyah Safety Class A, B, C, and D piping systems.

2. ANSI Standard B31.1 (1967) - Power Piping - Governed installation and inspection requirements for Sequoyah Safety Class G and H piping systems.

3. 10CFR50 Appendix B - Basis for QA program utilized at Sequoyah.

4. SNP FSAR Section 3.2.2, "System Quality Group Classification (Fluid Components)."

B. Findings

1. References 11, 12, and 13 define the governing codes and safety classifications for various structures, components, and systems at SQN. From these guidelines, the drawings for various systems and structures identify the safety class(es) for design and installation.

2. References 1 and 3 defined the welding and nondestructive testing (NDE) requirements for various safety classes and configurations. For each weld to be performed on a safety class system or structure, an attachment A from these procedures (which defined the appropriate welding and NDE requirements) was to be filled out by a cognizant individual.

3. Until 1977, all welding and subsequent NDE activities were documented on Attachment A (Weld History Record)) of reference 2.

4. In 1977, Sequoyah instituted the use of a universal computer program to monitor inspection and test status during the construction of the plant. Every uniquely identified weld joint was entered into the program. At this same time, weld records were changed to computer cards for each required operation, such as fit-up, visual examination, liquid penetrant examination, etc.

5. From interviews with various knowledgeable personnel who worked at Sequoyah during construction, all safety class A, B, C, and D welds were required to be inspected and documented in accordance with procedures and instructions. These individuals reported that class E and G socket welds were visually examined and logged but that no QA documentation was prepared.

6. In order for a particular weld joint to be designated acceptable, a reference 2, Attachment A (Weld History Record), and/or all weld record computer cards had to be completed,

reviewed, and accepted by the QC Records Unit for storage as a permanent plant record in accordance with the requirements of references 7 and 9.

7. Input to the universal program required that incomplete or missing inspection documentation be reconstructed if adequate supporting documentation was available or the item was reinspected in accordance with the requirements of reference 8.

8. A review of the universal computer program printout for those piping systems within the unit 1 accumulator and fan rooms revealed that the above documentation system was utilized for class A, B, C, and D socket welds and that required socket weld inspections had been performed and the results were acceptable. Class E or G welds were not tracked on the computer program.

IV. CONCLUSIONS AND RECOMMENDATIONS

A. Neither employee concern of record could be substantiated for the following reasons:

1. The universal computer status system required that all documentation be present before the system could be transferred to Nuclear Power. Any safety class welds that were not examined prior to the utilization of the universal program would have been examined at a later date to meet QA record requirements.

2. The construction instructions and procedures in place at the time of the concern did require inspections and documentation; therefore, an adequate program was in place. However, the use of the universal program provided a better method of determining the present status of any weld and what remained to be done. Although the universal program provided a more positive means of preventing oversights, the old manual system could have provided the same assurance but by a much more laborious method.

B. The concerned individual (CI) may not have been aware of the changes made later in the weld documentation tracking program. In addition, the CI may not have been aware that class E or G welds did not require the same level of inspection as class A, B, C, or D welds.

DOCUMENTS REVIEWED IN INVESTIGATION I-85-776-SQN
AND REFERENCES

1. SNP Construction Procedure M-3, Rev. 2, dated May 1, 1975, "Welding Surveillance and Weld Procedure Assignment"

2. SNP Construction Procedure M-7, Rev. 14, dated November 19, 1976, "Erection and Documentation Requirements for Piping Systems"

3. SNP Construction Procedure M-20, Rev. 3, dated December 15, 1975, "Pipe Support Installation and Documentation"

4. SNP Construction Procedure W-3, Rev. 3, dated December 4, 1978, "Weld Procedure Assignment and Welding Surveillance"

5. SNP Inspection Instruction No. 63, Rev. 13, dated May 20, 1983, "Piping Inspection"

6. SNP Inspection Instruction No. 66, Rev. 16, dated March 1, 1983, "Inspection of Supports"

7. SNP Construction Procedure No. P-8, Rev. 10, dated August 24, 1976, "Quality Assurance Records"

8. SNP Construction Procedure No. P-8, Rev. 16, dated February 17, 1983

9. SNP Standard Operating Procedure No. 550, Rev. 0, dated December 14, 1977, "Review of Qualiy Assurance Records"

10. SNP Construction Procedure No. P-24, Rev. 4, dated March 28, 1980, "Inspection and Test Status"

11. SNP Final Safety Analysis Report, Section 3.0, Design Criteria - Structures, Components, Equipment, and Systems

12. SNP General Design Criteria, SQN-DC-V-3.0, Rev. 0, dated December 12, 1975, "Classification of Piping, Pumps, Valves, and Vessels"

13. SNP Construction Specification N2M-865, Rev. 3, dated April 12, 1977, "Field Fabrication, Assembly Examination, and Tests for Pipe and Duct Systems"

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 27 1985

SUBJECT: NUCLEAR SAFETY REVIEW STAFF INVESTIGATION REPORT TRANSMITTAL

Transmitted herein is NSRS Report No. I-85-472-NPS

Subject DRAWING CONTROL MASTER INDEX SYSTEM

Concern Nos. IN-86-108-002 (WBN), XX-85-062-002 (BFN) and XX-85-062-002 (BLN)

and associated prioritized recommendations for your action/disposition.

It is requested that you respond to this report and the attached Priority 2 [P2] recommendations by January 17, 1986. The Priority 3 [P3] recommendation will be looked at for corrective action follow through by April 1, 1986. No response is required for this item. Should you have any questions, please contact R. C. Sauer at telephone 2277.

Recommend Reportability Determination: Yes ____ No X

R. C. Sauer

Director, NSRS/Designee

RCS:JTH
Attachment
cc (Attachment):
W. C. Bibb, BFN
P. B. Border, BLN
G. G. Brantley, WBN
J. P. Darling, BLN
R. P. Denise, LP6N35A-C
R. J. Griffin, SQN E-18
G. B. Kirk, SQN
R. J. Mullin, 1350 CUBB-C
T. F. Newton, BFN
D. R. Nichols, E10A14 C-K
QTC/EPT, Watts Bar Nuclear Plant
Eric Sliger, LP6N48A-C
J. H. Sullivan, SQN
W. F. Willis, E12B16 C-K (4)

0222U

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS INVESTIGATION REPORT NO. I-85-472-NPS

EMPLOYEE CONCERN NOS: IN-86-108-002 (WBN)
XX-85-062-002 (BFN)
XX-85-062-002 (BLN)

SUBJECT: DRAWING CONTROL MASTER INDEX SYSTEM

DATES OF INVESTIGATION: OCTOBER 2-30, 1985

LEAD INVESTIGATOR: W. R. Simonds — W. R. SIMONDS — 12/6/85 DATE

INVESTIGATOR: R. E. McClure — R. E. MCCLURE — 12/6/85 DATE

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

APPROVED BY: M. A. HARRISON — 12/26/85 DATE

I. BACKGROUND

A Nuclear Safety Review Staff (NSRS) investigation was conducted to determine the validity of two expressed employee concerns as received by Quality Technology Company (QTC)/Employee Response Team (ERT). The concerns of record, as summarized on the Employee Concern Assignment Request Forms from QTC and identified as XX-85-062-002 and IN-86-108-002, stated:

> Concern: The current TVA system of using a master card file of drawings (this system in place at Sequoyah) is inadequate and should not be implemented (if not already) at Watts Bar, Browns Ferry, and Bellefonte. This system allows for instant disarray, distruction, or loss of reference cards if a drawer is dropped or the system sabotaged. The data from the cards should be input to the computer and available to all applicable offices within TVA.

From this stated concern, NSRS made the following assumptions:

- Reference 1 states that the Drawing Control Center (DCC) was established at each nuclear plant site for handling configuration control drawings. Configuration control (CC) is defined as the system whereby drawings are as-constructed prior to transfer and upon completion of each workplan. The only drawings not under CC are those not used to build the plant or to verify equipment configuration. Therefore, it is assumed by the investigator that "drawings" as used in the above concern means "CC drawings."

- A master index card file system (master index) was generally accepted for use prior to 1983 within NUC PR to maintain as-constructed status and distribution information for CC drawings on file in the DCC. However, references 1, 2, 7, and 14 indicate that the Drawing Management System (DMS) has replaced the master index, in addition to other systems, and is currently intended to be the single CC drawing management system for the Office of Power and Engineering (P&E) (Nuclear). Therefore, it is assumed by the investigator that the above concern inadvertently refers to the master index as "the current TVA system. . . ."

II. SCOPE

A. The scope of the investigation is defined by the concerns of record to be two specific issues requiring investigation as follows:

   1. The master index used by the Sequoyah (SQN) DCC for CC drawings is inadequate and should not be implemented at TVA's other nuclear plants.

   2. Information contained in the SQN master index should be established as a computerized data base which is accessible to all applicable TVA offices.

B. NSRS reviewed correspondence to identify commitments and audit findings applicable to the drawing control program for TVA's nuclear power plants. Quality assurance program procedures and plant instructions were also reviewed to identify programmatic controls. Interviews were conducted with nine DCC employees and CC Task Force members to obtain details of current CC drawing control practices and requirements.

III. SUMMARY OF FINDINGS

A. Requirements and Commitments

1. NUC PR NQAM, Part III, Section 1.1, dated March 21, 1985, "Document Control."

2. NUC PR NQAM, Part V, Section 6.1 (ID-QAP-6.1), dated December 31, 1984, "Configuration Drawing Control."

3. Memorandum from H. J. Green to A. W. Crevasse, "Joint Quality Assurance Audit Report No. JA8100-06 (A24 820128 001)," dated March 8, 1982 (L16 820305 876), commits TVA to develop the DMS in accordance with the CC Task Force recommendations (ref. 14).

4. Memorandum from J. A. Coffey to Manual Holders, "BFN Regulatory Performance Improvement Plan (RPIP) Oversight Group Meeting Minutes No. 84-12," dated July 19, 1984 (L24 840719 800).

5. Memorandum from J. A. Coffey to R. J. Mullin, "BFN Deviation JA8100-06-A05 As-Constructed Drawings," dated February 2, 1985 (R25 850212 952), provides status and updated progress in implementing CC Task Force recommendations.

6. TVA Topical Report TVA-TR75-1A, Revision 8, Table 17D-3 commits TVA to US NRC Regulatory Guide 1.33, (Revision 2), dated February 1978, "Quality Assurance Program Requirements (Operations)," which endorses use of ANSI N18.7-1976.

7. ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants."

B. Findings

1. SQN Master Index

a. The master index was implemented through reference 19 for CC drawings on July 11, 1979. It contains an index card for each CC drawing on file in the DCC. The cards are intended to contain the as-constructed status and distribution information applicable to the corresponding drawings. SQN DCC personnel stated that they continue to maintain the master index in accordance with reference 19 and that they rely upon the index cards as a source of information to identify CC drawing revision level, as-constructed status, and distribution.

b. As a result of SQN DCC personnel interviews, it was determined that access to the master index is not controlled, nor is it periodically audited to verify the integrity of the information on the cards.

c. References 1 and 2 do not identify the master index as an approved CC drawing control system.

2. Drawing Management System (DMS)

a. References 1, 6, and 7 collectively identify the DMS as the single system to be used within P&E (Nuclear) to store, update, and retrieve construction status and other information for TVA and vendor drawings. The DMS data base is a controlled document which is periodically audited and is protected by security features.

b. Reference 23 states that the DMS replaces several systems that previously contained drawing information which were not generally accessible to all TVA organizations needing the information. As-constructed status, revision level, and distribution information can be retrieved by any authorized DMS user through online access to TVA's mainframe computer.

c. Browns Ferry converted its CC drawing control system to the DMS effective December 21, 1983. BFN DCC personnel update the DMS for BFN CC drawings in accordance with upper-tier requirements and utilize the DMS exclusively to obtain distribution and construction status information.

d. SQN has placed all of its CC drawing information onto the DMS but has not fully divorced its use of the Master Index card system. As a result, SQN has on occasion used the index as its official drawing record control system for distribution and construction status information.

e. Watts Bar DCC personnel use the DMS for CC drawing control according to the WBN DCC Supervisor.

f. Bellefonte DCC personnel use the DMS for CC drawing control according to the BLN DCC Acting Supervisor.

IV. CONCLUSIONS AND RECOMMENDATIONS

A. Conclusions

1. The employee concern is substantiated not on the specifics of the concern but its substance. The basis for this conclusion follows:

   a. The "current TVA system" for configuration control is intended to be the DMS not the master card index system as alleged. However, upper-tier instructions do not preclude the use of the master card file and, therefore, though SQN DCC personnel are complying with the updating of DMS for SQN CC drawings in accordance with upper-tier (reference 1 and 2) requirements, they are not meeting the literal intent of the requirements to use DMS exclusively, since SQN DCC personnel continue to maintain and utilize the master index.

   b. The SQN master index system is not a reliable CC drawing control system since it is not controlled nor is it an approved QA program method.

   c. The master index is inadequate for TVA use because it is a localized system which does not provide accessibility to other TVA organizations having a need to obtain the proper as-constructed status of SQN equipment.

2. Accordingly, P&E (Nuclear) decided to replace all the systems such as the master index with the DMS as the single CC drawing management system for P&E (Nuclear). Browns Ferry, Watts Bar, and Bellefonte do not use a master index system for CC drawing control but do use DMS.

B. Recommendations

1. I-85-472-NPS-01, <u>Configuration Drawing Control Requirements</u>

   The upper-tier requirements contained in references 1 and 2 do not preclude the continued use of master index systems for construction status and other information retrieval for CC drawings. NSRS recommends:

   a. That references 1 and 2 be revised to require DMS utilization exclusively by site DCC organizations for CC drawing information retrieval, and

   b. that the master index be placed under the QA program as a backup system, if desired, or until total conversion to DMS occurs. [P2]

2. I-85-472-NPS-02, SQN Implementation of DMS

SQN DCC has not implemented the DMS as the single CC drawing management system. NSRS recommends that SQN revise reference 19 to require exclusive utilization of DMS for CC drawing information retrieval. This action should establish the DMS as the single CC drawing management system and discontinue DCC utilization of the master index. [P2]

3. I-85-472-NPS-03, Need to Perform Audit/Survey Master Index System

Because SQN DCC is using the non-QA approved Master Card Index system on occasion as its official drawing control system for distribution and construction status information, DQA is requested to audit/survey the adequacy of this area in the near future and on a periodic basis until total conversion to DMS occurs.

No response to this issue is required. NSRS will moniter the progress of this issue based on the responses made and actions taken to 1 and 2 above. [P3]

DOCUMENTS REVIEWED IN INVESTIGATION NO. I-85-472-NPS
AND REFERENCES

1. NUC PR NQAM, Part III, Section 1.1, dated March 21, 1985, "Document Control"

2. NUC PR NQAM, Part V, Section 6.1 (ID-QAP-6.1), dated December 31, 1984, "Configuration Drawing Control"

3. Drawing Management System (DMS) Users Manual, Revision 0 (B42 850606 509)

4. Memorandum from H. L. Abercrombie to Those listed, dated August 24, 1983, "Document Control and Drawing Management Meeting Notes" (L68 830817 801)

5. Memorandum from M. M. McGuire to R. D. Guthrie, dated August 14, 1984, "DMS Subtask Group of the Configuration Control Task Force" (L99 841023 003)

6. Memorandum from R. D. Guthrie to Joe W. Anderson, dated May 3, 1985, "Request for Drawing Management System (DMS) Quality Control and Information Systems Analysis Services" (R25 850501 910)

7. BFN - Regulatory Performance Improvement Plant - Oversight Group Meeting Minutes - No. 84-12

8. Memorandum from R. H. Wright to PWR Project Files, dated April 17, 1984, "SQN - As-Constructed Drawing Meeting Notes" (BWP 840417 004)

9. Memorandum from R. H. Wright to PWR Project Files, dated May 30, 1984, "SQN - As-Constructed Drawing Task Force" (BWP 840530 001)

10. Memorandum from R. H. Wright to PWR Project Files, dated August 16, 1984, "SQN - As-Constructed Drawing Task Force" (BWP 840816 006)

11. Memorandum from R. H. Wright to PWR Project Files, dated March 21, 1984, "BFN - As-Constructed Drawing Task Force - Meeting Notes" (BWP 840321 004)

12. Memorandum from H. J. Green to A. W. Crevasse, dated March 8, 1982, "Joint Quality Assurance Audit Report No. JA8100-06" (A24 820128 001) (L16 820305 876)

13. Memorandum from J. A. Coffey to R. J. Mullin, dated February 12, 1985, "BFN - Deviations JA8100-06-A05 - As-Constructed Drawings" (R25 850212 952)

14. Task Force Report, dated June 3, 1983, "Report of Recommendations from the As-Constructed Drawing Task Force" (A43 830711 005)

15. OEDC and QA&AS Joint Quality Assurance Audit Report No. JA8100-06, dated January 28, 1982 (A24 820128 001)

16. ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants"

17. 10 CFR Part 50, Appendix B, Criterion VI, "Document Control"

18. SQN M&AI-3, Revs. 4 and 5, "Revision of As-Constructed Drawings"

19. SQN AI-25, Revisions 0 and 10, "Receipt, Filing, and Distribution of Drawings"

20. WBN AI-4.10, Revision 3, "Drawing Control Center for a Licensed Unit"

21. BLN BLA5.9, Revision 17, "Drawing Control Before Receipt of an Operating License"

22. BFN DCU-IMM-X.1, dated September 22, 1985, "Drawing Control"

23. Memorandum from R. D. Guthrie to James P. Darling, dated October 5, 1984, "BFN - Drawing Management System (DMS) Budet Request" (R25 841002 864)

24. Memorandum from Michael L. Scalf to R. D. Guthrie, dated July 15, 1985, "DMS Overview II" (B04 850715 902)

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

NRC

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : H. G. Parris, Manager of Power and Engineering (Nuclear), MR6N011

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

DATE : DEC 27 1985

SUBJECT: NUCLEAR SAFETY REVIEW STAFF INVESTIGATION REPORT TRANSMITTAL

Transmitted herein is NSRS Report No. I-85-590-SQN

Subject REACTOR COOLANT SYSTEM INSTRUMENT SENSE LINE SLOPE

Concern No. XX-85-046-001

and associated prioritized recommendations for your action/disposition.

It is requested that you respond to this report and the attached Priority 1 [P1] and 2 [P2] recommendations by January 17, 1986. Should you have any questions, please contact R. C. Sauer at telephone 2277.

Recommend Reportability Determination: Yes X No ____

R. C. Sauer

Director, NSRS/Designee

RCS:JTH
Attachment
cc (Attachment):

- H. L. Abercrombie, Site Director, SQN
- R. W. Cantrell, W12A12 C-K
- 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-85-590-SQN

EMPLOYEE CONCERN: XX-85-046-001

SUBJECT: REACTOR COOLANT SYSTEM INSTRUMENT SENSE LINE SLOPE

DATES OF INVESTIGATION: November 5-15, 1985

INVESTIGATOR: N.T. Henrich by MWA 12/20/85
N. T. Henrich — Date

REVIEWED BY: MW. Alexander 12/20/85
M. W. Alexander — Date

APPROVED BY: R.C. Sauer 12/23/85
R. C. Sauer — 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-85-046-001, stated:

> Instrument sensing lines for system 68 at Sequoyah may have slope deficiencies. Details known to QTC, withheld due to confidentially. Construction department concern. C/I has no further information.

Further information was requested from the ERT followup group to identify any specific instruments or panels which may have sensing line slope deficiencies. This information was withheld to protect the confidentiality of the concerned individual.

II. SCOPE

A. The scope of this investigation was determined from the stated concern to be that of a single specific issue requiring investigation:

- The reactor coolant system instrument sense lines may have slope deficiencies.

It should be noted that employee concern PH-85-001-002 (ref. 20) initiated at Watts Bar Nuclear Plant (WBN) alleged instrument sense line slope deficiencies in the Reactor Coolant System (RCS) associated with Watts Bar local panels L-226, -227, and -228. RCS flow transmitters are located on these panels. Due to the strong similarity between Sequoyah (SQN) and WBN, the scope of this investigation focused on an evaluation of instrument sense line slope for SQN reactor coolant system flow transmitters located on SQN local panels L-226, -227, and -228.

B. To accomplish this investigation, NSRS reviewed SQN design criteria and construction specifications related to the installation and inspection of instrument sense lines. A walkdown of selected RCS flow transmitter sense lines was conducted on unit 1 and progressed from the RCS elbow process taps to the transmitters on local panels L-226, -227, and -228. Plant surveillance and instrument maintenance instructions for these transmitters were also reviewed. To determine operational and maintenance history on these instruments, plant maintenance records were reviewed and interviews were conducted with instrument maintenance personnel.

III. SUMMARY OF FINDINGS

A. Requirements and Commitments

The only identifiable instrument sense line slope requirement is defined by a note on TVA drawing 47W600-24 (ref. 10) which states:

> All sensing lines to be field routed from local panel through proper sleeves to sensing point. All liquid service flow, level, or pressure sensing lines shall slope 1/8 inch per foot minimum downward to termination of line 3 inches above local panel. All condensable vapor service sensing lines shall slope upward 1/8 inch per foot minimum from sensing point to condensing chamber (high point) and shall then slope downward 1/8 inch per foot minimum to termination of line 3 inches above local panel. . . . Field to route all lines in a manner that will allow for thermal expansion and leave lines free of traps.

B. Findings

1. There are no US NRC Regulatory Guides or NUREGs which define or provide guidance on instrument sense line slope requirements.

2. The Instrument Society of American (ISA) has issued standard S67.02-1980 entitled "Nuclear Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants" (ref. 18). This standard covers the design, protection, and installation of instrument sense lines in light water reactors but does not define acceptance criteria for the slope of these lines. In addition, TVA was not committed to the implementation of the requirements of this standard in the design of SQN.

3. International Standard ISO 2186, "Fluid Flow in Closed Conduits - Connections for Pressure Signal Transmissions Between Primary and Secondary Elements (ref. 19), states "sense lines should be arranged so that their slope is always greater than 1 inch per foot in order that any gas bubbles may rise to vents and so that condensed liquids may drain to catchpots or water seals." It also allows sense lines to be run in a series of slopes provided that vents are installed at all high points and sealing chambers at all low points. TVA is not committed to this standard and has required an 1/8 inch per foot minimum slope.

4. ASME Research Committee on Fluid Meters recommends instrument sense lines be so arranged and installed so as to have a slope of 1 inch per foot or more (ref. 21). TVA is not committed to this recommendation and has required an 1/8 inch per foot minimum slope.

5. All instrument sense lines, including those in the reactor coolant system, were field routed by construction. No controlled TVA drawing was issued to show sense line routing and installation. Refer to general notes on TVA drawing 47W600-24 (ref. 10), which states:

> "All sense lines to be field routed from local panel through proper sleeves to sensing point."

Specifically, the instrument sense lines were to be installed in accordance with Sequoyah Construction Specification No. N2M-865, "Field Fabrication Assembly Examination and Tests for Pipe and Duct Systems" (ref. 5). This procedure defined the requirements for fabricating and installing instrument sense lines, but it did not establish sense line slope requirements.

6. After installation of instrument sense lines, Sequoyah Construction Mechanical and Welding Inspection Unit inspected and documented the installation, flushing, and pressure testing of the sense lines. These inspections were done in accordance with Inspection Instruction II-85 "Installation Verification, Flushing, and Pressure Testing of Instrumentation Sensing Lines, Sampling Lines, Control Supply Headers, and Signal Lines" (ref. 3).

This inspection included verification of color coding and tagging of the instrument sense lines; routing of the sense lines; installation of high point vents and condensate chambers; root, vent, and isolation valve installation and operation; size and type of material used; and sense line support.

No specific acceptance criteria was called out in II-85 with regard to sense line slope. However, Section 8.0, Acceptance Criteria, item A.2 stated that "Line routing shall be specified on the applicable I&C drawing and/or the applicable process piping diagram." TVA drawing 47W600-24 is applicable and specified the required slope for RCS flow transmitter sense lines.

7. Inspection Instruction II-85 data cards were reviewed for all 24 reactor coolant flow transmitters listed in Attachment 1.

The II-85 data cards for the unit 1 transmitters indicated the routing of the sense lines for each transmitter was acceptable. However, variances were noted for each transmitter in the remarks portion of the data card. All of these variances were written against discrepancies in the span between sense line supports. Each variance was reviewed and approved by Office of Engineering, Civil Engineering Branch. No variances were written against sense line slope.

The II-85 data cards for the unit 2 transmitters indicated that the routing of the sense lines for each transmitter was acceptable, but each data card referenced NCR 2358 (ref. 15).

This NCR was written to document a nonconformance related to sense lines from different protection sets run on the same support. Office of Engineering determined that the nonconformance was not a significant condition, and the installation was approved for use "as-is." The NCR did not address sense line slope.

8. A walkdown of the instrument sense lines on selected RCS flow transmitters on unit 1 was performed. The following observations are considered typical for all loops:

   a. The high pressure elbow taps on the reactor coolant system piping are approximately 81 inches above the reactor building elevation 679.78 feet. The low pressure elbow taps are approximately 98 inches above the same floor elevation.

      The RCS flow transmitters are mounted on local panels L-226, -227, -228. Panel arrangements are shown in Attachment 2. The sense lines to the upper transmitters are approximately 44 inches above floor elevation 679.78 feet; while the sense lines to the lower transmitters are approximately 22 inches above this elevation. Assuming the floor is level the process taps are at a higher elevation than the process transmitters which is proper.

   b. The sense lines run vertically from the process taps to high point vent valves which are approximately 220 inches above reactor building floor elevation 679.78 feet. From there the sense lines are routed to the local panels.

   c. An inspection of sense line slope in the vicinity of 6 high point vent valves inside the polar crane wall revealed incorrect slope for transmitters 1-FT-68-48D and -71A such that the vent valves were not at the high point in the sense line.

   d. The sense lines are not always routed continuously downward from the high point vent valves to the local panels. Each sense line has a low point between a high point vent valve located above the process tap and a second vent valve located above the local panel. This is not consistent with the stated requirement.

   e. Sense lines do not consistently slope a minimum of 1/8 inch per foot. Some runs are sloped in the wrong direction or are essentially level. This is not consistent with the stated requirements.

9. Plant Technical Specifications (STS) specify that RCS total flow rate indicators be subjected to a channel calibration at least once every 18 months (STS 4.2.3.4). In addition, RCS total flow rate is to be determined by measurement at least once every 18 months (STS 4.2.3.5).

a. Surveillance Instruction SI-155 (ref. 12) is used to determine reactor coolant flow by measuring heat inputs and outputs and feedwater flow rates to the steam generators. With this data and knowing fixed inputs and losses, reactor coolant flow can be calculated. With the RCS flow rate known, SI-246 (ref. 11) is then used to calibrate RCS flow transmitters on an as needed basis.

b. Surveillance Instruction SI-246 (ref. 11) is performed on an as needed basis after each refueling outage. It is implemented on request from the nuclear engineers during performance of SI-155 based upon data obtained during this SI. SI-246 recalibrates (rescales) the RCS flow transmitters with the reactor at power to ensure these instruments accurately indicate RCS flow as determined from calorimetric measurements and calculations.

c. Surveillance Instruction SI-94.2 (ref. 27) implements the RCS flow instrumentation channel calibration refueling requirements specified in STS 4.2.3.4 and applicable portions of STS 4.3.1.1.1. SI 94.2 accomplishes this by completing portions of Instrument Maintenance Instruction IMI-99 (ref. 13).

d. Instrument Maintenance Instruction IMI-99, Sections CC 6.13A through CC 6.24A (ref. 13), utilizes a wet calibration technique when calibrating the RCS flow transmitters. This technique uses a water box design which allows the calibration to be performed without draining any portion of the sense lines. Since RCS is not normally drained down to the RCS flow transmitter process taps during refueling outages, no air is in the process line to migrate into the sense line. This technique ensures that the sense line from the sense line isolation valve at the local panel to the transmitter remains solid during the calibration as well as when returning the transmitter to service.

e. Instrument Maintenance Instruction IMI-118 (ref. 9) is not routinely used during the calibration of the RCS flow instrumentation if the instrument sense line is filled. If instrument maintenance personnel suspect the sense line has entrapped air or may not be completely filled, the sense lines are backfilled prior to calibrating the transmitter in accordance with IMI-118. In most cases backfilling is not required. IMI-118 is also used to remove air from the sense line as part of maintenance activities when air entrapment is suspected.

Completion of these surveillance instructions on a routine basis ensures proper calibration and operation of the transmitters.

10. Air entrapped in instrument sense lines typically manifests itself as an erratic signal on differential pressure

measurements such as RCS flow. This is due to the low pressures applied to the transmitter. Attachment 3 shows a strip chart recording of a typical erratic flow signal (not RCS flow) caused by air entrapped in the sense line and then subsequently removed. Although it is possible that a large air bubble in a vertical section of sense line may cause a zero shift upscale or downscale depending on which line contains the bubble, it would also be characterized by an erratic transmitter output signal similar to that shown on Attachment 3. This erratic signal can be detected by observing flow indicators or recorders and comparing redundant flow channels during periodic channel checks.

11. If air entrapment in an RCS flow transmitter sense line is suspected, a Maintenance Request (MR) is prepared for Instrument Maintenance to investigate and resolve the problem. A review of MRs for the RCS flow transmitters (ref. 14) on both units 1 and 2 was performed. A total of 99 MRs were identified from November 17, 1978, to the present on these transmitters. Only five MR's could possibly be associated with air entrapment in the sense lines or sense lines not being filled during this period. A summary of these MR's is given below:

MR	Date	Transmitter	Unit Status
048281	4/15/79	1-FT-68-6A	Hot Functional Tests (venting RCS)
059641	1/09/80	1-FT-68-6B	Preoperational Tests (RCP 1 maintenance)
059642	1/09/80	1-FT-68-6D	Preoperational Tests (RCP 1 maintenance)
059643	1/09/80	1-FT-68-48B	Preoperational Tests (Low RCS level)
059644	1/09/80	1-FT-68-71D	Preoperational Tests (Low RCS level)

A review of Unit Operator, Assistant Shift Engineer, and Shift Engineer logs for April 15, 1979, revealed the RCS was being vented with RCS pumps being started and stopped periodically. A similar review of logs for January 9, 1980, revealed that reactor coolant pump 1 was being started and stopped to check repairs for an oil leak. These activities could have contributed to the instrument problems noted on these MRs. Investigation revealed there was no problem with the transmitters and the problem was corrected by equalizing the transmitters. MRs 059643 and 059644 were written when low RCS water level apparently resulted in incomplete filling of the sense lines to the RCS flow transmitters. System activities on

April 15, 1979, and January 9, 1980, are not typical of those conditions encountered during normal plant operation.

12. The investigation of Watts Bar Employee Concern PH-85-001-002 (ref. 20) by QTC identified some specific discrepancies with instrument sense line slope for WBN unit 1 RCS flow transmitters (ref. 21). These deficiencies were subsequently evaluated by TVA in WBN-NCR 6172, Revision 0 (ref. 22), dated July 9, 1985. A subsequent TVA inspection of a random sample of instrument sense lines revealed a generic deficiency with regard to WBN sense line slope requirements. This was documented by WBN NCR 6172, Revision 1 (ref. 23), dated September 12, 1985. The Engineering Report was prepared in accordance with Office of Engineering Procedure OEP-17 (ref. 24), and documents the corrective action proposed for WBN. The NCR was not transmitted to SQN for review of generic implications. Attachment 4 summarizes the events addressed in preparing, evaluating, and dispositioning NCR 6172, Revision 0 and Revision 1.

13. An overall engineering evaluation was verbally requested by NSRS of the SQN Site Director and is now underway by the SQN site design organization to assess slope conditions of all safety-related instrument sense lines which might be susceptible to air entrapment. This evaluation was also added as a licensing commitment to the NRC per reference 26.

IV. CONCLUSIONS AND RECOMMENDATIONS

A. Conclusions

The concern of record was substantiated. Several sense lines for the reactor coolant system flow transmitters were found to have slope conditions that are not presently in accordance with stated requirements. However, a review of operational histories for these instruments does not indicate SQN has experienced significant problems with air entrapment in their sense lines. Air entrapment in instrument sense lines is generally detectable during operation, and its removal can be accomplished with existing plant procedures.

B. Recommendations

1. I-85-590-SQN-01, <u>Engineering Evaluation of Instrument Sense Line Slopes</u>

   The engineering evaluation requested of SQN Site management in paragraph III.B.13 involving safety-related instrument sense line slope should be completed and any identified corrective actions implemented by the site. The results of this evaluation should be forwarded to NSRS for an independent evaluation. [P1]

2. I-85-590-SQN-02, High Point Vent Valves

The routing of reactor coolant system flow transmitter sense lines should be reverified to ensure high point vent valves are clearly the high point in the sense line. [P1]

3. I-85-590-SQN-03, Returning RCS Flow Transmitters to Service

Plant procedures should be revised or developed to require safety-related sense lines which may be sensitive to air entrapment be backfilled during unit outages just prior to entering the mode in which the associated instrument is required to be operable. The procedure should require a channel check to verify normal indication when these instruments are returned to service. In addition, channel checks to determine if backfilling is required whenever one of these instruments is calibrated and returned to service during unit operation should be incorporated into the procedure. [P1]

4. I-85-590-SQN-04, Sense Line Installation Procedure

Appropriate instrument sense line routing and slope criteria should be obtained from OE and incorporated in site procedures for installation of new sense lines or significant modification to existing sense lines. [P2]

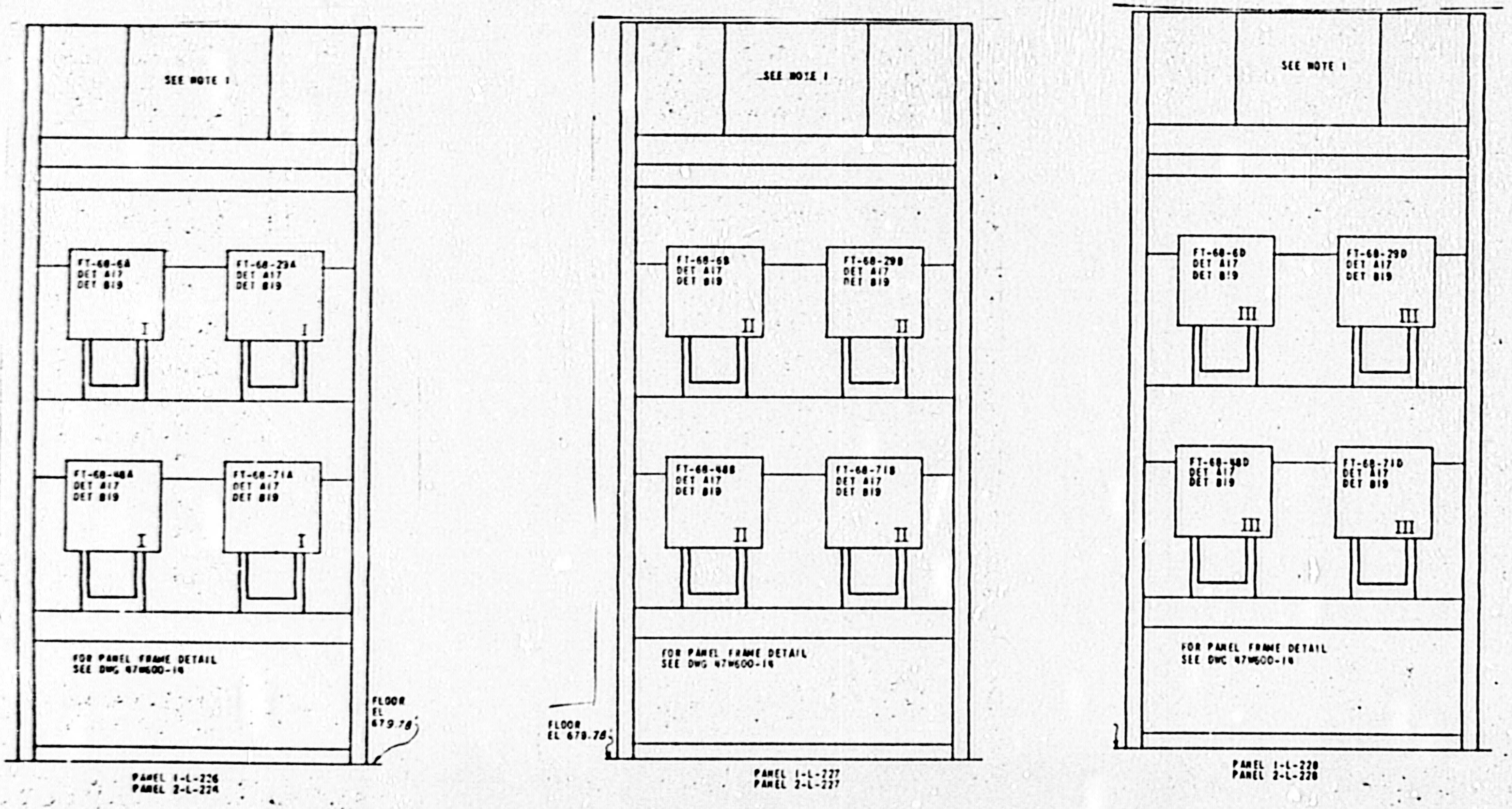
5. I-85-590-SQN-05, Nonconformance Report NCR 6172 Evaluation for Generic Applicability to Other TVA Facilities

Because of the demonstrated generic effects of WBN NCR 6172 (ref. 23) to SQN on improper instrument sense line slope, the WBN construction NCR should be sent through design to SQN and to the other TVA plants for a generic review for applicability in accordance with Office of Engineering Procedure OEP-17 (ref. 24). [P2]

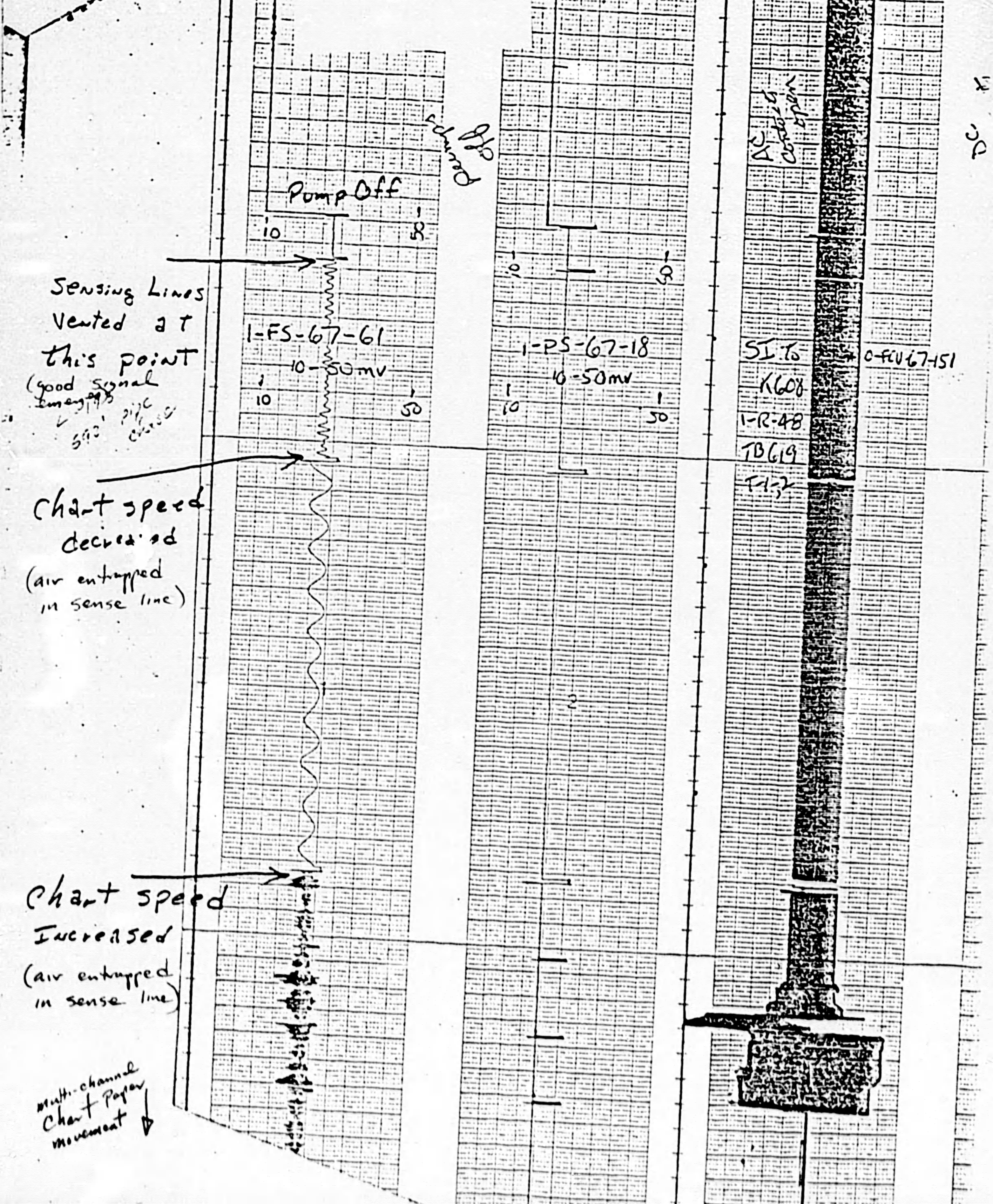
ATTACHMENT 1

Instrument No.	RCS Loop	Local Panel	TVA Drawing
1,2-FT-68-6A	1	L-226	47W600-80
1,2-FT-68-6B	1	L-227	47W600-80
1,2-FT-68-6D	1	L-228	47W600-80
1,2-FT-68-29A	2	L-226	47W600-80
1,2-FT-68-29B	2	L-227	47W600-80
1,2-FT-68-29D	2	L-228	47W600-80
1,2-FT-68-48A	3	L-226	47W600-80
1,2-FT-68-48B	3	L-227	47W600-80
1,2-FT-68-48D	3	L-228	47W600-80
1,2-FT-68-71A	4	L-226	47W600-80
1,2-FT-68-71B	4	L-227	47W600-80
1,2-FT-68-71D	4	[illegible]228	47W600-80

ATTACHMENT 2



ATTACHMENT 3



ATTACHMENT 4

Summary of WBN NCR 6172

- QTC's investigation of WBN Employee Concern PH-85-001-002 identified specific instances where RCS flow transmitter sense line slope was not in agreement with stated requirements (ref. 25).

- Nonconformance Report NCR 6172, Revision 0, was issued July 9, 1985, by the Office of Construction (OC) to document these deficiencies. Planned corrective actions include relocation of RCS flow instrumentation to reduce sense line length and installation of new sense lines to correct slope deficiencies.

- Subsequent inspections by TVA of a random sample of additional safety-related instrument sense lines revealed that a generic condition exists at WBN with regard to overall sense line slope requirements. This was documented by WBN NCR 6172, Revision 1, on September 12, 1985. It extended the planned corrective action to all safety-related instruments.

- Dispostion of NCR 6172, Revision 1, entails the following:

    1. OE will provide to OC a list of instrument sense lines essential to safe operation and shutdown which might not function reliably with air entrapped in their sense lines.

    2. Reinspection of these identified sense lines to ensure the required slope is met.

    3. Rework any identified sense lines which do not conform to slope requirements.

    4. The remaining essential instrument sense lines will be generally evaluated by OE for functional operability "as-is."

- In accordance with OEP-17 the Engineering Report for NCR 6172, Revision 1, was prepared on September 30, 1985, and sent to the Nuclear Engineering Branch Nuclear Licensing Staff who determined it was reportable as a 50.55(e) item. It was subsequently reported to NRC.

- The NCRs were not transmitted to SQN or evaluated for generic implications at SQN.

DOCUMENTS REVIEWED IN INVESTIGATION I-85-590-SQN AND REFERENCES

1. Sequoyah Nuclear Plant SQA 134, "Critical Structure Systems, and Components (CSSC) List," Revision 7, dated August 12, 1985

2. Sequoyah Technical Instruction TI-54, "Compliance Instruments unit 0,1," Revision 8, dated August 5, 1985

3. SQN Construction Inspection Instruction No. 85, "Installation Verification, Flushing, and Pressure Testing of Instrumentation Sensing Lines, Sampling Lines, Control Supply Headers, and Signal Lines," Revision 10, dated May 4, 1984

4. SQN Design Criteria No. SQN-DC-V-3.0, "General Design Criteria for The Classification of Piping, Pumps, Valves, and Vessels," Revision 1, dated December 12, 1975

5. SQN Construction Specification No. N2M-865, "Field Fabrication, Assembly Examination, and Tests For Pipe and Duct Systems," Revision 3, dated April 12, 1977

6. SQN Construction Specification No. N2E-883, "Routing and Separating Instrument Lines in The Vicinity of High Energy Process Piping Inside and Outside Containment," Revision 0, dated August 23, 1978

7. SQN Design Criteria No. SQN-DC-V-10.5, "General Design Criteria For Separation of Instrument Sensing Lines and Instrument Air Lines," Revision R1, dated August 24, 1984.

8. SQN Construction Inspection Instruction No. B1, "Inspection of Instrument Line Separation and Flexibility," Revision 4, dated January 15, 1984

9. Instrument Maintenance Instruction IMI-118, "Filling of Sealed Instrument Systems Backfilling, Venting, and/or Flushing of Instrument Sensing Lines," Revision 5, dated December 31, 1984

10. TVA Drawings
    47W610-68-1 R17
    -68-2 R15
    -68-3 R19
    -68-4 R22
    -68-5 R20
    -68-6 R11
    -68-7 R8

    47W600-24 R16
    -133 R8

DOCUMENTS REVIEWED (continued)

11. SQN Surveillance Instruction SI-246, "Recalibration Procedure for Reactor Coolant Flow Transmitters," Revision 7, dated March 5, 1984

12. SQN Surveillance Instruction SI-155, "Reactor Coolant Flow Verification," Revision 10, dated February 1, 1985

13. SQN Instrument Maintenance Instruction IMI-99, "Reactor Protection System," Sections CC 13.A through CC 6.24A as shown below.

Section	Revision Level	Effective Date	Instrument
CC 6.13A	2	10/2/84	FT-68-6A
CC 6.14A	2	10/2/84	FT-68-6B
CC 6.15A	2	10/2/84	FT-68-6D
CC 6.16A	3	10/2/84	FT-68-29A
CC 6.17A	2	10/5/84	FT-68-29B
CC 6.18A	2	10/2/84	FT-68-29D
CC 6.19A	2	10/2/84	FT-68-48A
CC 6.20A	2	10/2/84	FT-68-48B
CC 6.21A	3	10/2/84	FT-68-48D
CC 6.22A	3	10/2/84	FT-68-71A
CC 6.23A	2	10/2/84	FT-68-71B
CC 6.24A	2	10/2/84	FT-68-71D

14. Plant Maintenance requests for the following instruments on both units 1 and 2.

FT-68-6A	FT-68-48A
FT-68-6B	FT-68-48B
FT-68-6D	FT-68-48D
FT-68-29A	FT-68-71A
FT-68-29B	FT-68-71B
FT-68-29D	FT-68-71D

15. SQN Nonconformance Report NCR 2358 dated September 29, 1980

DOCUMENTS REVIEWED (continued)

16. SQN Construction Inspection Instruction 85 data cards for unit 1 and unit 2 RCS flow transmitters listed in Attachment 1, including referenced variance reports

17. US NRC Regulatory Guide 1.11 (March 1971), "Instrument Lines Penetrating Primary Containment"

18. Instrument Society of American (ISA) Standard S67.02-1980, "Nuclear Safety-Related Instrument Sensing Line, Piping and Tubing Standards For Use in Nuclear Plant"

19. International Standard ISO-2186, "Fluid Flow in Closed Conduits - Connections For Pressure Signal Transmissions Between Primary and Secondary Elements," issued in 1973

20. Watts Bar Employee Concern Assignment Request Form identified as PH-85-001-002

21. "Fluid Meters, Their Theory and Application," Report of ASME Research Committee on Fluid Meters, Sixth Edition, 1971.

22. Watts Bar Nonconformance Report NCR 6172, Revision 0, dated July 9, 1985

23. Watts Bar Nonconformance Report NCR 6172, Revision 1, dated September 12, 1985

24. Office of Engineering Procedure OEP-17, "Corrective Action," Revision 2, dated August 30, 1985

25. Quality Technology Company (QTC) Watts Bar Employee Concern Disposition Report - Concern No. PH-85-001-002 - prepared October 21, 1985

26. TVA Employee Concern Program (Nuclear Performance Plan) as submitted to NRC on November 20, 1985 (L44 851120 800)

27. SQN Surveillance Instruction SI-94.2, "Reactor Trip Instrumentation Refueling Outage Channel Calibration (RCS Temperature and Flow) Units 1 and 2," Revision 1, dated November 19, 1982

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

NRC

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : E. R. Ennis, Plant Manager, Watts Bar Nuclear Plant

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

DATE : DEC 27 1985

SUBJECT: NUCLEAR SAFETY REVIEW STAFF INVESTIGATION REPORT TRANSMITTAL

Transmitted herein is NSRS Report No. I-85-510-WBN

Subject EXCESSIVE PIPING VIBRATION

Concern No. IN-85-289-002

and associated recommendations for your action/disposition.

It is requested that you respond to this report and the attached recommendations by January 23, 1986. Should you have any questions, please contact J. H. Kincaid at telephone 3701-WBN.

Recommend Reportability Determination: Yes X No ____

Director, NSRS/Designee

JHK:GDM
Attachment
cc (Attachment):
R. P. Denise, LP6N35A-C
D. R. Nichols, E10A14 C-K
QTC/ERT, Watts Bar Nuclear Plant
E. K. Sliger, LP6N48A-C
W. F. Willis, E12B16 C-K (4)

--Copy and Return--

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

From: ________________

Date: ________________

I hereby acknowledge receipt of NSRS Report No. I-85-510-WBN
Subject EXCESSIVE PIPING VIBRATION for action/disposition.

________________ ________
Signature Date

0215U

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS INVESTIGATION REPORT NO. I-85-510-WBN

EMPLOYEE CONCERN IN-85-289-002

MILESTONE 5

SUBJECT: EXCESSIVE PIPING VIBRATION

DATES OF INVESTIGATION: October 2-November 12, 1985

LEAD INVESTIGATOR: JHKincaid 11/25/85
J. H. Kincaid Date

INVESTIGATOR: Charles E Elledge 11/25/85
C. R. Elledge Date

REVIEWED BY: for Mary Anne Holtewich 12/19/85
P. R. Washer Date

APPROVED BY: for Mary Anne Holtewich 12/19/85
M. R. Harrison Date

I. BACKGROUND

NSRS has investigated Employee Concern IN-85-269-002 which Quality Technology Company (QTC) identified during the Watts Bar Employee Concern Program. The concern is worded:

> Mini water #74 6" to 8" pipe vibrated dangerously during a test. This pipe would not hold up during operations. Pipe chase bldg 713' el 692' el 3/'85 unit #1.

II. SCOPE

The concern was determined to be that the piping had vibrated at a level which could have caused support damage and that the problem had not been identified and corrected. The piping loop observed by the CI was determined to be the 3-inch minimum flowline associated with the residual heat removal (RHR) pump B-B. The line appeared to be 6 inches to 8 inches due to insulation. The system was not in a test mode. The time of the event and line observed by the CI were verified by a callback to the CI through QTC.

III. SUMMARY OF FINDINGS

The pipe vibration reported by the CI was also reported in the daily journal of the shift engineer and the unit operator. The vibration occurred in the RHR B-B minimum flowline on March 1?, 1985 at 1945 hours and lasted off and on for approximately 15 minutes. The operator and shift engineer were made aware of the problem by observers positioned for loop transfer.

RHR pump A-A was in service with suction supply from the reactor vessel (RV) which was unpressurized and at a minimum water level. RHR pump B-B was brought online with suction from the same source. It is possible that two-pump operation with a minimum net positive suction head (NPSH) pulled air into the B loop. It is also possible that air had been trapped in the B loop prior to loop transfer. The vibration occurred when the RHR 1B-B loop was isolated in the minimum flow mode by closing valve FCV 74-28. The vibration would stop when valve FCV 74-28 was opened. Operations personnel made the decision to continue running in order to troubleshoot the problem.

The shift engineer observed the piping, pump, and operation of valve FCV 74-24. The piping vibration was characterized as a sustained water hammer at approximately 1 cycle per second (cps) with audible metal-to-metal impact in the pipe supports. The valve stem on valve FCV 74-24 stroked on command and did not appear to be tracking pressure pulsations. The 8-inch piping did not vibrate excessively, and the RHR B-B pump felt smooth (although measurements were not taken). The RHR B-B pump was shut down for a short time and restarted in the next shift to check for vibrations. The vibrations could not be induced and, based on interviews, has not occurred since. Routine pump-vibration diagnostics have not indicated any signs of damage, and valve FCV 74-24 was found to be in good mechanical condition. There have not been any procedural, instrumentation, or mechanical problems identified to date which could have caused the problem.

The event was not identified as an abnormal occurrence because there was no apparent or suspected damage to critical equipment. Consequently, the event was not reported per Administrative Instruction AI-7.3, "Adverse Conditions and Corrective Actions." The Reactor Engineering Section, Results Section, and Preoperational Test Section were not aware of the event.

An inspection of supports on the RHR 1B-B minimum flowline was conducted by the investigator. A standoff pipe clamp shown on drawing 63-1SIS-R199 (node M88 on isometric 47W432-202) was rotated approximately 15 degrees about the pipe. The support integrity was not degraded. The inspection did not reveal any other indications of support damage.

IV. CONCLUSIONS AND RECOMMENDATIONS

Conclusions

The concern that the piping vibrated at a level which could cause support damage has been substantiated. This concern was mitigated by the fact that operators were aware of the problem and took appropriate action to troubleshoot the problem. Although the event had been logged appropriately in the shift engineer's daily journal, other sections were not aware that a vibration problem had occurred.

An inspection of supports on the RHR 1B-B minimum flowline identified a standoff pipe clamp (shown on support drawing 63-1SIS-R199 and located at node M88 on piping isometric 47W432-202) rotated approximately 15 degrees about the pipe. Drawing 63-1SIS-R199 is attached with a note showing clamp rotation. The support integrity had not been degraded. The remaining supports were inspected but did not show any indications of support damage.

Recommendations

I-85-510-WBN-01 - Abnormal Event Reporting

The water-hammer event reported by the CI and recorded in the daily journal of the shift engineer on March 19, 1985 should be documented and handled per Standard Practice WB-11.3, "Reporting Adverse Conditions to Plant Superintendent."

I-85-510-WBN-02 - Support Damage

The standoff pipe clamp located at node 88 on isometric 47W432-202 should be rotated to be as-designed as shown on support drawing 63-1SIS-R199.

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 27 1985

SUBJECT: NUCLEAR SAFETY REVIEW STAFF INVESTIGATION REPORT TRANSMITTAL

Transmitted herein is NSRS Report No. I-85-652-SQN

Subject CONTAMINATION OF LAUNDERED ANTI-CONTAMINATION CLOTHING

Concern No. XX-85-101-004

and associated prioritized recommendations for your action/disposition.

It is requested that you respond to this report and the attached Priority 2 [P2] recommendation by January 17, 1986. Should you have any questions, please contact R. C. Sauer at telephone 2277.

Recommend Reportability Determination: Yes ___ No X

R. C. Sauer

Director, NSRS/Designee

RCS:GDM
Attachments
cc (Attachments):
- 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, E1 B16 C-K (4)

0226U

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS INVESTIGATION REPORT NO. I-85-652-SQN

EMPLOYEE CONCERN: XX-85-101-004

SUBJECT: CONTAMINATION OF LAUNDERED ANTI-CONTAMINATION CLOTHING

DATES OF INVESTIGATION: October 17-19, 1985

INVESTIGATOR: D. J. Hornstra DATE: 12/19/85
D. J. Hornstra

REVIEWED BY: MW Alexander DATE: 12/19/85
M. W. Alexander

APPROVED BY: R.C. Sauer DATE: 12/27/85
R. C. Sauer

I. BACKGROUND

A Nuclear Safety Review Staff (NSRS) investigation was conducted to determine the validity of an expressed employee concern as received by the 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-85-101-004, stated:

> Sequoyah - CI (concerned individual) expressed that insufficient attention to detail in regards to minimizing radiation exposure. Due to the policy of reusing outer gloves in radiation areas, CI has observed used gloves, available for reuse, which were contaminated to a level 5 times that of the area in which the employee was working. The CI has no further information.

II. SCOPE

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

1. Gloves available for reuse are sometimes more contaminated than the area in which the employee will be working.

2. Reuse of such contaminated gloves is an example of inadequate radiation exposure control.

B. In order to make a determination on the validity of these issues, the following areas related to the reuse of protective clothing were evaluated:

1. The acceptable level of fixed contamination in cleaned protective clothing.

2. Typical levels of contamination on gloves and overshoes.

3. The impact of the acceptable level on individual exposures.

III. SUMMARY OF FINDINGS

A. Requirements and Commitments

1. SQN Final Safety Analysis Report (FSAR) Section 12.3 (ref. 1) commits SQN to establish procedures and conduct activities which fulfill the policy of the Radiation Protection Program Manual (RPP).

2. Radiation Protection Plan (RPP - ref. 2) paragraph 3.3.1 requires that, "Prior to reuse of anticontamination equipment, excluding respirators, it should be decontaminated to a maximum of 300 dpm/100 cm^2 of alpha and 0.75 mr/h of beta-gamma and no transferrable contamination."

3. Radiation Protection Manual, Program Area 3, Procedure 0301.02 (ref. 3), paragraph 3.6.8, repeats the above requirement from the RPP.

B. Findings

1. The SQN radiological control procedures were found to satisfy the requirements of the RPP and procedure 0301.02. Radiological Control Instruction, RCI-1 (ref. 4), specifies the maximum acceptable contamination limits for laundered protective clothing that can be reused. Laundered protective clothing is limited to a direct survey level of 0.75 mrad/hour (beta-gamma) averaged over any 644 square centimeters (100 square inches). Using a thin window pancake probe, RCI-1 states that an exposure level of 0.05 mrad/hour would be equivalent to 300 counts per minute (cpm). Therefore, the above 0.75 mrad/hour would correspond to 4500 cpm. Since the 0.75 mrad/hour limit is an average over any 644 square centimeter area and the RM-14 frisker probe is only 15 square centimeters, spot readings with a frisker may exceed 4500 cpm and still have the protective clothing meet the RCI-1 average limit. In addition to the direct survey limit, RCI-1 requires that no transferable (smear) contamination be detectable on laundered protective clothing that is to be reused.

   Corresponding limits for personal clothing are 0.05 mrad/hour (300 cpm) with no detectable transferable contamination.

2. As a part of this investigation, a random sample of rubber gloves and overshoes that had been cleaned was surveyed by health physics (Individual C) in the SQN laundry. Approximately 20 gloves and 20 overshoes were surveyed with a frisker. The radiation level of one glove was 3400 cpm; one overshoe was 1800 cpm; and two gloves and one overshoe were between 500 cpm and 600 cpm. All other items were less than 500 cpm. Background was 150-200 cpm. These five items were then taken to a health physics station where it was determined that no transferable contamination was detectable on any of the items.

3. No information was found during interviews (Individuals D, E, and F) to indicate that laundered protective clothing had been declared clean and ready for reuse which did not meet the direct survey requirements of RCI-1.

4. Protective clothing bins are conspicuously labeled "Caution - Radioactive Material Area."

5. The SQN laundry performs a direct beta-gamma survey on all laundered protective clothing, but a determination of transferable contamination is not conducted on a routine basis. Health physics personnel (Individuals E and F) stated that the past performance of the cleaning process had demonstrated that an inspection for transferable contamination is not necessary on a routine basis.

6. Health physics personnel (Individuals E and F) stated that a direct alpha survey of laundered protective clothing is not necessary at this time due to the absence of alpha emitters in the reactor coolant system.

7. The lesson plans for General Employee Training (GET) 2.3, Level II health physics training, address anticontamination clothing but do not adequately explain allowable fixed contamination on laundered protective clothing which is permissible after cleaning.

   The current SQN GET instructors for GET 2.3 (Individuals A and B) stated that they have discussed in their training the potential for fixed contamination on laundered protective clothing.

8. Limits for fixed contamination (with an associated direct dose) and for transferable contamination (with a potential for personnel contamination) are not numerically comparable. The activity levels of transferable contamination found in work areas in the plant are extremely low when compared to direct radiation which may come from either fixed contamination or process system inherent sources. Therefore, it is possible that the indicated counts per minute from fixed contamination on clothing may exceed the observed level of transferable contamination in a work area. This condition is not an unexpected or unusual event and is within acceptable limits of personnel exposure.

IV. CONCLUSIONS AND RECOMMENDATIONS

A. The concern of record was substantiated in that cleaned gloves were found with fixed contamination significantly above the level of transferable contamination of many work areas. However, this level of fixed contamination was found to be within prescribed levels for cleaned protective clothing as promulgated by the RPP. Reuse of such gloves with fixed contamination up to 0.75 millirad per hour was not found to represent "insufficient attention to detail" as alleged; this allowable level of fixed contamination was the result of a policy decision by the TVA Radiological Health Staff.

B. IN-85-652-SQN-01, Training on Acceptable Levels of Contamination on Laundered Protective Clothing

   The concerned employee may not understand plant procedure regarding contamination on protective clothing cleaned for reuse.

Recommendation

The lesson plans for health physics GET should be revised to better explain acceptable fixed contamination levels on protective clothing and the associated radiation exposure to the individual as part of the initial training and periodic retraining. [P2]

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DOCUMENTS REVIEWED DURING INVESTIGATION I-85-652-SQN AND REFERENCES

1. SQN FSAR, Section 12.3, "Health Physics Program"

2. Office of Power Radiation Protection Plan, R1, dated November 2, 1983

3. Nuclear Power Radiation Protection Manual, Program Area 3, Procedure 0301.02, dated December 7, 1984

4. SQN Radiological Control Instruction RCI-1, R27, "Radiological Hygiene Program," dated September 12, 1985

5. GET Lesson Plans 2.3, Level II, "Health Physics Training," June 1985

6. SQN Health Physics Section Instruction Letter, HPSIL-2, "Contamination Surveys," Rev. 12, dated July 15, 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 27 1985

SUBJECT: NUCLEAR SAFETY REVIEW STAFF INVESTIGATION REPORT TRANSMITTAL

Transmitted herein is NSRS Report No. I-85-513-SQN

Subject RADIATION EXPOSURE OF OLDER PERSONNEL

Concern No. XX-85-009-001 and XX-85-009-002

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

R.C. Sauer
Director, NSRS/Designee

RCS:JTH
Attachments
cc (Attachments):
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--For response to employee
H. S. Sanger, Jr., E11B33 C-K
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 REPORT NO. I-85-513-SQN

EMPLOYEE CONCERNS: XX-85-009-001 (partial)
XX-85-009-002

SUBJECT: RADIATION EXPOSURE OF OLDER PERSONNEL

DATES OF INVESTIGATION: SEPTEMBER 18 - OCTOBER 4, 1985
NOVEMBER 8-12, 1985

LEAD INVESTIGATOR: D. J. Hornstra — D. J. HORNSTRA — 12-18-85 DATE

INVESTIGATOR: C. L. Wilson — C. L. WILSON — 12-18-85 DATE

REVIEWED BY: MW Alexander — M. W. ALEXANDER — 12/18/85 DATE

APPROVED BY: R.C. Sauer — R. C. SAUER — 12/20/85 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 initial concern of record, as summarized on the Employee Concern Assignment Request Form from QTC, and identified as XX-85-009-001, stated:

> No regard for safety at operating plants (Sequoyah). Management (names known) told Craft Supervisor (name known) to assign oldest employees to 'hot work' until they exceeded dose levels and then get rid of them. The supervisor refused to follow the instruction and got blamed for it. Also, management (names known) fired a group of employees (number and one person's name known) in 1978 for refusing to work 2 hours overtime in early 1978.

On November 29, 1985, during the investigation of this concern the investigators were provided with a new Employee Concern Assignment Request Form from QTC identified as XX-85-009-002. The new concern stated:

> Sequoyah: There is nonregard for personal safety at operating plants. Management (known) directed that the oldest employees be assigned to 'hot' work in order for them to reach their radiation exposure levels first. A supervisor (known) made the statement that 'older folks won't be long around.' Details known to QTC, withheld due to confidentiality. Construction Department concern. CI has no further information.

In a discussion between the investigators and the ERT on December 2, 1985, ERT stated that in a follow-up interview the CI had informed them that XX-85-009-001 had mischaracterized his concern in that he had not stated that a craft supervisor (craft foreman) had been instructed to assign his oldest employees to "hot work" and that administrative action had not been taken against a craft foreman for refusal to follow such an instruction. Based upon this follow-up information, ERT replaced concern XX-85-009-001 with XX-85-009-002 (as stated above) and XX-85-009-003 (relative to the alleged firing of employees). ERT identified the man- agement and supervisor (also management) referred to in XX-85-009-002. This investigation is limited to employee concern XX-85-009-002 only.

II. SCOPE

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

1 Management directed that the oldest employees be preferentially assigned "hot work."

2. The above direction was implemented, causing the oldest personnel to receive higher accumulated doses.

B. Based upon the identification of the management individuals in the concern of record, the timeframe of the concern was narrowed down to October 1979 to March 1981.

III. SUMMARY OF FINDINGS

A. Requirements and Commitments

1. 10CFR20.101 (ref. 1) limits the whole body exposure of radiation workers to 1.25 rem/quarter. However, this may be increased to 3 rem in a quarter provided the lifetime exposure does not exceed 5(N-18), where N is the age at the last birthday, and the individual's accumulated occupational dose has been determined on form NRC-4. Thus an older worker may have a larger remaining allowable dose (lifetime) than a younger worker. Using the lifetime dose constraint would allow a worker to receive up to 3 rem in consecutive quarters and thus receive 12 rem in a calendar year.

2. SQN Technical Specifications (ref. 2) require that "Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10CFR20 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) identified Radiation Control Instructions (RCIs) which are consistent with 10CFR20. Table 13.5.7-1 identifies 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 Plan (RPP) established by the Radiological Health Staff.

5. RPP Section A3.4 (ref. 5) allows TVA employees 19 years of age or over and who have submitted a radiation history to TVA to receive 3.0 rem each calendar quarter up to an accumulated dose of 5(N-18) rem (where N is the age) but not more than 4 rem each calendar year. The annual limit of 4 rem is a TVA policy and supersedes the use of the 10CFR20 permitted 3 rem per quarter up to 5(N-18) lifetime dose. Thus no radiation limit advantage exists to assign older workers, who may have a higher remaining allowable dose (lifetime), to "hot work."

6. The Radiation Protection Manual (RPM) Area Plan 3, Procedure 0301.03 (ref. 6), Attachment 1, limits the whole body dose to 3.0 rem per quarter, 4.0 rem per year, and lifetime limit of 5(N-18) for TVA employees over 19 years old.

7. The RPM (ref. 6) Attachment 3 provides TVA policy tracking action levels for radiation exposures. When 70 percent of the quarterly or annual limit is reached, the responsible section supervisor is to use the individual in regulated areas only where no other qualified individual is available. When 85 percent of the quarterly or annual limit is reached, the responsible section supervisor is to use the individual in the regulated areas only on rare occasions and only if the Plant Health Physics Section is fully satisfied that the employee will not exceed any applicable exposure limit.

8. SQN Radiological Control Instruction, RCI-1 (ref. 7), effective during the time period in question, provided maximum dose limits for the whole body of 3 rem per quarter and 4 rem per year. No additional administrative restrictions existed in RCI-1 to limit the exposure of individuals as they approached this maximum dose limit. However, RCI-1 stated that "Work assignments shall be made to equalize exposure of plant personnel as much as practical without causing substantial increases in total overall exposure for all employees."

9. RCI-3 (ref. 8) currently establishes SQN administrative action levels as follows:

   a. Action level 1 - 70 percent of quarterly or annual exposure limit. The responsible supervisor shall not use the individual in radiation or high radiation areas unless no other qualified personnel with lower exposures are available.

   b. Action level 2 - 80 percent of quarterly or annual exposure limit. The individual shall be restricted from regulated areas. Removal of this restriction shall require written justification from the individual's section supervisor and approval of the Health Physics Section Supervisor.

   c. Action level 3 - 90 percent of quarterly or annual exposure limit. The individual shall be restricted from regulated areas.

   These limits are more restrictive than the limits in the RPM Procedure 0301.03 (ref. 6).

B. Findings

1. A review of radiation exposure records of 179 craft workers and foremen assigned to SQN during the period from October 1979 to March 1981 revealed that none of them had received a dose which would have prevented or restricted their work in regulated areas. A review of doses for subsequent periods for these same individuals indicated that one individual had received a quarterly exposure above the currently imposed 70 percent administrative limit, thus influencing the work assignments made by the supervisor, but not limiting the employment of the individual.

2. SQN exposure records were reviewed for the period of January 1980 to June 1985 to determine if any personnel had exceeded 70 percent of either quarterly limits or annual limits. 36 individuals exceeded a quarterly dose of 2.1 rem or an annual dose of 2.8 rem, of which 20 were TVA employees. Of the 20 TVA employees, 10 were engineers/technicians and 10 were craft personnel. Of the 10 craft personnel, 6 were currently employed at SQN. A comparison of the employment records and exposure records of the other 4 individuals who had exceeded the 70 percent administrative limit revealed the following:

   a. An employee (craft personnel number 1) exceeded 70 percent of his quarterly exposure limit in the period January through March 1984. He was terminated at the end of his temporary appointment on April 13, 1984--into the next quarter for exposure limits. There was no indication that the employee's termination was effected by his exposure.

   b. An employee (craft personnel number 2) exceeded 90 percent of his annual limit in 1983. However, his temporary appointment at SQN was terminated in February 1983, with a first quarter dose at SQN less than 70 percent of the quarterly limit. There was no indication that the employee's termination was effected by his exposure at SQN. Based upon the exposure records reviewed, it is believed that the employee was subsequently employed at Browns Ferry Nuclear Plant (BFN) where he received additional radiation exposure.

   c. An employee (craft personnel number 3) exceeded 90 percent of his annual limit in 1984 and resigned at SQN to accept other employment. The employee had been previously employed in 1984 at BFN and subsequently returned to BFN during 1984. He remained a TVA employee into the second calendar quarter of 1985. Almost all of his 1984 dose was received at BFN. There was no indication that this employee's resignation from SQN was effected by his radiation exposure.

d. An employee (craft personnel number 4) exceeded 90 percent of his annual limit in 1983 and resigned at SQN to accept other employment. The employee left SQN during the first quarter of 1983 and had received less than 70 percent of the quarterly dose at that time. Although the employee subsequently received radiation exposure in 1983, there was no indication that the employee's resignation was effected by his exposure.

3. Based upon the exposure record of 179 craft personnel for the period October 1979 to March 1981, no pattern of selection of personnel for hot work based upon age was found in any of the craft sections.

4. Based upon an interview with Individual A, plant management had discussed in the 1979-1980 time period options that could be taken if employees approached the quarterly or annual dose limits established by RCI-1. No information was received from Individual A or one of the craft supervisors of that timeframe (Individual B) that any direction was provided to preferentially expose older workers.

5. The supervisor who was alleged to have made the statement that "older folks won't be long around" is no longer a TVA employee, could not be located from his last known address, and thus could not be interviewed.

6. A craft foreman from the 1980 time period (Individual B) was unaware of any "management direction" regarding the assignment of personnel to "hot work" based upon age.

IV. CONCLUSIONS/RECOMMENDATIONS

The concern of record was not substantiated. NSRS could find no objective evidence that SQN management told supervisors in the 1980 timeframe to assign older personnel to work in high radiation areas ("hot work"). There is no evidence that older personnel were preferentially assigned "hot work." During the period in question, no individual received a dose high enough to require any consideration of work restrictions, even using the more conservative TVA policy exposure limits.

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DOCUMENTS REVIEWED IN INVESTIGATION OF I-85-513-SQN
AND REFERENCES

1. 10CFR20.101, "Radiation Dose Standards for Individuals in Restricted Areas"

2. SQN Technical Specifications, Section 6.11, "Radiation Protection Program"

3. SQN Final Safety Analysis Report, Section 13.5.7, "Radiation Control Instruction"

4. SQN Final Safety Analysis Report, Section 12.3 "Health Physics Program"

5. Office of Power Radiation Protection Plan (RPP), dated August 18, 1983, Section A3.4 "External Exposure Control"

6. Nuclear Power Radiation Protection Manual, Area Plan 3, Procedure 0301.03, dated December 29, 1983, "External Radiation Exposure Limits and Controls"

7. SQN Radiological Control Instruction, RCI-1, Revision 8, "Radiological Hygiene Program," dated October 1979

8. SQN Radiological Control Instruction, RCI-3, Revision 21, "Personnel Monitoring" dated September 30, 1985

9. Personnel Exposure Monitoring Report for 1980, 1981, 1982, and 1984

10. Radiation Exposure Record (Muscle Shoals computer printout) for personnel assigned to outage group in period October 1979 through March 1981

11. Muscle Shoals listing of all personnel who had received greater than 70 percent of allowable radiation exposures

12. Various personnel employment records