> affected by the procedure no less frequently than every two years to determine if changes are necessary or desirable. A revision of a procedure constitutes a procedure review.

N-OQAM, part III, section 1.1, paragraph 5.0, states:

> Procedures and instructions shall be reviewed by an individual knowledgeable in the area affected by the procedure/instruction no less frequently than every two years to determine if changes are necessary or desirable. A revision of a procedure/instruction constitutes a review. (Note: Some procedures/instructions may require more frequent reviews depending on the type and complexity of the activity covered.)

The N-OQAM restated the ANSI requirement that a review of procedures and instructions be made at least every two years. AI-3.1 states:

> Responsible section supervisors shall ensure that CSSC-related plant instructions (other than rarely used maintenance and modification instructions) are reviewed by an individual knowledgeable in the area affected by the instruction no less frequently than every two years to determine if changes are necessary or desirable. The reviewer shall have adequate understanding of the requirements and intent of the original document. Successful documented performance of the instruction or a "General" or "All" revision made to the instruction satisfies this requirement.

Substitution of "successful documented procedural performance" for the required review was not identified by upper-tier documents as an acceptable alternative.

Plant management stated, as an example of use of the successful document performance, that when the section supervisor reviewed the completed data package of a SI that any errors in the SI would be discovered. According to Planning and Scheduling personnel, only 30 to 40 SIs are performed monthly at WBN. SQN routinely performs approximately 1000 SIs monthly. In February 1984, SQN reported 992 performances of 220 periodic SIs

and 433 performances of 45 conditional SIs. A conditional SI is not performed on any set time basis but is performed when plant parameters require it. When WBN becomes a licensed plant, the large numbers of SIs performed daily will make it difficult to adequately review the SI at each performance. The proof of adequacy and currency of a procedure or instruction is more appropriately demonstrated by the actual performance and not from an after-the-fact review.

The SI was only one example of instructions and procedures used at the plant. Administrative Instructions, System Operating Instruction, General Operating Instructions, Standard Practices, and so forth, were all included in the guidance given by RG 1.33. At SQN, AI-4 R41, Section 4.3.2, "Periodic Review," a biennial review was required for each instruction, but there was no provision to allow a successful documented use of the instruction to replace an actual review. At BFN, Standard Practice BF 2.3, "Review, Approval, and Use of Instructions," contained no requirement for a biennial review of instructions, but it referred to N-OQAM, part III, section 1.1, which did require a biennial review. BF 2.3 does state that "Each use of an instruction serves as a review" but does not directly stated that this replaces a biennial review. The NSRS does not believe that a documented performance of a procedure or instruction meets either the intent or the letter of the guidance given by any upper-tier document. NSRS recognized that site procedures may impose more stringent controls than upper-tier documents but should not be more lenient.

b. Field Quality Engineering Review

N-OQAM, part III, section 1.1, paragraph 4.4.4 contained the following statements:

> 4.4.4.1 New and revised plant standard practices and Administrative Instructions (AIs) which implement the quality assurance program shall be reviewed and concurred with by the Field QE Section prior to use. The purpose of this review shall be to implement the division quality assurance program as documented in the OQAM, DPM, and interdivisional quality assurance procedures. This review shall be documented and maintained to show concurrence.

4.4.4.2 Each plant superintendent shall prepare a list of instructions requiring the above mentioned review. This list shall be submitted for the concurrence of the Field QE Section.

4.4.4.3 The Field QE Section supervisor shall assure that site quality assurance instructions that detail his assigned OQAM functions are prepared and implemented.

AI-3.1, paragraph 4.5.3.16 contained the following statements:

> Any revision to existing instructions listed in Appendix H or new instructions which implement the division's quality assurance program (as documented in the OQAM, DPM, and interdivisional quality assurance procedures) shall be reviewed by the Quality Engineering Section (QE) prior to formal PORC review.
>
> For those instructions which receive informal PORC review, the QE review shall take place after all informal PORC review comments have been resolved, but prior to formal PORC review.
>
> The QE review shall ensure that the plant instructions correctly implement the divisional quality assurance programs as documented in the OQAM, DPM, and interdivisional quality assurance procedures.
>
> QE shall document this review by signing and dating the Appendix B coversheet in the QE review space.
>
> NOTE: This Quality Engineering review shall be performed in addition to the normal review performed by QE of all CSSC instructions.

The cause of the NSRS concern in the area of FQE review was the fact that AI-3.1, "Plant Instructions - Control and Use," was written by an engineer in FQE, it was then revised by the same engineer, and the QE review was made by the same engineer documented by his signature in the FQE signoff space. Thus, the documented FQE review was made by its author. Although there was no requirement in the N-OQAM or AI-3.1 for an independent review, it does appear that the basic concept

behind an FQE review is to get a different person to review the document. In this case, the review was for a very specific reason: to ensure the plant instructions implement the divisional quality assurance programs. The FQE reviewer might not be able to identify errors in his own procedure simply because he was required to review it again. AI-3.1, Appendix H, "Plant Instructions Reviewed by the Field QE Section," included AI-3.1 as one of the instructions that was required to be reviewed by FQE.

NSRS understood that in practice the FQE supervisor also reviewed the instruction, but the documented review was not performed by the supervisor. This concern only applied to instructions and revisions originated by the FQE Section.

2. Observation of Work Activities

For the second part of the procedures and instructions review, the NSRS reviewer observed work activities in the field. This review was conducted to determine if the personnel performing the activities had the correct instructions with them to direct their work, if the instructions were adequate to perform the work, if the instructions were followed, and if the proper level personnel were assigned to the task.

The performances of four SIs were witnessed on two radiation monitors. Those were SI 3.3.3-I and SI-3.3.3-II, "Radiation Monitoring Instrumentation Fuel Pool Radiation Monitors Channel Calibration," and SI-3.3.4-I and SI-3.3.4-II, "Radiation Monitoring Instrumentation Fuel Pool Radiation Monitors Channel Functional Test." Three instrument mechanics performed the SIs. One was a senior instrument mechanic, one was a journeyman instrument mechanic, and one was an apprentice. The journeyman instrument mechanic and the senior instrument mechanic switched roles between the tests on the two radiation monitors to avoid the possiblity of making a common-mode type error in the tests. The SIs were available at the location where they were being used in the red folders indicating they were official SIs. The instructions were followed step-by-step. The instrument mechanic checked out the required instruments, verified their calibration dates, used two separate voltmeters for the two radiation monitors, verified that they had not been used on the previous performances of the SIs, and switched instruments between the two radiation monitor SIs, again to avoid common-mode type errors. The SI itself appeared to meet the format requirements for procedures and was technically correct.

In conclusion, the personnel who performed these four SIs knew their job and performed it. They were of the grade level required to perform their functions, and the test instruments they used were within their calibration period. Also, the SIs they were using were technically correct, formatted correctly, and met all administrative requirements.

While monitoring the SIs being performed on the radiation monitors, it was noted by NSRS that an Instrument Maintenance Instruction (IMI) was to be conducted on a water level switch in the spent fuel pit. This switch was installed in response to a past event in which water was found in what should have been a dry spent fuel pit. An Operations Section employee noted that the annunciator in the control room, indicating water in the pit, was on. His check of the pit revealed that there was no water in the pit but that the level switch (a float-type switch) was lying on top of the fuel racks. Apparently during previous construction activities in the pit the switch had been placed there. Operations personnel should have responded to the annunciation immediately by removing the spent fuel pit covers and checking for water rather than waiting for the spent fuel pit covers to be removed for the performance of an SI. Failure to respond to this alarm was a failure to follow an instruction.

NSRS also reviewed the use of a temporary change to an SI. SI-4.0.5.3.C3, "Check Valve Testing During Refueling Outage - Feedwater System," was reviewed in detail. A change was being made to the SI to allow it to be used as a vehicle for generating a pump curve for the motor-driven auxiliary feedwater pumps. A temporary change, form Appendix G from AI-3.1, was included in the SI package and was approved by a plant superintendent. Everything was handled in accordance with AI-3.1 for a temporary intent change to an SI on a piece of CSSC equipment. The SI itself met all the requirements of AI-3.1.

3. System Walk-Through

A walk-through was conducted of portions of the residual heat removal system (RHRS). The inplant equipment and piping was compared to the "as-constructed" physical drawings and flow diagrams. Also, several hangers were randomly selected and compared to the drawings. No major discrepancies were found during the comparisons. Minor problems were brought to the attention of the plant staff.

H. Unit Interface Controls

TVA interface control requirements were defined in ID-QAP-2.3, "Physical Interfaces Between Licensed and Unlicensed Units."

NUC PR implemented the requirements for which it has responsibilities, in Area Plan 1103.01, "Physical and Functional Interfaces." WBN implemented the Area Plan requirements in AI-1.6, "Interface Establishment and Control." Also, the Preoperational Test Section had issued section instruction letter IL-9, "Preoperational Test Program - Unit Interface Program," and the Operations Section had issued section instruction letter OSLA-36 for interface controls.

Each level of definition or implementation generally incorporated the requirements of the higher-tier document. However, the Division of Nuclear Power Test Staff Program Manual, N82A10, TPMP-c.7, March 17, 1983, was made an attachment to IL-9 and was the main body of the section instruction letter. In the Area Plan, this document, N82A10, TPMP-c.7, was no longer in effect, but the division procedures listing indicated N82A10, Division of Nuclear Power Preoperational Test Program Manual, had been superseded by 1102.02, 1104.01, and 1105.01, but Area Plan Procedure No. 1103.01 best described the interface plan.

In reviewing IL-9 with attachment, there did not appear to be any contradiction between it and 1103.01. If it was the intent of the Preoperational Test Section to use the superseded document as its program basis because of the lack of contradiction and the greater detail that was included in the superseded document, there is no problem. Otherwise, IL-9 should be revised to include the latest document as an attachment.

In the review of the implementation of the interface control points, emphasis was placed on the electrical and mechanical interface points that had been established, that were to be established later, or that should have been established, rather than on physical security boundaries. In 1980 an interface study was conducted by the Preoperational Test Section at WBN. In September 1980, a report was issued for the study. In that report electrical and mechanical interface points were identified with justification for their use. All formal interface points were to be established within a preoperational or non-critical system test. That report (with a few changes since) identified 159 interface points in 13 plant systems that were to be controlled by the interface program. Also, in the report were recommendations that certain valves not included in the formal interface control be administratively controlled and that the output cabinets of the solid state protection system (SSPS) for unit 2 be transferred to NUC PR before unit 1 operation because several wire lifts and jumpers were to be installed in these cabinets to prevent unit 2 testing from interfering with unit 1 operation.

Previous to the NSRS review, the plant FQE had conducted a survey of the unit interface. The survey was conducted by comparing the plant procedures, AIs and SIs, to the Area Plan Procedure 1103.01 and by sampling the established interface control points. The

findings of this review were issued in NSI-84-50 which was signed on February 20, 1984. Several deficiencies were found by FQE during its review resulting in WBN CAR 84-08, DR-84-23-R, and DR-84-24-R being written. WBN CAR 84-08 stated that (1) interface points were not accurately addressed on "as-constructed" drawings, (2) marked-up drawings were not being submitted to CONST, (3) test director name, for the test installing the interface point, was not listed on the interface control point log sheet as required by AI-1.6, and (4) AI-1.6 did not adequately implement the area plan in that there was no requirement for unique numbering of interface holdpoints.

The DRs involved an OSLA-36 requirement that drains that were sealed as interface points be painted yellow but had not been, and the identification of some interface temporary alterations which were numbered incorrectly--the year of installation was used instead of the system number. The two DRs had been closed but the CAR was still open with the Preoperational Test Section working on resolving it.

NSRS reviewed the survey, the method used in the survey, its findings, and the correction actions. DR-84-23-R was closed when OSLA-36 was rewritten deleting the requirement that drains capped for interface control were to be painted yellow. Not painting the drain caps had been the philosophy for one and one-half years, but the OSLA was not changed until the FQE survey. DR-84-24-R was closed when the Operations Section said they would reinstruct their Shift Engineer on the numbering system for interface points. The interface points that were already installed were not changed to show the system number rather than the year of installation and this was agreeable to FQE because the control was still intact. NSRS found the survey and its methods to be adequate, the findings valid, and the corrective action acceptable.

NSRS took a different approach to the interface control. NSRS reviewed the interface study report and the preoperational tests that would install the control points plus sampling some areas already surveyed by FQE. It was found that 57 control points had been installed in 5 systems with the use of 5 preoperational tests. From the interface study report there were 36 points still to be installed in the 5 systems and 66 more to be installed in 8 other systems using 8 preoperational or NCS tests. Table II shows the results of this review.

Interface points are installed using two control methods, the Temporary Alteration and Control Form (TACF) and the Hold Order (HO). Standard forms and tags are used for the TACFs and HOs, but a stamp is used on each one that installs an interface point. This stamp indicates that the TACF or HO is being used for an interface control point and that a workplan is required to remove the TACF or HO.

From a review of the test procedures, several discrepancies were found between the study and the procedures. See Table II for more details, test name, and systems. NCS-21 had been completed in September of 1980 with no interface points installed while the study called for five. W-10.9 had no interface points included, but the study had listed seven. The test had not been completed. TVA-28 had no interface points included, but the study indicated there should be six. The test was partially complete. TVA-1 had no interface points in the procedure, but two were listed in the study. W-2.2 included the required electrical interfaces, wire lifts, and jumpers; but the valves listed in the study were not included as interface control points. All pumps for this system had been transferred to NUC PR so the interfaced points were possibly no longer needed. TVA-1 did not include the nine valves listed in the study as interface control points. The test instruction had closed them using System Operating Instruction SOI-65.1, "Emergency Gas Treatment System," but did not have them controlled as interface points. During the week of the NSRS review, the test directors were told by their supervisor to write change sheets to include the omitted interface control points.

Also, in W-2.2, interface control points requiring wire lifts and jumpers were to be installed according to the procedure without the required two-part signoff. These operations were to be performed as temporary conditions which required the two-party verification. Interfaces were to be installed in steps 5.11.1 through 5.11.6. Step 5.11.7 stated, "Implement the administrative procedures necessary to identify, document, and maintain these interface points." The NSRS does not feel that this statement replaces the double signoff required for temporary conditions and which were found in an appendix to the procedure in most preoperational test procedures reviewed which installed interface control points.

AI-1.6 required that the test director submit a set of marked drawings to the shift engineer when interface points were established. In reviewing documents in the shift engineer's office, the NSRS found that 15 HOs and 11 TACFs had been issued to control interface hold points. Only one marked-up drawing from a preoperational test director was available in the shift engineer's office.

From the review of the interface study, it was noted that several points had been deleted and several more had been added. The points were included in the test procedure, and the changes were coordinated between the test director and the interface coordinator and included in the interface log. Also, it could be seen that many of the points overlooked in the test procedures were valves that should have interface HOs put on them. Instead of using interface HOs, the test directors were relying on the simple statement that the valve be closed as sufficient to control the interface point.

Hold Order 20004 was reviewed. It was installed on the essential raw cooling water (ERCW) system. Several valves were included on the HO. It had been partially removed twice (one valve opened each time), and each time a workplan was used. It was also noted that all changes to interface points were made with a safety-related change sheet. These parts of the interface control appeared to be working properly. The interface program coordinator in the Preoperational Test Section stated that he had committed to plant management that a walk through of the physical interface control points would be conducted two weeks prior to fuel loading. In a conversation with plant management, it was estimated that there would be a period of approximately two years between unit 1 and unit 2 fuel load. From that it could be seen that the interface points installed now would remain in place for up to two years. Thus, it is important that the program control points be installed and that they remain in place.

In conversations with the Safety Section Supervisor, it was determined that a review had been conducted by that section of the fire protection system. Their study found locations where the fence between units obstructed access to fire hoses. Because of this obstruction, extra fire hoses have been provided. Also, emergency and fire drills were successfully conducted to prove their ability to respond to an emergency situation.

In summary, it appeared that AI-6.1 is adequate to control the interface program if it were implemented correctly. The interface log in the interface study should be reviewed against test procedures and updated. The preoperational test directors should be better informed about the interface control program. Hold Orders on valves appear to be a point of confusion. Many valves which should be interface valves are closed with no documented interface control. Most preoperational test engineers properly used the TACFs as interface controls, but the HOs were not widely used.

VI. DOCUMENTS REVIEWED

1. TVA Topical Report, TVA-TR75-1A R7

2. N80A27, "Interdivisional Procedure ID-QAP-2.5 R1, "Major Modifications," 6/29/82

3. DPM-1300, "Field Services," 3/14/83

4. N-OQAM, Part II, Section 3.1, "Plant Modifications: Before Issuance of the Operating License," 3/18/82

5. N-OQAM, Part II, Section 3.2, "Plant Modifications: After Licensing," 6/3/83

6. N-OQAM, Part II, Section 5.3, "Maintenance and Modification Inspection Program," 6/23/82

7. N-OQAM, Part II, Section 5.3A, "Training and Certification Program for Quality Control Inspectors," 8/9/83

8. N-OQAM, Part III, Section 3.1, "Control of Measuring and Test Equipment," 1/23/84

9. WBN-AI-5.4 R7, "Material Issue, Transfer, and Traceability," 3/1/84

10. WBN-AI-7.1 R5, "Quality Control (QC) Inspection Program," 7/26/83

11. WBN-AI-8.5 R9, "Control of Modification Work on Transferred Systems Before Unit Licensing," 2/9/84

12. WBN-AI-8.1 R0, "Control of Modification Work After Unit Licensing," 8/9/83

13. WBN-AI-9.2 R9, "Maintenance Program," 1/27/84

14. WBN-TI-10 R19, "Calibration Program for Measuring and Test Equipment," 2/2/84

15. FSGL-A16 R4, "The Handling of Field Services Group Measuring and Test Equipment," 5/28/83

16. QA-SIL-4.2 R7, "Quality Control (QC) Inspection Program," 1/20/84

17. FQE-SIL-5.1 R16, "Survey Program," 3/7/84

18. Informal memorandum from W. T. Cottle dated March 10, 1982, "Watts Bar Nuclear Plant - Maintenance and Field Services at Watts Bar Nuclear Plant"

19. Memorandum from T. G. Campbell to Those listed dated August 18, 1982, "All Nuclear Plants - Design Change Request Processing and Modification Control" (L35 820804 858)

20. Informal memorandum to W. T. Cottle dated March 27, 1984, "Independent Safety Engineering Group Report - Plant Training on Administrative Procedures"

21. Memorandum from J. F. Bledsoe to W. T. Cottle dated June 3, 1983, "Employee Training" (OQA 830614 700)

22. Memorandum from T. L. Howard to W. T. Cottle dated March 14, 1984, "Watts Bar Nuclear Plant - FQE Section Monthly Report"

23. Memorandum from G. W. Killian to J. A. Coffey dated March 19, 1984, "Office of Quality Assurance Audit Report No. SQ-8400-03 - Plant Staff Performance Training and Qualification" (OQA 840319 702)

24. OQAB Surveillance Report No. XWB-S-84-0013, "Preop Test/Workplan Control," 3/29/84

25. FQE Survey Report No. 13MT(F)-82-1 "M&TE (Field Services)," 4/2/82

26. FQE Survey Report No. 13MT(F)-83-1, "Field Services Measuring and Test Equipment," 6/2/83

27. WBN-CAR-84-06, WBN-CAR-84-08

28. WB-DR-84-12-R, WB-DR-84-37-R, WB-DR-84-38-R, WB-DR-84-39-R, WB-DR-84-40-R, WB-DR-23-R, WB-DR-24-R

29. MR-224689

30. Workplans - 2393, 2419, 2432, 2441, 2443, 2682, 3704, 3837, 2332, 4173, 3762, 3384, 3665, 494, 720, 3906, 2749, 3230, 3915, 3578, 2827, 2294, 2637, 2440, 2683, 3292, 4124, 3293, 2860, 3126, 2706, 3656, 1738, 2797, 3441, 3911, 3918, 2897, 4163, 4154, 4051, 4132, 2023, 3593, 3590, 3652, 2870, 3270, 3901

31. WBN-QCI-1.30 R5, "Control of Work in Transferred Systems, Equipment, and Architectural Features," 9/17/82

32. WBN-AI-4.1 R6, "Quality Assurance Records"

33. 10CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"

    Appendix A - General Design Criteria for Nuclear Power Plants
    Appendix B - Quality Asurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

34. 10CFR Part 70, "Domestic Licensing of Special Nuclear Material"

35. DPM WB79E1 (1806.01), "Fuel Exposure and Isotopic Accounting for WBN," 9/18/81

36. WBN Final Safety Analysis Report (FSAR)

37. Regulatory Compliance Program Manual

38. WBN Draft Technical Specifications

39. WBN Fuel Handling Instruction Manual (FHIM)

40. WBN Technical Instruction TI-2, "SNM Control and Accountability System"

41. WBN Technical Instruction TI-28, "Physical Verification of Core Load Prior to Vessel Closure"

42. WBN Technical Instructions TI-2, "Initial Fuel Receipt and Storage"

43. WBN-AI-2.8.7 R2, "Report of Accidental Criticality or Loss or Theft or Attempted Theft of Special Nuclear Material"

44. WBN-AI-2.8.8 R2, "Report of Unaccounted for Shipments, Suspected Theft, Unlawful Diversion of SNM or Industrial Sabotage"

45. WBN-AI-5.2 R3, "Receipt Inspection of Materials, Parts, or Components"

46. WBN-GOI-2 R5, "Plant Startup from Cold Shutdown to Hot Standby"

47. WBN-SOI-99.1 R4, "Reactor Protection System"

48. WBN-SI-3.1.1 R3, "Functional Test of Manual Reactor Trip Channels (Prior to Startup)"

49. AOI-1 R5, "Reactor Trip"

50. EOI-12, "Emergency Shutdown Procedure"

51. MI-57.2 R5, "Annual 480-Volt Switchgear Inspection"

52. IE Information Notice 83-81, "Failures of the Undervoltage Reactor Trip System Breakers"

53. Westinghouse WATM-10709 Field Change Notice

54. Regulatory Guide 1.13, Revisions 1 and 2

55. Regulatory Guide 1.33, "Quality Assurance Program Requirements"

56. ANSI N18.7/ANS-3.1, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants"

57. N-OQAM, Part III, Section 1.1, "Document Control"; Section 5.0, "Periodic Review of Procedures and Instructions"; and Section 8.1, "Preparation, Maintenance, and Implementation of the Manual"

58. WBN-AI-3.1 R4, "Plant Instructions - Control and Use"

59. SI-3.3.3-I R4 and SI-3.3.3-II R4, "Radiation Monitoring Instrumentation Fuel Pool Radiation Monitors, Channel Calibration"

60. SI-3.3.4-I R4 and SI-3.3.4-II R4, "Radiation Monitoring Instrumentation Fuel Pool Radiation Monitors Changel Functional Test"

61. SI-4.0.5.3.C.3, "Check Valve Testing During Refueling Outage - Feedwater System"

62. ID-QAP-2.3 R1, "Physical Interfaces Between Licensed and Unlicensed Units"

63. Area Plan 1103.01, "Physical and Functional Interfaces," 9/27/83

64. WBN-AI-1.6 R2, "Interface - Establishment and Control"

65. Preoperational Test Section Instruction Letter No. 9, IL-9, "Preoperational Test Program - Unit Interface Program"

66. Operations Section Instruction Letter, OSLA-36 R1, "Interface Controls"

67. Interface Study Report, September 1980

68. FQE Survey NSI-84-50

69. Preoperational Test Instructions:

    a. TVA-9A, "Auxiliary Gas Treatment System and Door Status Indication and Interlock"

    b. TVA-10, "Control Building Air Conditioning System"

    c. TVA-28, "Sampling System"

    d. W-10.9, "Ice Condenser Reactor Containment System"

    e. W-2.2, "Boric Acid System"

    f. TVA-1, "Shield Building Inleakage Rate Tests, Emergency Gas Treatment System"

70. Hold Orders 20001, 20002, 20004, 20009, 20010

71. Temporary Alterations Control Forms 2-2000-70, 2-2002-82, 2-2003-82, 2-2004-83, 0-2005-80, 2-2006

72. Memorandum from H. G. Parris to G. F. Dilworth, "Nuclear Safety Review Staff Review No. R-83-22-NPS on Training of Plant Management" (GNS 830907 100)

73. Memorandum from H. G. Parris to G. F. Dilworth, "Nuclear Power Position Regarding Training Plant Management - NSRS Report No. R-83-22-NPS" (GNS 830809 050)

74. NUC PR Procedure No. 0202.07, "Shift Technical Advisor (STA) Training," 9/15/83

75. NUC PR Procedure No. 1202S01, "Shift Technical Advisor (STA) Program Responsibilities," 10/5/83

76. WBN-AI-2.16, "Shift Technical Advisors"

77. WBN Engineering Section Instruction Letter No. ENSL R1, "Reactor Engineering Unit Personnel Training"

78. WBN Engineering Section Instruction Letter No. ENSL R4, "Shift Technical Advisor Plant Familiarization Walk Throughs"

79. 10CFR50.34 and .36-1982

80. NUREG-0737, "Clarification of TMI Action Plan Requirements"

81. 10CFR50.54M, "Conditions of Licenses"

82. Regulatory Guide 1.8 R1-R-75, "Personnel Selection and Training"

83. ANSI 18.1-1971, "Standard for Selection and Training of Personnel for Nuclear Power Plants"

84. Area Plan 9, Procedure No. 0901.01, "Organization and Staffing"

85. N-OQAM, Part III, Section 6.1, "Selection and Training of Personnel for Nuclear Power Plants"

86. NUC PR Nuclear Training Program Area Plan No. 2, Procedure 0202.05, "Nuclear Plant Operator Training Programs," 8/8/83

87. OP-QAP-2.6 R0, "Selection and Training of Nuclear Power Plant and Support Personnel"

88. WBN-AI-2.1 R8, "Authorities and Responsibilities for Safe Operations and Shutdown"

89. WBN-AI-2.4 R5, "Shift Manning and Recall of Personnel to Plant"

90. WBN-AI-10.1 R3, "Plant Training Program"

91. NSRS Report No. R-81-03-WBN

92. NSRS Report No. R-82-16-WBN

93. NSRS Report No. R-83-07-WBN

94. NSRS Report No. R-83-22-WBN

95. SQN-AI-4 R41, "Plant Instructions - Document Control"

96. BFN Standard Practice, BF 2.3, "Review, Approval, and Use of Instructions," April 11, 1984

VII. LIST OF PERSONNEL CONTACTED

Name/Title	Attended Entrance Meeting	Contacted During Review	Attended Exit Meeting
Rita Aikens, QCRU CONST Engineer		X	
James G. Adair, Civil Engineer, Civil Support Branch		X	

Name/Title	Attended Entrance Meeting	Conducted During Review	Attended Exit Meeting
Daniel L. Anderson, Electrical Maintenance Engineer		X	
Robert H. Anderson, Contracts Civil Engineering Branch		X	
J. M. Ballard, CONST Engineer		X	
R. A. Beck, Health Physics Supervisor		X	
J. F. Bledsoe, OQAB Site Representative			X
Ralph J. Blevins, Document Control Staff		X	
Vincent M. Burzese, OMMM Engineer		X	
W. L. Byrd, Compliance Section Supervisor	X	X	
L. N. Calahan, OMMM Engineer		X	X
C. Richard Cook, Operations Section - SRO		X	
Dennis Collins, OMMM Engineer		X	
J. L. Collins, Mechanical Maintenance Supervisor		X	
W. T. Cottle, Plant Manager	X	X	X
M. E. Cutlip, Compliance Engineer			X
Grady Davis, Operations SPO		X	
W. C. Delk, Reactor Engineering Supervisor		X	X
G. T. Denton, Operations Section Supervisor		X	X
J. E. Englehardt, Compliance Engineer	X	X	X
E. R. Ennis, Assistant Plant Manager			X
Craig S. Faulker, Reactor Engineer/STA		X	
Randall R. Gibbs, Reactor Engineer/STA		X	
R. J. Griffin, Mechanical Engineering Supervisor		X	
C. A. Hearr, Preoperational Test Engineer		X	
D. S. Heidwich, OMMM Engineer			X

Name/Title	Attended Entrance Meeting	Contacted During Review	Attended Exit Meeting
George R. Hendricks, Engineering Aide		X	
J. E. Hoffert, OMMM Engineer		X	
T. L. Howard, Field Quality Engineering Supervisor		X	X
Mario Hug, National Nuclear Corporation, California		X	
T. F. Huth, Reactor Engineer/STA		X	
Joseph Ineer, OQAB Engineer		X	
G. L. Johnson, mechanical Engineer		X	
Ken Jones, Engineering Supervisor		X	X
G. T. Jordan, OMMM Engineer			X
William S. Karsner, Reactor Engineer/STA		X	
Dale Kaulitz, Preoperational Test Engineer		X	
M. E. King, Chemical Engineer		X	
J. T. Kirkpatrick, OMMM Assistant Supervisor		X	
H. F. Koehler, Preoperational Test Engineer		X	
L. B. Kuehn, Test Section Supervisor		X	X
J. P. Law, OMMM Engineer		X	X
James E. Lee, Instrument Maintenance Engineer		X	
D. L. Lester, Preoperational Test Group Supervisor		X	
Jim Loud, Safety Section Supervisor		X	
Zlata I. Martin, Reactor Analysis Group		X	
Robert T. McCollom, Compliance Section Engineer		X	
L. N. McIntosh, OMMM Superintendent		X	X
Ben Mears, Preoperational Test Engineer		X	

TABLE I

WATTS BAR
OPERATIONAL READINESS REVIEW

Phase	Review Area
I 2/13/84 - 2/17/84 (Completed - Report R-84-02-WBN Issued)	1. General Employee Training 2. Employee Awareness of Regulatory and TVA Requirements and Policies Relating to Nuclear Safety Issues and Expression of Staff Views 3. Preoperational Testing (Partial)
II 3/26/84 - 4/6/84 (Completed - Report R-84-05-WBN)	1. Organization 2. Qualifications of Personnel in Key Management Positions 3. Shift Technical Advisers 4. Control of Licensed Activities 5. Plant Procedures (Partial) 6. Unit Interface Control 7. Reactor Safety and Criticality Control (Partial) 8. Modifications and Outage Control
III 6/19/84 - 6/28/84	1. Mini Hot Functional Test - Operations Section and Test Section personnel activity will be reviewed during this time. Adequacy of and adherence to instructions and procedures will be stressed. 2. Maintenance 3. Reactor Safety and Criticality Control
IV 7/24/84 - 7/27/84	Initial Fuel Load

Note 1. Plant staffing and organization will be further evaluated during subsequent reviews due to changes caused by the reorganization.

Note 2. Regulatory compliance is a part of all reviews.

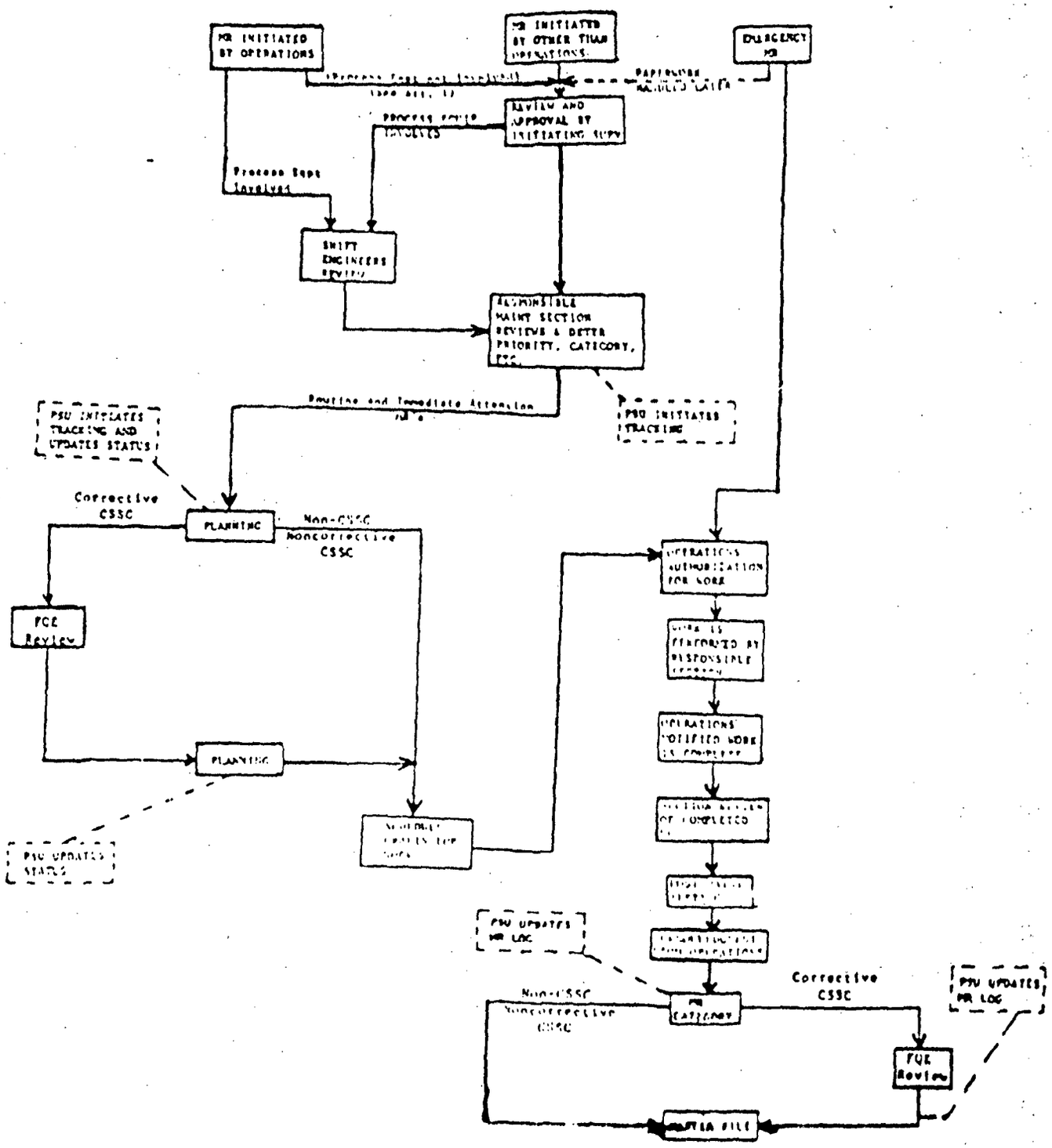
Name/Title	Attended Entrance Meeting	Contacted During Review	Attended Exit Meeting
Redford Norman, Assistant Operations Section Supervisor		X	
H. L. Pope, FQE Engineer		X	X
Thomas A. Shelton, Nuclear Engineer, Nuclear Engineering Branch		X	
L. J. Smith, FQE Engineering Supervisor		X	X
James Swallows, FQE Engineer		X	
G. V. Tippens, FQE Quality Control Supervisor		X	
Bill D. Varga, Training Officer		X	
Lynn Wallace, FQE Engineer		X	
R. L. Warren, Reactor Engineer/STA		X	
Luther Welsh, National Nuclear Corporation, California		X	
J. R. Werkler, Preoperational Test Engineer		X	
Steve Woods, Acting Instrument Maintenance Supervisor		X	
Joe Yarborough, FQE Engineer		X	

TABLE II

Test No.	Title	System	Points Installed	Points to be Installed	Total
TVA-9A	Auxiliary Gas Treatment System & Door Status Indication & Interlock	30	6		6
TVA-18C	Essential Raw Cooling Water Flow Balance	67	39	10	49
TVA-20A	Component Cooling System	70	9	9	18
TVA-44A	Liquid Waste Drains, Collection, & Transfer Facilities	77	1	17	18
TVA-14E	Diesel Generators and Supporting Auxiliaries	81	2		2
SUBTOTAL			57	36	93
TVA-10	Control Building Air Condition System	31		10	10
NCS-21	Gland Seal Water System	37		5	5
TVA-28	Sampling System	43		6	6
W-10.9	Ice Condenser Reactor Containment System	61		7	7
W-2.2	Boric Acid System	62		19	19
TVA-1	Shield Building Inleakage Rate Tests, Emergency Gas Treatment System	65		11	11
TVA-46	Primary Makeup Water System	81		5	5
TVA-51	Flood Protection Provisions	84		3	3
SUBTOTAL				66	66
TOTAL			57	102	159

ATTACHMENT A

MAINTENANCE REQUEST ROUTING



ATTACHMENT B

WORKPLAN CONTROL FORM

I. Identifying Information

Prepared by _____ Date _____ Phone _____

Outstanding Work Item Number: _____ _____ _____
Unit System Type - Number

II. Prework Review

A. Originating Section

Technical specification change required? Yes ____ No ____

☐ Technical verification and review complete.

☐ Affected section supervisors considered and listed in II.C below.

☐ Instructions and/or vendor manuals requiring revision listed on page 5.

☐ The CSSC list needs revision as a result of this modification.

☐ If work affects the pressure boundary of an ASME code component on which the N-5 Code Data form is signed, an ANI Instruction Review sheet is included.

_____ / _____
Section Supervisor Date

B. Workplan Coordinator

Tracking numbers assigned, appropriate reference documents included.

_____ / _____
Workplan Coordinator Date

ATTACHMENT A (CONTINUED)

C. Affected Sections (excluding Operations, Safety, and QA)

Review for technical accuracy; instructions and/or vendor manuals requiring revision listed on page 5 of attachment A.

Section	Supervisor's Signature	Date
* Prcop		

D. Safety Engineer Review (may be performed by fire brigade captain in absence of the safety engineer).

Safety Engineer _____ / _____ Date

E. Reviewed for effect on operating systems (check one block in paragraph III.B).

/ / Operating instruction requiring revision listed on page 5 of attachment A.

Operations Supervisor _____ / _____ Date

* F. For CSSC workplan [illegible], initial review by:

Supv, Field QE Section _____ / _____ Date

G. For CSSC workplan [illegible] recommended for approval of this modification.

PORC Chairman _____ / _____ Date

H. This modification [illegible] authorized to be performed when appropriately scheduled.

Power Plant Superintendent _____ / _____ Date

ATTACHMENT B (CONTINUED)

III. Performance of Work

A. Workplan has been scheduled at daily meeting and work may begin.

_____ / _____
Modification Coordinator Date

B. Removal of equipment from service. Signature on this paragraph is required before equipment is taken out of service. Other work (prefabrication, hangers, non-equipment-related work, etc.) may proceed without further authorization.

☐ This workplan required no equipment to be removed from service.

☐ This workplan requires equipment to be removed from service.

_____ / _____ _____
Shift Engineer Date Time

C. List equipment removed from service.

1. _____ 2. _____ 3. _____

IV. Work Completion

A. Field work, including functional tests, is complete. Shift engineer's control copy drawings updated by cognizant engineer to show modifications.

_____ / _____
Cognizant Engineer Date

B. This signoff required only if equipment was removed from service.

☐ Operator training on modified systems completed, if required.

☐ Temporary/permanent revisions have been made to affected operating instructions (see list on page 5).

Item IV., A. and B., above must be complete before returning equipment to service.

_____ / _____ _____
Shift Engineer Date Time

C. Drawing requirements complete and verified for correctness on attachment D.

_____ / _____
Cognizant Engineer Date

ATTACHMENT B (CONTINUED)

D. Vendor manual and/or instruction changes listed on page 5 complete.

_____ / _____
Cognizant Section Supervisor Date

E. Spare parts inventory revised as required for modified equipment, spare parts ordered for new equipment.

_____ / _____
Cognizant Section Supervisor Date

F. Nameplate data collected for each component affected on Nameplate Data Form, Attachment F.

_____ / _____
Cognizant Engineer Date

G. ASME Section XI Summary Report filled out, if required.

_____ / _____
Cognizant Engineer Date

H. Post-modification test required by EN DES scoping document has been evaluated and given field approval. Copies of test results have been sent to EN DES as required.

_____ / _____
Cognizant Section Supervisor Date

I. Workplan complete.

1. L-DCR file copy with marked-up drawings transmitted to EN DES if appropriate.
2. Nameplate data form transmitted in accordance with instructions.
3. CSSC list revised as necessary.
4. Section XI Summary Reports filed in appropriate places.
5. Workplan reviewed by ANI when required in accordance with attachment I.
6. OWIL updated.

_____ / _____
Modification Coordinator Date

TVA 84 (03-9-83)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

GNS '840406 050

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

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

DATE : April 6, 1984

SUBJECT: SPECIAL REVIEW OF NCR WBNSWP8303 RELATED TO MISSING PIPE SUPPORT CALCULATIONS - WATTS BAR NUCLEAR PLANT - UNITS 1 AND 2 - NUCLEAR SAFETY REVIEW STAFF REPORT NO. R-84-07-WBN

The final report of the subject review is attached for your information and action. The review was initiated when the subject NCR disposition was sent to NSRS for review. This occurred after the independent review of the corrective action was completed.

We believe this review indicates a lack of independence in the classification of NCRs as to whether they are adverse to quality or not. We also believe the requirement for maintaining quality records is a fundamental part of the TVA Quality Assurance (QA) program. The approach should be to meet all requirements of the QA program and not take them lightly when the result will be an admission of incorrect decisions or error.

The report contains two specific recommendations concerning this particular NCR. You are requested to provide us with your plan for resolving the two recommendations within 30 days of the date of this memorandum. It is expected that appropriate action to correct these conditions will be completed in a timely manner.

If you have any questions concerning the report, please contact P. R. Washer at extension 6860.

H N Culver

H. N. Culver

PRW:LML
Attachment
cc (Attachment):
J. W. Anderson, M155G MIB-K
R. W. Cantrell, W11A9 C-K
MEDS, W5B63 C-K

NSRS FILE

GNS '840406 051

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS REPORT NO. R-84-07-WBN

SUBJECT: WATTS BAR NUCLEAR PLANT - UNITS 1 AND 2 - SPECIAL REVIEW OF NCR WBNSWP8303 RELATED TO MISSING PIPE SUPPORT CALCULATIONS

DATES OF REVIEW: MARCH 5 - MARCH 13, 1984

REVIEWER: P R Washer — P. R. WASHER — 4/5/84 DATE

APPROVED BY: J. F. Murdock — J. F. MURDOCK — 4/5/84 DATE

MEDS, W5B63 C-K

TABLE OF CONTENTS

I. SCOPE

The purpose of this review was to evaluate NCR WBNSWP8303, which identifies missing pipe supports calculations that were performed by EDS on safety-related systems in the reactor building at Watts Bar Nuclear Plant (WBN) units 1 and 2. The review was to determine whether the calculations were to be retained for the life of the plant as quality records as defined in TVA's QA program and ANSI Standard N45.2.9-1974. The review also included an evaluation of whether this condition should be classified as "significant condition adverse to quality" and reported to NRC under 10CFR50.55(e).

II. BACKGROUND

This item was brought to the attention of NSRS on March 5, 1984, when a copy of NCR WBNSWP8303 was sent to NSRS for review of the proposed disposition. The document was sent to NSRS after the independent review of the proposed corrective action was completed by T. C. Cruise on March 2, 1984. The NCR was written on February 23, 1983, and has been open since that time.

III. CONCLUSIONS AND RECOMMENDATIONS

The following paragraphs contain the conclusions followed by recommendations, if applicable.

A. R-84-07-WBN-01

The calculations for pipe supports on safety-related systems must be kept as quality records for the life of the plant. These are required as permanent records to meet the TVA QA program as defined in Topical Report TR75-1A and the quality records requirements as defined in N45.2.9-1974.

Recommendation

The calculations for the pipe supports on safety-related systems should be recreated and stored as a part of quality records. During the creation of these design calculation, if any pipe supports require changes, then a copy of the calculations for those supports should be sent to NSRS. See IV.A for details.

B. R-84-07-WBN-02

This NCR is a "significant condition adverse to quality" since it reflects an overall breakdown in the Watts Bar QA program related to records retention.

Recommendation

The NCR should be upgraded to a "significant condition adverse to quality" and reported to NRC as a deficiency under 10CFR50.55(e)(1)(i). See IV.B. for details.

IV. DETAILS

A. R-84-07-WBN-01

EDS Nuclear, Incorporated, was contracted by TVA to do piping analysis and pipe support design on safety-related systems at WBN and Sequoyah Nuclear Plant (SQN). In reference (1), EDS wrote to TVA to confirm verbal instructions to destroy the stored records of this work. In reference (2), TVA confirmed that EDS could destroy all stored records except for "hard copy code compliance computer runs." OQA concurred with this letter based on the understanding that a seperate copy of the calculations existed at TVA. They would not have agreed to destroy the calculations if they had known that EDS had the only copy of the calculations. As a result of that direction, EDS destroyed all the record engineering calculations on the pipe supports that they had done. The only thing that exists is the input loads from the piping analysis and the end product, pipe support drawings. In reference (3), EDS documented several conversations between EDS and TVA personnel. TVA had requested that EDS estimate the costs to provide TVA with a copy of the support calculations. EDS confirmed in this reference that they had carried out TVA's direction and destroyed the calculations.

An NCR regarding the missing calculations was transmitted by reference (4) from the project to CEB. After stating lack of knowledge of ever having received the NCR, CEB transmitted the NCR by reference (5) back to the project for processing. In the process of dispositioning the NCR, the project, after an independent review, sent the NCR to NSRS for review.

NSRS, during their review, came to specific conclusions and recommendations based on the following information. Chapter 17 of the WBN FSAR states that the QA program for WBN shall be as presented in TVA Topical Report TVA-TR75-01, section 17.1A. In TVA-TR75-1A, paragraph 17.1.17, "Quality Assurance Records," it is stated that "the typical types of records to be generated and retained are listed in Appendix A to ANSI N45.2.9-1974." In Appendix A to N45.2.9, design calculations and records of checks are shown to be stored for the lifetime of the plant, if they are classified as "Life-time Quality Assurance Records."

In paragraph 2.2.1 of ANSI N45.2.9, there are four criteria, any one of which qualifies records as lifetime records. The pipe support calculations for safety-related systems qualify under the first three of these criteria. Since these calculations meet the criteria for lifetime records, as defined in the TVA QA program, the calculations must be kept for the life of the plant. Since the calculations have been destroyed, they must be recreated and stored as quality records.

B. R-84-07-WBN-02

The destruction of the safety-related piping support calculations is a major breakdown in the TVA QA program for vendors. This is a breakdown in the implementation of the requirements of TVA Topical Report TVA-TR75-1A, Paragraph 17.1.17., "Quality Assurance Records." It is also a nonconformance to Criterion 1 of the NRC General Design Criteria. As such, this NCR should be upgraded to a "significant condition adverse to quality" and reported to NRC as a deficiency under 10CFR50.55(e)(1)(i).

V. REFERENCES

1. EDS letter 0060-300-090, S. B. Hosford to R. O. Barnett dated June 4, 1981, "SNP, WBNP, Disposition of Backup Documentation" (CEB 810609 273)

2. TVA letter R. O. Barnett to EDS Nuclear, Incorporated, dated August 19, 1981, "Disposal of Records Stored by EDS Nuclear, Incorporated (EDS)." (CEB 810819 023)

3. EDS letter 0060-30-182, S. B. Hosford to R. O. Barnett dated November 1, 1982, "Watts Bar Nuclear Plant, Copies of Support Calculations"

4. Memorandum from J. C. Standifer to R. O. Barnett, C. Bonine, L. J. Cooney, R. A. Costner, J. C. Key, J. J. Nash, and G. Wadewitz dated February 22, 1983, "Watts Bar Nuclear Plant Units 1 and 2 - Nonconformance Report WBNSWP8303" (SWP 830225 060)

5. Memorandum from R. O. Barnett to J. C. Standifer dated July 11, 1983, "Watts Bar Nuclear Plant Units 1 and 2 - EDS Nuclear Engineering, Incorporated (EDS), Support Calculations - NCR WBNSWP8303" (CEB 830711 027)

TVA 64 (09-9-68)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

CMS 840627 053

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

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

DATE : June 27, 1984

SUBJECT: BELLEFONTE NUCLEAR PLANT (BLN) - NUCLEAR SAFETY REVIEW STAFF (NSRS) REVIEW OF INPO FINDING QP-5.1 - NSRS REPORT NO. R-84-09-BLN

Attached is the NSRS report for the review conducted at BLN concerning INPO finding QP-5.1. This review consisted of an examination of the three conditions identified by the INPO reviewer: (1) some inspectors were being encouraged not to write nonconformance reports (NCRs), (2) nonconforming conditions had been dispositioned by invalidating or voiding the NCR, and (3) NCRs closed before corrective action had been completed.

NSRS found that there was no indiciation to support the position that inspectors were being encouraged not to write NCRs. Some administative or procedural problems with the NCR process may have caused some of the inspectors to perceive a problem. With regard to the other two identified conditions, NSRS found sufficient indication to support the INPO finding.

Four recommendations were made in the report for BLN response. NSRS requests a written response by August 1, 1984. If there are any questions concerning this report, please contact C. M. Key at extension 4815 in Knoxville.

H N Culver

H. N. Culver

CMK:LML
Attachment
cc (Attachment):
W. R. Brown, 102 ESTA-K
MEDS, W5B63 C-K

NSRS FILE

GNS 840627 054

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

REVIEW

NSRS REPORT R-84-09-BLN

SUBJECT: REVIEW OF INPO FINDING QP-5.1

DATES OF REVIEW: APRIL 23-27 AND MAY 10-11, 1984

REVIEWER: C. M. Key — C. M. KEY — DATE 6/21/84

APPROVED BY: M. S. Kidd — M. S. KIDD — DATE

MEDS, W5D63 C-K

TABLE OF CONTENTS

I. BACKGROUND

During the Institute of Nuclear Power Operations (INPO) review (CP-84-02) of Bellefonte Nuclear Plant (BLN), item QP-5.1 was identified by an INPO evaluator. This finding was in the area of "corrective actions." The evaluator's recommendation was that "controls need to be implemented to ensure that conditions adverse to quality are being identified and resolved in an effective manner." Three conditions that required attention were cited:

1. Some inspectors were being encouraged not to write nonconformance reports (NCRs).

2. Nonconforming conditions had been dispositioned by invalidating or voiding the nonconformance report.

3. Nonconformance reports had been closed before the corrective action had been completed.

Due to the possibly serious nature of this item, W. R. Brown, BLN Project Manager, requested that the Nuclear Safety Review Staff (NSRS) perform an independent review of this finding. Per this request, the NSRS initiated a review that commenced on April 23, 1984, and was concluded on May 11, 1984.

II. SCOPE

The review involved examination of the three cited conditions: some inspectors were being encouraged not to write NCRs; nonconforming conditions had been dispositioned by invalidating or voiding the NCR; and NCRs had been closed before the corrective action had been completed. The review was conducted by interviewing personnel, reviewing procedures and records, and using other established review techniques. The NSRS review was limited to interviewing personnel from the Electrical Quality Control (EQC) and Instrumentation Quality Control (IQC) Units. This limitation was imposed based on information received from the INPO evaluator and W. R. Brown.

III. CONCLUSIONS AND RECOMMENDATIONS

A. R-84-09-BLN-01, Inspectors Encouraged Not to Write NCRs

Conclusion

The NSRS found no evidence to indicate that there was a pervasive, systematic attempt by BLN management to discourage the reporting of nonconforming conditions. Some administrative or procedural problems with the NCR process may have caused some of the inspectors to perceive a problem. See section IV.A for details.

Recommendation

The NSRS recommends that: (1) definitive guidelines be issued to provide instructions for the usage of "Reply" memos, (2) appropriate action be taken to emphasize to all employees the importance of proper identification and handling of nonconformances, and (3) the nonconformance procedure be revised to require the NCR be numbered prior to the review and approval cycle.

B. R-84-09-BLN-02, Nonconforming Conditions Dispositioned by Invalidating or Voiding the NCR

Conclusion

Nonconformance reports had been invalidated or voided improperly. See section IV.B for details.

Recommendation

NSRS recommends that all invalidated NCRs be reviewed to determine action necessary to correct nonconformances that have been improperly invalidated or voided. For action to prevent recurrence, the NSRS recommends that the nonconformance procedure be revised to provide a detailed explanation of the invalidation process and to require an independent review of all invalidated NCRs. Also, appropriate action (e.g., training) should be taken to ensure that all personnel have a thorough understanding of what constitutes a valid NCR.

C. R-84-09-BLN-03, NCRs Closed Before Corrective Action Completed

Conclusion

Nonconformance reports have been closed before corrective action to rectify the nonconforming condition has been completed. See section IV.C for details.

Recommendation

NSRS recommends that this condition adverse to quality be documented on a nonconformance and that appropriate corrective action (e.g., a sampling program) be taken. NSRS recommends that the nonconformance procedure be revised to ensure that NCRs are not closed prior to completion of corrective action to rectify the nonconforming condition to prevent recurrence.

D. R-84-09-BLN-04, Evaluation of "Offsite-Generated NCRs"

Conclusion

The evaluation of "offsite-generated NCRs" allowed by BNP-QCP-10.4 and QAP 15.1 violates upper-tier requirements. See section IV.D for details.

Recommendation

NSRS recommends that the site perform a review to determine if any items with "offsite-generated NCRs" have been received and that nonconformance reports be initiated for items not covered by site NCRs. In addition, the NSRS recommends that BNP-QCP-10.4 and QAP 15.1 be revised to require the site to initiate NCRs to track "offsite-generated NCRs."

IV. DETAILS

A. Inspectors Encouraged Not to Write NCRs

The INPO evaluator's basis for this concern appeared to be interviews of quality control (QC) inspectors. The following accounts of interviews were included in the supporting details for item QP-5.1:

> During an interview with two inspectors both individuals expressed a concern that not all of the nonconformance reports that they prepared were approved and issued. One individual voiced a concern that some construction supervisory personnel had been placed in QC inspection supervisory positions. The perception expressed by the individual was that construction was taking over Quality Control.
>
> During an interview with an individual from one of the QC inspection units the individual expressed a concern that he and some of his subordinates were being encouraged not to write nonconformances reports because of the cost involved and how it looked bad for QC if the reported deficiency turned out not to be valid.

Since this type of concern cannot be identified or substantiated by a records review, the NSRS utilized the interview process to gather information used in making a determination of the validity of the concern. A total of 36 quality control inspectors and 3 management personnel were interviewed during the NSRS review.

The Quality Manager's Organization (QMO) became effective January 23, 1983, and was implemented February 20, 1983. This organization was formed to separate the QC functions from production support units. Previously to this all engineering and QC personnel reported to the same first line supervisor (M-5). The reorganization removed the QC inspection and related quality assurance (QA) functions from the CONST engineering organization (CEO) and placed them under the QMO. The QMO is headed by a Quality Manager who reports directly to the Project Manager. The QMO was staffed similarly to the CEO and contained management positions at the appropriate levels. Two of the supervisory positions were filled by personnel from the CONST (craft). These two individuals had held supervisory positions in their respective areas before becoming

supervisors in the QMO. Each individual was offered and accepted a supervisory position in the same area (electrical and instrumentation) he worked with in CONST. During the course of interviewing QC personnel, no inspector expressed the opinion or perception that the movement of these two supervisors into the QMO was an effort by CONST to take over quality control. As a result of the absence of input concerning the issue of CONST taking over quality control, NSRS did not pursue this area any further.

On November 1, 1983, a significant change was made in the QA program. This program change was the replacement of the quality control investigation report (QCIR) with the inspection rejection notice (IRN). The QCIR was an integral part of the nonconformance program. It was used to document, disposition, and control known or suspected conditions adverse to quality (CAQ). The procedure (BNP-QCP-10.4) required that upon institution of a QCIR all affected items be tagged, if practical. After the QCIR tag was attached to an item, it was "not to be relocated, reworked or repaired except as designated by the approved disposition on the QCIR form." All QCIRs were evaluated to determine if a nonconformance report should be generated. Upon completion of the recommended disposition, the QCIR was closed and kept as a QA record. An IRN as defined by BNP-QCP-10.43 is "a communication tool used by inspection personnel to inform craft and/or engineering of a failed inspection." In accordance with the procedure an IRN should not be used for identification of possible nonconforming conditions. It does not prevent work from being accomplished on an item and is closed when the failed inspection condition is corrected. The IRN is a unit record and is required only to be maintained in the unit files until closure or voiding and completion of applicable trend analysis. Interviews of the QC personnel revealed that the inspectors as a group had a good understanding of the IRN process. As required by procedure, an IRN is only written on QA activities. Normally QC personnel do not inspect non-QA related activities, but will do inspections of this type if requested by engineering. Any deviations discovered on non-QA inspections are forwarded to the engineering unit via an informal means, such as a "Reply" memo.

The "Reply" memo as described by the Quality Manager was an "in-house" communication tool to be used on nonsafety related, non-QA items. It was to be used by engineering units to request quality control units to perform inspections on those activities not normally inspected by QC. In turn QC would give engineering the results of those inspections by returning the memo. The "Reply" memo could also be used by inspection to request information from engineering on possible drawing problems. It was stressed by the Quality Manager that the use of the memo was limited to matters dealing with non-QA, nonsafety related items. However during interviews with inspection personnel (inspectors and supervisors), it appeared to the NSRS reviewer that the "Reply" memo was not being strictly limited to non-QA, nonsafety related activities. (Due to time constraints, the NSRS reviewer was

unable to obtain specific examples of the "Reply" memo being used for safety-related activities.) As indicated earlier the memo was being used by engineering to solicit an inspection by QC. In addition to using them to pass information to engineering, QC personnel were usi : the "Reply" memo to make engineering aware of problems (i.e., questionable conditions of items in the powerhouse and drawing discrepancies). At the time of the NSRS review there was no procedure governing the use of a "Reply" memo. Therefore, as indicated by QC personnel, there was (1) no requirement for the engineering unit to answer the memo (however, inspectors indicated most were answered), (2) no formal documentation of potential deficiencies, (3) no tracking capability, (4) no required follow-up by inspectors, and (5) no requirement to escalate a memo to the NCR status if a nonconforming condition was identified. A nonconforming condition should be documented on an NCR and not a "Reply" memo.

BNP-QCP-10.4 defined nonconformance as "a deficiency in characteristic, documentation or procedure that renders the quality of an item or activity unacceptable or indeterminate." Questioning of inspectors on when to write an NCR revealed a very consistent response. The response was that QC personnel primarily wrote a nonconforming condition report on an incorrect item when it had been inspected and accepted and the records were in the vault (bought off). Questioning of inspectors on the method to handle incorrect or questionable items that had not been "bought off" revealed the following answers: (1) leave the condition as-is and wait for an inspection request to write up the condition, (2) prepare a "Reply" memo to engineering, (3) write an IRN, and (4) use any available method other than an NCR to rectify the situation. From the interviews, it appeared that inspectors had been indoctrinated to ensure that a condition was definitely nonconforming before initiating an NCR report and had been exposed to conversation concerning the cost of a nonconformance report. Although an unacceptable item is a nonconformance, by definition an indeterminate item is also a nonconformance. However, with the deletion of the QCIR procedure and the indoctrination to NCR only items positively identified as nonconforming, it appeared that inspectors had no "formal" method of addressing a questionable condition. This could have resulted in the alternative methods (cited earlier) being utilized to handle potential nonconformances. Contrary to the response given by the inspectors, when asked by the NSRS reviewer when to write an NCR, the QC supervisors essentially repeated the definition given by the procedure.

When questioned about the procedure for writing an NCR, the QC inspectors appeared to be following the same procedure. The following is a generalization of the procedure utilized by the inspectors:

1. Identification of potential nonconformance.

2. Check to ensure item had been "bought off".

3. If undecided ask opinion of fellow workers, group leader, and/or unit supervisor.

4. Write NCR.

5. Submit NCR to unit supervisor for review and approval.

6. Upon return of NCR from supervisor, obtain number from the Document Control Unit (DCU).

7. Submit to appropriate unit for disposition.

From interviews and review of NCRs, it was apparent that the NCRs were receiving another review and approval by the Assistant Quality Manager (AQM) in addition to the one performed by the unit supervisor. One unit supervisor indicated that the review was being performed and the review by the Assistant Quality Manager was for clarity of the NCR. The unit supervisor also stated that the additional review of NCRs by the AQM was performed because the Quality Manager had directed it be done. All of this review and approval are accomplished prior to the nonconformance being numbered. When the NCR was numbered by DCU then it became a recognized nonconformance.

In November and December 1983 the Office of Quality Assurance (OQA) performed an audit, C00-A-84-0001, at the Watts Bar (WBN) and BLN sites. Deviation number 10 of that audit cited BLN for conflicting requirements for initiation of NCRs. Attachment D of BNP-QCP-10.4 indicated that the report was numbered prior to review and approval by the supervisor. Paragraph 6.2.2.1 of the procedure required that the NCR be reviewed and approved before being numbered (currently the method being utilized). In the details of the report OQA recommended that the procedure as outlined in Attachment D be the site practice to control potential nonconformances since the responsible supervisor had the discretion to invalidate or void the initiated nonconformance report. The site response to this deviation was that BNP-QCP-10.4 would be revised to eliminate the conflicting requirements for initiation of NCRs by March 9, 1984. There was no statement as to the position that BLN would take on the issue (i.e., whether the NCR would be numbered before or after review and approval process). On March 9, 1984, a memo (BLN 840309 302) from L. S. Cox to R. W. Diebler stated that the revision to BNP-QCP-10.4 had not been initiated since the site was awaiting revision to higher tier documents (QAPs 15.1 and 16.1). The memo indicated that the procedure would be changed upon approval of these higher tier documents. Revision 11 to BNP-QCP-10.4, which was in the review cycle, stated explicitly that the NCR was not to be numbered until the responsible supervisor reviewed and approved it. This position is opposite to the recommendation made by OQA.

In continuance with the questioning the NSRS reviewer asked 25 of the 36 QC personnel if at any time they had been "encouraged" not to write a nonconformance report. (NOTE: This question was used to determine if an inspector had been "encouraged" not to write an NCR for any reason. This included the INPO examples of NCRs possibly not written because of high cost as well as damage to the QC unit reputation if the reported deficiency turned out not to be valid.) Twenty-four of the inspectors indicated that they had not been encouraged not to write an NCR. Most inspectors recounted that the group leader(s) and the supervisor were used as a source of advice on whether or not a condition was nonconforming but the final decision to write (or not write) the NCR had been left to the inspector. However, there were two questionable incidents revealed by this inquiry. The first incident was recalled by an inspector who stated that on one occasion he had been persuaded not to write an NCR. The inspector felt that the problem was a "gray" area and still wasn't sure that the condition was conforming but the problem had been corrected. Upon interviewing the responsible supervisor, he did not recall the problem and stated that he had never encouraged anyone not to write an NCR. Information on the other incident was obtained from a unit supervisor. He had encouraged a QC inspector not to write an NCR on one inspection. The explanation given by the supervisor was that the work had already been done and that an NCR would not accomplish anything else. However, the supervisor stated that the inspector had been instructed to write the NCR if in the inspector's opinion one was deemed necessary. No inspector from this unit bought this incident to the NSRS reviewer's attention during the interviews. It was also noted that no inspector recalled any occasion of being "encouraged" not to write an NCR because of cost or possibly damaging the unit's reputation if the NCR was not valid.

In connection with this question the inspection personnel were asked if any NCRs generated by them had not been approved and issued. With one exception the inspectors related that there had been no incidents where the NCR was not approved and issued. The majority revealed that there had been questions about the NCRs they had written from group leader(s) and supervisor(s). These questions had been clarification-type inquiries. In some cases inspectors recalled that some NCRs had not been written after being questioned, but the decision not to write the nonconformance report had been their own. The one exception occurred when an inspector accompanied an engineer to perform an activity. The results of the test were unacceptable. Instead of nonconforming the item, the engineer troubleshot the item, corrected the problem, and successfully completed the test. The inspector related that the supervisor felt that the documentation from the test activity was sufficient to cover the item. Also the supervisor said it was not an NCR because the problem did not exist anymore. The inspector believed that the documentation was barely adequate to justify the action taken. At a later date, during a discussion of what was an NCR, this situation was described to the Assistant Quality Manager and the inspector was told the condition was

nonconforming. In the interview the inspector stated if the problem ever arose again, he would initiate an NCR. While interviewing the responsible supervisor, the NSRS reviewer related the details of the incident to the supervisor and asked the supervisor his account. The supervisor did not recall the problem and stated that he had never disapproved an NCR.

As a part of the review the NSRS reviewer asked the QC personnel if there had ever been any discussion in their unit(s) as to why the QCIR was deleted. The typical comments made were: (1) IRN was cheaper than QCIR, (2) too many QCIRs being written, and (3) QCIRs were not being utilized properly. Some QC personnel perceived that deletion of QCIR was a mistake and that it should be reinstituted. On the subject of cost of QCIRs, one supervisor indicated that economics had been a factor in determining the fate of the QCIR. However, he indicated that quality had not been sacrificed when the IRN replaced the QCIR.

In summary, two supervisors from the CONST (craft) were placed in supervisory positions in the QMO. The disciplines to which the supervisors were assigned were the same as the ones they were involved with in CONST. Interviews revealed that the inspectors did not view this event as CONST trying to take over Quality Control. The "Reply" memo as described by the Quality Manager (i.e., to be used on non-safety related, non-QA activities) could be used to cover all activities at the site. Inspectors had been indoctrinated to ensure that a condition was definitely nonconforming before initiating an NCR. As a result of this indoctrination inspectors indicated that "indeterminate conditions" could be handled by: (1) leaving condition as-is until item was inspected, (2) preparing a "Reply" memo, (3) writing an IRN, or (4) using any alternative other than writing an NCR. In addition, cost of QCIRs and NCRs had been discussed in the units. Although two cases were identified that could be classified as "inspector being encouraged not to write an NCR," it did not appear that either was influenced by cost or how it might make the unit look if the NCR was not valid. With the exception of one incident, no evidence was found that NCRs prepared by the inspectors were not approved and issued.

NSRS determined that the transfer of the two CONST managers into the QMO had not led to the belief by the inspectors that CONST was taking over Quality Control. In addition, there was no evidence found to substantiate that inspectors were being "encouraged" not to write NCRs because of high cost or because it would damage the unit's reputation if the NCR was not valid.

The following factors when considered in total, do support the INPO contention that some inspectors could have perceived a supervisory attitude that discouraged the reporting of deficiencies using an NCR.

1. The "Reply" memo could be used to cover all activities.

2. Inspectors had been indoctrinated to write NCRs only on conditions that were definitely nonconforming.

3. "Indeterminate conditions" could be handled by alternate methods other than writing an NCR.

4. Cost of NCRs and QCIRs had been discussed in the units.

5. NCRs had to be reviewed and approved by management before being numbered.

6. Three incidents were identified that could be labeled "inspector encouraged not to write an NCR" or "potential NCR disapproved."

However, NSRS concluded that even though these factors did exist, there was no pervasive, systematic attempt by BLN management to discourage the reporting of nonconforming conditions.

In order to strengthen the program and to address the identified factors, NSRS recommends: (1) definitive guidelines be issued to provide instructions for the usage of the "Reply" memo, (2) appropriate action be taken to emphasize to all employees the importance of proper identification and handling of NCRs, and (3) revise the nonconformance procedure to require the NCR be numbered prior to the review and approval cycle.

B. Nonconforming Conditions Dispositioned by Invalidating or Voiding the NCR

The INPO evaluator listed four examples in the supporting details of NCRs that were dispositioned by invalidating or voiding. These nonconformance reports were numbers 765, 913, 2300, and 2839. The following paragraphs are excerpts from the INPO report that explain the INPO evaluator's position on the four NCRs.

NCR 765 - Temperature in a class "B" warehouse had dropped below the minimum requirements. The NCR was invalidated because it was accomplished by a QCIR. However, no specific QCIR was referenced.

NCR 913 - A 4-inch crack in the base material of a piping elbow. The NCR was invalidated because the process specification was revised.

NCR 2300 Some modules could not be calibrated. The NCR was determined to be significant, but was later invalidated.

NCR 2839 - Two installed category I conduit supports had the same unique identification number on them. The NCR was invalidated because it did not meet the definition of a nonconforming condition.

Prior to the start of the NSRS review, a quality assurance engineer on the Quality Manager's staff conducted a review of NCRs that had been invalidated from October 4, 1983 to March 21, 1984. His findings and conclusions were contained in a memorandum dated March 26, 1984, sent to the Quality Manager. The memo indicated during this time period there had been 18 invalidated NCRs (this number did not include the NCRs mentioned in the INPO report). Attached to the memo was a copy of these 18 NCRs (and a copy of the 4 noted in the INPO report) with the QA engineer's opinion of whether or not the nonconformance had been properly voided noted on each NCR. Prior to and/or during the course of the review, another evaluation of invalidated NCRs was completed by this same QA engineer. This review listed all the NCRs that he found to have been invalidated at BLN. The total number was 120. In the opinion of the QA engineer, approximately 47 percent of those nonconformances had been invalidated or voided improperly. The NSRS reviewer performed a review of selected voided NCRs. The identification of the NCRs and the results of that review are detailed in Appendix 1. (Note: The NSRS review included 8 invalidated NCRs that had not been reviewed by the site.)

BNP-QCP-10.4, paragraph 6.7.2 states:

> If the supervisor responsible for approving the disposition determines that further action on the NCR is not warranted, the supervisor shall mark the NCR "INVALID" or "VOID," state the reason, and sign and date the NCR in section 3. All invalid or voided NCRs receive the same approval and distribution as the original.

As shown by Appendix 1, NCRs 177 and 2807 did not state a reason for the invalidation. The reason given for voiding NCRs 2732 and 2733 was that the condition was not nonconforming in accordance with BNP-QCP-10.4 or QAP 15.1. This is a blanket statement with no specifics for invalidating the NCR. NCRs 192, 808, and 2147 were voided because another document, a Field Change Request (FCR), was generated. If problems did not exist, then FCRs would not have needed to be written. NCRs 1508 and 2845 were voided by saying that the items would be reworked. NCR 2698 was invalidated because the problem of grease (lubricant) separation was determined not to be a nonconforming condition since a significant amount of oil had not leaked into the switch compartment. However, no criteria was given as to what constituted a significant amount. It appeared to the NSRS reviewer that NCRs 177, 192, 765, 808, 913, 1508, 2147, 2374, 2539, 2553, 2698, 2732, 2733, 2735, 2807, and 2845 were all improperly voided. Although there is no procedural requirement for interface, interviews with quality control (QC) inspectors indicated that there was no interface between engineering units and QC units before invalidation of NCRs occurred. The QC inspectors related that if the reason for voiding the NCR was unsatisfactory, then the NCR could be taken to the group leader or unit supervisor for discussion. However, no inspectors related any examples of having the invalidated NCRs reinitiated.

NSRS concluded that nonconformance reports were invalidated or voided improperly. This is confirmed both by the review made at BLN and by NSRS. The high percentage of the NCRs that were invalidated or voided improperly in the samples taken by BLN and NSRS indicate that the problem may be widespread. To correct this situation BLN should review all invalidated or voided NCRs to determine if nonconforming conditions still exist. Where nonconforming conditions are identified the site should initiate NCRs and properly correct the nonconformances. The NCR procedure should be revised to provide a detailed explanation of the invalidation process and to require an independent review of all invalidated NCRs. Appropriate action (e.g., training) should be taken to ensure all personnel have a thorough understanding as to what constitutes a valid NCR.

C. Nonconformance Reports Closed Before Corrective Action Completed

In the supporting details for item QP-5.1, the INPO evaluator stated that 35 closed NCRs were randomly selected for review. Of these 35, 10 nonconformance reports had been signed off as having been completed based upon a commitment to take action in the future. For four NCRs, documentation supporting or indicating that the corrective action had been accomplished could not be located. The INPO evaluator did not list any specific nonconformance reports. Therefore, the NSRS reviewer randomly selected completed NCRs for review. Thirty-two nonconformance reports were analyzed in detail. Of the 32 NCRs reviewed, 14 appeared to have been closed in accordance with procedures. Five NCRs involved support problems and were closed by initiation of another document to correct the nonconformance (similar to items 9 and 10). The following paragraphs contain the results of the review for the remaining 13 NCRs all of which involved NCRs which were closed before corrective action was completed or where documentation was not available.

1. NCR 995

Problems: (1) Core flooding tanks A and B could not be installed due to an interference between the lower manway of the tanks and the cross bracing of the tank supports and (2) the attaching bolts between the tank and the supports could not fully engage due to insufficient thread length on the bolts. On December 4, 1978, QCIR 1139 was written to document these problems. The disposition of the QCIR was to initiate an NCR for the first problem and an FCR to correct the second problem. NCR 995 and FCR M-521 were initiated. The QCIR was closed on May 24, 1979. EN DES agreed to the rework disposition submitted by CONST on the NCR and was to revise drawing 1RN0430-X2-19 to reflect the necessary changes. NCR 995 was closed by the site on August 16, 1979. Drawing 1RN0430-X2-19R7, which included the changes made, was issued on November 23, 1979. This action occurred approximately three months after the NCR was closed.

2. NCR 2344

Problem: Embedded plates for supports OWD-MPHG-0028 and 2KC-MPHG-0808 Sheet 1 were not installed per drawings 4AW0824-X2-21 and -30. The problem was initially documented on QCIR 32,564. The QCIR was dispositioned to prepare a NCR and was closed on April 22, 1983. NCR 2344 was written on April 21, 1983, and was dispositioned to use surface-mounted plates in lieu of the embedded plates that were omitted. EN DES agreed with the recommended disposition on May 31, 1983. Drawings 4AW0824-X2-21 R4 and 4AW0824-X7-30 R4 were issued on August 4, 1983. These revised drawings changed the embedded to surface-mounted plates as requested by the NCR. The site closed NCR 2344 on August 25, 1983. However, support drawing 2KC-MPHG-0808 Sheet 1 still shows the support attached to an embedded plate.

3. NCR 2464

Problem: Indications of galling were found on the north key (B&W part number 20-4) and guide of the core support cylinder (INC-MRCT-001B) for the reactor pressure vessel. QCIR 35,164 was initiated on August 2, 1983, and was dispositioned to prepare a NCR. The QCIR was closed on September 23, 1983. NCR 2464 was written August 31, 1983, to document the problem. EN DES and Babcock and Wilcox (B&W) provided the site with repair instruction on December 13, 1983. The NCR was closed on January 24, 1984. Documentation indicating that the corrective action had been accomplished could not be located.

4. NCR 2480

Problem: Various discrepancies on B&W supplied core supports. These discrepancies were documented on QCIR 36,020 dated September 1, 1983. The recommended disposition was to issue an NCR. NCR 2480 was initiated on September 20, 1983, and the QCIR was closed on September 21, 1983. EN DES and B&W provided the site with corrective action. The site closed the NCR on April 17, 1984. Review of records indicated that sequence control chart (SCC) No. INC-W007 had been initiated on March 13, 1984. However, no records were located to indicate that the work had been accomplished.

5. NCR 2564

Problem: Wedge bolts holes for support 1CA-MPHG-0237 R3 were drilled in the wrong location. CONST initiated the NCR on November 25, 1983. The recommended disposition was to use-as-is and to write a support modification request (SMR). BNP-QCP-10.4 required that NCRs dispositioned use-as-is be approved by EN DES. However, the NCR was not reviewed by

EN DES and was closed by the site on December 8, 1983. Per the NCR, SMR 15285 was generated on December 8, 1983. At the time of this review the SMR was still open.

6. NCR 2397

Problem: Elevation west on drawing 1NB-MPHG-0658F R4 should be elevation east. QCIR 33305 identified this problem on May 13, 1983, and was dispositioned to initiate an NCR. The QCIR was closed on June 20, 1983. NCR 2397 was opened on June 10, 1983. The NCR disposition was to rework the drawing by initiating a field modification (FM). The NCR was closed by the site on August 1, 1983. FM 18848 was opened on June 20, 1983, but was not closed until September 13, 1983.

7. NCR 2574

Problem: Seismic support 1NV-MPHG-0642 damaged. NCR 2574 was written to document this problem on November 29, 1983. The recommended disposition was to rework the support by initiating a sequence control chart (SCC). The NCR was closed by the site on February 9, 1984, without EN DES review. SCC 1NV-H1853, which was generated by this NCR, was opened on November 29, 1983. The support was inspected and accepted on March 21, 1984, approximately six weeks after closure of the NCR. During examination of this NCR, the NSRS reviewer was unable to locate support drawing 1NV-MPHG-0642 R3.

8. NCR 2577

Problem: Hanger span violated for hangers OEA-EHNG-43-/1, OEA-EHNG-69-/1, OEA-EHNG-71-/1, OEA-EHNG-68-/1 and OEA-EHNG-70-/1. The NCR was written on November 30, 1983, and the disposition was for EN DES to resolve the problem. On January 23, 1984, EN DES dispositioned NCR 2577 by stating that drawings would be issued showing new hangers for OEA-EHNG-43-/1 and OEA-EHNG-44-/1. The other hangers were to be used-as-is. The site closed the NCR on January 27, 1984. Revision 4 of the affected drawing (5AW0206-EA-1) was issued on March 19, 1984.

9. NCR 2579

Problem: Support load table 3BH0462-WE-01F and support drawing OWE-MPHG-0018F R0 did not agree on elevation location of support. The NCR was opened on November 29, 1983, and dispositioned to initiate an FM. NCR 2579 was closed on February 8, 1984. FM 19760 was issued on December 6, 1983 and at the time of the NSRS review was still open.

10. NCR 2580

Problem: Support load table 3BH0471-RF-148 R2 and support drawing ORF-MPHG-2965 R1 did not agree on elevation location of support. The problem was documented on NCR 2580 November 28, 1983. The recommended disposition was to initiate an FM. The FM number was not recorded on the NCR. NCR 2580 was closed on January 25, 1984. The NSRS identified the FM by reviewing the revision block on the support drawing. By this method it was determined that drawing ORF-MPHG-2965 R2 had been revised as a result of FM 19735. The FM was opened on December 1, 1983, and was closed on March 14, 1984.

11. NCR 2590

Problem: Anchor spacing violation on seismic pipe support 1NV-MPHG-1018. The NCR was written on December 9, 1983 and was dispositioned use-as-is by initiating an anchor spacing variance (ASV). NCR 2590 was closed by the site on December 21, 1983, without EN DES review as required by the nonconformance procedure for use-as-is dispositioned NCRs. Anchor spacing variance H2412 was initiated on December 16, 1983, and was still in the review cycle at the time of the review.

12. NCR 2795

Problem: One of four embedded studs for support anchorage MK9-3 (unit 1) was broken off. On January 25, 1984, an NCR was initiated to document the problem. The site's recommended disposition was to use-as-is. EN DES agreed with the disposition on February 2, 1984, and was to revise drawings 1RN0430-X2-27 and 1RN0433-X2-9 per ECN 2484. The site closed NCR 2795 February 15, 1984. Drawing 1RN0430-X2-27 R7 was issued on May 26, 1983. Drawing 1RN0433-X2-9 R5 was issued on August 16, 1983. At the time of the review neither drawing had been revised per ECN 2484.

13. NCR 2811

Problem: Hanger OYP-MPHG-0011F was not welded per the drawing. The NCR was written on February 7, 1984, and dispositioned to issue a repair card on the hanger so that weld "C" could be completed. The NCR was closed on February 10, 1984. An operation checklist was issued on February 10, 1984; however, the weld was not inspected and accepted until February 21, 1984.

The preceding examples can be divided into three categories: (1) NCR closed on a future commitment, (2) NCR closed without documentation being located, and (3) NCR closed by initiating another document. NCRs 995, 2344, 2577 and 2795 fall into category 1. At the time of the NSRS review all necessary corrective action had been accomplished for NCRs 995 and 2577.

(NOTE: As indicated in the details of these NCRs, the corrective action was not completed prior to closure of the NCRs.) However, drawing revisions required by NCRs 2344 and 2795 had not been issued at the time of the review, although the NCRs had been closed by the site. Nonconformance reports 2464 and 2480 fall into category 2, that is documentation indicating or supporting that the corrective action had been accomplished could not be located. The remaining NCRs (2564, 2377, 2574, 2579, 2580, 2590 and 2811) can be placed in category 3. All these NCRs were dispositioned to generate another document to correct the nonconformance. The nonconforming condition had not been corrected before closure of the NCR. The documents generated by NCRs 2564, 2579, and 2590 were still in the "open" status at the time of the NSRS review. NSRS concluded that this method of closing NCRs was improper because there is no assurance that the generated documents will be processed to completion.

In addition to being identified by the INPO evaluator, the problem of closing NCRs without completing corrective action was documented by deviation 11 of OQA audit COO-A-84-0001. As a part of the reply to this deficiency, BLN CONST stated that it had "taken the position that nonconformances written against drawing discrepancies or hardware discrepancies that require drawing revision as corrective action where the hardware is dispositioned to use-as-is, only require initiation of necessary documents to correct the drawing prior to closure of the nonconformance report." This position is scheduled to be incorporated into BNP-QCP-10.4 R11.

The NSRS conclusion was that NCRs were closed without corrective action to rectify the nonconforming item being completed. The BLN site should document this condition adverse to quality on a nonconformance report and take action (e.g., a sampling program) to determine the magnitude of this problem. To prevent this condition from occurring again, BNP-QCP-10.4 should be revised to preclude closure of NCRs prior to completion of corrective action. Contrary to BLN's position, this revision should also prohibit closing NCRs requiring drawing or procedural changes.

D. Evaluation of "Offsite-Generated NCRs"

In the course of the review, NSRS observed an area in the site nonconformance procedure that had been changed by Addendum 2 to BNP-QCP-10.4 on December 15, 1983. This addendum revised the BLN procedure to conform to the requirements of QAP 15.1 R9 concerning initiation of NCRs. Paragraph 6.2.1.2 of BNP-QCP-10.4 R10 had required that the site generate an NCR to track any "offsite-generated NCR" until closure. By the addendum the requirement was revised to state that a site NCR may be initiated if an evaluation indicated a need to tag or segregate to prevent inadvertent use or installation of nonconforming items. Criterion XV of 10 CFR 50 Appendix B states:

Measures shall be established to control materials, parts, or components which do not conform to requirements in order to prevent their inadvertent use or installation. These measures shall include, as appropriate, procedures for identification, documentation, segregation, disposition, and notification to affected organizations. Nonconforming items shall be reviewed and accepted, rejected, repaired or reworked in accordance with documented procedures.

It appeared to NSRS that the procedural change made to QAP 15.1 and BNP-QCP-10.4 violates upper tier requirements, and the NSRS recommends these procedures be revised to require the site to initiate NCRs to track "offsite-generated NCRs." An example of a problem that could develop with only the requirement to perform an evaluation was given by deviation 6 of OQA audit S-A-84-0001. Details of the deviation indicated that a vendor had shipped approximately 1600 valves to SQN, WBN, HTN, and PBN. After shipment, the vendor notified TVA that the hydrostatic shell tests for the valves had been performed at a pressure lower than required. The valves were not identified or tagged as nonconforming. Subsequently, PBN transferred the valves to BLN. At BLN these valves were not received as nonconforming (because PBN had not identified the valves as nonconforming), thus allowing corrective action proposed by EN DES to be incomplete and not providing proper control over the valves.

E. Potential Problems

1. Management Action/Reaction

After concluding their review of BLN, the INPO team held an exit meeting. This meeting was held to discuss all the findings (weaknesses and good points) with site management.

During the interview process of QC personnel, the NSRS reviewer became aware of the fact that after the INPO review was concluded one of the QC units held a meeting to discuss item QP 5.1. The Assistant Quality Manager and unit supervisor were both present. It was the perception of some inspectors that the meeting was held to determine who, which inspector(s), had voiced concerns to the INPO evaluator. The unit was also told during the meeting that there would be an investigation because of the allegation. It appeared from the interviews with the QC inspectors that the two supervisors used this meeting to express their opinion that the allegations were unwarranted and that the system had been circumvented because someone may have voiced a concern to an outside organization without using onsite channels. When asked about this meeting, the unit supervisor confirmed that the meeting had been held, but he stated that it had been held to reassure the inspectors rather than to chastise.

NSRS finds it a normal and desirable practice to have an organizational unit meeting and discuss a finding regardless of the source of the finding, (i.e., NRC, INPO, NSRS). The meeting should be one to discuss the findings and obtain clarification and understanding, certainly not to identify the individual voicing a concern. This meeting would also be an appropriate time to emphasize the employee concern program and to encourage employees to discuss any concerns with their supervisor or the designated organization at the site to handle employee quality or safety concerns. It was imprudent on the part of the supervisors to hold a meeting and announce there would be an investigation into the allegations and at the same time indicate the allegations were unwarranted. In fact, subsequent investigation both by BLN staff and the findings of this investigation support this contention.

V. PERSONNEL CONTACTED

Abernathy, K. A.	EQC Unit, CONST
Bell, W. C.	EQC Unit, CONST
Black, T. R.	EQC Unit, CONST
Bowlin, T. L.	EQC Unit, CONST
Claiborne, C. M.	IQC Unit, CONST
Coffman, C. O.	IQC Unit, CONST
Cox, P. R.	EQC Unit, CONST
Curry, D. A.	EQC Unit, CONST
Davis, W. M.	EQC Unit, CONST
Dulaney, M. L.	EQC Unit, CONST
Farmer, J. W.	EQC Unit, CONST
Fletcher, M. E.	EQC Unit, CONST
Ford, L. M.	EQC Unit, CONST
Goggans, M. J.	EQC Unit, CONST
Gross, S. W.	IQC Unit, CONST
Hill, J. L.	EQC Unit, CONST
Holder, C. M.	EQC Unit, CONST
Johnson, C. A.	EQC Unit, CONST
Jones, W. A.	EQC Unit, CONST
Killingsworth, D. D.	IQC Unit, CONST
Kindred, J. F.	EQC Unit, CONST
Leeth, W. K.	IQC Unit, CONST
Lott, J. L.	EQC Unit, CONST
Lowe, L. E.	EQC Unit, CONST
Mann, P. C.	Supervisor, Nuclear Licensing Unit, CONST
Martin, R.	Supervisor, EQC Unit, CONST
McCutchen, J. H.	IQC Unit, CONST
Mitchell, J.	EQC Unit, CONST
Nix, A. J.	IQC Unit, CONST
Pankey, T. R.	IQC Unit, CONST
Parde, V. L.	IQC Unit, CONST
Price, S.	IQC Unit, CONST
Richardson, M. R.	Supervisor, IQC Unit, CONST
Sanders, D. A.	EQC Unit, CONST

Smith, J. M.	EQC Unit, CONST
Starcznski, C. E.	IQC Unit, CONST
Thomas, B. J.	Quality Manager, CONST
Thompson, M. B.	EQC Unit, CONST
Torrie, T. B.	IQC Unit, CONST
Yockel, D. E.	EQC Unit, CONST

VI. DOCUMENTS REVIEWED

QAP 15.1, "Reporting and Correcting Nonconformances," R10 (Proposed)

QAP 16.1, "Evacuation of Nonconformances Condition Reports," R4 (Proposed)

QAP 15.1, "Reporting and Correcting Nonconformances," R9 (Addendums 1, 2, and 3), 9/19/83

BNP-QCP-10.4, "Nonconforming Condition Reports," R8 (Addendums 1, 2, and 3), 6/5/80

BNP-QCP-10.4, "Nonconforming Condition Reports," R9, 11/18/82

BNP-QCP-10.4, "Nonconforming Condition Reports," R10 (Addendums 1, 2, and 3), 11/1/83

BNP-QCP-10.26, "Quality Control Investigation Reports," R4 (Superseded by R5), 3/20/81

BNP-QCP-10.29, "Quality Assurance Training Program," R5 (Addendum 1), 8/24/83

BNP-QCP-10.35, "Employee Concerns and Differing Opinions," R2, 12/23/83

BNP-QCP-10.43, "Inspection Rejection Notice," R0 (Addendum 1), 11/1/83

BNP-QCP-10.29, R5, Attachment E, "Bellefonte Nuclear Plant Unit Certification/Training Requirements" for EQC and IQC units

Personnel Certification and Training Program for EQC and IQC Personnel

Memorandum from R. W. Diebler to C. Bonine, Jr., "Office of Quality Assurance Audit Report No. COO-A-84-001, Nonconformance Control and Corrective Action," 12/30/83 (OQA 831230 601)

Memorandum from L. S. Cox to R. W. Diebler, "Bellefonte Nuclear Plant - Office of Qualtiy Assurance Audit Report No. COO-A-84-0001, Nonconformance Control and Corrective Action," 1/26/84 (BLN 840126 303)

Memorandum from L. S. Cox to R. W. Diebler, "Bellefonte Nuclear Plant - Office of Quality Assurance Audit Report No. COO-A-84-0001," 3/9/84 (BLN 840309 302)

Memorandum from R. W. Diebler to L. S. Cox, "Deviation Report Closure - Audit COO-A-84-0001, Nonconformance Control and Corrective Action," 5/9/84 (OQA 84 0509 601)

Memorandum from J. W. Davenport to B. J. Thomas, "Bellefonte Nuclear Plant - INPO Construction Project Evaluation, Finding QP-S.1, Corrective Actions," 3/26/84

Nonconformance Reports - 0117, 0177, 0192, 0438, 0471, 0505, 0548, 0593, 0639, 0738, 0759, 0765, 0808, 0833, 0838, 0913, 0919, 0955, 0991, 1003, 1021, 1132, 1177R1, 1247, 1302, 1378, 1458, 1508, 2024, 2058, 2080, 2084, 2094, 2109, 2147, 2155, 2210, 2269, 2296, 2300, 2344, 2357, 2363, 2369, 2370, 2374, 2374, 2395, 2397, 2412, 2415, 2464, 2478, 2480, 2482, 2526, 2539, 2548, 2553, 2554, 2564, 2574, 2577, 2579, 2580, 2586, 2588, 2589, 2590, 2605, 2607, 2613, 2615, 2617, 2624, 2641, 2674, 2675, 2686, 2698, 2699, 2701, 2710, 2721, 2728, 2732, 2733, 2735, 2737, 2738, 2739, 2751, 2752, 2757, 2761, 2763, 2773, 2775, 2777, 2778, 2779, 2789, 2792, 2795, 2799, 2807, 2811, 2817, 2824, 2830, 2832, 2839, 2840, 2845, 2888, 2908, 2953, 2981, 3013

Miscellaneous quality assurance records

Memorandum from D. R. Bridges to Those listed, "Bellefonte Nuclear Plant - E, I, C&M Units - Late (Tardiness) and Sick Leave Policies," 12/9/83

APPENDIX I

CHART - REVIEW OF INVALIDATED NCRs

NCR Number	Nonconforming Condition	Reason for Invalidation	NSRS Opinion	BLN Opinion
117	Material received without material test report	Not code material	Agree	Agree
177	Pneumatic tests were substituted for hydrostatic tests subassemblies O-KE-01-7 part B and O-KE-01-8 part A	None	Disagree	Disagree
192	Vertical steel dowels (rebars) have been welded	Handled by FCR, no FCR number referenced	Disagree	Agree
438	Tape received without documentation	Not required	Agree	Agree
471	Removed three bearing plates without approval of procedure	Approval of procedure obtained prior to activity being performed	Agree	Agree
505	COC not received on material	Covered on NCR 507	Agree	Agree
639	Crack in grease can for rock anchors	Covered by NCR 24	Agree	Agree
738	Damaged sections (MK 100) for traveling water screens	Duplicate of NCR 759	Agree	Agree
765	Temperature in class B warehouse dropped below minimum	Accomplished by QCIR, no QCIR number referenced	Disagree	Disagree
808	Incorrect reinforcing steel cut	Dispositioned on FCR 0-920	Disagree	Agree
913	Base material cracked	Process specification revised	Disagree	Disagree
1508	Anchor bolt on box 2ED-EJB-26 during torque test	Item was reworked to conform to design specifications	Disagree	Disagree
2147	Flex conduits cannot be installed as required per drawing	The condition can be corrected within scope and requirements of the drawing. FCR E2792	Disagree	No comment

APPENDIX I (Continued)

NCR Number	Nonconforming Condition	Reason for Invalidation	NSRS Opinion	BLN Opinion
2374	Documentation not complete for welding activities	Disposition of 33571 sufficient to document problem	Disagree	No review
2539	Incorrect welds for whip restraint	Deficiency will be dispositioned by EN DES	Disagree	No review
2553	NCR to track EN DES NCR	Not needed since material does not need to be tagged and segregated	Disagree	Agree
2698	Internal grease separation in valves	The separation of lubricant does not become a nonconforming condition unless a significant amount of oil has leaked into the switch compartment.	Disagree	Disagree
2732 2733	Drawings do not address supports for installation of electrical boxes	This condition is not a nonconforming condition in accordance with BNP-QCP-10.4 or QAP 15.1	Disagree	No review
2735	Completed NCR on ASME item not signed by authorized nuclear inspector (ANI)	No further action required. NCR 2561 corrected by revision	Disagree	No review
2807	Carbon steel pipe (ASME) contains pitted indications	None	Disagree	No review
2845	Digital isolator output state does not change when the input's state is changed	Modules to be reworked onsite. Failure is isolated occurrence	Disagree	Disagree

GNS '840801 051

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

INVESTIGATION

NSRS REPORT NO. I-84-12-SQN

SUBJECT: SEQUOYAH NUCLEAR PLANT - INVESTIGATION OF UNIT 1 INCORE INSTRUMENTATION THIMBLE TUBE EJECTION ACCIDENT ON APRIL 19, 1984

DATES OF INVESTIGATION: APRIL 25 THROUGH MAY 18, 1984

INVESTIGATORS: Gerald G. Brantley — GERALD G. BRANTLEY — DATE August 1, 1984

MICHAEL D. WINGO — DATE 8/1/84

APPROVED BY: for MSK — MICHAEL S. KIDD — DATE 8/1/84

MEDS, W5B63 C-K

TABLE OF CONTENTS

LIST OF ATTACHMENTS

1. Trojan Nuclear Plant Flux Thimble Tube Cleanout at Full Power
2. INPO Entry

LIST OF FIGURES

I. SCOPE

This investigation was conducted to identify the causal and event factors that precipitated the ejection of a highly radioactive thimble tube from its respective guide tube and the unit 1 reactor core into an adjacent instrument room containing eight employees. Additionally, an assessment was made of the actions taken to recover the ejected thimble tube, the Office of Nuclear Power (NUC PR) investigation and reporting of the accident, the efforts to determine the operational readiness of the unit for restart and return to service, and long-term planned corrective actions. During the investigation established accident investigation techniques were utilized in obtaining information from personnel interviews, document and record reviews, and accident scene observation.

II. MANAGEMENT SUMMARY

The thimble tube ejection accident subjected eight Sequoyah Nuclear Plant (SQN) employees to hazardous energy sources of water/steam at 545° F and high radiation levels but caused no injuries, and caused no danger to the general public or the environment. Approximately 16.5 man-rem of radiation exposure and 21 days were required to return the unit to its state prior to the accident (30 percent power).

After the accident the SQN operators took appropriate immediate and subsequent actions in accordance with established procedures to classify, mitigate the consequences of the accident, place the affected unit in a safe shutdown condition, and report the events as they occurred. The operator actions and the design of the plant systems prevented uncovering the reactor core and endangering the health and safety of the general public. The operator efforts were enhanced by prompt notification by the workers of the nature of the reactor coolant leak and conditions in the work area.

No physical injuries were reported as a result of the accident. This is attributed to coincidence, luck, and the prompt egress from the work area which was promoted by the increased awareness of some of the radiological hazards of the job. The increased awareness of the workers can be attributed to the actions by the plant health physics staff to question and slow the job down as the radiological hazards increased and the response of the workers to heed the warnings and stop and discuss the safety aspects of the job.

The causal factors that precipitated the accident were determined by NSRS to be associated with allowing the degraded conditions of the thimble tubes to progressively worsen without taking decisive and effective actions to restore the tubes to their fully operational status, an inadequate decisionmaking process to clean the tubes at power, and assignment of the work activity to a plant organization that was normally accustomed to working on the system while shut down, cooled down, and depressurized without providing sufficient information and management involvement. The assignment of a timeframe of less than 48 hours in which to plan and accomplish the job created an atmosphere of urgency as opposed to safety.

The workers were aware that if the job was not accomplished in that timeframe the reactor was going to be shut down and they were working hard to prevent that from happening.

Those factors discussed above promoted the subsequent breakdown in program controls that were established to regulate maintenance activities of this nature. These breakdowns resulted in the direct causal factors of the accident and include the following:

- Inadequate control of the maintenance activity in that planning, job safety analysis, and review phases were not adequate.

- Breakdown in the procedure process in that inappropriate work instructions were proposed, reviewed, approved, used, and violated.

- Inadequate controls over modification of tools used on the system in that tools were modified without performing adequate evaluations and testing to determine the effects on the system.

Indirect causal factors for the accident include the following:

- The ineffectiveness of the Independent Safety Engineering Group (ISEG) in executing their responsibilities for maintaining surveillance of plant maintenance activities to verify that known system deficiencies are identified and corrected.

- Failure to use all available resources for input into the decisionmaking process to do the job with the reactor at power.

There were other observed program weaknesses that were not causal factors for the accident but could have made the consequences of the accident worse or indicate possible program weaknesses. These include the following:

- Noncompliance with the requirements of a Radiation Work Permit (RWP).

- Improper issuance of hold orders.

- Lack of control of egress routes from the work area.

- Inoperative communication equipment.

On a more positive note the recovery effort was well planned and executed using available industry, TVA, and plant resources, approved instructions, and well-informed personnel. Those involved with the planning and execution of the recovery effort made themselves acutely aware of the hazards they were up against and exercised ingenuity in devising special tooling and simulated exercises to keep radiation exposures as low as reasonably achievable.

It should be emphasized that the TVA health physics organization performed well prior to and after the accident and their efforts can be credited with minimizing the possible serious consequences of this accident.

The actions taken to assure that unit 1 was safe for restart involved inspections, repair, and restoration of affected equipment along with special testing and evaluations. These actions were considered appropriate to ensure that the plant was safe for restart when the decision was made to proceed with returning the unit to operation.

TVA accident reporting and investigation requirements were not adhered to after the accident, and an accident investigation performed by NUC PR did not address important causal factors and respective corrective actions. The report submitted to the NRC describing the nature of the accident, causes, and needed corrective actions was misleading and revisions of that report have been recommended.

TVA's and SQN's policies for safety first before schedule and providing a safe work environment for our employees was not properly executed primarily because the plant staff did not take the time to carefully identify and evaluate the hazards of the job. This led to the subsequent breakdown of established program controls intended to prevent an accident of this nature from occurring. Realizing the hazards associated with the recovery, that effort was carefully evaluated, planned, and executed, and made good use of available resources and established program controls.

Management attention should be focused on evaluating and improving the execution of TVA policy and correcting direct and indirect causal factors and other identified program weaknesses of this accident. This was the second undesirable event involving radiation hazards that has occurred at SQN in less than two years, the last more serious than the first, that were precipitated by similar causal factors.

III. CONCLUSIONS AND RECOMMENDATIONS

A. Background

1. I-84-12-SQN-1, Inadequate Corrective Measures to Alleviate the Degraded Condition of the Thimble Tubes

Conclusion

The degraded condition of the thimble tubes had existed for a period of four years prior to the accident. Effective cleaning efforts had not been accomplished nor changes made in the methods prescribed by documented instructions to correct the problem despite the importance of the system.

Responsibilities for the different aspects affecting system operability (operation and maintenance) were dispersed among several organizations with no one central figure responsible or accountable for overall system operability allowing the degraded condition of the system to remain uncorrected (see sections IV.A.4 through IV.A.11 for details).

Recommendation

Responsibility for overall systems operability should be formally assigned to plant engineers and those engineers held accountable for periodically assessing the adequacy of the performance of the systems, the adequacy of instructions affecting the operation, maintenance or testing of the systems and for assuring that problems are promptly identified and corrected in a quality manner. The responsible engineers should be required to keep informed of industry and TVA information relating to the different aspects of the systems and to periodically formally update plant management on the status of the system.

B. The Decisionmaking Process to Clean the Thimble Tubes at Power

1. I-84-12-SQN-2, Inadequate Industry Survey and Feedback to Field Services Group (FSG) Personnel

The industry survey performed by the Engineering Section was limited in scope and appeared to attempt to determine if the thimble tubes could be cleaned at power rather than how they could be cleaned safely. The engineer performing the survey did not use available information sources (INPO), had not read the cleaning instruction, had not cleaned thimble tubes, and did not interface with FSG personnel after the survey (see section IV.B.1 for details).

Recommendation

In the future, work assignments of this nature should be given to those who are knowledgeable of and will be responsible and accountable for the success and safety of the operation to be accomplished. All available information should be identified and used.

2. I-84-12-SQN-3, Inadequate Decisionmaking Process

Conclusion

The decisionmaking process for the conduct of the cleaning of the thimble tubes while at power was less than adequate. The process used to acquire information was inadequate, readily available information sources and input resources were not used, no independent hazard analysis was performed, and the magnitude of the hazards was not realized or identified (see section IV.B.3 for details).

Recommendation

For unique activities plant management should take the time necessary to identify and thoroughly evaluate hazards associated with the activities using readily available inputs and obtaining information from knowledgeable personnel who will be responsible and accountable for the activity to be performed. Techniques such as a systematic hazard analysis methodology to identify and derive an independent assessment of the hazards involved should be used.

C. Assignment of Work Functions and Job Planning Prior to Beginning the Cleaning Operation

1. I-84-12-SQN-4, Assignment of Work Function to the FSG as an Ordinary Work Activity

Conclusion

The supervision, coordination, and execution of the cleaning operation were assigned as if the activity was an ordinary maintenance activity when in reality it was a unique activity with unique hazards identified. The coordinators and workers were unaccustomed to working on the system when the reactor was operating at rated temperature and pressure and with the dose rates that would likely be encountered and had little if any feedback from the industry survey and management discussion process. A sense of urgency was established as the supervisors, coordinators, and workers knew that the work would have to be done or the unit would be brought off the line (see sections IV.C.1 and IV.L for details).

Recommendation

Emphasize to plant management that it is a fundamental responsibility of management to assure that the knowledge and background of workers assigned to work functions is adequate and that sufficient time and information are provided to properly plan and execute the work activity.

2. I-84-12-SQN-5, Selection of an Inappropriate Instruction for the Control of the Work Activity

Conclusion

Special Maintenance Instruction SMI-0-94-1 was a poor quality instruction and inappropriate for the activity to be controlled. However, the instruction was selected during the planning process as the primary procedural control for the cleaning activity apparently because those performing the planning and coordination function were not aware of what quality elements an instruction should contain, the

change process for inadequate instructions, or had a careless attitude toward procedural compliance (see section IV.C.2.b.(1) for details).

Recommendation

Conduct an awareness program to reaffirm supervisor, engineer, and worker knowledge of the importance of procedure controls, compliance with procedural requirements, and the proper change process for inadequate procedures. Emphasize the SQN policy as stated in SQA129, which states that following instructions and taking the time to correct those which are inadequate are methods to achieve nuclear safety.

3. I-84-12-SQN-6, Inadequate Job Safety Analysis and Hazards Assessment

Conclusion

The job safety analysis and hazards assessment program associated with maintenance activities at SQN is inadequate for identifying, evaluating, preventing, and mitigating accidents of this nature. Similar findings had been identified to SQN as causal factors of an inadvertent radiation exposure at SQN in December 1982, but recommendations in that report (I-82-21-SQN) had not been implemented (see sections IV.C.b.2 and IV.O for details).

Recommendation

The job safety analysis program should be upgraded. An effective hazards assessment methodology should be established as a tool to be used to analyze the identified radiological and industrial aspects of the job, the probability of an accident, and the impact on the workers, plant, and the public. Additionally, implement the recommendations of NSRS Report No. I-82-21-SQN.

4. I-84-12-SQN-7, Inadequate Field Quality Engineering (FQE) Review of Maintenance Request (MR) and Referenced Work Instruction

Conclusion

SMI-0-94-1 was referenced and attached to the MR when sent to FQE for review. The poor quality of the instruction was not identified nor was the fact that the instruction could not be used to perform the cleaning activity with the reactor at power. The FQE review process had not been effective in initiating quality improvement of the instruction since its original issuance in July 1981 (see section IV.C.2.c for details).

Recommendation

Improve the quality of the FQE review process of MRs to assure the quality of the referenced work instructions, the proper program controls are identified, and the instructions are appropriate for the activity being performed.

5. I-84-12-SQN-8, Noncompliance With Requirements of RWP No. 01-1-00102

Conclusion

RWP No. 01-1-00102 specified the following requirement: "Verify hold order is in effect on incore probes prior to entering Reactor Building lower compartments and the Annulus." On April 18 and 19 FSG evening and day shift employees and a HP technician entered the reactor building lower compartment while the hold order was not in effect (see sections IV.C.3.a-c for details).

Recommendation

Emphasize to plant employees that compliance with the requirements of RWPs is essential for their own protection.

6. I-84-12-SQN-9, Noncompliance With Requirements of Section 5.1.4 of AI-3, "Clearance Procedures"

Conclusion

Hold Order No. 1 was issued only to the Assistant Shift Engineer (ASE) and not as required by AI-3 to the persons responsible for work being performed in the instrument room between 0220 on April 17 and 0400 on May 1. This is contrary to the requirements of section 5.1.4 of AI-3 (see section IV.C.3.d for details).

Recommendation

As the hold order system is the method used at SQN for the protection of workers, the public, and equipment, strict compliance with the requirements of AI-3 should be emphasized and enforced.

D. Work Activities Related to the Thimble Tube Cleaning Prior to the Incident

1. I-84-12-SQN-10, Modification of Cleaning Tool Base Supports Without Performing a Technical Evaluation or Testing

Conclusion

The cleaning tool base support was modified and a temporary base was constructed and used without a technical evaluation

of the effect on the mechanical seals. No testing was performed before use. Use of the tool and its support was determined during postaccident testing to impose forces of considerable magnitude on the mechanical seals and those forces were found to cause strain sufficient that the thimble tube separated from the mechanical seal (see section IV.D.1.a. and b for details).

Recommendation

Emphasize to the plant staff that changes to tools and equipment affecting work on critical structures, systems, and components (CSSC) can be made only after a thorough technical evaluation has been made on the effect it will have on the system and used only after the modified tool or equipment has tested satisfactorily.

2. I-84-12-SQN-11, Violation of Work Instruction

Conclusion

SMI-0-94-1 clearly stated that the Teleflex-supplied equipment and the instruction were not to be used at power. Using the equipment and instruction for that operation was a violation of work instruction and the unit 1 SQN Technical Specifications. If the responsible engineers had written an adequate procedure appropriate for the activity and that procedure had been Plant Operation Review Committee (PORC) reviewed the result of the cleaning operation may have been different (see section IV.D.2.a for details).

Recommendation

Emphasize to the plant staff that adherence to PORC-reviewed, plant manager-approved plant instructions is mandatory and a requirement of the Technical Specifications and that instructions are controls established to assure nuclear and industrial safety. Periodic assessments of compliance with instructions should be initiated and corrective actions taken to correct weaknesses observed.

3. Health Physics (HP) Technicians Expression of Concern for Radiation Safety of the Job

Conclusion

The health physics technicians providing coverage for the job expressed concern for safety when they realized the potential for high dose rates. They made recommendations that as low as reasonably achievable (ALARA) preplanning should be performed and that further discussions should be conducted with management about the hazards. These recommendations were heeded by the workers and as a result the workers had an increased awareness of the hazards for the

job before entering the containment to commence work on the evening of April 19 (see sections IV.D.2.a and d and IV.D.3 for details).

4. I-84-12-SQN-12, Lack of Control of Egress Capability from Containment

Conclusion

For approximately 30 minutes during the morning of April 19, the inner door of the personnel airlock was made inoperable without the knowledge of some of the workers cleaning the thimble tubes. This would have hindered egress from the room if the mechanical seal had failed at this time. The FSG workers were unaware of the Technical Specification requirements for maintaining containment integrity and that leaving the inner door of the airlock open would enter the unit into a limiting condition for operation. Leaving the inner door open would have hampered rescue efforts if needed (see sections IV.D.2.b. and IV.D.3 for details).

Recommendation

Establish a policy and methodology requiring an evaluation of the effect on work in progress and notification of affected workers as necessary before granting permission to incapacitate egress routes from the reactor building containment. Emphasize to plant managers and workers that working in the reactor building containment involves some risks and controls for containment integrity are established. Identify the risks involved and established controls to the employees.

5. I-84-12-SQN-13, Breakdown in the ALARA Preplanning Program

Conclusion

The responsible supervisor is required to initiate and complete an ALARA preplanning report prior to job commencement. Even though the cleaning job was expected to involve unusually high dose rates, ALARA preplanning was not conducted until the cleaning operation was well underway on the day shift on April 19, and some recommendations made in the Trojan report to reduce the radiation dose to workers were not incorporated in the cleaning instruction or the work process. The responsible supervisor was not involved in the preplanning effort (see section IV.D.2.c for details).

Recommendation

Emphasize to the plant staff that compliance with ALARA preplanning requirements as specified in RCI-10 must be accomplished.

6. I-84-12-SQN-14, Need for Formal Documentation for Upper Plant Management Approval to Work in Radiation Dose Rate Fields Greater than 50 Rem/Hour

Conclusion

There are no requirements for formal documentation for authorization to work in dose rate fields greater than 50 rem/hour (see section IV.D.3 for details).

Recommendation

Establish formal requirements and a method to document authorization to work in dose rate fields greater than 50 rem/hour.

E. The Accident

1. Failure Mode of the Mechanical Seal

Conclusion

Based upon observations of the workers immediately prior to the accident, a kink in the cleaning cable entered the cleaning tool and resulted in more force being exerted by the worker turning the handle. Additional force was transmitted to the mechanical seal resulting in strain of the seal metal allowing separation of the seal and the thimble. When separation occurred, the thimble tube started out of the guide tube immediately. SMI-0-94-1 had no restrictions or warnings on the use of the cleaning tool or the cable to alert the workers to the potential for causing a failed seal (see section IV.E.3 and IV.K for details).

2. Nature of the Leak

Conclusion

The leak occurred as a sudden spray of relatively cool water in the immediate vicinity of the workers (slightly wetting two of the workers) and rapidly developed into a "gusher" type leak flashing to steam above the workers constituting a life threatening hazard (see section IV.E.3 for details).

3. Egress From the Work Area After the Accident

Conclusion

The egress was rapid and orderly with the exception that one HP technician fell over the handrail a distance of approximately seven feet, there was some crowding and pushing at the door, and one worker was late getting into the airlock. The

rapid egress can be attributed to the fact that by the time the workers entered the work area on the evening of April 19 they were acutely aware and alert to some of the hazards associated with the cleaning operation (see sections IV.E.2 and 4 for details). However, had welding in the airlock been in progress, or if the HP technician had been hurt in his fall and required assistance, the potential for catastrophic consequences is evident. NSRS attributes the fact that severe personal injury was not sustained during the accident to coincidence and luck as well as to the heightened sensitivity of the group to the hazardous conditions.

4. Head Counts of Employees

The FSG day shift coordinator had the presence of mind to conduct a head count in the airlock and again immediately after exiting the airlock. Had someone been injured and left behind in the instrument room it is probable that the head count would have initiated immediate rescue efforts and improved the chances for a successful rescue (see section IV.E.4 for details).

5. I-84-12-SQN-15, Availability of Communications Following the Accident

Conclusion

When the workers entered the airlock after the accident, they discovered that the telephone in the airlock was inoperable (see section IV.E.4 for details).

Recommendations

Anytime the telephone is out of service in the airlock, alternate communications methods should be considered and employed. Additionally, availability of communications should be considered during the performance of the job safety analysis and job planning.

6. Reporting of Accident Conditions to the Control Room

Conclusion

Immediately after exiting the airlock the FSG day shift coordinator told the Public Safety Officer controlling access to reactor building containment to notify the control room of what was happening. The officer was unsuccessful in getting through to the control room (reason not determined by NSRS). The coordinator exited the contamination zone immediately and notified the control room operators of the accident and the nature of the leak. This early notification was helpful to the operations staff in properly classifying the degree of the problem (see sections IV.E.4 and IV.F.1 for details).

F. Operator Actions to Mitigate the Accident

1. Immediate and Subsequent Operator Actions

Conclusion

Using the information provided by the FSG coordinator and properly analyzing the system responses, the operations staff classified the nature of the leak and took proper action in accordance with established procedures to shut the unit down, report the accident, and mitigate the leak. Reactor coolant charging capacity compensated for the leak rate. The core was never uncovered even though the leak was nonisolable and no core damage was sustained. Public health and safety were not jeopardized (see section IV.F for details).

G. Initial Actions Taken to Evaluate Conditions in the Instrument Room

1. Establishment of Upper Plant Management Direction and Control of the Recovery Effort

Conclusion

Realizing after the accident that the radiation levels in the instrument room were unusually high, one RWP (RWP No. 02-1-0005) was established to track total radiation dose acquired by the workers during the recovery effort and to establish plant manager control of all activities relating to the recovery effort. Considering the magnitude of the hazards in the room this was an appropriate decision (see section IV.G.2 for details).

H. The Recovery of the Thimble Tube and Actions Taken to Ensure Unit 1 Was Safe to Return to Power

1. Prior NUC PR Planning for Emergency Project Management

Conclusion

NUC PR had issued in November 1983 a procedure to delineate a program for emergency project management that enhances the ability of normal plant forces to ensure that nuclear safety and remaining plant capacity and availability are not affected. The plant manager elected to use the established concept for the recovery effort at SQN. The prior establishment of this concept and its use proved useful and effective during the recovery effort (see sections IV.H.1 and IV.H.2.a and b for details).

2. Effective Use of TVA and Industry Resources

Conclusion

Personnel were brought in from other industry, TVA, and NUC PR organizations to assist in obtaining ideas, planning, oversight, and execution of the recovery effort to ensure that the recovery was conducted in a safe manner and that the radiation doses to the workers involved were kept ALARA. This action proved useful to a successful recovery effort (see section IV.H for details).

3. Use of Ingenuity in the Planning and Execution of the Recovery Effort

Conclusion

The recovery effort of the highly radioactive thimble tube was carefully thought out, evaluated, planned, simulated, practiced, and executed using available resources, appropriate procedures for the activities, and remote handling tools. The radiation dose to individuals involved in the effort was closely monitored, controlled, and was very close to the projected man-rem dose for the job. Personnel involved in the effort demonstrated excellent ingenuity during the recovery effort (see sections IV.H.2.c and d for details).

4. I-84-12-SQN-16, Effective Cleaning of the Thimble Tubes by Nuclear Utilities Services (NUS) Corporation

Conclusion

The method used by NUS as prescribed in SMI-0-94-2 to clean the thimble tubes after the accident was effective in eliminating the material causing the blockage in the thimble tubes. This effectiveness is primarily due to the pressure of the new backflush process (200 psi) versus that of the old method (40 psi) and the controlled application of NEOLUBE as prescribed in SMI-0-94-2 (see sections IV.H.4 and IV.I for details).

Recommendation

Advise Watts Bar Nuclear Plant (WBN) of the effectiveness of the NUS cleaning method over the Teleflex method.

5. I-84-12-SQN-17, Poor Quality Cleaning Procedures and Inadequate PORC Review

Conclusion

As noted in section III.C.2, SMI-0-94-1 was not adequate for its intended use. SMI-0-94-2 was written after the accident

to clean the tubes via the NUS method. It too was a poor quality instruction and could promote accidents of a similar nature in the future. This conclusion is based upon the facts that SMI-0-94-2 had no cautions or warnings to prevent damage to the mechanical seals, no administrative barriers to prevent cleaning the tubes at pressure, no instructions for disassembly and reassembly of the detector drive system, no postmaintenance inspections after cleaning and before pressurizing the reactor, and postmaintenance testing to ensure operability was optional.

Despite the poor quality of the instructions both were recommended for approval by PORC. In these instances, PORC failed to adequately fulfill its responsibilities to the plant manager on these matters relating to nuclear safety (see sections IV.H and IV.N.2 for details).

Recommendation

Evaluate the PORC procedure review process and consider supplementing the review process with expert subcommittees to properly evaluate procedures and advise the plant manager on their adequacy before he approves or disapproves.

Additionally, cancel SMI-0-94-1 and do not use SMI-0-94-2 again until it has been revised to include at least the quality elements listed above. Perform a generic review of all maintenance and special maintenance instructions to ensure adequacy.

6. Inspection, Testing, and Repair of Affected Equipment Before Returning the Unit to Power

Conclusion

The actions taken by SQN to inspect and repair the thimble tubes high pressure seals, evaluate various combinations of SWAGELOK/GYROLOK fitting hardware, and other equipment possibly affected by the accident were appropriate to ensure the unit was safe to return to power (see sections IV.H.6 through IV.H.9 for details).

I. Accident Investigations (Other than NSRS)

1. I-84-12-SQN-18, Noncompliance with Serious Accident Reporting and Accident Scene Preservation Requirements

Conclusion

Corporate and SQN procedures require that serious accidents be reported immediately and that the accident scene be preserved until released by the chairman of an appointed Accident Investigation Team (AIT). The accident was not reported

as a serious accident until approximately three weeks after the accident occurred, nor was the accident scene preserved as restoration of equipment was essentially complete before the accident was reported (see section IV.J.2 for details).

Recommendation

Determine the cause of the noncompliance and take corrective actions as necessary to ensure future compliance with established requirements.

2. I-84-12-SQN-19, Limited NUC PR Accident Investigation

Conclusion

The appointment of the SQN FSG supervisor to the NUC PR investigation team was inappropriate for this investigation as it created a potential conflict of interest. The NUC PR investigation did not address any breakdown of program controls such as job planning, job safety analysis, inadequate procedures, or the nuclear safety and radiological aspects of the accident. Overall the accident investigation performed by NUC PR is considered limited in scope, somewhat misleading, and did not address what NSRS determined to be the nature of the causes of the accident (see section IV.J.2.a for details).

Recommendation

During future accident investigations appropriate personnel should be appointed to eliminate any potential conflict of interest; the investigation should be initiated as soon as possible after the accident as prescribed by established procedures; sufficient time should be allowed for conduct of the investigation; and it should encompass all aspects of the accident including programmatic weaknesses or breakdowns, and nuclear and radiological safety.

Recommendation No. 5 of the NUC PR report should be revised to delete the recommendation that consideration should be given to leaving the inner door open during such activities.

J. Employee Expression of Concerns for Safety

1. I-84-12-SQN-20, Needed Reemphasis on the TVA and SQN Employee Expression of Concerns for Safety and Safety-First Policies

Conclusion

The employees should have but did not relate their increasing concerns for the safety of the job to upper plant management, and an expression of concern for the adequacy of

the design of the new tool support base was not followed up. The workers felt that they had to a complish the job to prevent shutdown of the unit. It is probable that the workers are not acutely aware of TVA's and SQN's policies and their related responsibilities for expression of concerns for safety and safety first before schedule (see section IV.M for details).

Recommendation

Emphasize to all SQN employees that they are actually responsible for voicing their views concerning safety, that these views are valuable management tools to prevent accidents of this nature from happening, and that management is responsible for addressing the views in a satisfactory manner. Emphasize to all supervisors, engineers, and foremen that responsible concerns expressed to them by their employees must be evaluated regardless of how insignificant they may seem. The TVA and SQN safety-first policy should be emphasized to all SQN employees that nuclear safety is the number one SQN objective and that safety first means before schedule and before production.

K. Program Controls Established by SQN Unit 1 Technical Specifications

1. I-84-12-SQN-21, Ineffective SQN ISEG Activities

Conclusion

The SQN ISEG organization had been ineffective in performing the function that was originally intended for the organization. This is due in part to the dual responsibilities for compliance/ISEG activities and lack of true independence from line responsibilities and pressures (see sections IV.N.1 and IV.Q for details).

Recommendation

Reorganize or reassign functions as necessary to provide ISEG personnel adequate independence from line responsibilities and pressures. Additionally, functions should be limited to ISEG-type duties as required by Technical Specifications.

2. I-84-12-SQN-22, Significant Breakdown in the SQN Procedure Process for Maintenance Activities

Conclusion

There is an apparent breakdown in the procedure process at SQN for maintenance activities as PORC reviewed and recommended approval of two poor quality instructions used for

cleaning thimble tubes (one after the accident); the biennial review did not correct poor quality in one instruction; instructions being used were inappropriate for the activities being performed; an instruction was violated; and some engineers and managers interviewed did not seem to understand what quality elements should be in a maintenance instruction, were not aware of the procedure change process, or expressed a careless attitude toward procedure compliance (see section IV.N.2 and 3 for details.)

Recommendation

The procedural process for maintenance activities at SQN should be thoroughly evaluated. Corrective actions including procedure verification should be initiated as necessary to improve the (1) knowledge of those personnel preparing and using procedures of what constitutes an appropriate procedure, the quality elements that should be incorporated into a procedure, and the change process for existing procedures; (2) quality of the PORC and biennial reviews; and (3) compliance with procedures.

L. SQN Licensee Event Report (LER) No. SQRO-50-327/84030

1. I-84-12-SQN-23, Inadequate Reporting of the Event to NRC

Conclusion

The subject LER was misleading in that the true nature of the leak was not described, there was no mention of an inadequate procedure or violation of procedures as causal factors, and the long-term corrective actions are not adequate to correct the true causal factors of the event (see section IV.P for details).

Recommendation

Revise the LER to reflect the true nature of the leak, the adequacy and violation of SMI-0-94-1, and effective long-term corrective action.

IV. DETAILS

A. Background

This accident occurred during the performance of maintenance activities on the unit 1 incore instrumentation system. The following is a description of that system along with a discussion of background information considered pertinent to the accident itself.

1. Incore Instrumentation System Description

This system was designed to measure temperatures and neutron densities at 58 different locations in the reactor core. The process of measuring the neutron density at different locations in the core is referred to as flux mapping. The flux mapping data is used to confirm nuclear design parameters and ascertain that the nuclear fuel is properly loaded and oriented.

a. Neutron Detectors and Drive System (Refer to figure 1 for the basic system schematic)

The neutron instrumentation portion of the system consists of six movable miniature fission chamber detectors (0.188 inches in diameter and 2.1 inches long). Each detector is welded to the end of a 0.188-inch-diameter helical (spiral wound) drive cable. Each detector and cable is inserted into the reactor core by an electric drive unit through interconnecting tubing via path transfer units which direct the detectors to the desired core location through an isolation valve and one of 58 stainless steel tubes known as "thimble tubes." The thimble tubes are terminated at a common header-type device known as the "seal table" (see figure 2) and are physically held stationary against reactor pressure by mechanical seals (SWAGELOK/GYROLOK fittings).

b. Thimble Tubes (Refer to figures 1, 3, and 4)

There are 58 stainless steel thimble tubes each having an outside diameter (od) of 0.300 inch and an inside diameter (id) of 0.201 inch. The last 1.5 inches of each thimble tube at the seal table is expanded from 0.300 inch od to 0.314 inch od to facilitate installation of the mechanical seal. The thimble tubes vary in length between 103 and 117 feet depending upon the distance between their respective position at the seal table and the core to their respective position in the reactor core. The clearances between the detectors and the inside of the thimble tubes is 0.013 inch. The ends of the thimble tubes in the reactor are sealed, the tubes are dry on the inside, and they serve as a reactor coolant system pressure boundary and thus are a "critical system, structure, or component" (CSSC). The tubes are designed for service at 2500 psig. Each thimble tube is individually routed from the seal table to the reactor vessel through its respective guide tube. The configuration of the thimble tubes as designed and installed creates a loop at the lowest portion of the system which is a natural collection or concentration point for any loose substances in the tube.

Approximately 12 to 14 feet of each thimble tube is in place in the reactor core region during normal plant operation. This portion of the tube is normally exposed to an intense neutron flux causing activation of the stainless steel tubing into long-lived radioactive nuclides.

The radiation from these long-lived nuclides caused high dose rates in the instrument room after the thimble tube was ejected during the accident, complicating recovery of the tube.

c. Guide Tubes

The guide tubes are 1 inch od stainless steel and are essentially extensions of the reactor vessel with no isolation valves. The thimble tubes are routed through the guide tubes which extend from the bottom of the reactor vessel through the concrete shielded area to the seal table (see figure 1). The space between the thimble tube and the guide tube contains approximately four gallons of reactor water at reactor pressure. The water in this space is relatively cool rather than at reactor water temperature (~545° F) as it is normally stagnant and there is approximately 100 feet of thimble and guide tube between the seal table and the reactor pressure vessel.

d. Mechanical Seals (Refer to figures 5 and 6)

The thimble tube is held in place at the seal table against reactor pressure (~2250 psig normal operating pressure) by two mechanical seals connected to the guide tube and thimble tube by a SWAGELOK union, ferrules, and nuts. The guide tube is reduced in size from 1 inch od to 0.625 inch od at the seal table and is welded in place at the seal table surface. The end of the thimble tube passes through the end of the guide tube at the seal table.

The high pressure fitting on the thimble tube involved in this accident contained a two-piece GYROLOK ferrule assembly in a SWAGELOK fitting. Figure 6 shows a photograph of a piece of thimble tube and a typical SWAGELOK fitting. Once tightened, the unit compresses the two-piece ferrule assembly against the thimble tube forming a reactor pressure boundary seal and holding the thimble tube in place against reactor pressure within the guide tube. The lower and larger portion of the fitting forms a reactor pressure boundary seal between the guide tube and the SWAGELOK union in a similar fashion.

2. Physical Arrangement of the Incore Instrument System Equipment

This accident occurred inside the lower compartment of the unit 1 reactor containment building in a room called the instrument room. Figure 7A is a top view of the lower compartment of reactor containment showing the instrument room, the relative position of the seal table, personnel airlock, a submarine hatch allowing access into the containment raceway, and a door allowing access to inside the polar crane wall (a wall supporting the polar crane and providing a radiation shield from the radiation produced by the reactor during operation). Figure 7B is a view of the WBN personnel airlock door as viewed from inside the reactor building containment. Figure 8 is an elevation drawing of the reactor building and illustrates the relative position of the seal table to the top of the reactor core. The drawing depicts the location of the raceway below the instrument room.

Figure 9 is an elevation drawing that illustrates the location of the incore instrumentation system equipment in the instrument room. The portion of the system directly over the seal table is on rollers and can be disconnected and rolled back out of the way allowing overhead access to the seal table.

Figure 10 is a top view of the location of the incore instrumentation equipment in the instrumentation room. The neutron detectors can be stored in cavities in the polar crane wall for radiation shielding while personnel are working in the area.

3. Access to the Instrument Room in the Reactor Building Through the Personnel Airlock at Elevation 690 (See figures 7A and 7B)

The personnel airlock is the primary means of entrance and egress to and from the instrument room where the seal table is located. This airlock is normally locked to prevent uncontrolled entry into the containment. Access is administratively controlled by Administrative Instruction AI-8, "Access to Containment." AI-8 establishes requirements that entry into containment will be controlled by the shift engineer with lock and key and strict personnel accountability by a public safety officer who formally tracks personnel entering and leaving containment on a "Containment Entry Checklist."

The personnel airlock is equipped with two doors that close to form a gastight seal. These two doors are interlocked with one another so that during unit operation both doors cannot be opened at the same time thus breeching containment

integrity. Although infrequent, problems have been encountered with these interlocks and personnel have been prevented from exiting containment through that route because one or both doors would not open. On at least one occasion personnel have been caught in the airlock and could not get out without assistance. A telephone is provided in the airlock for communication.

4. Lubrication of the Incore Detectors, Cables, and Thimble Tubes

The lubricant selected for use in the portions of the system involved in this accident was a colloidal graphite alcohol mixture with a product name of "NEOLUBE." This lubricant was approved for use for this application and was selected because of its compatibility with the component constituents, its lubrication properties (described by those interviewed as not being the very best), its resistance to damage from radiation and temperature, and its low neutron activation properties. The lubricant works properly for this application only when used sparingly and properly applied. If used in excess in this environment (high radiation and temperatures), corrosion products from the system (thimble tubes and detector drive cables) mix with the lubricant and cause it to harden and lump resulting in thimble tube blockage.

5. Initial Installation, Cleaning, and Lubrication of Unit 1 Thimble Tubes

The thimble tubes for unit 1 were installed by TVA construction forces using Westinghouse specifications. After the thimble tubes were installed it was observed that they were significantly blocked. The reason for the blockage was not determined by the plant staff but was thought possibly to be caused by improper storage of the thimbles prior to installation causing the buildup of corrosion products or dirt on the inside of the tubes. Teleflex Corporation was contracted to clean the tubes prior to operation. Resistance was met during initial attempts to insert a cleaning cable into the thimble tubes. Copious amounts of NEOLUBE were added to the tubes by Teleflex personnel to facilitate insertion of the cleaning cable. The tubes were then brushed, backflushed, and dried using methods similar to those prescribed in Special Maintenance Instruction SMI-0-94-1 discussed in section IV.A.8 of this report.

During the performance of the "Incore Movable Detectors Preoperational Test W-11.4, Unit 1," in April 1980, blockage was encountered while attempting to insert test cables. Further cleaning efforts by the FSG was conducted along with attempts to "polish" the tubes by driving the test cables in

and out of the tubes at a fast speed. When the unit 2 thimble tubes were installed they, were not blocked and no NEOLUBE was added to the tubes. Problems with thimble tube blockage on that unit have been minimal.

6. Maintenance History of the Incore Instrumentation System Thimble Tubes Prior to the Accident

The detailed history of prior cleaning activities was not determined by NSRS other than it was related to NSRS by plant management that they were not very successful since blockage problems continued to worsen. Prior to the shutdown for the cycle 2 outage, a maintenance request was written in December 1983 to clean all 58 thimble tubes during the outage. However, due to manpower limitations, time restrictions, and low priority only nine thimble tubes were cleaned. The personnel performing the cleaning reported that they had difficulties getting the brush and the backflush tubing to the ends of the thimble tubes due to the severe blockage and restrictions on the use of NEOLUBE in the thimble tubes.

7. NUC PR Requirements Applicable for the Control of Plant Maintenance

NUC PR requirements applicable for providing control over maintenance activities on CSSC equipment were delineated in Part II, Section 2.1, "Plant Maintenance," of the NUC PR Operational Quality Assurance Manual (N-OQAM). This section of the N-OQAM contained the following requirements:

- Paragraph 1.3 - Specified that maintenance on CSSC shall be properly preplanned and performed in accordance with written procedures or documented instructions appropriate to the circumstances.

- Paragraph 3.3.1.3 - Specified that the instructions shall contain requirements for verifying the quality of maintenance or repair and shall include appropriate quantitative or qualitative acceptance criteria.

- Paragraph 3.3.1.4 - Specified that upon completion of maintenance on any item of the CSSC list and before release for service, appropriate testing shall be performed to verify operational acceptability.

- Paragraph 4.4.2 - Specified that if generic problems are suspected, equipment maintenance history files should be consulted to determine the frequency, cause, and mode of previous failures. If evidence indicates that equipment of the same type has performed unsatisfactorily, corrective measures shall be planned and carried out.

8. Special Maintenance Instruction SMI-0-94-1, "RPV Bottom Mounted Instrument Thimble Tubes Cleaning," Issued July 10, 1981

The thimble tube cleaning process consisted of five steps, only three of which were discussed in SMI-0-94-1. SMI-0-94-1 established the primary administrative controls that had been used on past thimble tube cleaning operations at SQN. These steps and controls are discussed below.

a. Thimble Tube Cleaning Steps.

(1) Disconnecting the Overhead Drive Assembly (Not Discussed in SMI-0-94-1). The thimble tubes and interconnecting tubing were disconnected at the SWAGELOK union flare fittings between the high pressure fittings and the isolation valves (see figures 1 and 5). The overhead assembly was then rolled out of the way allowing access to the top of the seal table and the thimble tubes.

(2) Dry Brushing (Refer to figures 11A, 11B, 12A, and 12B). The dry brushing step involved the use of a brush assembly which consisted of a 0.200 inch od brass wire brush welded to a 0.187 inch od carbon steel helical (spiral wound) cable driven by a handcranked drivebox. The brush assembly was provided by the same vendor (Teleflex) that provided the detector drive system. The upper and lower supports for the handcrank device were fabricated by TVA. The lower support was equipped with a 90° base support that fit over a boss on a seal table providing additional stability to the support assembly as the handcrank was turned. The fit of the base support over the bosses for all the thimble tubes was not always secure. The brush assembly was used to "dry" brush each of the thimble tubes to dislodge particles and dried lubricant attached to the thimble tube wall by the scrubbing action of the brush. The brush was driven into the thimble 10 inches for each revolution of the handcrank. The brushing motion was strictly linear without any rotation of the brush.

(3) Demineralized Water Backflush. After the thimble tubes were dry brushed, pressurized water from the plant demineralized water supply system at approximately 40 psi was injected into each of the thimble tubes via a nylon fluid injection tubing (0.156 inch od) inserted into each thimble tube. It was intended that the turbulent waterflow backflushing out through the void between the inside of the thimble tube (0.201 inch id) and the

outside of the injection tubing would carry the particles dislodged by the scrubbing action of the dry brushing step out of the thimble tubes to waste. Backflushing was to continue until water leaving the tube was visually clear. The clearance between the backflush tube and the inside of the thimble tube is 0.045 inch. Note: The NUS system used to backflush the thimbles after the accident used demineralized water at approximately 200 psi.

(4) Drying of the Thimble Tubes. After the demineralized water backflush, the remaining water in each thimble tube was removed by injecting nitrogen or control air through the nylon injection tubing until there was no evidence of moisture in the nitrogen or air backflushing from the tubing.

(5) Reconnecting the Overhead Drive Assembly (Not Discussed in SMI-0-94-1). After the cleaning operation was complete, the interconnecting tubing and the thimble tubes were reconnected at the SWAGELOK union flare fitting.

b. Administrative Controls. The administrative controls for the thimble tube cleaning process as prescribed by SMI-0-94-1 are discussed below.

(1) Precautions and Warnings. SMI-0-94-1 contained cautions and warnings indicating that the cleaning equipment and the instructions were not to be used at power (reactor operating). These limitations were placed in the instruction because of contamination hazards created from the neutron activation of foreign matter in the thimble tubes. The materials removed from the thimble tubes would be extremely radioactive thus subjecting the workers to additional radioactive contamination.

With the reactor shutdown the normal radiation dose rate level in the work area (seal table) was approximately 10 millirem/hour. Since the special maintenance instruction was not to be used during power operations, no warning or cautions were included in the instruction addressing any unique radiation dose rate hazards that would be encountered due to activation of the cleaning equipment (cable and brush). The instruction did not address any special precautions or unique actions that should be taken if the thimble tubes were being cleaned at elevated reactor pressure regardless of

the operating status of the reactor. The instruction did not address any special precautions that should be taken to prevent damage to the mechanical seals when disconnecting the drive system from the thimble tubes at the seal table, during the cleaning operation, or when connecting the drive system back up to the thimble tubes.

(2) Disconnection/Connection Instructions. There were no instructions provided for the disassembly and reassembly of the drive system from and to the thimble tubes at the seal table.

(3) Acceptance Criteria. The instruction contained no acceptance criteria other than section 5.2.E which stated "when all 20 thimbles are clean, as evidenced by continued clear fluid passing through the discharge base assembly, clean the remainder of the thimbles in the same manner." Note: If the backflush was ineffective in removing the loose materials in the tube the water backflushing would appear clean while the loose materials remained in the tubes.

(4) Postmaintenance Inspections. The instruction contained no postmaintenance inspections to verify that the mechanical seals had not been degraded during the cleaning activity.

(5) Postmaintenance Testing. The instruction contained no postmaintenance testing requirements of the thimble tubes to ensure operability after cleaning was performed.

In summary, the methods employed in the past cleaning operations including those during the outage had been ineffective in removing solid matter from the thimbles. This is due in part to the design of the system (minimal clearances between thimble tubes and guide tubes and low point collection of solid matter) and to the backflush method using demineralized water at 40 psig rather than at 200 psig as with the NUS method that was eventually used to adequately clean the system after the accident. The primary controlling document for the activity (SMI-0-94-1) did not promote thoroughness or prevent damage to the system as it contained only a marginally effective acceptance criteria to establish when the thimble tubes were clean, no postmaintenance testing requirements to ensure the thimble tubes were functional before reassembly and use, and no postmaintenance inspections to assure that the mechanical seals could perform their functions against full

reactor pressure. The instruction contained no restrictions on the use of cleaning tools or cleaning cable other than those prohibiting the use of the tools and methods established in the instruction during power operations. Despite the historical ineffectiveness of the cleaning methods no changes had been made to the instruction (and thus the cleaning methods) since its original issuance in July 1981.

9. Plant Restart Testing Program

After a refueling outage the plant restart testing program as defined in Restart Test Instruction RTI-1, "Restart Sequence," revised April 13, 1984, required that a reactor core neutron flux map be performed prior to exceeding 30 percent reactor power. Section 3.3.3.2 of the SQN Technical Specifications required that 75 percent (44 of 58) of the detector thimble tubes must be operable (i.e., capable of passing the detector into the core) in order to perform the flux mapping.

10. Plant Responsibilities for Different Aspects Relating to the Operability of the Incore Instrumentation System

At SQN the incore detector system was operated by operators, nuclear engineers, and shift technical advisors (STAs). The system drive units were maintained by the Electrical Maintenance Section and thimble tube cleaning was performed by the FSG.

The operators of the system were aware that it would be required for the startup testing program but were not involved with the cleaning activities. Those involved with the cleaning activities were not involved with the startup program and were possibly not aware of the importance of the system to that program. There was no apparent central figure who seemed to be cognizant of the system as a whole to recognize and coordinate resolution of problems affecting the system. Efforts to clean the thimble tubes were not effectively accomplished until after the accident when it was recognized that the tubes must be cleaned and cleaned properly to continue the restart of the unit.

11. Unit 1 Operational History After the Startup From the Cycle 2 Refueling Outage

After the 56-day Cycle 2 refueling outage, initial criticality occurred on April 15, 1984. Low power physics testing commenced on April 15 in accordance with RTI-1. With the first attempts to insert the incore detectors into the reactor core for testing purposes, it was noted that the detectors could not be inserted through the required minimum number of thimbles (less than 75 percent of the thimble tubes were operable). Five of the nine thimbles cleaned

during the refueling outage were still inoperable. Engineers and craft personnel from the FSG, the Reactor Engineering Unit (REU), and the Electrical Maintenance Section (EMS) performed testing and maintenance to try to determine if the blockages were unique to certain detector cables and drive units thus indicating problems with the cables and drive units, or if indeed the thimble tubes were blocked. From these testing and maintenance activities it was determined that 23 out of the 58 thimble tubes were blocked, leaving only 60 percent of the tubes operable. By 1700 April 18, the unit had reached 30 percent power and could proceed no further because of the blocked thimbles and the requirements of the restart testing program. Also, problems were being encountered with secondary water chemistry and a leaking power-operated pressurizer relief valve (PORV).

In summary, the unit 1 incore instrumentation system had been in a degraded condition since initial installation, preoperational testing, and subsequent power operations (approximately four years). The cleaning methods employed by the plant personnel as described by SMI-0-94-1 were ineffective in removing the material causing the blockage from the tubes. The cleaning instruction was of poor quality and did not meet the requirements as specified by the N-OQAM. The inadequate instruction was PORC reviewed and plant manager approved but had not been revised since the original issuance in 1981. Despite the importance of the system for the restart testing program to confirm nuclear design parameters and ascertain that the nuclear fuel was properly loaded and oriented and periodic verification of calculated parameters, cleaning of the tubes was given a low priority during the outage. Attempts were made to clean only 9 out of 58, and only 4 of these 9 were successfully cleaned. It is apparent that assigned cognizance responsibility for the overall system operability is less than adequate or improperly executed in that no decisive action was taken to correct the program inadequacies until the degraded condition of the thimble tubes prevented the plant restart process after the refueling outage and the occurrence of the accident. The less than adequate cognizance of system operability was determined by NSRS to be due in part to the dispersion of the assigned responsibilities for operation and maintenance of the system.

For conclusions and recommendations relating to this section, refer to section III.A.1.

B. <u>The Decisionmaking Process to Clean Thimble Tubes at Power</u>

During the restart from the refueling outage, plant management had recognized that if a neutron flux map could not be successfully obtained, the normal restart testing and power escalation

of the unit could not proceed. The Engineering Section Supervisor had discussed cleaning thimble tubes at power with a representative from the Trojan Nuclear Plant during a reactor engineers conference he had attended in the past. Thimble tubes had been cleaned (dry brushed only) at the Trojan Nuclear Plant while operating at 100 percent power. The SQN plant management had a copy of a report of that particular cleaning activity (see attachment 1). This report was a brief outline of the Trojan cleaning operation, contained some recommendations, and related the problem encountered with the high dose rates at the seal table (170 rem/hour maximum and 60 rem/hour average) when the cleaning cable and brush were withdrawn. As a cleaning operation of this nature had been performed at Trojan, SQN upper management directed the Engineering Section Supervisor to perform an industry survey to obtain further knowledge of industry experience in cleaning thimble tubes at power. Additionally, they directed him to inquire about the possibility of acquiring the services of a contractor experienced in thimble tube cleaning to come to the plant and perform the cleaning operation at power. The Engineering Section Supervisor assigned these jobs to the Reactor Engineering Unit (REU) Supervisor who in turn assigned them to two nuclear engineers in his unit. The following are the results of the surveys and inquiries:

1. Industry Survey of Operating Nuclear Plants to Determine Their Experience in Cleaning Thimble Tubes at Power

   During the course of the survey the following nuclear plants were contacted by a SQN nuclear engineer:

   a. Trojan Nuclear Plant. Thimble tubes had been cleaned (dry brushed) at 100 percent power at Trojan in 1979. The major problem encountered during the cleaning operation was the high radiation dose rates (170 rem/hour maximum; 60 rem/hour average) at the seal table when the brush and cable were being withdrawn after they had been inserted into the reactor core. Teleflex, the vendor of the incore instrumentation drive system, assisted Trojan in the brushing operation.

   b. Beaver Valley Nuclear Plant. Beaver Valley had cleaned (dry brushed) thimble tubes at power and did not have any problems. However the cleaning operation was not effective since only one out of six tubes that were blocked was made operable by the dry brushing operation.

   c. Kewaunee Nuclear Plant. Kewaunee did not clean thimbles at power because their technical specifications were not as restrictive as the SQN technical specifications on the use of the incore instrumentation system. They had never had the need arise to clean the thimble tubes at power.

d. North Anna Nuclear Plant. North Anna did not clean thimbles at power because they have a subatmospheric containment of 10 psia which restricted access to containment during power operations.

e. Ginna Nuclear Plant. Ginna had contracted Nuclear Utilities Services (NUS) to clean their thimble tubes in 1978 using a water backflush method while they were shutdown. They hadn't experienced any problems with their thimble tubes since that time.

None of the people contacted at these plants indicated any problems with thimble tube ejections.

The nuclear engineer performing this survey was told which plants to call, had not read the special maintenance instruction (SMI-0-94-1) prior to making the survey, had no experience cleaning thimble tubes, and did not interface with the FSG personnel doing the cleaning after the survey. The information received from this survey was passed on to the Engineering Section Supervisor.

NSRS consulted the INPO "Nuclear Network" for industry experience with thimble tubes. The Network contained an entry made May 3, 1983 concerning incore thimble tube blockage (see attachment 2). The entry indicated that Salem Nuclear Plant had experienced problems with thimble tube blockage over the years at the point where the thimble tubes enter the reactor vessel. To discover the source of the blockage two tubes were removed and a contract awarded to Battelle Columbus Laboratories. The entry stated that the blockages had been successfully removed at Salem with the unit at full power by probing the thimble tubes with a test cable. Salem removed the input tube from a 10-path transfer device and attached a Teleflex hand drive with a cable loaded into it. They then drove the cable to the area of the blockage and pushed it out of the way. They had found it unnecessary to drive the cable into the core region. In fact they took precautions to prevent that from happening so as not to activate the cable to ~ 100 rem/hour. They counted the revolutions of the handcrank and drove the cable to within 6 feet of the core. They then retracted the cable, rotated the 10-path to the next path and repeated the process. The method used by Salem did not subject their workers to high dose rates and did not subject the mechanical seals to any forces greater than those encountered during normal operation. The name and number for a contact at Salem for further information was given. SQN did not consult the INPO Network or talk to Salem during the survey.

The industry survey performed by the Engineering Section was limited in scope in that it did not identify any significant hazards or better methods to perform the cleaning operation

and did not result in any changes to the cleaning instruction to improve the safety and efficiency of the operation. The engineer was told exactly who to call and did not use readily available information sources, had no experience in cleaning the thimble tubes, had not read the cleaning instruction prior to performing the survey, and was not responsible for performing the cleaning operation. The survey appeared to attempt to determine if the thimble tubes could be cleaned at power rather than how they could be cleaned safely.

For conclusions and recommendations relating to this section, refer to section III.B.1.

2. Inquiries of Contractors for Acquiring Services to Clean the Thimble Tubes at Power

During the course of the inquiries the following contractors were contacted by another SQN nuclear engineer:

a. Nuclear Utilities Services (NUS). NUS indicated that their method of cleaning thimble tubes (water flush) was not acceptable for cleaning at power because of temperature considerations (water would flash to steam and injection tubing would melt at 545° F). The NUS procedure did not include dry brushing thimble tubes.

b. Teleflex Corporation. Teleflex indicated that they would not dry brush the thimble tubes at power. They did indicate that they would send a representative from their company to SQN to advise and assist the plant staff during the cleaning operation if they did elect to clean at power. Plant management decided that they had people with sufficient experience in cleaning thimble tubes and thus elected not to acquire the services of the Teleflex adviser. NSRS was informed on May 7 by a representative of Teleflex that they had assisted Trojan with a dry brushing cleaning operation of thimble tubes at power and had decided after that operation not to do it at power again because of the radiation exposure received by their personnel during that operation.

3. Assessment of the Results of the Survey and Inquiries and Risks of the Job

The survey and inquiry information was relayed to the Assistant Plant Manager and on April 18 meetings were conducted to evaluate the results of the survey and to decide whether to clean the tubes or not. Those in attendance and providing input included the following:

- Assistant Plant Manager
- Engineering Section Supervisor
- Electrical Maintenance Supervisor
- Field Services Group Supervisor
- Field Services Group Maintenance Specialist

There were no health physics, safety section, or Independent Safety Engineering Group (ISEG) members present during these meetings. The ISEG organization was aware that the decisionmaking was in progress but that group was not involved.

The following is a summary listing of the pertinent information available to management at that time to support the decision to clean thimble tubes at power:

- The objective of a nuclear power plant is to produce maximum electrical power at the lowest practical cost consistent with maintaining a high degree of nuclear safety.

- The plant could not proceed past 30 percent power because 23 thimble tubes were blocked (9 out of 23 had to be cleared to meet 75 percent required by Technical Specifications).

- Trojan Nuclear Plant had cleaned thimble tubes at 100 percent power reportedly with no problems other than high radiation dose rates (170 rem/hour maximum; 60 rem/hour average).

- SQN had qualified and experienced health physics personnel along with approved radiation control procedures to assist during the cleaning operation and control radiation exposures to ALARA and below any plant limits.

- Plant management had a report from Trojan giving a brief outline of the cleaning method, the results, and containing some recommendations.

- Beaver Valley Nuclear Plant had cleaned thimble tubes at power. Even though 5 out of the 6 tubes cleaned were still blocked after the operation, they reported no problems during the cleaning itself.

- SQN had an established system of procedures that had been reviewed by PORC.

- SQN had an established method (Standard Practice SQM2) for the control of the planning, work instruction preparation, FQE review for quality

assurance criteria, performance of job safety analyses, and work authorization to ensure no Technical Specification criteria were violated (MR process).

- SQN had an established plant operational review committee (PORC) to review any required work instruction to ensure it would not endanger the health and safety of plant personnel, the general public, and safe operation of the plant. PORC would recommend approval or disapproval of the instruction to the plant manager.

- SQN had an ISEG group that routinely reviewed maintenance activities to ensure that unsafe conditions were minimized.

- The plant had a trained and experienced operations crew with approved instructions to handle off-normal situations with plant operations.

- The nature of operating reactor and associated power conversion systems creates the necessity to perform maintenance on systems and components at elevated pressures and temperatures. Maintenance on pressurized systems at temperature can be and had been performed safely with proper planning, good procedures, and trained personnel.

- The probability that a thimble tube would rupture was minimal because of the material and metal thickness.

- SQN had previously performed cleaning operations on the tubes without creating leaks or problems.

- While cleaning the tubes the steam generator water chemistry problems could be stabilized and minimized.

- They had people on the staff who had experience cleaning thimble tubes while the plant was shutdown.

The following is a summary listing of the pertinent information available to management at that time to counter the decision to clean thimble tubes at power:

- SQN cleaning operations including both dry brushing and water backflushing had been only temporarily successful in the past in alleviating the blocked tube problem. Dry brushing was not a permanent fix to the problem.

- Five out of nine tubes cleaned (dry brushed and backflushed) during the outage were still blocked.

- Beaver Valley Nuclear Plant had cleaned (dry brushed) 6 thimble tubes at power and were unsuccessful as 5 out of 6 tubes were still blocked after the operation. Details of their operation were not known.

- Dose rates during the Trojan cleaning operation were 170 rem/hour maximum and 60 rem/hour average. They could expect the same at SQN. These dose rate levels are not encountered during normal plant maintenance activities and could result in higher than normal exposures.

- The Trojan report was brief and did not provide the details of how the cleaning operation at that plant was conducted. There was no real basis to compare the SQN and Trojan operation from a safety review standpoint.

- SQN had no appropriate procedure for performing the work at power.

- Ginna Nuclear Plant had contracted NUS to clean their thimble tubes in 1978 using a water backflush method while they were shutdown and they had not experienced any problems since. This represented a permanent fix.

- NUS indicated that they would not clean thimble tubes at power as their method involved a water backflush method (would flash to steam at reactor operating temperatures of 545° F and the injection pressure of their system 200 psig).

- Teleflex Corporation indicated they would not perform the dry brushing operation at power for TVA but would send an engineer in to advise TVA personnel. Teleflex had assisted Trojan with their cleaning operation at power.

- If a leak occurred in the thimble tube during the dry brushing operation, the leak could not be isolated.

- A thimble tube had been ejected at SQN during the initial cold hydro or hot functional testing probably due to a missing ferrule in a mechanical seal.

- They did not have anyone onsite who had experience cleaning thimble tubes at power operations.

- The job involved some risks to personnel both from the radiological aspects (high dose rates from brush and cable) and from the industrial hazards (working in containment during operation on systems pressurized and at temperature in contamination zone clothing including full face mask for respiratory protection).

- The commitment to maintain the safest work environment practical for employees is inherent in all TVA plant operating philosophy.

- The job involved some risk to the safety of the plant in the event a thimble tube leak occurred.

- Unit 1 had a PORV leaking and it would eventually have to be repaired.

- Unit 1 had problems with steam generator chemistry and they could clean up the water while shutdown.

During the meetings, the results of the industry survey and contractor inquiries and the potential hazards were discussed. The discussion included the increased radiation hazards, the inability to isolate the system should a leak develop through a ruptured thimble tube, and the fact that the work would involve working on a pressurized system (2250 psig) at temperature (545° F). The probability associated with rupturing a thimble tube was considered minimal because of the material of construction (304 stainless steel) and the wall thickness of the tube. The probability that the mechanical seal would leak was not considered because the tubes had been dry brushed before without creating leaks. No one in attendance recognized or discussed the probability that a thimble tube could be ejected in the event something happened to the mechanical seal. Note: A thimble tube had been ejected at SQN during initial hydro testing or hot functional testing of unit 1. Most of the managers interviewed were aware of this event but were unsure of the causes (some thought it was due to a missing ferrule in one of the fittings of the seal table.)

The dry brushing cleaning method was recognized by plant management as only a temporary fix but the goal at this point was not to provide a permanent fix to the problem but only to clear a sufficient number of tubes to facilitate continuing the restart program.

It was considered acceptable to work on a pressurized system at temperature because there are frequent maintenance requirements to do so and it had been done safely before. The primary hazard was considered to be due to the high radiation dose rates that would be encountered at the seal

table during the cleaning activity, but it was felt that the dose rates and worker doses could be controlled by assigning constant health physics coverage during the cleaning activity. Management at this point did not recognize that the procedure was inadequate to perform the work and any potential hazard associated with use of the cleaning tool in promoting failure of the mechanical seal. Management did not recognize that a failed seal could cause a thimble tube to eject. The opinion of those in the meetings was that dry brushing the thimble tubes at power was an accepted industry practice as it had been performed at power at Trojan and Beaver Valley and there were no unusual risks involved in the process other than the high radiation dose rates. With this in mind the decision to clean the thimble tubes was made by the Assistant Plant Manager and the decision was approved by the Plant Manager in the afternoon of April 18. The Plant Manager established that if the thimble tubes were not cleaned by noon on Friday, April 20 that he was going to shut the unit down over the weekend to clean the tubes and resolve the other problems they were encountering (steam generator chemistry and a leaking PORV) during the restart. The weekend was considered a desirable time for the shutdown because of the lighter system load.

In summary, plant management made the decision to clean the tubes with a false sense of security and without the realization or knowledge of the magnitude of the hazards involved. Even though the radiation dose rates were unusually high, the operation involved working on a system pressurized at 2250 psig at 545° F, and the operation was to be conducted inside the reactor containment, the health physics supervision and the plant safety section were not consulted to provide an independent hazard analysis and to get a head start on job planning.

For conclusions and recommendations relating to this section, refer to section III.B.2.

C. Assignment of Work Functions and Job Planning Prior to Beginning the Cleaning Operation

The cleaning operation was assigned and planned as follows:

1. Assignment of Work Functions

The task of dry brushing the thimble tubes was assigned to the FSG as this group had cleaned or coordinated the cleaning of the thimble tubes in the past while the units were shutdown. The assignment was made in the afternoon of April 18 after normal working hours. The FSG mechanical supervisor was notified to make assignments for the cleaning operation. This supervisor had not been involved in the decisionmaking process nor had he interfaced directly with the REU for feedback from the utility survey and contractors

inquiries. He in turn assigned a mechanical engineer on the evening shift the task of planning the preparation for the cleaning operation. (For purposes of this report this individual will be referred to as the "evening shift coordinator.") The evening shift coordinator had never cleaned thimble tubes prior to this assignment. He had been involved in the maintenance and testing activities associated with the incore instrumentation system since the startup on April 15 and had interfaced with the Shift Technical Advisors (STAs) and the nuclear engineers during these activities.

The FSG mechanical supervisor notified a mechanical engineer (for purposes of this report this individual will be referred to as the "day shift coordinator") assigned to the day shift to come in to work at 0315 on April 19 to relieve the evening shift and continue the work of dry brushing the tubes. The day shift coordinator was experienced in thimble tube cleaning as he had been involved in cleaning activities during prior outages. However, his experience was limited to cleaning while the units were shutdown, cooled down, and depressurized.

Management recognized that this was a unique activity as they had identified that the operation involved working on a system at pressure and temperature in the reactor building containment with the reactor operating, if a leak developed it could not be isolated, and the job would involve unusually high dose rates. Management had taken the trouble to have an industry survey performed and had tried to get the activity performed by a contractor. Neither contractor would do the job at power. Discussions concerning the activity had been held involving engineers and plant managers. However, the job assignment was made to the FSG as if the activity was an ordinary maintenance activity in that the supervision and coordination were assigned to a supervisor and engineers who had not participated in the surveys, inquiries, and management discussions, were not aware of the unique hazards, and were normally accustomed to working on systems while the unit was shutdown, cooled down, and depressurized. The routine process to plan and execute the activity was to be used when in reality this was not a routine job. Upper plant management involvement from that time on was minimal. Additionally, a sense of urgency was established as the work was to be planned and performed in less than 48 hours. Planning and work commenced almost immediately on the evening shift, one crew was called in at 0315 for around-the-clock efforts and coordination, and workers knew that the job had to be accomplished or the unit would be shut down.

For conclusions and recommendations relating to this section, refer to section III.C.1.

2. The Maintenace Request Form (MR) for Initiating, Planning, and Controlling the Work Activity

The methodology for managing the initiation, planning, scheduling, and execution of maintenance activities at SQN is depicted in Standard Practice SQM2, "Maintenance Management System," revised April 18, 1984. The primary mechanism for control of these functions is form TVA 6436, "Maintenance Request Form," commonly referred to as the MR.

a. MR Origination. MR A-238084 was initiated by a STA on April 18 and described the work requested as "dry brush blocked thimbles listed below: See attached.** Use no water or NEOLUBE.**" The attachment had 23 thimble tubes listed. The MR was assigned to FSG for planning, scheduling, and execution of the activity. The MR was initialed by the STA's supervisor signifying that the request was needed and that sufficient information had been given to allow FSG to plan the work to be done. The STA supervisor had been involved with the recent maintenance and testing activities of the incore instrumentation system and the industry survey and contractor inquiries.

The priority of the MR was classified as requiring immediate attention indicating that the maintenance activity was to be started expediently. The "Equipment Category" was classified as CSSC by the evening shift coordinator which ensured that the MR would be directed to the plant FQE for a quality assurance (QA) review to ascertain that required QA controls were in place. (QA controls are necessary when working on CSSC to assure that the quality of the system is not degraded by the operation being performed. QA controls include proper work instructions appropriate to the work being performed, qualification of workers, acceptance criteria, postmaintenance inspections, and postmaintenance testing to ensure the system is suitable to return to service.) The MR was forwarded to the FSG evening shift coordinator.

b. MR Planning

(1) Work Instruction. The evening shift coordinator referenced SMI-0-94-1 as the work instruction to be used in the performance of this work activity because that procedure had been used in the past for cleaning activities. He recognized that the instruction stated that the cleaning equipment and the instruction was not to be used at power but thought that the restriction was placed in the instruction to prevent the use of water for backflushing because of the high temperature water

flashing problem. For this reason he added the additional instructions to the MR "dry brush only following applicable sections of SMI-0-94-1." The applicable sections of SMI-0-94-1 were not specified on the MR. A copy of SMI-0-94-1 was attached to the MR. The selection of this instruction was inappropriate as it was a poor quality instruction for the activity to be performed and contained administrative barriers stating that the instruction and the equipment (including dry brushing equipment) were not to be used during power operations (see section IV.A.8 of this report).

The QA or postmaintenance test requirements were specified as "per SMI-0-94-1." SMI-0-94-1 did not contain any postmaintenance test requirements.

When asked how SMI-0-94-1 should have been changed to make it appropriate for the dry brushing cleaning operation at power, managers and engineers interviewed responded that a temporary change to the instruction should have been issued to delete the words concerning not using the instruction or equipment at power. NSRS determined that a temporary change would not be in order as a change of that nature would be an "intent" change and would thus be disallowed by section 6.8.3 of the SQN Technical Specifications. It was apparent that those managers and engineers interviewed were not aware of what quality elements procedures should contain and the procedural change process, or were expressing a careless attitude about procedure compliance. This lack of awareness or careless attitude toward procedural compliance allowed the unique activity to be initiated with inadequate procedural controls.

For conclusions and recommendations relating to this section, refer to section III.C.2.

(2) Job Safety Analysis. The MR was routed to an FSG second shift steamfitter foreman for performance of a job safety analysis. Section B.4 of SQM2 indicated that the responsible first line foreman (or engineer) will review each job for the safety aspects. The review was to include the need for transient fire load considerations, special work permits (replaced by the radiation work permit at SQN), and the need for a hold order. Section B.4 states "The MR supplement form should be used when one or more of the MR supplement (Form 6436D, Figure 2) safety/work control considerations are

required. If any 6436D item is required, Form 6436D should be filled out, signed by the planner, and attached to the MR."

Safety/work safety control considerations on the supplement that were applicable for this work activity included the following:

- Operations Authorization
- Hold Order Clearance
- Special Work Permit (SWP)
- Health Physics Assistance
- Respiratory Protection
- Special Processes

The supplement was not filled out and attached to the MR. Supervisors, engineers, and foremen in the FSG interviewed indicated that these forms were seldom used and attached to MRs. On the MR the foreman wrote the words "perform work safely." This was the statement normally used by the foremen unless there was some special precaution that should be observed.

Guidance provided in Standard Practice SQM2 for performing a job safety analysis addresses only transient fire load considerations, RWPs, hold orders, and special processes. There was little or no guidance for identifying and evaluating the safety hazards (radiological, industrial, and potential impact on safe plant operation) and prescribing unique accident preventive and mitigation measures for the following:

- Working on a system at primary or secondary temperatures and pressures that cannot be isolated, cooled down, and depressurized.

- Identifying unique safety hazards (such as use of improper tools and instructions) that might promote system failures.

- Performing an evaluation of how the job may promote failures of the system or components that might endanger the safety of workers, plant, and the public.

- Work performed in a harsh environment.

- Work in containment while the reactor is at power.

- Designation and control of primary and alternate egress routes during hazardous activities.

- Communications for emergencies.

- Evaluation of work instructions versus system/component hardware to ensure that they are compatible and the instructions contain adequate precautions to prevent degrading the system or component to the point of failure.

- Prejob meetings and briefings with supervisors, engineers, foremen, and crafts to seek out ideas for unique hazard identification, expressing safety concerns, and if concerns are identified, ideas for performing the work safely.

- Involvement of the plant safety engineering group for workplace hazard identification and assessment.

- Involvement of a plant cognizant authority for related industry and plant experience pertaining to the job and the system.

- Guidance on how to openly express any responsible concerns relating to the safety aspects of the job.

- Methodology for a hazards assessment of the identified industrial and radiological aspects of the job for their impact on the workers, the plant, and the public.

In summary, the unique hazards associated with this job were not recognized or adequately addressed in the preplanning phase for the job at the plant management, engineer, first line supervisor, FQE, operator, or craft level. The thought process that went into the safety analysis was not documented on the attachment to the MR as suggested. Interviews with managers and engineers indicated that the attachments were seldom used. The foreman that performed the safety analysis had never cleaned the thimble tubes, had not read the work instruction, and his experience was primarily construction and outage working on systems when the reactor is shutdown and systems are cooled down and depressurized.

In general the job safety analysis and hazards assessment program at SQN is inadequate for identifying and evaluating an operation of this

nature. Similar findings had been previously identified to SQN as causal factors of an inadvertent 10-rem extremity exposure in December 1982 (see section IV.O of this report).

Note: SQN Hazard Control Instruction (HCI) G29, "Workplace Hazard Assessment," establishes a methodology that can be used to evaluate and establish priorities to correct identified hazards. The methodology evaluates such items as proximity to hazardous condition/operation, number of employees exposed to the hazardous conditions, and the length of exposure and uses a point system (1-10) to establish a basis for determining the accident probability (highly likely, predictable, remote) and the hazard severity (catastrophic, serious, minor, negligible).

For conclusions and recommendations relating to this section, refer to section III.C.3.

c. MR Review. The MR was routed to the FQE unit for a QA review to assure the format and controls were in compliance with quality assurance requirements and that the preparation and initial planning guidelines for MRs had been consulted. Guidelines for review of MRs were specified in Appendix C of Standard Practice SQM2 and Quality Engineering Section Instruction Letter No. 5.3, "Maintenance Requests - FQE Section Review," revised January 20, 1984. SQM2, Appendix C guidelines included the following:

- Include appropriate clearance and permits [e.g., hold orders, temporary alterations, SWP (RWP), drilling and chipping permit, etc.]. Note: Hold Order 1 was normally required for any work on the detector drive system of thimble tubes to prevent exposing workers to the highly radioactive incore detectors. RWP No. 92-1-00102 was posted at the entrance to the personnel airlock. Therefore an RWP Timesheet was required to enter the lower compartment of the reactor building.

- Include appropriate controls for special processes (e.g., welding, NDE, hydro, cleaning, protective coating, etc.). Note: Appropriate controls include work instructions appropriate to the special process.

- Determine whether the work is within the skills of qualified maintenance personnel or if detailed instructions need to be included or referenced.

The MR and the attached work instruction had none of the following:

- No indication that a hold order was needed.
- No indication that a SWP (RWP) was required.
- No applicable acceptance criteria.
- No postmaintenance inspections.
- No postmaintenance testing.

Although not followed, the attached work instruction did contain cautions and warnings not to use the Teleflex supplied equipment and SMI-0-94-1 at power.

The MR was reviewed by an FQE engineering associate assigned to the evening shift, signed, and routed to the Operations Section for work authorization. The FQE review failed to identify that the MR and referenced work instruction SMI-0-94-1 (which was attached to the MR) had no indication that a hold order was required, no indication that an RWP was required, no acceptance criteria, no postmaintenance inspection and testing requirements were specified, and the equipment was not to be used at power. During an NSRS interview the FQE unit supervisor indicated that the MR and attached work instruction would probably have been approved even if reviewed by an engineer on day shift.

For conclusions and recommendations relating to this section, refer to section III.C.4.

d. Work Authorization. An assistant shift engineer on the second shift authorized the work in the evening of April 18. This authorization signified that the work would not violate technical specifications.

3. Radiation Work Permit (RWP) and Clearances (Hold Orders)

a. RWP and RWP Timesheets. The RWP is an administrative control used for radiation protection of workers and establishes requirements for entry or work in an area of known or potential radiological hazards. The RWP Timesheet is a subset to the RWP and is used to set protective clothing requirements, list specific instructions, and document personnel entry and exit date, time, and radiation exposure received for specific jobs. The work supervisor initiates the RWP Timesheet after discussion of the work to be performed with the HP representative.

b. Clearance Procedures. The clearance procedure process is the method used at SQN for the protection of workers, the public, and equipment. The shift engineer or designated assistant shift engineer (ASE) are the

only persons authorized to issue a clearance. A clearance is established by the use of protective tags placed so as to indicate the main point of control and the boundary of isolation.

The hold order is a subset of the clearance procedure and is a red tag normally used as a master tag for the clearance. It is usually installed on the main control point to isolate equipment from all sources of energy and to permit work to be safely performed.

Hold Order No. 1, "Unit 1 Incore Probes," is the clearance used to assure that the highly radioactive incore detectors are stored in their storage cavities for radiation protection of personnel working in the reactor building lower compartments and the annulus (which includes the instrument room).

RWP No. 01-1-00102 was issued on January 1, 1984, for the seal table location for the job of "Inspection and Maintenance." The requirements established for entry were included in the RWP. One of the requirements stated "Verify hold order is in effect on incore probes prior to entering reactor building lower compartments and the annulus."

The FSG evening shift coordinator initiated an RWP Timesheet at 2000 on April 18 to "break loose thimble connections @ seal table, remove selector path from seal table, and dry brush blocked thimbles." The protective clothing requirements were specified on the RWP Timesheet.

The RWP Timesheets specified the following "Special Instructions:"

- Obey all instructions on the RWP
- Do not exceed 700 mrem per day
- Sign in and dress out to enter containment
- Do not enter high RAD areas (A high RAD area is an area where the radiation dose rate exceeds 100 mrem/hour.) Note: This special instruction was deleted on April 19 after high dose rates were encountered.
- HP to be present during job
- Protective requirements subject to change at the discretion of HP covering the job

- HP to instruct workers on proper placement of dosimeter, multibadging, and extremities.

c. Hold Order No. 1 Issue and Release Versus Entry and Exit to and From the Instrument Room Before the Accident. A comparison of the issue and release of Hold Order No. 1 versus entry and exit to and from the instrument room is depicted below. All times are Eastern Standard Time (EST).

- At 1910 on April 18 Hold Order No. 1 was released.
- At 2300 on April 18 five FSG evening shift personnel and an HP technician entered the instrument room.
- At 2330 on April 18 Hold Order 1 was issued to the ASE.
- At 0006 on April 19 Hold Order 1 was released.
- At 0020 on April 19 Hold Order 1 was issued to the ASE.
- At 0030 on April 19 Hold Order 1 was released.
- At 0330 on April 19 two FSG day shift employees entered the instrument room.
- At 0430 on April 19 two FSG day shift employees entered the instrument room.
- Between 0435 and 0525 on April 19 all employees exited the instrument room.
- At 0530 on April 19 Hold Order 1 was issued to the ASE.

The hold order was released while workers were in the instrument room to accomodate work being performed by FSG to free two detectors that were stuck in thimble tubes and could not be retracted using the drive units. At 2300 on April 18 and 0330 and 0430 on April 19, FSG and HP personnel entered the instrument room while Hold Order No. 1 was released and not in effect. This represents noncompliance with requirements of RWP 01-1-00102 and the respective RWP Timesheet.

For conclusions and recommendations relating to this section, refer to section III.G.5.

d. Issue of Hold Order to Person Responsible for the Work

Section 5.1.4 of AI-3, "Clearance Procedures," states that "no actual work shall begin on the equipment to be included in the clearance until the clearance has been issued to the person responsible for the work." Between 0220 on April 17 and 0400 on May 1 Hold Order No. 1 was issued only to the ASE while work was in progress in the instrument room before and after the accident. The ASE was not the person responsible for the work. This represents noncompliance with the requirements of section 5.1.4 of AI-3.

For conclusions and recommendations relating to this section, refer to section III.C.6.

D. Work Activities Related to the Thimble Tube Cleaning Prior to the Accident

The following is a discussion of the work activities conducted after the planning process to the time the accident occurred:

1. Work Activities During the Evening of April 18 to Approximately 0830 on April 19

a. Fabrication of New Support for the Cleaning Tool

The dry brushing tool (handcrank) and its support mechanism that had been used in past thimble tube dry brushing operations had been inadvertently discarded in radwaste. A handcrank device had been acquired from WBN. The support for the handcrank was not supplied from the vendor that supplied the dry brushing tool. The FSG second shift coordinator consulted with an FSG maintenance specialist who had been involved with prior thimble tube cleaning activities to determine what type of base support was needed for the new dry brushing tool. It was suggested that a new support device be fabricated somewhat differently than the one that had been used on previous cleanings. The change involved removing the right angle support on the base support (see figures 11A and 11B) to allow the base support to make better contact with the surface of the seal table. The problem with the old tool was that the support did not always fit up well with some of the "bosses" on the seal table and allowed the tool to move around during the turning of the handcrank. Figures 12A and 12B depict the tool and the base support in use when the accident occurred (part of the ejected thimble tube D12 is still attached to the upper portion in figure 12B).

The evening shift coordinator requested the FSG machine shop to fabricate the new base support pieces for the cleaning tool. Note: The new base support pieces were

not finished and used until approximately 1500 on April 19. The day shift coordinator and his crew used the tool with the new base supports and he felt it offered much better support for the tool than the supports that had been used in the past. The steamfitter on the evening shift that was using the tool and support when the accident occurred and had experience with the old support was of the opinion that the new base support was not as good as the old ones used in past cleaning operations. He had expressed some concern about the design of the new support to the evening shift coordinator (see section IV.M.2 of this report).

The change to the tool base support was made without any technical evaluation of its effect on the mechanical seals. The new base support was not tested before use on the thimble tubes.

b. Disassembly of the Incore Instrumentation System Drive Paths and Initial Assembly of Dry Brushing Cleaning Equipment. The evening shift coordinator, one steamfitter foreman, three steamfitters, and an HP technician entered the instrument room through the personnel airlock at 2300 on April 18 (without verifying that Hold Order No. 1 was in effect) and worked until approximately 0430 on April 19 freeing two detectors stuck in their thimble tubing, disassembling the overhead drive paths at the SWAGELOK union flare fitting, and rolling the path transfer units and associated tubing back out of the way allowing access to the seal table. The high pressure fittings were reportedly not disturbed during this process. During this 5.5 hours activity in the instrument room, the maximum whole body radiation dose received (based on pocket dosimeters) was 15 millirem.

At approximately 0315 on April 19 the day shift coordinator, three steamfitters, and a steamfitter foreman reported to work. The day shift coordinator and a steamfitter entered the instrument room at approximately 0330 (without verifying that Hold Order No. 1 was in effect) and worked with the evening shift coordinator and his crew until the evening shift exited the instrument room through the personnel airlock. At approximately 0430 two steamfitters entered the instrument room (without verifying that Hold Order No. 1 was in effect) and the composite day shift crew removed deck grating from above the seal table and assembled the dry brushing equipment. It was noted at this time that there was no base support for the Teleflex-supplied dry brushing tool. The day shift coordinator and the three pipefitters exited the instrument room at approximately

0530 on April 19 to fabricate a temporary base support to be used until the new base support device being fabricated by the machine shop was finished and ready to use. During this two-hour activity in the instrument room, the maximum whole body radiation dose received (based on pocket dosimeters) was 3 millirem.

A temporary base support for the cleaning tool was fabricated out of angle iron. No technical evaluation was performed on this temporary support to assess the effect it would place on the mechanical seals. The temporary base support was not tested before use on the thimble tubes.

At approximately 0800 on April 19 it was announced at the morning meeting normally attended by most plant managers that the decision had been made to clean the thimble tubes at power. No objections were offered or concerns expressed.

For conclusions and recommendations relating to this section, refer to section III.D.1.

2. Work Activities from 0830 on April 19 until Approximately 1700 on April 19

a. Initial Cleaning of Five Thimble Tubes - 0830-1115 April 19. At approximately 0830 the day shift coordinator, a steamfitter, and an HP technician entered the instrument room and began to assemble the cleaning tool with the temporary base support. (At 0945 another steamfitter joined the group.) When the cleaning tool was assembled they connected the tool to the SWAGELOK union flare fitting on one of the tubes identified as blocked on the MR. The cleaning tool was assembled as depicted in figure 12A with the exception that the tool support base was at that time constructed of angle iron. As they had not previously had success with getting the cable and brush through the thimble tubes the workers decided to try a cable without a brush. They ran the cable without the brush into the first tube approximately 85 turns ($\sim$ 70 feet) and encountered severe resistance. They repeated this technique with the other four thimble tubes with the same approximate results. The day shift coordinator at this point thought that probably something was wrong with the cleaning cable. The dose rate on the cable when it came out of the thimble tube was approximately 10-15 mrem/hour at contact.

Note: The cleaning operation at this point had been initiated using SMI-0-94-1 as the primary procedural control for the activity. Section 1.1 of SMI-0-94-1

states "this system is not to be used at power." "This system" is in reference to the thimble cleaner, Teleflex part number 43679 which includes the brushing assembly. Section 4.3.A of SMI-0-94-1 states "This procedure is not to be used while the plant is at power. If cleaning at power is necessary contact Teleflex, Inc." Teleflex was contacted by the plant but they would not clean the tubes at power. Using the Teleflex-supplied equipment and SMI-0-94-1 to perform the cleaning operation at power was a violation of procedure and section 6.8.1.a of the SQN Unit 1 Technical Specifications (see section IV.N.3.a of this report).

The workers stopped the cleaning operation and exited the instrument room via the personnel airlock at 1115 on April 19.

During this 2½-hour activity in the instrument room, the maximum whole body radiation dose received (based on pocket dosimeters) was 22 millirem. The HP technician suggested that before resuming the cleaning operation that ALARA preplanning should be performed. After leaving the instrument room the HP technician covering the job went to the ALARA engineer and discussed the job and recommended that ALARA preplanning be performed. This action by the HP technician initiated the concern for the radiation safety of the job and resulted in an increased awareness of the hazards of the job. It should be noted that the workers and HP technicians did not have an awareness of the hazards to this point in the work process.

For conclusions and recommendations relating to this section, refer to sections III.D.2 and III.D.3.

b. Welding Operation in Personnel Airlock During Work Being Performed in the Instrument Room. Section 3.6.1.3.a of the SQN unit 1 Technical Specifications states that each containment airlock shall be operable with both doors closed except when the airlock is being used for normal transit entry and exit through the containment, then at least one airlock door shall be closed with one containment door inoperable. The operable airlock door is to be maintained closed. At 1050 on April 19 the shift engineer entered unit 1 into a Limiting Condition for Operation (LCO) for section 3.6.1.3 of the Technical Specifications because FSG personnel were welding in the personnel airlock with a welding lead running through the outer door rendering it inoperable because the door could not be shut. The door was made operable at 1121, and unit 1 went out of the LCO. While the

outer door was open the inner door could not have been opened in an emergency because of the interlock which will not allow both doors to be open at the same time. The workers were cleaning thimble tubes at that time and the day shift coordinator was not aware that the outer airlock door was open thus hindering their egress from the area in the event of an emergency.

When the FSG welders requested permission from the shift engineer (SE) to do the work in the airlock, he informed them that people were working in containment and asked them how long it would take them to get their equipment out of the door. They told him that it would take approximately 15 seconds. Some workers did enter and exit while the welders were working. The workers would shake the handle or tap on the door when they wanted out.

For conclusions and recommendations relating to this section, refer to section III.D.4.

c. ALARA Preplanning 1115-1520 on April 19

(1) SQN ALARA Policy. Radiation Control Instruction RCI-10, "Minimizing Occupational Radiation Exposure," revised June 7, 1983, provides policy guidance to management and supervisory staff involved in the operating and maintenance of SQN so that occupational radiation exposures may be kept as low as reasonably achievable. The RCI states that maintaining occupational radiation exposures at the lowest level reasonably achievable requires as a minimum the following:

- Management commitment and support
- Careful design of the facility and equipment
- Good radiation protection practices, including good planning and proper use of appropriate equipment by qualified, well-trained people.

Section VI.C of RCI-10 states that jobs with potentially greater than 5 man-rem exposure (total radiation exposure accumulated by all persons involved in the job) shall require an ALARA preplanning report to be completed by the responsible supervisor. The report is to be submitted to the designated ALARA coordinator for review and approval prior to job commencement.

(2) Processing of Attachment I to RCI-10

At approximately 1130 on April 19 an ALARA HP technician along with a maintenance specialist (not the responsible supervisor) who was knowledgeable of the cleaning process with the reactor shutdown and who had been involved in the decision-making process initiated an Attachment I to RCI-10, "ALARA Preplanning." They calculated that there would be a total of 154 RWP man-hours at a radiation exposure rate of 20 millirem/hour and that the estimated man-rem for the job would be 3.08 rem (whole body dose). The feasible considerations for reducing exposure were as follows:

- Temporary shielding - "Take shielding in - might can be used during job."

- Special tools - "Use of improved drive box mounting device."

- Remote operations - "Use of teletector for survey" Note: A "teletector" is a radiation (X-ray, gamma, high energy beta) dose rate measuring instrument with an extendable detector which provides for increasing the distance between the person making the radiation dose rate measurement and radiation source thus reducing the dose rate to the person.

- Decontamination - "Use of vacuum cleaner with HEPA unit during job to minimize contamination." Note: A HEPA filter is a high efficiency filter for particulate activity (99.97 percent efficient for a 0.3 micron size particle.)

- Remove source - "Special precaution will be used when removing vacuum cleaner from area."

- Improve work instructions - "Reviewed Trojan Nuclear Plant's suggestions from when they did job at 100 percent power."

  Note: The Trojan method used a 10-foot conduit and funnel on the end of the cleaning tool so as to enable the worker turning the handcrank to be positioned above the seal table and away from the high dose rates when the cable and brush came out of the thimble tube and to ease transfer to the other tubes. The Trojan report suggested the use of a

12-foot rigid conduit, a motorized helical drive, and a support platform above the seal table for the helical drive operator. SMI-0-94-1 was not revised to incorporate these revisions nor was the Trojan technique used.

- Additional supervision - "HP and engineer at all times."

- Shift turnover discussion - "Turnover is scheduled."

- Proper Ventilation - "Use of vacuum cleaner with HEPA unit to reduce contamination."

- Reduce reactor power level - "Unit at 30 percent - trying to prevent reactor shutdown."

- Others:

  - "ALARA zone - when not performing work stay in ALARA area - per HP on job."

  - "Hold order - Insure hold order on incore probes " Note: Hold Order No. 1 is the applicable hold order.

Attachment I of RCI-10 was completed sometime after 1200 on April 19. The Trojan report was attached to the completed attachment, and the ALARA preplanning was discussed with the day shift coordinator and the recommendations implemented.

With the expected high dose rates the potential exposure would have been greater than 5 man-rem. However, the ALARA preplanning was only conducted after the job was in progress and after the HP technician expressed concern for the job. The lack of awareness of the potential high dose rates on the part of the FSG coordinators promoted this oversight. The lack of awareness was due to poor transfer of information to the coordinators from those making the decision to do the job at power. The responsible supervisor was not involved in the planning and the suggestions made in the Trojan report were not incorporated. However, even though the total man-rem whole body dose calculated out to be less than 5.0 man-rem (3.08 man-rem) the ALARA preplanning was implemented and the ALARA technician covered the job in addition to the HP technician assigned to the job.

(3) Preparations for Resuming Work in the Instrument Room

After lunch the day shift coordinator and his crew collected the additional equipment needed for implementing the ALARA plan. In addition, he acquired the new base support for the handcrank from the machine shop.

For conclusions and recommendations relating to this section, refer to section III.D.5.

d. Resumption of Work in the Instrument Room 1520-1705, April 19. At 1520 the FSG day shift coordinator, two HP technicians (one was the ALARA technician who had assisted in the ALARA preplanning), and two FSG steamfitters entered the instrument room to resume the cleaning operation. They changed to the new base support for the dry brushing tool. They continued to insert the cable into the blocked thimble tubes with the same lack of success as they had encountered in the morning. On the fourth thimble tube the cable inserted approximately six feet into the reactor core. As they were withdrawing the cable the HP technicians were measuring the dose rate from the cable as it came out. The dose rate started increasing rapidly and at 15 rem/hour the HP technician stopped the withdrawal process. The cable was reinserted into the thimble tube until background dose rates (~10 millirem/hour) were achieved at the seal table. The workers clipped the cable and tied it off so it could be retrieved later after the radiation levels decreased due to the decay of the activation products.

At this point the HP technicians prescribed the use of multidosimeters to ensure that the whole body and extremity radiation dose profile was properly measured. The workers were equipped with the dosimeters at various positions on the whole body (head, trunk, groin, upper legs, etc.) and extremities (forearms, hands, feet, and ankles).

The cable with the brass brush was connected to the dry brushing tool and the tool was connected to another thimble. The brush and cable were inserted into the thimble tube but met resistance during the insertion. The brush and cable entered the core but did not go to the end of the thimble tube. As it was being withdrawn a dose rate of 40 rem/hour was measured. The tool base support was shielded with some lead blankets that had been carried in for that purpose and the cable and brush were withdrawn and inserted into thimble tube

D-12. Note: Subsequent processing of the extremity dosimeters revealed that one steamfitter involved in the transfer of the tool from one thimble tube to the other accrued an extremity dose of 5 rem in the process.

The decision to try thimble tube D-12 was made by the day shift coordinator as he knew D-12 was a thimble tube that had not been identified as blocked and he wanted to determine if the resistance being encountered during insertion of the brush and cable was due to blocked tubes or kinks in the cleaning cable.

The cable brush and cable were inserted into thimble tube D-12 but again not to the end of the tube. As it was being withdrawn the HP technicians stopped the withdrawal when the dose rate increased to 40 rem/hour and instructed the workers to reinsert the brush and cable until the dose rate at the table was approximately background (approximately 15 feet). At this point the HP technicians, the day shift coordinator, and the workers were very concerned with the high dose rates being encountered. The day shift coordinator had not expected and had never worked with dose rates of this magnitude. He and the HP technicians decided that the work should be stopped and discussed with management before continuing. The workers exited the instrument room via the personnel airlock at 1705. The highest radiation whole body dose encountered during this portion of the cleaning operation was 145 millirem as measured by pocket dosimeters.

3. Work Activities from 1700 April 19 to 2120 on April 19

After the workers exited the instrument room, the day shift coordinator and his crew reported the problems they had encountered with the high radiation dose rates to their supervisor (the FSG mechanical supervisor). The HP technician reported the events to the HP shift supervisor. As a result a meeting was scheduled in the FSG office to discuss the progress of the cleaning activity, and the problems being encountered, and to do some further planning to better handle the high radiation dose rates. Those in attendance were the following:

- FSG assistant supervisor
- FSG mechanical supervisor
- FSG day shift coordinator
- FSG evening shift coordinator
- FSG mechanical maintenance specialist
- FSG evening shift mechanical general foreman
- FSG evening shift steamfitter foreman
- HP shift supervisor
- HP ALARA technician

During the meeting safety factors were discussed concerning performing the cleaning operation at full reactor pressure and temperature and the fact that if a leak developed the unit would have to come off the line to stop it. The problems being encountered with the radiation dose rates were addressed at length. Note: The HP group during the meeting reported that one of the steamfitters involved in the cleaning activity during the day had received an extremity dose of 5 rem (quarterly dose limited to the extremities is 18.75 rem as specified in SQN RCI-1, "Radiological Hygiene Program"). The supervisors and personnel in the meeting became very concerned with the safety aspects of the job. The primary concern was the radiation dose rates that were being encountered. The following additional decisions were made to improve the safety aspect of the job:

- After insertion the cables would be withdrawn until the dose rate began to increase, cut and tied off, and kept in the thimble tubes to be removed later after the dose rate had decreased.

- The decision was made to only clean all 10 blocked thimble tubes in C path as they were running short of time. After cleaning these tubes the path transfer units would be hooked back up and the detectors inserted. If all 10 tubes were clear, the flux map could be run as 83 percent of the tubes would be operable.

- The evening shift coordinator was very close to his legally allowable quarterly whole body radiation dose limit (3 rem). The majority of the dose had been received during the Cycle 2 refueling outage. The coordinator was equipped with a radiation dose rate meter to alarm if the dose rate increased. The coordinator was instructed to remain out of the high radiation dose rate areas.

- The inner door on the personnel airlock would be left open to allow for quicker egress in the event a leak developed. Note: The personnel involved were not aware that this would enter the unit into a limiting condition for operation (LCO). Additionally, leaving the door open would have hampered entry into the instrument room because of the interlocks in the event rescue efforts were required.

The ALARA HP technician questioned the advisability of using so many people from FSG (six) for the cleaning activity. He was informed that the additional personnel were necessary to provide additional management oversight for the activity and to provide additional training for this activity to some of the FSG craftsmen.

Section IV.B.6 of RCI-14 requires that the plant superintendent (Plant Manager) review the RWP when the dose rate exceeds 50 rem/hour. The HP shift supervisor notified the Assistant Plant Manager by phone (the Plant Manager had been absent from the plant April 19), the shift engineer, and the plant Assistant HP Supervisor temporarily in charge of unit 1 activities (the plant HP Supervisor was on annual leave) of the dose rate conditions and that it may be necessary to work in a dose rate field of over 50 rem/hour during the cleaning operation. Authorization to continue work was given. The six FSG workers then proceeded to the HP laboratory to pick up the protective equipment to be used during the work activity.

During the course of the work to this point the HP technicians covering the job and the FSG personnel took actions commensurate with the increasing hazards that they had identified. These actions were as follows:

- HP technician suggested work stoppage and ALARA preplanning - FSG responded.
- ALARA implementation even though the calculated total man-rem exposure was less than 5 man-rem.
- Additional ALARA technician coverage during the job (two HP technicians covering the job).
- Health Physics prescribed multidosimeters for measuring whole body radiation dose profile.
- Health Physics suggested work stoppage and further discussions with management about hazards of job - FSG responded.
- ALARA technician questioned the use of so many workers for the job.
- Health Physics shift supervisor responded to concerns when identified and participated in discussion with FSG workers and supervision.
- Health Physics notified upper plant management and shift engineer of increasing dose rates as prescribed by RCI-14 and was given permission to continue the cleaning process. Note: There are no requirements in RCI-14 that formal documentation be made for authorization for working in dose rate fields greater than 50 rem/hour. Legal actions being brought against corporations for radiological matters are increasing. Authorization to work in dose rate fields greater than 50 rem/hour should be formally documented.

The actions of the Health Physics staff and the FSG personnel involved in the cleaning activity to address increasing concerns for the radiological safety aspects of the job stimulated discussions about other safety aspects increasing the worker awareness of some of the hazards involved. When the accident occurred the workers in the instrument room were primed for exit.

For conclusions and recommendations relating to this section, refer to sections III.D.4 and 6.

E. The Accident

The following is a discussion of the worker activities immediately prior to the accident, work area and worker conditions, the accident, and the worker actions immediately after the accident:

1. Worker Activities Immediately Prior to the Accident

Between 2120 and 2145, FSG and HP personnel donned their contamination protective clothing (including face masks for respiratory protection) and radiation dosimeters and entered the instrument room in a staggered fashion (not all at once). An FSG craftsman was stationed outside the airlock to assist the workers inside if needed. A public safety officer was stationed at the outer airlock to control access to the reactor building containment as per AI-3.

The evening shift coordinator was one of the first workers to enter. He marked the thimble tubes that were to be cleaned (C group) with duct tape. At this time he noticed that the cleaning tool was on tube D-12 and that there was a small gap ($\sim$1/2 inch) between the upper and lower portions of the cleaning tool base support. Being aware that the base support had been modified to provide solid support from the cleaning tool to the seal table, he acquired two shims from the FSG worker stationed outside the airlock and shimmed the lower portion of the base support to make contact with the upper portion. As the last of the FSG employees entered the instrument room they shut the inner airlock door out of force of habit. This action was contrary to their contingency planning. At this time there were eight workers in the instrument room. Refer to figures 13A and 13B for their assigned functions and respective positions for the cleaning operation.

2. The Work Area and Worker Alertness

When work was initiated at 2120 on April 19 the work area was well lighted and reasonably uncluttered. The temperature of the work area was reasonably cool. The radiation dose rate in the area around the seal table was approximately 10 millirem/hour. The workers were in contamination

zone clothing with respiratory equipment (coveralls, rubber gloves, plastic booties, shoe covers, surgeon caps, canvas hoods, and full face masks). The workers were reportedly fairly well rested and very alert because of the increased concerns for the safety of the job. When they entered the instrument room, the workers involved were acutely aware of the hazards from the high radiation dose rates being emitted from the cleaning cable and the possibility that in the event of a leak the water would be coming straight from the reactor. The workers cleaning the tubes on the day shift did not have the same level of alertness as they had not had benefit of the same level of concerns and discussions prior to beginning work.

3. The Accident

The workers assembled around and above the seal table as depicted in figures 13A and 13B for performing their assigned tasks. The evening shift coordinator noted that the cleaning tool was on thimble tube D-12 which was not included on the list to be cleaned. The cable was inserted approximately 15 feet into the thimble tube. The coordinator decided that as long as they were connected to thimble tube D-12 they would go ahead and clean it one more time to make sure it was clean. Steamfitter (D) on the cleaning tool turned the handcrank one complete revolution. Coordinator (A) measured the length of insertion to verify that the insertion was 10 inches per one complete revolution. Steamfitter (D) continued to turn the crank and stopped at 50 revolutions and called out the number of revolutions. The number of cranks was verified by steamfitter foreman (C). Steamfitter (D) continued to crank the tool inserting the cable into tube D-12. At approximately 70 cranks a kink was noted in the cleaning cable coming out of the cable container. The workers stopped and examined the kink and decided to proceed. After a total of approximately 79 cranks the cleaning tool offered some resistance to being turned. As the crank started its upward stroke it was noted that additional effort was being required to turn the handcrank. Some movement of the cleaning tool was observed. At this moment the leak occurred spraying water at ambient temperature and slightly wet two of the workers. The cleaning tool pulled loose from the the grasp of steamfitter (D). He reached up, grabbed the tool and pitched it out of his way to the left so he could get out. The water by this time was blowing straight up at a significant rate and was described as hanging up in the overhead. Someone yelled "Let's go."

One of the eight workers (the one farthest from thimble tube D-12) described the first indication of the leak as a bubbling action from around the tool support base. The remaining seven workers assembled around and above D-12

described the leak first as spraying of water from between the upper and lower tool support pieces followed by the leak rapidly developing into a "gusher" blowing straight up and hanging up in the overhead. As there is approximately four gallons of relatively cool water in the guide tube it is apparent that initially the spraying water would not burn the workers. However, after it started blowing straight up at 545° F/2250 psi, it was flashing to steam above the workers and constituted a life threatening hazard.

The seal failed and the leak occurred suddenly with little warning and the tool was pulled away from the worker turning the handcrank. This indicates that the thimble tube started out of the guide tube almost simultaneously with development of the leak.

It is evident that kinks were not uncommon in the cleaning cables as workers looking for kinks were stationed at the point where the cable left its container and that kinks caused problems with the cleaning process in that they were difficult to get through the cleaning tool or insert properly into the thimble tubes. Some of the workers interviewed felt that the extra effort required to turn the handcrank immediately prior to the development of the leak was caused by the kink entering the cleaning tool. SMI-0-94-1 had no restrictions addressing kinks in the cable.

For conclusions relating to this section, refer to sections III.E.1 and III.E.2.

4. <u>Worker Actions Immediately After the Accident</u> (see figures 13A and 13B for exit routes)

Workers (A), (C), (D), (E), (F) and (G) moved hurriedly onto the platform and started down the stairs. HP technician (G) noted HP technician (H) falling backwards towards the handrail. HP technician (H) dropped the teletector he was using to measure dose rates and fell over the handrail, hitting a toolbox on elevation 693. He started running toward the airlock.

When the seven workers reached the airlock, several tried to open the door together. One worker was pushed away by another worker. The door was opened and seven workers entered the airlock. HP technician (G) remembered seeing HP technician (H) falling backwards toward the handrail and became concerned that they had left him behind. He started asking if anyone had seen him. [HP technician (H) was in the airlock.] A head count was conducted by the coordinator (A) and the workers realized they were one worker short. The airlock door was being pulled shut when general foreman (B) stuck his arm in and stopped the door from closing. The

door was opened, general foreman (B) entered the airlock, and the door was closed. The HP technician (G) noted that the dose rate inside the airlock was approximately 200 millirem/hour. The coordinator (A) went to the telephone in the airlock with the intention of calling the control room but noted that the telephone had a MR tag on it indicating it was out of sevice. The time elapsed from the incident until everyone was in the airlock was estimated by the workers to be approximately 20 seconds.

A few seconds prior to the incident the coordinator (A) looked at his dose rate meter and noted that the dose rate was approximately 2 millirem/hour. As he entered the airlock the alarm on the dose rate meter activated and he noted that it indicated 25 millirem/hour.

The outer door of the airlock was opened and the workers exited the airlock. The coordinator (A) yelled instructions to the public safety officer to call the control room and notify them that a leak had developed at the seal table. All workers started surveying themselves for radioactive contamination. The coordinator (A) conducted another head count to ensure that everyone was out of the airlock. The public safety officer was unsuccessful in contacting the control room (reason not determined by NSRS). The coordinator (A) exited the contamination zone, called the control room, and contacted the ASE for unit 1. He informed him that a leak had occurred at the seal table and that it could not be isolated.

The workers removed their protective clothing, surveyed for radioactive contamination (none was detected), and dressed in their personal clothing. The coordinator and the mechanical general foreman proceeded to the control room to inform the operators and the STA of the conditions inside the instrument room. The time was 2215.

The highest radiation dose recorded on the RWP Timesheet was 200 millirem (determined from pocket dosimeters). This dose was received by general foreman (B) who was the last one to enter the personnel airlock.

All workers were subsequently analyzed by whole body count to determine if they had ingested any radioactive materials during the incident. The whole body counts for all eight indicated that no detectable radioactive materials were ingested.

At approximately 0100 on April 20 the FSG evening shift coordinator and the mechanical general foreman submitted written statements of what they had observed before, during, and immediately after the accident.

In summary, the egress from the work area was rapid (20 seconds from when the leak occurred until everyone was in the airlock) and orderly with the exception that the HP ALARA technician was startled to the point that he fell over the handrail by the seal table and there was some crowding and pushing at the door. The general foreman who was located above the seal table was the last to enter the airlock. The day shift coordinator conducted a head count in the airlock and had identified that they were one short. He instructed the public safety officer outside the airlock to count heads again immediately after exiting the airlock. It is probable that the general foreman would not have been left behind because of the head count. As the workers entered the airlock they noted that dose rates were substantially higher than usual. After exiting the airlock the workers recorded their radiation dose received on the RWP Timesheet. The last person out, the general foreman, had received a radiation dose of 200 millirem which is amost twice the dose received by any of the other workers (50-125 millirem). The only action with the cleaning tool and thimble tube immediately prior to the accident was driving the cable and brush into the thimble which reduced the background radiation. The normal background was described as being approximately 10 millirem/hour and the general foreman was in the area for approximately one hour. His radiation dose received prior to the incident should have been 10-20 millirem. The general foreman therefore received approximately 180 millirem in 20 seconds. It is apparent that the thimble tube was out of the guide tube within 20 seconds of the break and before the workers were out of the instrument room.

For conclusions and recommendations relating to this section, refer to sections III.E.3, 4, 5, and 6.

F. Operator Actions to Mitigate the Accident

1. Immediate Operator Action

At 2200 the ASE/SRO on unit 1 was notified by the FSG coordinator that the seal on thimble tube guide D-14 (actually was D-12) at the seal table was severed and a high energy steam blow existed. Concurrently the "Pressurizer Pressure Low - Backup Heaters On" alarm on the unit 1 alarm panel activated. The unit operator noted a decreasing pressurizer water level and increased charging water flow to 130 gallons per minute (gpm) per section III.A. (Immediate Operator Action) of Abnormal Operating Instruction AOI-6, "Small Reactor Coolant System Leak." (A small leak is defined as one for which pressurizer level can be maintained by the charging system and a reactor trip or safety injection does not occur.) Prior to the leak the charging waterflow had

been 85 gpm. At 2215 the pressurizer water level began to increase. The additional charging waterflow required to maintain pressurizer level was approximately 40 gpm.

2. Subsequent Operator Action

At 2217 the SE informed the ASE and unit operator to begin a shutdown of the unit at 1 percent per minute. At 2220 the SE noted in his journal that the leak was a pressure boundary leak and classified the event as an "Unusual Event" in accordance with SQN Radiological Emergency Plan - Implementing Procedure IP-1, "Emergency Plan Classification Logic," because the primary system leak rate was greater than 10 gpm and the source of the leak was identified. The Unusual Event is the emergency classification used by TVA to provide early and prompt notification of minor events which could develop into or be indicative of more serious conditions which are not yet fully realized. The purposes of Notification of Unusual Event are to (1) assure that the first steps in activating emergency organizations have been carried out and (2) provide current information on the event.

At 2220, IP-2, "Notification of Unusual Event" was initiated. IP-2 provides a method for timely notification of appropriate individuals when the SE has determined by IP-1 that an incident has occurred which is classified as an Unusual Event and provides a method for periodic reanalysis of the current situation by the Site Emergency Director to determine whether the Notification of Unusual Event action should be cancelled, continued, or upgraded to a more serious classification.

At 2233 with steam generator level controls in manual and the reactor at 12 percent power, the reactor tripped on low-low level in steam generator No. 1. At 2305 the reactor coolant system was at 500° F and 1900 psig (Hot Standby-Mode 3).

At 0110 on April 20 a surveillance instruction (SI 137.1) was completed and indicated 33.25 gpm leakage from unit 1.

3. Cooldown, Depressurization, and Draining of the Reactor Coolant System (RCS)

Cooldown and depressurization of the RCS continued and at 0508 on April 20 the temperature of the RCS was 350° F (Hot Shutdown-Mode 4).

At 0755 the residual heat removal (RHR) system was initiated and at 1032 the temperature of the RCS was ~200° F (Cold Shutdown-Mode 5). At 1214 the leak rate from unit 1 was

approximately 18 gpm at 250 psig. At 1505 on April 20 the Unusual Event was cancelled as the identified leak rate had decreased below 10 gpm (estimated to be approximately 5.4 gpm at 40 psig).

At 0235 on April 21 the operators started draining the reactor coolant system and at 0815 the water in the reactor vessel was at elevation 701 (one foot below the top of the seal table) and the leakage was essentially stopped.

4. Technical Specification Requirements for Reactor Coolant System Operational Leakage

Section 3.4.6.2 of the SQN unit 1 Technical Specifications states that RCS leakage shall be limited to "no pressure boundary leakage." If a pressure boundary leak develops while the reactor is in Mode 1 (power operation) the reactor is required to be in at least Hot Standby (Mode 3) within six hours and in Cold Shutdown (Mode 5) within the following 30 hours. These actions are considered necessary as pressure boundary leakage of any magnitude is considered unacceptable since it may be an indication of an impending gross failure of the pressure boundary. Therefore, the presence of any pressure boundary leakage requires the unit to be placed in Cold Shutdown.

5. Operator Actions Specified by Abnormal Operating Instruction AOI-6, "Small Reactor Coolant Leak"

AOI-6 is an instruction that provides guidelines for RCS leakage where pressurizer level can be maintained with the charging system and does not increase containment pressure to the point of safety injection (SI) activation. Section IV.B.9, "Subsequent Operator Action" of AOI-6 states that if the pressurizer level stabilizes by additional charging pumps the operator is to determine the leakage source; and if the leak is not identified and isolated, and it is apparent the leak rate is greater than Technical Specification 3.4.6.2 (without running SI-137.1), and a trip will not be generated, the operator is to trip the reactor and proceed to cold shutdown. The source of the leak was identified to the operators by the FSG personnel, therefore a controlled shutdown was initiated.

Using the information provided by the day shift coordinator and properly analyzing the system responses the operations staff classified the nature of the leak and took immediate and subsequent action in accordance with established procedures to shut the unit down, declare an Unusual Event, cool down, depressurize, and drain the water level in the reactor below the seal table elevation thus stopping the leak.

For conclusions relating to this section, refer to section III.F.1.

G. Initial Actions Taken to Evaluate Conditions in the Instrument Room

1. Plant Management Decision to Enter the Instrument Room After the Accident

After the leak was stopped, plant management considered their priorities at that point were the following:

- To find out how much water was in the room.
- To find out the extent of the damage from the water, steam, and radioactive contamination.
- To determine the radiation levels in the room.

They knew that they had the following conditions that would prevent them from returning the unit to operation:

- An ice-bed temperature monitoring system was inoperable.
- A containment sump level transmitter was inoperable.
- A leak at the seal table that had to be repaired.

Plant management at this point did not know that a thimble tube had been ejected. They had reviewed the written statement submitted by the FSG Mechanical General Foreman which stated that before he left the work area immediately after the accident he observed the cleaning cable starting to lay back on the grating at the head of the stairs where he was located. He estimated that approximately 30 feet was laid out when he turned to exit. They assumed that the cleaning cable had been ejected from the thimble tube during the incident and the unusual radiation readings were from the cable.

A radiation survey and some pictures of the area were considered to be the first step necessary to determine the extent of the damage and the radiation levels in the room.

2. Radiation Work Permit (RWP) 02-1-00005

RWP 02-1-0005 was issued April 20, 1984, for the lower containment and seal table to provide radiological controls for all activities related to recovery from the seal table accident and to track total radiation dose acquired by the workers during the recovery effort. The RWP contained an instruction that no entry would be made into the seal table (instrument) room without prior knowledge and approval of the Plant Manager and/or the project supervisor that would be assigned from the Nuclear Central Office (NCO) to direct

the recovery effort. The Plant Manager signed the RWP. This action established upper plant management direction and control of the recovery effort.

For conclusions relating to this section, refer to section III.G.1.

3. Initial Entry After the Accident into the Instrument Room

At 0935 on April 21, four members of the plant health physics staff made the initial entry into the instrument room for the purpose of assessing the damage to the room and to determine the radiation levels.

They found the thimble tube completely ejected from the guide tube and twisted throughout the room. A small amount of water was observed to still be flowing from the fitting for thimble D-12. This water was determined to be flowing from the system because of the pressure exerted by the nitrogen blanket in the pressurizer. The temperature and humidity in the room was very high making conditions difficult for the workers. The radiation dose rates at various locations and a contamination survey taken at one location while the workers were in the room is depicted in figure 14. The initial radiation surveys indicated dose rates of 1-2 rem/hour at the airlock, 300 rem/hr at approximate elevation 708 above and to the right of the seal table and 1000 rem/hour measured 8 inches away from a bend in the ejected thimble tube located at the surface of the seal table. Several pictures were taken of the area. The four individuals were in the area approximately two minutes. The total collective radiation dose received by the four individuals was approximately 3 rem. The highest dose received by one individual was approximately 1.2 rem.

4. Management Assessment of the Conditions Found in the Instrument Room During the Initial Entry

When plant management looked at the pictures taken during the initial entry and evaluated the radiation dose rates measured, they realized that they had a problem of greater magnitude than they had previously thought. They decided that they needed to make another entry and make more detailed pictures using a telephoto lens (to reduce the radiation dose to the photographer) to get as much detail as they could of the ejected tube and a more detailed idea of the condition of the room. They decided that they needed an experienced photographer to take the pictures because of the unusual conditions.

5. The Second Entry into the Instrument Room

Plant management located a photographer at the Power Operations Training Center. When he arrived onsite, he was

briefed by the plant management and Health Physics Staff concerning the conditions in the room and radiological aspects of the work. At approximately 1830 on April 21 the photographer and a HP shift supervisor entered the instrument room and took photographs of the seal table area. They were in the instrument room for approximately seven minutes and received radiation doses of 1.97 rem and 1.94 rem.

6. Preparation of Drawing Depicting the Configuration of the Ejected Thimble Tube

The film was developed and the photographs returned to the plant. From the photographs the plant staff composed a drawing of the thimble tube configuration (see figure 15). An entry into the instrument room was made on April 23 at 1300 by the plant HP section supervisor, an HP shift supervisor, and an HP technician to confirm that the actual configuration was as depicted in the drawing. In addition, contact dose rates were taken at various locations on the ejected thimble tube with a radiation measuring instrument with an extendable radiation detector (see figure 16 for contact dose rates.) They determined that the actual configuration of the thimble tube was in agreement with that depicted in the drawing. The highest radiation dose received (based on high-range dosimeters) during the entry was 0.4 rem.

H. The Recovery of the Thimble Tube and Actions Taken to Ensure Unit 1 was Safe to Return to Power

The following actions were taken by NUC PR to recover the ejected thimble tube and to ensure unit 1 was safe to return to power operation:

1. Assignment of Responsibilities

The Nuclear Production Manager and the SQN Plant Manager assigned a project manager from the NCO to direct the overall effort of recovering and disposing of the ejected thimble tube. This assignment was made in accordance with NUC PR Area Plan Procedure No. 1200A12, "Emergency Project Management."

The Plant Manager made the following additional assignments to the members or organizations of his staff:

- Mechanical Maintenance - Coordinate the preparation and installation of the new thimble tube, examine the affected guide tube for damage, and examine the remaining thimble tube mechanical seals at the seal table for proper installation.

- Electrical Maintenance - Examine and evaluate the electrical equipment in the instrument room and affected areas to determine if any damage occurred and to repair any damage to that equipment.

- Instrument Maintenance - Examine and evaluate the instrumentation in the instrument room and affected areas to determine if any damage occurred and to repair any damage to that equipment.

- Plant Compliance Section - Collect and maintain any information and documents pertaining to the accident to preserve the historical account of the accident.

- Engineering Section - Coordinate the acquisition of NUS Corporation services to clean the thimble tubes.

- Maintenance Superintendent - Coordinate the decontamination efforts of the instrument room.

Additionally, the Plant Manager requested that the NCO Mechanical Branch assist in the examination of the fitting involved in the accident and an assessment of the other fittings on the seal table.

2. Recovery of the Ejected Thimble Tube

a. NUC PR Area Plan Procedure No. 1200A12, "Emergency Project Management". The current revision of the emergency project management procedure was issued in November 1983. The stated purpose of the procedure was to ensure that major components or other emergency maintenance projects receive proper expediting, coordination, procedural compliance, and documentation with the result being maximum efficiency in the use of resources and minimum errors in implementation. The procedure ensures that normal plant forces remain available to perform normal maintenance and ensure that remaining plant capacity and availability are not affected. The procedure is applicable to any major component project of a critical nature with respect to plant availability or nuclear safety.

The activities to be performed by the project manager were to be within the scope of the emergency project management procedure.

b. Project Manager's Initial Interface with Plant Management. At approximately 1200 on April 21, the Manager of Nuclear Production contacted an NCO senior engineer and assigned him as the project manager for the ejected thimble tube recovery from the instrument room at SQN. He was to report directly to the Plant Manager during

the execution of his duties. The assigned project manager immediately proceeded to SQN and at approximately 1400 on April 21 met with the Plant Manager and was briefed on the incident, the activities in progress, and the scope of his assignment.

For conclusions relating to this section, refer to section III.H.1.

c. Planning and Preparation for the Recovery Effort

On April 22, after the configuration of the thimble tube was determined, a meeting was held for the purpose of obtaining ideas for the recovery process. The participation of those at the meeting was reportedly very good. Ideas were discussed and evaluated; and during the afternoon of April 22, the general actions that would be taken to recover the tube were established.

Note: Personnel from NUC PR (Emergency Preparedness and Protection and Mechanical Branches), Office of Power (Radiological Hygiene Staff), and EN DES along with the project manager and the plant staff participated in planning and preparation for the recovery effort. The NRC (site resident and Region II inspectors) observed the planning and preparations.

On the morning of April 23, the project manager began directing the planning and preparation for the recovery effort. These activities were conducted with the goal of developing the safest method of recovering the ejected tube while maintaining the radiation dose to those involved in the process as low as possible. The planning and preparation activities involved the following:

- Made arrangements with WBN to use their unit 1 instrument room to simulate the existing conditions in the SQN instrument room.

- Designed and fabricated special tooling necessary to cut and move the tubing to shielded containers.

- Conducted recovery team trial runs at WBN with simulated conditions and mocked up thimble tubing using the special fabricated tooling.

- Health physics personnel projected the radiation dose for the first phase of the operation (cutting and removing the highly radioactive portion of the thimble tube from the instrument room). The projected dose for this portion of the recovery was 0.6 man-rem.

- Installed temporary shielding at SQN.

- Obtained a remotely operated robot from the Department of Energy (Y-12) to assist in the recovery effort.

- Prepared the following Special Maintenance Instructions incorporating the experiences gained during the WBN exercise and while installing temporary shielding at SQN:

  SMI-1-94-3, "Retrieval of Approximately 25 Feet of Unit 1 D-12 Incore Thimble to Acceptable Work Location," PORC reviewed and Plant Manager approved April 24.

  SMI-1-94-4, "Retrieval of Approximately 100 Feet of Unit 1 D-12 Incore Thimble From U-1 Containment to a Barrel Shield in U-1 El 690 Penetration Room," PORC reviewed and Plant Manager approved on April 25.

- Established maximum stay times for personnel in the instrument room.

- Established emergency personnel response teams in the event of injury or unforeseen circumstances during the tube recovery.

- Established alternate escape routes.

- Established that recovery team members would immediately exit the area if conditions were encountered that were different than those at the simulated WBN exercise.

- Established a communication link between the control point and the Plant Manager's office to allow the Plant Manager to monitor the recovery effort. Provided the link with a tape recorder to record the dialogue of the recovery effort.

- Members of POWER's Radiological Health Staff were onsite and reviewed the procedures and plans to ensure radiation doses to personnel would be ALARA during the recovery.

d. <u>Recovery of the Ejected Thimble Tube and Cleaning Cable</u>

(1) <u>Recovery of the 25-Foot Section With the Highest Radioactive Levels</u>

This portion of the recovery was conducted in accordance with SMI-1-94-3.

Dry runs on the final plans for the operation were conducted at WBN for practice. The recovery team members were briefed on the morning of April 25.

The recovery team leader (an NCO health physicist) entered the personnel airlock on elevation 690 and inspected the instrument room for obstructions with a mirror. (The airlock was shielded.) He noted an air sampler on the stairs by the seal table. The location of the air sampler was made known to the recovery team members that were going to enter the instrument room. The team leader stayed in the airlock to observe the operations with a mirror.

(a) First Entry to Cut the Thimble Tube

The team member designated to cut the thimble (an SQN HP shift supervisor) entered the instrument room through the airlock equipped with a pair of cutters. He proceeded to the stairs leading to the seal table and noted a portion of the tube laying across the railing on the stairs. He immediately exited the instrument room through the airlock as instructed since the tube in that position was unexpected and he was only wearing a surgeon's cap as specified on the applicable RWP Timesheet. He donned a canvas hood which affords better protection of the head and neck against radioactive contamination and reentered the instrument room. He proceeded to the stairs, ducked under the tube, and climbed the access steps to the 10-path trolley elevation and cut off approximately 25 feet of the most radioactive portion of the tube with the cutters. The 25-foot portion of the tube fell exactly as had the mocked-up portion during the practice sessions at WBN. He exited the instrument room through the airlock. During this process he received a radiation dose of approximately 100 millirem.

(b) Second Entry to Attach a Clamping Mechanism to the Thimble Tube and to Pull the Tube Into the Raceway Below the Instrument Room. Team members had been stationed in the raceway to pull the cut portion of the thimble tube into the raceway. One team member placed the clamping mechanism with the cable attached through the submarine hatch on the instrument room floor. Two team members (plant HP shift supervisors) entered the instrument room through the airlocks, picked up the clamp and cable, attached the clamp

and cable to the 25-foot portion of the tube, and immediately left the instrument room through the airlock.

During this process one team member attaching the cable received a radiation dose of approximately 170 millirem and the other member approximately 10 millirem.

All personnel exited the airlock, both airlock doors were closed, and the team members in the raceway pulled the cut portion of the thimble tube from the seal table across the instrument room through the submarine hatch into the raceway. The thimble tube was then pulled to a predetermined location that had been marked on the floor with tape. The team members in the raceway exited the raceway and reactor building containment.

The accumulated radiation dose for all team members involved in this portion of the recovery was 0.7 man-rem.

(2) Recovery of the Remaining Portion of the Thimble Tube and Cleaning Cable from the Instrument Room. This portion of the recovery was conducted in accordance with SMI-1-94-4.

After the most radioactive portion of the thimble tube was in the raceway, the radiation dose rates in the instrument room were lowered substantially. A team leader for this portion of the recovery had been appointed and the team members briefed. On April 25 HP personnel entered the instrument room and located the portion of the remaining thimble tube and cleaning cable with the highest radiation levels. Team member personnel entered the instrument room, cut the most radioactive portions of the remaining thimble tube into 18- to 24-inch sections, placed these cut sections in specially fabricated buckets, and transported the buckets to the airlock. Team members outside the airlock retrieved the buckets and placed them in a barrel shield outside the airlock. These sections of the thimble tube and cleaning cable were transported to radwaste and prepared for shipment to an offsite burial site. This portion of the recovery was completed by 2000 on April 25.

(3) Cutting and Storage of the 25-Foot Section of Thimble Tube in the Raceway. From April 25 to the afternoon of April 26 the following actions were taken to prepare for cutting and storage of the thimble tube in the raceway:

- Erected scaffolding and special shielding and installed lifting devices in the raceway.
- Placed and secured a shielded cask to receive and store the cut sections of tubing in the raceway.
- Moved the robot to the raceway.
- Installed video equipment in the raceway to aid in the cutting and storage operation.
- Designed and fabricated special tools to be used in the cutting operation.

In the afternoon of April 26 a simulation of the cutting and storage operation was conducted, the process finalized and adjustments of tools and equipment were made.

On the morning of April 27, SMI-1-94-6, "Relocation, Cutting, and Storage of 25 to 40 Feet (approximately) of Unit 1 D-12 Incore Thimble," was prepared, PORC reviewed and approved by the maintenance superintendent for the Plant Manager. In the afternoon of April 27 in accordance with SMI-1-94-6 equipment placement and operability were verified, a practice run was completed, and a final briefing was conducted for all team members. The section of thimble was pulled using the cable previously attached around the raceway to a predetermined position for the cutting and storage operation.

With the aid of installed video equipment the team members controlled the robot and the hydraulically operated cutter from a remote location. The robot picked up the thimble tube and transported it to a cutting table. The robot then positioned the thimble tube, and the hydraulic cutter severed approximately 6 feet of the tube believed to have a low radiation level. This section of tubing was then put aside for survey and disposal as low level waste at a later time. The robot then picked up the remaining tubing, positioned the tubing on the cutting table, and the hydraulic cutter severed an approximate 18-inch section. The severed portion of the tubing was then transferred by the robot to the shielded storage cask. The robot then returned to the cutting table and picked up the remaining portion of the thimble tube and repeated the process until all of the

tubing had been cut and placed in the cask (19 cuts were required). The cask was topped off with lead shot for additional shielding and sealed.

The cask containing the highly radioactive portion of the ejected thimble tubing will remain stored in the raceway until removal and disposal at a later date (probably the next refueling outage). The dose rate at the surface of the cask is approximately 6 millirem/hour.

For conclusions relating to this section, refer to sections III.H.2 and 3.

3. Decontamination of the Instrument Room

After the ejected thimble tube and cleaning cables had been removed from the instrument room, preparations were made for decontaminating the surfaces and equipment in the room. An instruction (SMI-1-317-22, "Decontamination of Seal Table and Other Components and Structures Located Inside Incore Instrument Room") was prepared, reviewed by PORC, and approved on April 25. The instruction prescribed the cleaning methods to be used in reducing the radioactive contamination in the room to acceptable levels, disposal methods for cleaning fluids and equipment, and analytical methods and final acceptance criteria for chlorides and boron concentrations on the surfaces of equipment.

Personnel from the FSG and HP groups began removing temporary shielding and commenced the decontamination effort at approximately 2200 on April 25 and completed the effort at approximately 2200 on April 26.

4. NUS Cleaning of Unit 1 Thimble Tubes

SQN contracted NUS Corporation to perform the cleaning operation of the thimble tubes. On April 26 an instruction (SMI-0-94-2, "Incore Flux Thimble Cleaning and Lubrication") was reviewed by PORC and approved for the Plant Manager. This procedure was essentially the NUS-supplied procedure applicable to their method for cleaning and their equipment used in the process. The NUS procedure was changed to the SQN format for special maintenance instructions and changes incorporated to adopt the procedure to specific SQN circumstances and requirements.

The primary steps of the instruction were as follows:

- Flush foreign material from the thimble tube with demineralized water at approximately 200 psig through a flexible tube assembly which is inserted the full length of the thimble.

- Remove the majority of the flush water from the thimble by applying instrument air or nitrogen through the flexible tube assembly.

- Perform a vacuum drying of the thimble tubes to remove all residual moisture.

- Application of a thin film of NEOLUBE lubricant to the thimble bore along the entire thimble length. Note: The lubrication method utilizes a metered fine spray lubricator nozzle which is withdrawn from the thimble at a controlled rate while spraying the lubricant.

- Performance of a final air drying operation to remove the alcohol vehicle from the lubricant and produce a thin uniform film of lubricant for the entire base length.

- Optional performance of a "dummy" test cable insertion of all thimbles to the "dead end" of the thimble to verify no obstructions or problems.

Using the instruction and the NUS equipment, the thimbles were all cleaned by NUS personnel during the timeframe of April 26-April 30. The cost of NUS cleaning operation was approximately $40,000, of which approximately $12,000 was for the purchase of the NUS cleaning system and training TVA personnel on its use.

SMI-0-94-2 was a better quality instruction for the activity to be performed and it is apparent that the method of back-flushing at 200 psi and lubrication with NEOLUBE was effective because after the startup of the unit the blockage in the tubes was removed. However, the instruction still had no cautions or warnings to prevent damage to the mechanical seals, no administrative barrier to prevent cleaning the thimble tubes at pressure, no instructions for disassembly and reassembly of the detector drive system, no postmaintenance inspections after cleaning and before pressurizing the reactor, and optional postmaintenance testing to assure operability is acceptable. For these reasons the new instruction for cleaning the thimble tubes with the NUS equipment is considered a poor quality procedure and should not be used again until it is upgraded.

For conclusions and recommendations relating to this section, refer to sections III.H.4 and 5.

5. Installation of a New Thimble Tube Into Guide Tube D-12
On April 26 an instruction (SMI-1-94-5, "Thimble Tube Installation") was PORC reviewed and approved. Using this instruction a new thimble tube was prepared and inserted into guide tube D-12 on April 28.

6. Inspection of the Seal Table High Pressure Seals

a. Inspection and Results. All of the high pressure seals (fittings) on the seal table were examined for apparent damage or were gauged for proper tightness. During the course of the inspection, 174 high pressure fittings were examined. One fitting was found loose when gauged and 48 fittings were discovered made up with a combination of SWAGELOK and GYROLOK components (SWAGELOK and GYROLOK fitting components are designed for similar applications but manufactured by different companies). The cause of the loose fitting is not known.

b. Testing and Examination of Various Combinations of SWAGELOK and GYROLOK Brands of Fitting Hardware. Various combinations of SWAGELOK and GYROLOK brands of fitting hardware were cross-sectioned and examined by the NCO Mechanical Branch to determine if any combinations would render the assembled fittings unfit for service. The results of the study stated that the various combinations of fittings tested appeared to be satisfactory for the intended service (see reference IV.F.1 for details).

c. Repair of Loose Fitting. SMI-1-94-7, "Seal Table High Pressure Seal Repair," was reviewed by PORC and approved for the Plant Manager on April 30. The loose fitting was repaired in accordance with this instruction.

d. Inspection of Guide Tube D-12 at the Seal Table. The portion of guide tube D-12 at the seal table was visually examined and dye penetrant checked for damage. No damage was discovered.

7. Inspection of the Containment Ice Condenser

Inspection of the containment ice condenser indicated that the ice condenser doors never opened during the accident and steam did not enter the ice beds. Additionally, drain papers inspected were intact which indicated that no ice melted.

8. Inspection of Electrical, Mechanical, and Instrumentation Equipment

All electrical, mechanical, and instrumentation possibly affected by the event were inspected, cleaned, repaired, and recalibrated if necessary.

Note: A telephone located on the polar crane wall and approximately five feet to the right of the seal table was discovered melted and deformed by the heat generated from the leak from guide tubes.

SQN reported in Reportable Occurrence Report SQRO-50-327/8430, an evaluation of all class IE equipment was made to determine if the environmental conditions experienced during this event could be detrimental to their qualified life. The evaluation determined that no deterioration of qualified life was experienced. NSRS did not evaluate this area.

9. NSSS Vendor (Westinghouse) Assessment of Acceptability of the Seal Table for Startup

The plant management requested that Westinghouse perform an assessment of the seal table with the various combinations of SWAGELOK and GYROLOK fittings to determine if the configurations at the seal table were safe to restart the reactor and resume normal operations.

Westinghouse recommended that the reactor could be safely restarted and operated with the existing configuration of the fittings at the seal table for the following reasons:

- The thimble ejection accident occurred during a cleaning operation of the thimble and not during normal operation.
- There was no indication that the thimble ejection was due to mixed fitting components.
- Westinghouse conducted tests at 4250 psi on various fitting combinations with no leakage.
- SQN fitting design is standard and is the same as at many other plants with thousands of hours of operating experience.
- Adequate safeguards exist at SQN to achieve a safe shutdown following ejection of one thimble tube.

For conclusions relating to section IV.H.6 through 9, refer to section III.H.6.

I. Return of SQN Unit 1 to Power Operations

On May 5, unit 1 reached rated temperature and pressure with no problems encountered at the seal table with thimble tubes. The unit was returned to cold shutdown again on May 6 to repair a leaking pressurizer safety valve. The reactor was taken critical and brought to 30 percent power on May 10. Unrelated to seal table repairs, however, the reactor tripped due to low steam generator water level late in the evening on May 10. The reactor was again brought critical on May 11 and the flux mapping testing was successfully completed May 12 and 13. All thimble tubes worked well (no leakage and no evidence of blockage).

A period of 21 days and a man-rem exposure of 16.5 man-rem was required to restore the unit to the operational status (30 percent) that existed prior to the accident.

J. Accident Investigations (Other than NSRS)

1. NRC Inspection Efforts

The NRC performed an announced inspection of the accident onsite in the areas of radiation protection, preplanning and ALARA considerations in the removal of the highly activated incore thimble during April 23-April 28. The inspection involved one inspector.

Per the inspection report the preplanning and consideration for maintaining exposures ALARA were observed by NRC to be adequate for the operation involving the retrieval and storage of the thimble tube.

The NRC site resident inspector observed some of the planning and practice sessions for the thimble tube recovery effort.

Within the scope of the NRC inspections of the accident, no violations or deviations had been identified by the NRC as of June 1, 1984.

2. TVA Investigation Efforts

a. Reporting the Accident and Preservation of the Accident Scene

The TVA "Serious Accident Investigation Procedure" issued in January 1984 requires that in the event of a serious accident the senior management official in charge of the site will follow notification procedures established in his organization.

The procedures are to provide for notification of the Office Manager, the Designated Agency Safety and Health Officer (DASHO), and the Director of the Division of Occupational Health and Safety (OC H&S) as promptly as possible. Definition of a serious accident includes accidental damage to TVA properly with an estimated value of $100,000 or more excluding operating losses.

In the event of a serious accident, an Accident Investigation Team (AIT) is to report to the accident scene no later than the day following the accident. The senior management official in charge of the site where the accident occurred is responsible for securing the accident scene to prevent any disturbance of the evidence

and protect people and property from any hazards associated with the accident until the scene is released by the AIT chairman.

At approximately [illegible] on April 19 an ALARA HP [illegible] of specialist [illegible]

SQN Standard Practice SQS 29, "Accident Reporting and Investigation," revised January 27, 1984, states that during regular work hours, the Plant Manager or the senior plant official present shall report the accident immediately by telephone to the Manager of Nuclear Production. The Manager of Nuclear Production is required to report the accident immediately to the Division Director and the Division Director is required to report the accident within two hours to a designated Office of Power representative. SQS 29 states that the accident scene shall be preserved in the accident configuration until released by the chairman of the AIT.

Notification of the declared Unusual Event was made to the Office Manager's office on April 19 or 20. However, the accident was not immediately reported as a serious accident by plant management in accordance with the TVA procedure as the extent of the damage was not realized until after the initial entries into the instrument room and assessment of the damage had been made. Serious accident notification to the Office Manager, OC H&S, and the DASHO was not made until approximately three weeks after the accident occurred and an investigation had been conducted by NUC PR.

The accident scene was not preserved by the Plant Manager as required by TVA and SQN procedures in that restoration of the area was completed before the serious accident notification was made.

The failure to promptly report the accident as a serious accident after the extent of the damage was realized and the failure to preserve the accident scene represents noncompliance with SQN and TVA procedures.

For conclusions and recommendations relating to this section, refer to section III.1.1.

b. Conduct of the NUC PR Accident Investigation. Standard Practice SQS29 specifies that the Director of Nuclear Power shall establish a division accident investigation committee as soon as practical. The committee shall be responsible for fully investigating all circumstances relating to the accident and shall submit a written report to the division director not later than 15 days after the accident.

[illegible] in Line rates when the [illegible] and [illegible] out of the thimble tube and [illegible] to the other tubes. The [illegible] recommended the use of a

(1) Assigned Goals of the NUC PR Committee. A NUC PR accident investigation committee (AIC) was appointed to conduct an investigation and review of the industrial safety aspects of the thimble tube ejection on May 2, 1984. The committee consisted of a chairman who was a manager from the Industrial Safety Engineering Section, another member of the NCO staff, and the SQN FSG supervisor. The committee was directed to accomplish the following:

- Determine if the event should be investigated in accordance with the TVA "Serious Accident Investigation Procedure."
- Identify lessons learned as a result of the event.
- Provide any recommendations which should be considered in the future when performing similar activities.

(2) Committee Investigation. The committee completed the assigned investigation and reported their findings on May 17, 1984 (L05 840517 800). The investigation consisted of the following:

- Inspection of the seal table area.
- Review of procedures, sketches, and drawings.
- Discussions with Westinghouse.
- Interviews with five of the eight employees in the instrument room when the accident occurred.

(3) Committee Findings. The findings of the committee were as follows:

- Adequate prior warning of bubbling and low-volume flow of relatively cool water allowed egress from the most remote point prior to total seal failure and subsequent thimble tube ejection.

  Note: This description of the nature of the leak before the workers began their egress from the area contradicts information obtained by NSRS from the interviews with the workers (see section IV.E.3 of this reaport).

- There were three paths of egress, two of which were remote from each other, and the

individuals involved were knowledgeable of them. The airlock was the most desirable and the one used.

Note: While this is true, alternate routes of egress were not discussed in prejob planning. In addition one of these paths involved hazards as it was through the polar crane wall where the workers would be exposed to high radiation dose rates due to the gamma radiation from nitrogen 16 produced while the reactor is operating.

- The airlock had been out of service for periods of time during the day making the inner door inoperative. Had the incident occurred during this work, egress through the airlock would have been delayed or primary egress would have been through the submarine hatch.

  Note: Some of the workers in the instrument room while the airlock was out of service (including the FSG coordinator) were unaware that the airlock was out of service. Egress through the submarine hatch was not discussed in any prejob planning.

- The incident would exceed $100,000 in property damage, cleanup, and restoration. The majority of costs would result from the radiological aspects of the incident. (The DASHO and the Office Manager were notifed of the accident).

  Note: No distinction is made between radiological and industrial accidents in the corporate accident investigation procedure. The DASHO and Office Manager were notified three weeks after the accident.

- The investigation was not significantly hindered due to the restoration of the area prior to their involvement.

  Note: The corporate procedure for accident investigation requires that the accident scene be preserved until released by the AIT appointed by the Office Manager and the DASHO. Restoration of the work area before reporting the accident is a violation of TVA procedure.

- The sequence of events - In the sequence of events the committee stated, "The tube was not observed being ejected, nor was steam observed at this time." Looking back through the airlock portholes they could see steam begin to build in the room. Exit time from the platform to safety in the airlock was no greater than 20 seconds. Under the circumstances, the exit appeared very orderly and there were no injuries.

  Note: The start of the ejection of the thimble tube was almost simultaneous with the development of the leak as the cleaning tool was pulled away from the steamfitter when the leak developed and the tool was connected to the thimble tube. The water was flashing to steam above the workers prior to the beginning of their exit from the platform (see section IV.E.3 of this report). The exit was not altogether orderly (see section IV.E.4 of this report).

(4) Committee Conclusions. The committee concluded the following:

- The reason for the failure was not evident. Four possibilities involving the hardware of the seals were listed.

- The flexing activity of the brushing could have aggravated the hardware conditions leading to the failure.

- The instruction (SMI-0-94-1) states that the procedure is not to be used at power. Since the unit was in Mode 1, the procedure was violated.

(5) Committee Recommendations. The committee included the following recommendations:

- Recommendation No. 1. Cleaning and brushing of thimble tubes should be done with the reactor in cold shutdown (Mode 5).

- Recommendation No. 2. If brushing is required past Mode 5, a prejob safety analysis should be performed and the procedure approved by PORC. A mechanism should be installed to preclude tube ejection and leakage and a clear path of egress should be established.

Note: A prejob safety analysis is required by SQM2 for all maintenance activities performed by an MR, and all work performed on CSSC is required to be performed by PORC-reviewed, plant manager-approved procedures. The quality of the job safety analysis and the procedure that was in use and compliance with existing requirements are the true issues. Improving the quality of the job safety analysis and procedure and compliance with existing requirements should be stressed.

- Recommendation No. 3. The brushing mechanism should be modified to eliminate any stress or flex on the thimble tube connection.

- Recommendation No. 4. All work on any system where there is no secondary pressure boundary should be evaluated on a case-by-case basis and adequate means to mitigate an inadvertent pressure failure should be applied.

- Recommendation No. 5. Ensure the constant availability of the primary egress route, i.e., the airlock. Consideration should be given to leaving the inner door open (with the SE's permission) or providing a person to man the door.

  Note: This recommendation should be revised to delete the consideration to leaving the inner door open as the doors are interlocked and having the inner door open would prevent or delay someone from opening the outer door and entering the containment in an emergency for rescue purposes.

- Ensure that all emergency notification systems are in constant operation.

- Commend the eight employees for their coolness under pressure and their ability to reason through egress options under the stressful situation.

  Note: The eight employees did not have to reason through egress options under the stressful situation since the door to the airlock was opened by the employees.

For conclusions and recommendations relating to this section, refer to section III.I.2.

K. NUC PR Special Testing of Thimble Tube Fittings and the Dry Brushing Tool

The NUC PR Mechanical Branch performed postaccident inspection testing to provide insight to the thimble tube ejection accident and to assist in the determination if SQN unit 1 was safe for restart after the accident. The tests involved the following:

- Inspection of hardware from thimble tube D-12.
- Cross sectioning and examination of various combinations of SWAGELOK and GYROLOK brands of fitting hardware.
- Tensile testing of similar hardware.
- Examination of an alleged identical assembly.

The postaccident inspections of the seal from D-12 indicated that the seal had been properly installed (all components were in place and the nut was reasonably tight after the ejection of the thimble tube). Postaccident testing also indicated that the cleaning tool imposed unusual forces on the assembly and that strains of considerable magnitude resulted from reasonably applied forces on the fixture handle. These strains were of sufficient magnitude to cause separation of the thimble tube from a properly installed mechanical seal at reactor operating pressure of 2250 psig.

It should be noted that the cleaning tool supports were designed by TVA and the use of the tool was unrestricted by procedure. The control over the change of design of the tool was very loose as a temporary base support was fabricated and used during the day shift. Additionally, the base support for the tool in use when the accident occurred was modified prior to use. No technical evaluation or testing was performed to assess the effect of the tool on the mechanical seals. The failure to design, evaluate, and test a proper tool and support and the failure to provide restrictions for the tool, support, and cleaning cable in use are the contributors to the failure of the mechanical seal and the accident and not the tool itself.

For conclusions and recommendations relating to this section, refer to section III.D.1 and III.E.1.

L. Worker Background

The work backgrounds of the eight workers involved in the accident are shows in Table 1 and are summarized as follows:

- Three of the six FSG employees involved in the cleaning activity had not read the work instruction prior to the accident including the steamfitter foreman who performed the job safety analysis.

TABLE I

BACKGROUND OF WORKERS INVOLVED IN THE THIMBLE TUBE EJECTION INCIDENT

Worker Identification	Job Title	Read SMI-0-94-2 Prior to Incident	Previously Cleaned Thimble Tubes	Past Work Experience	Experience Working on Systems at Pressure & Temp
A - Evening shift coordinator in charge of activity	Mechanical Engineer	Yes	No	Primarily construction and outage work	Knew alternate egress routes. Had not normally worked on systems at pressure and temperature. Knew pressure, temperature, and configuration of system.
B - Observer	Mechanical General Foreman	NI	No	NI	Knew alternate egress routes. Knew pressure, temperature, and configuration of system.
C - Counting number of revolutions on handcrank	Steamfitter Foreman	No	No	5 years construction and outage work	Knew alternate egress routes, had not worked at these temperatures and pressures. Knew pressure, temperature and configuration of the system.
D - Turning the handcrank	Steamfitter	NI	Yes (only while unit shutdown)	Steamfitter 15 years, construction and outage	Knew alternate egress routes. Did not normally work on systems at these temperatures and pressures. Knew pressure, temperature, and configuration of the system.
E - Monitoring cable as it came out of container looking for rough spots on kinks	Steamfitter	No	Yes (only while unit shutdown)	Steamfitter 13 years, construction and outage	Knew alternate egress routes. Had worked on systems at temperature and pressure but not that much. Knew pressure, temperature and configuration of the system.

TABLE I (Continued)

BACKGROUND OF WORKERS INVOLVED IN THE THIMBLE TUBE EJECTION INCIDENT

Worker Identification	Job Title	Read SMI-0-94-2 Prior to Incident	Previously Cleaned Thimble Tubes	Past Work Experience	Experience Working on Systems at Pressure & Temp
F- Feeding cable into guide tube	Steamfitter	No	No	Steamfitter 5 Years construction and outage	Knew alternate egress routes. Had worked on systems at temperature and pressure but not that much. Knew pressure, temperature and configuration of the system.
G - Taking dose rates	Health Physics technician	NI	NI	HP technician at power plants for 7 years	NI
H - Taking dose rates	Health Physics technician	NI	NI	HP technician at power plants for 5 years	NI

Note: No information (NI) means that the background in this area was not assessed by NSRS.

- Two of six FSG employees had cleaned thimble tubes prior to the event but only while the unit was shutdown. The evening shift coordinator in charge of the cleaning operation and the steamfitter foreman who did the job safety analysis had never cleaned thimble tubes before the incident.

- Five of the FSG employees involved in the activity had primarily a construction and outage background with units shutdown and depressurized (the general foreman's background was not assessed).

- All six FSG employees knew the alternate egress routes before the incident from past experience (the alternate egress routes were not discussed before the accident).

- Even though some of the FSG had worked on some systems at temperature and pressure this type of work this was the exception and not the rule.

- All six FSG employees knew the pressure, temperatures, and configuration of the system before the accident from past experiences or because they had heard it discussed that evening before they entered the instrument room to do the work.

- The two HP technicians were permanent staff members with at least five years experience each at power reactors.

For conclusions and recommendations relating to this section, refer to sections III.B.2 and III.C.1.

M. Employee Expression of Concerns for Safety

1. TVA Policy on Expression of Staff Views

TVA's policy on expression of staff views is delineated in TVA Code II "Expression of Staff Views." It is TVA policy to encourage and protect the differing views of employees on policy and execution of policy. TVA believes that every responsible view is valuable and ensures that such views are heard and appropriately considered in all decisionmaking processes. TVA encourages expression of safety views involving all aspects of its operations, particularly those associated with the design, construction, and operation of TVA nuclear plants. Responsible views may be voiced without fear of recrimination or retribution. TVA employees are responsible for voicing views about significant issues and are encouraged to deal directly with line management so that corrective action may be handled promptly and at the working level. If the views are not resolved at the line management levels, TVA has established methods for handling the views at higher levels which include referring the views to the NSRS for investigation.

2. SQN Employee Expression of Concerns Before and During the Cleaning Activity

Essentially all employees interviewed by NSRS were asked if they openly expressed any concern for safety (nuclear and industrial) to their supervisors before and during the cleaning operation of the thimble tubes. One worker that had experience cleaning the system did express some concern to the steamfitter foreman and the evening shift coordinator about the new design of the base support system because it was different from the base support they had used before. The response to him was that they had used a tool like this in the past. He indicated that he knew the procedure said not to perform the cleaning operation with the reactor operating, but that they really did not have any "gripes" about it. They knew "the situation of the reactor," in that if they performed the work with "no power you have got to take the reactor off the line." He felt in his opinion that what they were going to do was relatively safe.

The concern for safety increased (primarily radiological concerns) as the job progressed. The FSG supervisor was contacted and further planning conducted. All workers interviewed indicated that they felt that there were no hazards that would have justified not performing the work. Some indicated that the work had to be performed to prevent removing unit 1 from operation. No expression of concern for the safety or the job was related to upper plant management.

For conclusions and recommendations relating to this section, refer to section III.J.1.

N. Program Controls Established by SQN Unit 1 Technical Specifications

Technical Specification requirements applicable to review and control of maintenance activities include the following:

1. Section 6.2.3, "Independent Safety Engineering Group (ISEG)". Section 6.2.3 states that the ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, licensee event reports, and other sources which may indicate areas for improving plant safety. Section 6.2.3 further states that ISEG shall be composed of at least five dedicated full-time engineers located onsite and shall be responsible for maintaining surveillance of plant activities to provide independent verification that these activities are performed correctly and that human errors are reduced as much as practical. The ISEG at SQN was not composed of five engineers devoting full attention to ISEG functions and had not been effective in providing independent verification that maintenance activities were performed

correctly and that human errors were reduced as much as practical. (See section IV.Q for details on ISEG activities).

For conclusions and recommendations relating to this section, refer to III.K.1.

2. Section 6.5.1, "Plant Operations Review Committee (PORC)"

The PORC shall function to advise the plant superintendent on all matters related to nuclear safety and is composed of the following members of the plant staff:

- Plant Superintendent (Manager)
- Operations Supervisor
- Results (Engineering) Supervisor
- Maintenance Supervisor
- Assistant Plant Superintendent (Manager)
- Health Physicist
- Supervisor, Quality Assurance Staff (FQE)

PORC responsibilities include the following:

- Review of all procedures required by section 6.8.1 of the Technical Specifications and changes thereto.

- Review of unit operations to detect potential nuclear safety hazards.

SMI-0-94-1 was originally PORC reviewed and approved for the plant superintendent in July 1981 and had not been revised since that time. The quality of the procedure was poor when submitted to PORC. SMI-0-94-2 that was written to clean thimble tubes after the accident and was also of poor quality in that it contained no instructions for disassembling and reassembling the detector drive system from the thimble tubes, no precautions or warnings to alert personnel of the sensitive nature of the mechanical seals and restrictions for working on the system with the reactor pressurized, no postmaintenance inspections to ensure the quality of the seals had not been degraded during the maintenance process, and postmaintenance testing was optional. Use of this instruction could degrade the mechanical seals and if performed at pressure could cause a thimble tube to eject or if not inspected, detected, and corrected could cause an ejection during pressurization and startup of the reactor. Despite these inadequacies and even after the accident the instruction was PORC reviewed and recommended for approval to the Plant Manager. It is apparent that the PORC review was ineffective in identifying the procedure inadequacies in the original instruction and in the instruction recommended for approval by PORC after the accident.

For conclusions and recommendations relating to this section, refer to section III.H.5.

3. Section 6.8, "Procedures and Programs"

a. Section 6.8.1.a. Section 6.8.1.a states that written procedures shall be established, implemented, and maintained covering applicable procedures recommended in Appendix A of Regulatory Guide (RG) 1.33, Revision 2, February 1978. Appendix A, section 9.C of RG 1.33 states that procedures for the repair of the incore flux monitoring system should be prepared prior to beginning work.

As discussed in section IV.D.2.a of this report, SMI-0-94-1 was violated and thus not properly implemented.

b. Section 6.8.2. Section 6.8.2 states that each procedure of section 6.8.1 and changes thereto shall be reviewed by PORC and approved by the plant manager prior to implementation and that each procedure shall be reviewed periodically as set forth in administrative procedures. Administrative Instruction AI-4, "Plant Instructions - Document Control," revised March 9, 1984, states in section 5.3.2 that each instruction shall be reviewed biennially after issuance to determine if changes are necessary or desirable.

Inadequate PORC review of SMIs is discussed in section IV.N.2 above. Additionally, the biennial review process established by AI-4 was inadequate in that the poor quality of SMI-0-94-1 was not corrected and the instruction was almost three years old when the accident occurred and had not been revised since its original issue.

c. Section 6.8.3. Section 6.8.3 states that "temporary changes" to procedures of paragraph 6.8.1 may be made provided:

- The intent of the original procedure is not altered.
- The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operators License on the unit affected.
- The change is documented, reviewed by PORC and approved by the plant manager within 14 days of implementation.

When asked how SMI-0-94-1 should have been changed to make it appropriate for the dry brushing cleaning operation at power, managers and engineers interviewed responded that a temporary change should have been issued to delete the words concerning "do not use the equipment or procedure at power." A change of that nature would be inappropriate as the intent of the instruction would be changed. This type of response is an indication that the people interviewed were not aware of what quality elements are necessary for a good instruction for assuring that the quality of a CSSC is not degraded during the maintenance process, were not aware of the procedure change process, or were expressing a careless attitude toward procedural compliance. The fact that this lack of awareness or careless attitude was expressed (toward procedures) after review of the accident indicates an alarming lack of appreciation of the importance of adequate procedures and procedural compliance. Effective preventive action to reduce procedure violation errors will not be successful unless and until the lack of awareness or such attitudes are changed.

In summary, there was a significant breakdown in the controls for maintenance activities established by the unit 1 Technical Specifications in that (1) ISEG activities did not comply with the intent of the Technical Specifications and had been ineffective, (2) PORC review of special maintenance instructions for the cleaning of thimble tubes before and after the accident had been inadequate, and (3) there was a significant breakdown in the SQN procedure process for maintenance activities.

O. Prior Findings and Recommendations Following NSRS Investigation of 10-Rem Extremity Exposure at SQN

During September and October 1982 NSRS conducted an indepth investigation into the causal factors associated with a 10-rem extremity exposure at SQN. The findings as reported in NSRS Report No. I-82-21-SQN issued December 1, 1982, indicated that the causal factors for the 10-rem extremity exposure were on inadequate hazard assessment, inadequate prejob planning, lack of training, and inadequate adherence to the TVA safety-first policy. Some of the causal factors for that incident are similar to some of the causal factors for this accident. Recommendations were made by NSRS in December 1982 to correct the causal factors of that incident. It is apparent that some of these recommendations had not been implemented.

For conclusions and recommendations relating to this section, refer to section III.C.3.

P. SQN Licensee Event Report (LER) No. SQRO-50-327/84030

This LER, prepared by the plant Compliance Staff and transmitted to the NRC on May 18, 1984, provided the details concerning ejection of the incore thimble tube.

Paragraph b.(2).ii.I of 10CFR50.73, "Licensee Event Report System," states "the narrative description must include the following specific information as appropriate for the particular event: The method of discovery of each component or system failure or procedural error."

Under "the Event" of the LER the method of discovery was stated as "water was noticed on the seal table."

Paragraphs b.(2)ii.(J)(2)(ii) of 10CFR50.73 states "for each personnel error the licensee shall discuss: whether the error was contrary to an approved procedure. . . or was associated with an activity or task that was not covered by an approved procedure."

There was no mention of inadequate or violation of procedures in the narrative of the LER.

Paragraph b.(4) of 10CFR50.73 states "The Licensee Event Report shall contain: a description of any corrective actions planned as a result of the event, including those to reduce the probability of similar events occurring in the future."

The "corrective actions" stated in the LER were "all short-term corrective action taken has been described in the above text. Per vendor recommendations, the seal table and associated fittings were inspected. This inspection determined that no additional corrective action was required. For long-term corrective action, management has made the decision that future thimble tube cleaning will not be performed during power operations."

LER No. SQRO-50-327/84030 transmitted to the NRC on May 18, 1984, was misleading and did not meet the specified requirements of 10CFR50.73 in that the leak was described as "water was noticed on the seal table." (While this is true it does not accurately describe the true nature of the leak as described to NSRS by the workers.) There was no mention in the LER that the primary work instruction for the activity, SMI-0-94-1 was inadequate, was violated, and the long-term correction specified does not address corrective actions to correct the causal factors of the event that may reduce the probability of an event of a similar nature.

For conclusions and recommendations relating to this section, refer to section III.L.1.

Q. SQN Compliance Staff/ISEG Activities

1. Background

NUREG 0737, "Clarification of TMI Action Plan Requirements," issued November 1980 specified post-TMI requirements for operating reactors and applicants for operating licenses to be incorporated into plant design and methods of operation for the purpose of minimizing the probability of a serious reactor accident. One of those items (I.B.1.2) was the requirement of the establishment of an "Independent Safety Engineering Group (ISEG)." The principal function of the ISEG would be to examine plant operating characteristics, NRC issuances, and other appropriate sources of plant design and operating experience information that may indicate areas for improving plant safety. The ISEG would perform independent review and audits of plant activities including maintenance, operational problems, and aid in the establishment of programmatic requirements for plant activities. Where useful improvements could be achieved, it was expected that this group would develop and present detailed recommendations to corporate management for such things as revised procedures or equipment modifications. Another intended function of the ISEG was to maintain surveillance of plant operations and maintenance activities to provide independent verification that these activities were performed correctly and that human errors were reduced as far as practicable. ISEG would then be in a position to advise utility management on the overall quality and safety of operations.

The ISEG was to be an additional independent group of a minimum of five dedicated, full-time engineers, located onsite but reporting offsite to a corporate official who held a high level, technically oriented position that was not in the management chain for power production. The ISEG would increase the available technical expertise located onsite and would provide continuing systematic and independent assessment of plant activities.

The requirement for the ISEG was made a licensing requirement by NRC for the SQN license and included in the Technical Specifications as discussed in section IV.N.1 of this report.

2. SQN Implemention of the ISEG Requirement

SQN and NUC PR management elected to assign the ISEG function to the existing Plant Compliance Staff. SQN Standard Practice SQA117, "Responsibilities of Nuclear Plant Compliance Staff for Nuclear Safety Engineering" revised March 1984, defines the responsibilities of the Compliance Staff at SQN in meeting the NRC requirement for a safety engineering group. The Standard Practice does not cover all of the

responsibilities of the Compliance Staff not related to the ISEG function. The defined responsibilities for fulfilling the safety engineering function and providing an independent onsite assessment of nuclear plant activities include review of plant operation and maintenance activities, review of potential reportable occurrences (PROs), and generation of LERs as applicable. (As of May 18 the Compliance Staff had generated 30 LERs for unit 1.)

Additionally, as a compliance function the Compliance Staff logs and tracks regulatory as well as other commitments. They provide the investigations and the responses to findings by NRC, Office of Quality Assurance, and others and coordinate the interface between the plant staff and the inspection, review, investigation, and audit groups. All of these are considered ISEG functions by the plant Compliance Staff in that they get involved with problems they or others have identified. They stated that they ensure that in the process of investigating and writing the reports, the right corrective actions have been taken, both short and long term, to prevent recurrence. The Compliance Staff advises the plant management and others on regulatory matters including interpretation of Technical Specifications.

The ISEG concept used at SQN had diverged from the original NRC and Technical Specification intent as interpreted by NSRS in that it is not composed of five full-time senior level engineers located onsite dedicated full time to ISEG functions, is involved in line production functions, is not independent from the power production organization to ensure objectivity, and is not in the position to assess and advise utility management on the overall quality and safety of operations.

At SQN the ISEG function was assigned to the Compliance Staff which performed line functions for the Plant Manager. These functions performed by the Compliance Staff do afford the opportunity to review plant operation and maintenance activities but do not afford the opportunity to perform the reviews thoroughly and with independence from pressures of operation of the facility. Additionally, the performance appraisals, and thus the promotability in the organization, are performed by the site management. The compliance functions performed by the Compliance Staff are line functions and are subject to operational pressures.

The accident was investigated by the SQN Compliance Staff (ISEG) and the description of the event, the cause of failure and the long-term corrective action specified in LER SQN-50-327/84030 were determined by that group. The Compliance Staff/ISEG conclusions concerning the accident as reflected in the LER failed to recognize any programmatic problems that may adversely impact the safety of plant personnel or plant operations in the future.

In general, the Compliance Staff/ISEG personnel interviewed expressed that their thoughts concerning the accident were that it was an unfortunate event. They thought that the plant had demonstrated through the outage that they had made tremendous headway in conducting outages and getting through them, and this accident was an unfortunate event that occurred and kept the unit from going back to power. Based on what they had seen and what the engineering section had done prior to making the decision to clean at power, they did an adequate evaluation, at least talked to industry people that had experience in this area, and came up with a decision that cleaning at power could and had been done.

The thoughts expressed by the Compliance Staff/ISEG personnel interviewed reflected a line supervisor's attitude and one that was concerned with schedule and not one that was concerned from an independent standpoint for nuclear safety.

The Compliance Staff at SQN has been ineffective in performing the ISEG functions of maintaining surveillance of plant activities to provide independent verification that activities (including maintenance activities) were performed correctly and that human errors were reduced as much as practical. This lack of effectiveness in identifying problem areas with program controls is in itself a program weakness which thus promoted conditions that allowed the accident to occur.

For conclusions and recommendations relating to this section, refer to section III.K.1.

V. PERSONNEL CONTACTED

A. Industry

1.	G. Black	Teleflex Corporation
2.	A. Burger	Beaver Valley Nuclear Plant
3.	R. Cockrell	INPO
4.	M. Garton	North Anna Nuclear Plant
5.	D. Kane	Beaver Valley Nuclear Plant
6.	M. Kwitck	Kewaunee Nuclear Plant
7.	R. Mathieson	Westinghouse (SQN Site Rep.)
8.	W. Mullet	NUS
9.	J. Perry	Trojan Nuclear Plant
10.	A. Stough	NUS
11.	R. Wells	INPO

B. TVA Corporate

1.	J. Thompson	OGM (DASHO)

C. Division of Occupational Health and Safety

1.	H. Linder	OC H&S

D. Office of Power (PWR)

1.	S. Bugg	RHS
2.	H. Kemp	PWR
3.	J. Lobdell	RHS

E. Division of Nuclear Power

1.	H. Abercrombie	NCO
2.	T. Campbell	NCO
3.	J. Fox	NCO
4.	L. Ellis	NCO
5.	R. Kitts	NCO
6.	R. Sessoms	NCO
7.	P. Wallace	NCO

F. Sequoyah Nuclear Plant

1.	D. Albury	FSG
2.	L. Alexander	FSG
3.	C. Baker	FSG
4.	R. Byrant	FSG
5.	J. Clift	FSG
6.	M. Cooper	Compliance Section
7.	D. Crawley	HP Section
8.	M. Edwards	HP Section
9.	R. Fortenberry	Engineering Section
10.	H. Gammage	FSG
11.	M. Harding	Compliance Section
12.	S. Harrison	HP Section
13.	S. Holderford	HP Section
14.	D. Jackson	Safety Section
15.	G. Kirk	Compliance Section
16.	J. Krell	Maintenance Section
17.	D. Love	Maintenance Section
18.	C. Mason	SQN
19.	S. Martin	Document Control
20.	B. McKay	Engineering Section
21.	L. Nobles	SQN
22.	J. Osborne	HP Section
23.	D. Paschal	FSG
24.	J. Record	Master Files
25.	J. Robinson	FSG
26.	B. Schofield	Engineering Section
27.	B. Simpson	Engineering Section
28.	J. Stiegleman	HP Section
29.	V. Taylor	Safety Section
30.	B. Turner	FSG
31.	J. Walker	Operations Section
32.	K. Whitty	Engineering Section

G. Watts Bar Nuclear Plant

1.	W. Byrd	Compliance Section
2.	R. Sauer	Compliance Section

VI. DOCUMENTS REVIEWED

A. Regulatory

1. U.S. NRC Report Nos. 50-327/84-14 and 50-328/84-14, received July 2, 1984

2. U.S. NRC Report Nos. 50-327/84-13 and 50-328/84-13, issued June 21, 1984

3. U.S. NRC NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980

4. Code of Federal Regulations
   10CFR50.73, "Licensee Event Report System," September 30, 1983
   10CFR50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," January 1, 1983

5. U.S. NRC Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," February 1978

6. U.S. NRC IE Information Notice NO. 84-55, "Seal Table Leaks at PWRs," July 6, 1984

7. SQN LER No. SQRO-50-327/84030

8. U.S. NRC NUREG/CR-1369, "Procedures Evaluation Checklist for Maintenance, Test and Calibration Procedures Used in Nuclear Power Plants," September 1982

B. Industry

1. Trojan Nuclear Plant, "Flux Thimble Tube Cleanout at Full Power"

2. Management Oversight and Risk Tree Users Manual, EG&G/DOE, Idaho National Engineering Laboratory, ERDA-76/45-4, November 1976

3. Westinghouse Electric Corporation, "Topical Report - Safety Related Research and Development for Westinghouse Pressurized Water Reactors Program Summaries," WCAP-7856, Fall 1971 - Spring 1972

4. Westinghouse Electric Corporation, "Topical Report - In-Core Instrumentation (Flux Mapping System and Thermocouples)," July 1971

5. Westinghouse Nuclear Energy Systems, "Technical Manual for In-Core Instrumentation - Tennessee Valley Authority Sequoyah Nuclear Plant Unit No. 1 and No. 2"

6. Occupational Safety and Health, Standards and Interpretations, "Subpart E - Means of Egress"

7. Westinghouse Correspondence from R. Howard to R. Mathieson, "Seal Table Fittings Intermix - SEQ 1," May 2, 1981

8. Letter to M. D. Wingo from M. Cuppula, Superintendent of Technical Services, Duquesne Light, "Incore Thimble Maintenance," May 14, 1984

C. Corporate

1. Memorandum from H. N. Culver to W. F. Willis, "Sequoyah Nuclear Plant - Notification of an Unusual Event," April 20, 1984 (GNS 840423 100)

2. Memorandum from H. N. Culver to H. G. Parris, "Sequoyah Nuclear Plant - NSRS Investigation of the Unusual Event on April 19, 1984 - NSRS Report No. I-84-12-SQN," April 25, 1984 (GNS 840425 051)

3. Tennessee Valley Authority, "Severe Accident Investigation Procedure," January 1984

4. Memorandum from H. N. Culver to E. A. Belvin and H. G. Parris, "Sequoyah Nuclear Plant Investigation of 10 Rem Extremity Exposure - Nuclear Safety Review Staff (NSRS) Report No. I-82-21-SQN," December 1, 1982 (GNS 821203 050)

D. Office of Power

1. Office of Power Radiation Plan, Section A, "Nuclear Power Plants," November 2, 1983

2. Memorandum from H. G. Parris to W. F. Willis, "Sequoyah Nuclear Plant - Notification of an Unusual Event," April 20, 1984 (GNS 840423 100)

E. Division of Nuclear Power

1. Operational Quality Assurance Manual Procedure No. N-OQAM, Part II, Section 2.1, "Plant Maintenance," February 7, 1983

2. Divison of Nuclear Power, "Plant New and Escalated Operational Event Report - Sequoyah Plant Status," April 17-30, 1984

3. Division of Nuclear Power, "Directives Manual," November 15, 1983

4. Area Plan Procedure No. 0604.05, "Responsibilities of Nuclear Plant Independent Safety Engineering Group/ Compliance Staff," October 31, 1983

5. Area Plan Procedure No. 0604.04, "Unreviewed Safety Question Determination (USQD - Intent, Method, Review, and Approval," October 13, 1983

6. Operational Quality Assurance Manual Procedure No. SQ-OQAM, Appendix A, "Critical Structures, Systems, and Components (CSSC) List"

7. Operational Quality Assurance Manual, Part III, Section 7.3, "Common-Mode Failures, Maintenance Initiated," January 15, 1981

8. Letter from J. A. Coffee to Mr. Larry Sinter, Director, Tennessee Emergency Management Agency, "Sequoyah Nuclear Plant Notification of Unusual Event - April 20, 1984," April 25, 1984 (GN8 840430 100)

9. Memorandum from R. A. Sessoms to L. C. Ellis, "Sequoyah Nuclear Plant Unit 1 - Incore Thimble Ejection - Investigation and Review of Events for Industrial Safety Implications," May 2, 1984 (L01 840502 802)

10. Memorandum from L. C. Ellis to R. A. Sessons, "Sequoyah Nuclear Plant Unit 1 - Incore Thimble Ejection - Investigation and Review of Events for Industrial Safety Implications," May 17, 1984 (05 840517 800)

F. Sequoyah Nuclear Plant

1. Draft - "Sequoyah Nuclear Plant Unit 1 D-12 Traveling Incore Probe Thimble Tube Separation Special Tests," May 17, 1984

2. Special Maintenance Instruction SMI-0-94-1, "RPV Bottom Mounted Instrument Thimble Tubes Cleaning and Flushing," July 10, 1981

3. Special Maintenance Instruction SMI-0-94-2, "Incore Flux Thimble Cleaning and Lubrication," Revision 0, April 26, 1984

4. Maintenance Request Form, A-238084, April 18, 1984

5. Radiation Work Permit No. 02-1-00102, January 1, 1984

6. Radiation Work Permit Timesheet No. 02-1-00102-0090, April 18, 1984

7. Radiation Work Permit No. 02-1-00005, Issued April 20, 1984

8. Radiation Work Permit Timesheet Nos. 92-1-00005-0002 through 0062, Issued April 20, 1984 through May 1, 1984

9. Whole Body Analysis Records for the following SQN personnel:

J. Clift, FSG	D. Albury, FSG
H. Gammage, FSG	C. Baker, FSG
B. Turner, FSG	S. Harrison, HP
B. Simpson, FSG	M. Edwards, HP
D. Paschal, FSG	

10. Radiological Control Instruction RCI-10, "Minimizing Occupational Radiation Exposure," Revision 8

11. Radiological Control Instruction RCI-14, "Radiation Work Permit (RWP) Program," Revision 2

12. Radiological Control Instruction RCI-10, Attachment 1, "ALARA Preplanning," April 19, 1984

13. Potential Reportable Occurrence, PRO No. 1-84-159, April 20, 1984

14. SQN Technical Specifications - Unit 1, Sections:

3.3.3.2	"Movable Incore Detectors"
3/4.3.3.2	"Movable Incore Detectors"
3/4.4.10	"Structural Integrity"
6.2.3	"Independent Safety Engineering Group (ISEG)"
6.5.1	"Plant Operations Review Committee (PORC)"
6.8	"Procedures and Programs"

15. SQN Final Safety Analysis Report, Sections:

3.6	"Protection Against Dynamic Effects Associated With the Postulated Rupture of Piping"
5.2	"Integrity of the Reactor Coolant System Boundary"
7.7.1.9.2	"Movable Neutron Flux Detector Drive System"
13.5	"Plant Instructions"

16. Administrative Instruction AI-4, "Plant Instructions - Document Control," March 9, 1984

17. Administrative Instruction AI-3, "Clearance Procedures," Revision 23

18. Administrative Instruction AI-8, "Access to Containment," Revision 10

19. Administrative Instruction AI-13, "Control of CSSC Equipment," Revision 25

20. Administrative Instruction AI-30, "Nuclear Plant Method of Operation," Revision 6

21. Administrative Instruction AI-8, "Containment Entry Checklists," April 18, 1984 - April 19, 1984

22. Clearance Sheets, Hold Order No. 1, "Incore Probes," January 1, 1984

23. Standard Practice SQA119, "Unreviewed Safety Question Determinaion," Revision 3

24. Standard Practice SQA 128, "Method of Operation - Policy," Revision 0

25. Standard Practice SQA129, "Objectives in Plant Operation - Sequoyah Nuclear Plant," Revision 2

26. Standard Practice SQA 131, "Recovery From a Spill of Radioactively Contaminated Liquid," Revision 1

27. Standard Practice SQS29, "Accident Reporting and Investigation," Revision 3

28. Abnormal Operating Instruction AOI-6, "Small Reactor Coolant Leak," Revision 13

29. Hazard Control Instruction HCI-G1, "Hazard Control Instruction Manual," April 21, 1976

30. Hazard Control Instruction HCI-G2, "The Supervisor," May 26, 1983

31. Hazard Control Instruction HCI-G3, "The Employee," January 31, 1984

32. Hazard Control Instruction HCI-G6, "Clearance Procedure Requirements," May 26, 1983

33. Hazard Control Instruction HCI-G15, "Initial Accident Reporting and Emergency Actions," March 22, 1983

34. Hazard Control Instruction HCI-G16, "General Safe Work Rules and Employee Conduct," May 26, 1983

35. Hazard Control Instruction HCI-G26, "Buddy System in Hazardous Low Accessibility Areas," March 22, 1983

36. Hazard Control Instruction HCI-G29, "Workplace Hazard Assessment," February 14, 1984

37. Quality Engineering Section Instruction Letter No. 5.3, "Maintenance Requests - FQE Section Review," Revision 9

38. SQN Shift Engineers Journal, April 17, 1984 - April 25, 1984

39. SQN Assistant Shift Engineer (SRO) Journals (Unit 1), April 17, 1984 - April 26, 1984

40. SQN Unit Operator Journals (Unit 1), April 17, 1984 - April 23, 1984

41. SQN Health Physics Journals for 690 HP Lab, April 19, 1984 - April 26, 1984

42. "Superintendent's Letter," Sequoyah Nuclear Plant, Volume 1, No. 6, April 30, 1984

G. Watts Bar Nuclear Plant

1. Standard Practice WB6.5.1, "Engineer Assignment to Plant Systems and Equipment," March 14, 1984

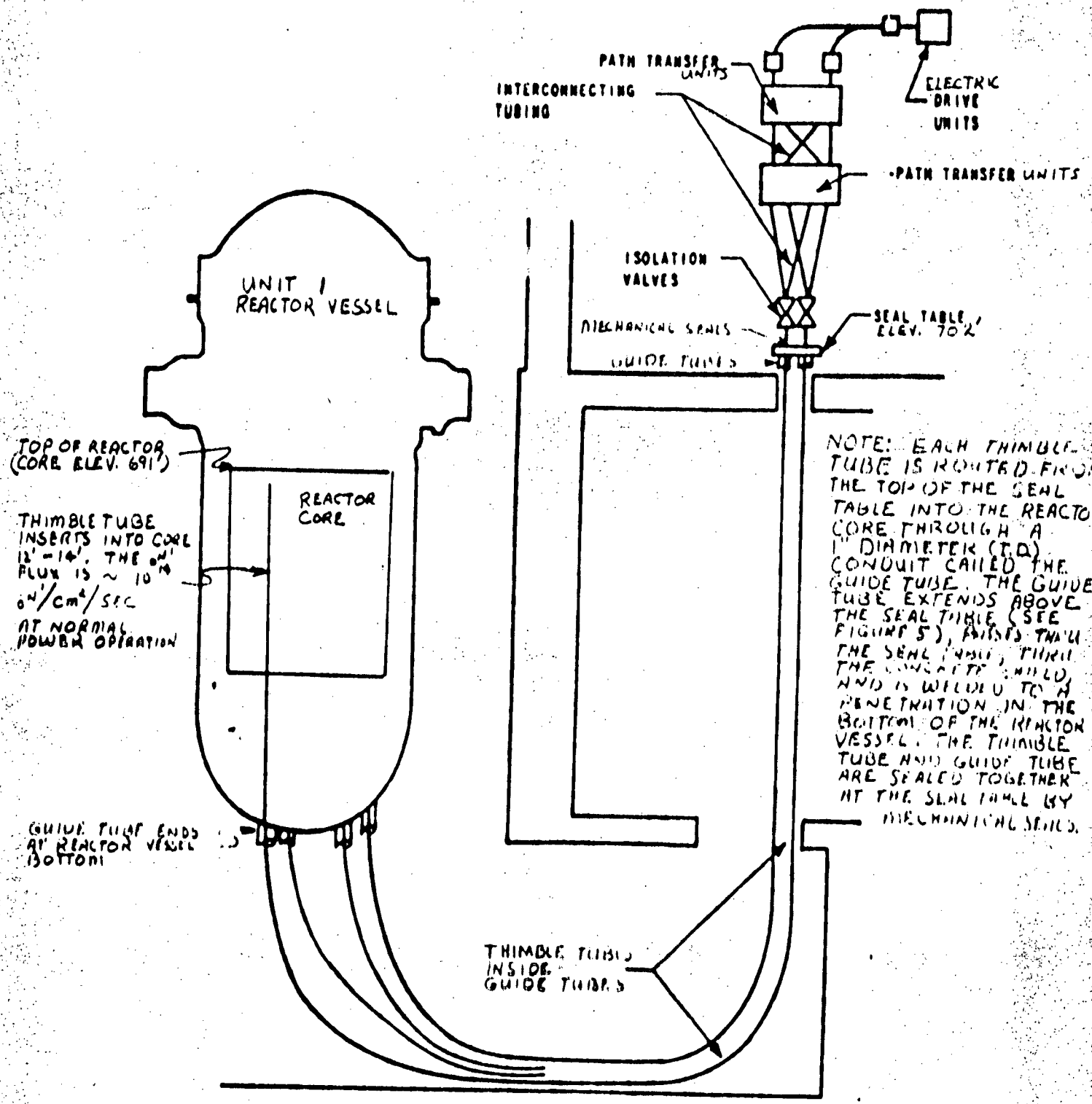
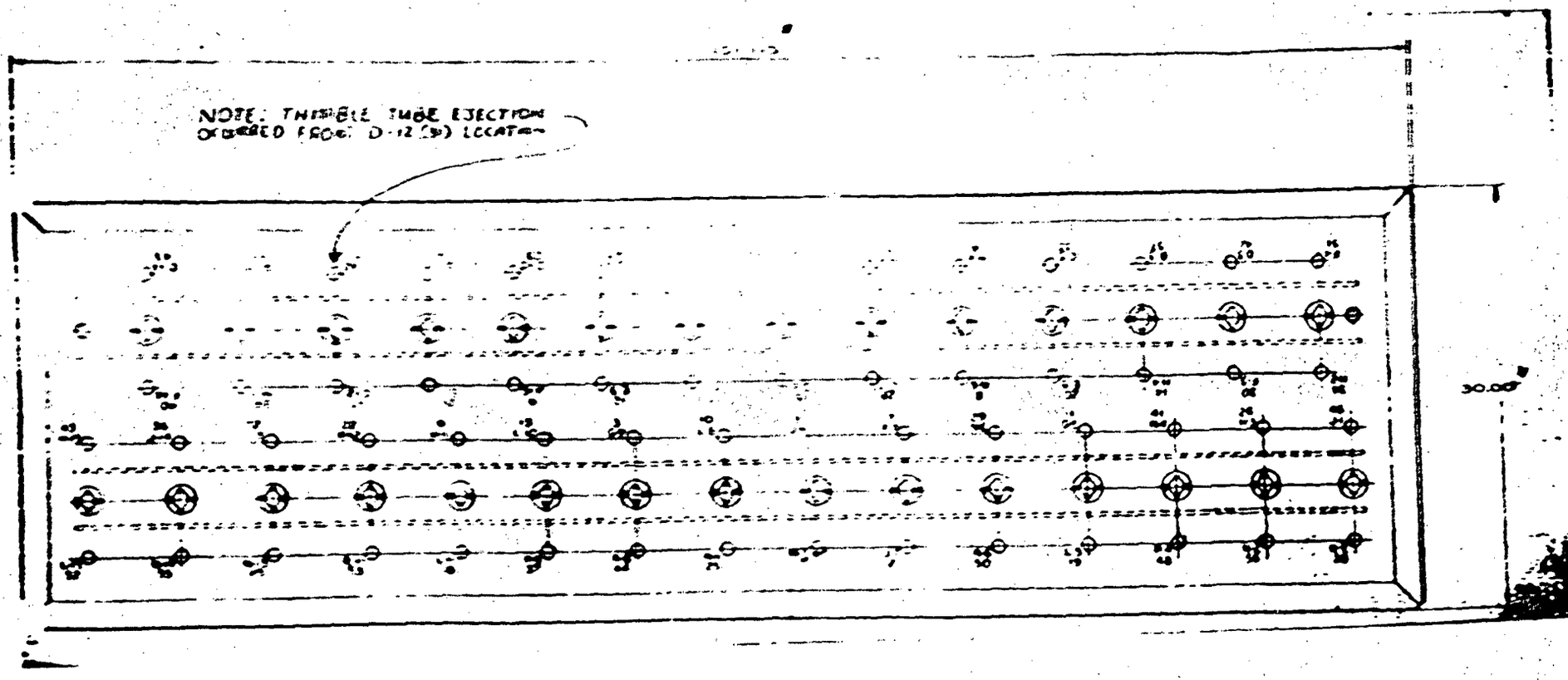


FIGURE 1 INCORE DETECTOR SYSTEM

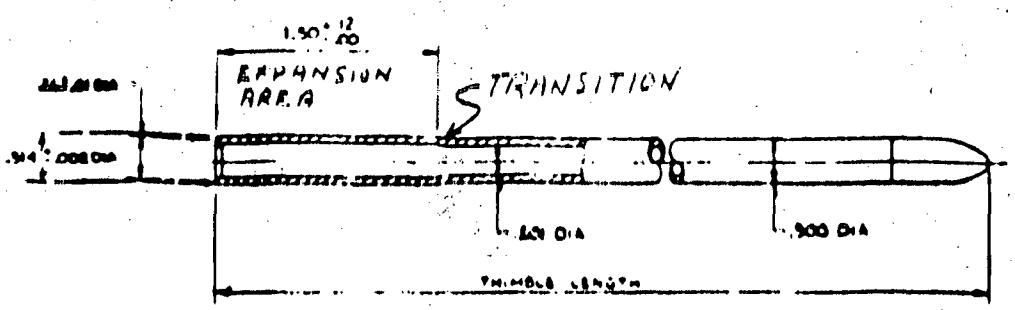


POLAR CRANE WELL

SEAL TABLE
FIGURE 2

SEAL TABLE END

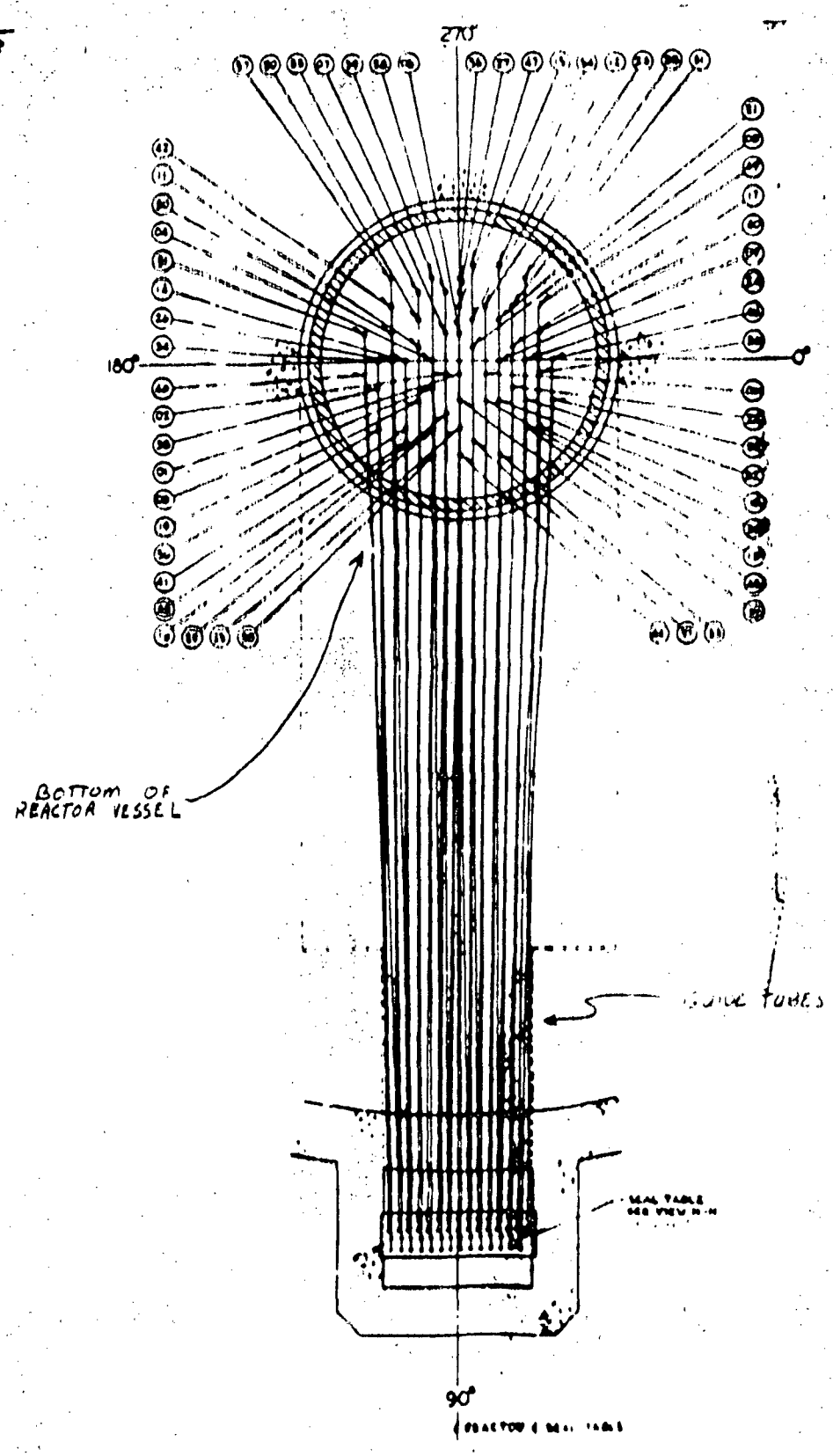
REACTOR CORE END



LENGTHS VARY FROM 164 TO 177 FEET

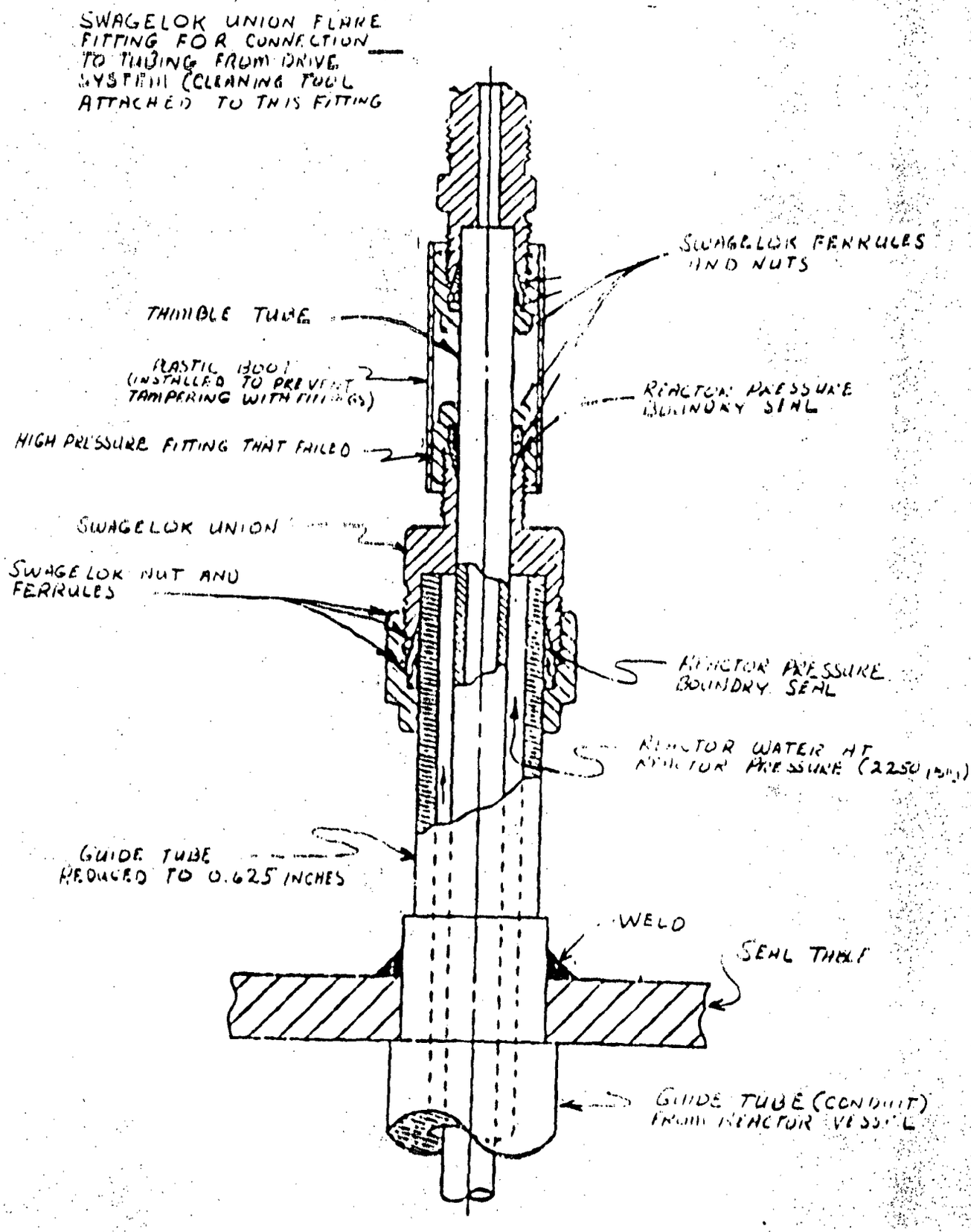
THIMBLE TUBE (SHOWING DESIGN AND EXPANSION FOR MECHANICAL SEALS AT THE SEAL TABLE)

FIGURE 3



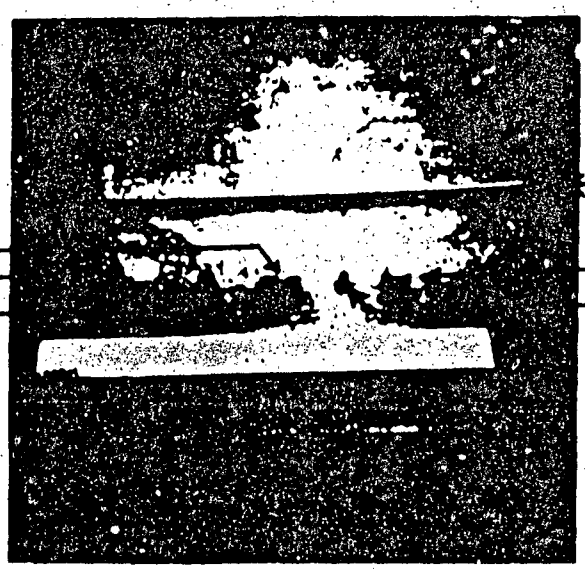
CONFIGURATION SHOWING GUIDE TUBES
AND RESPONSIVE THIMBLE TUBES

FIGURE 4



THIMBLE TUBE MECHANICAL SEALS

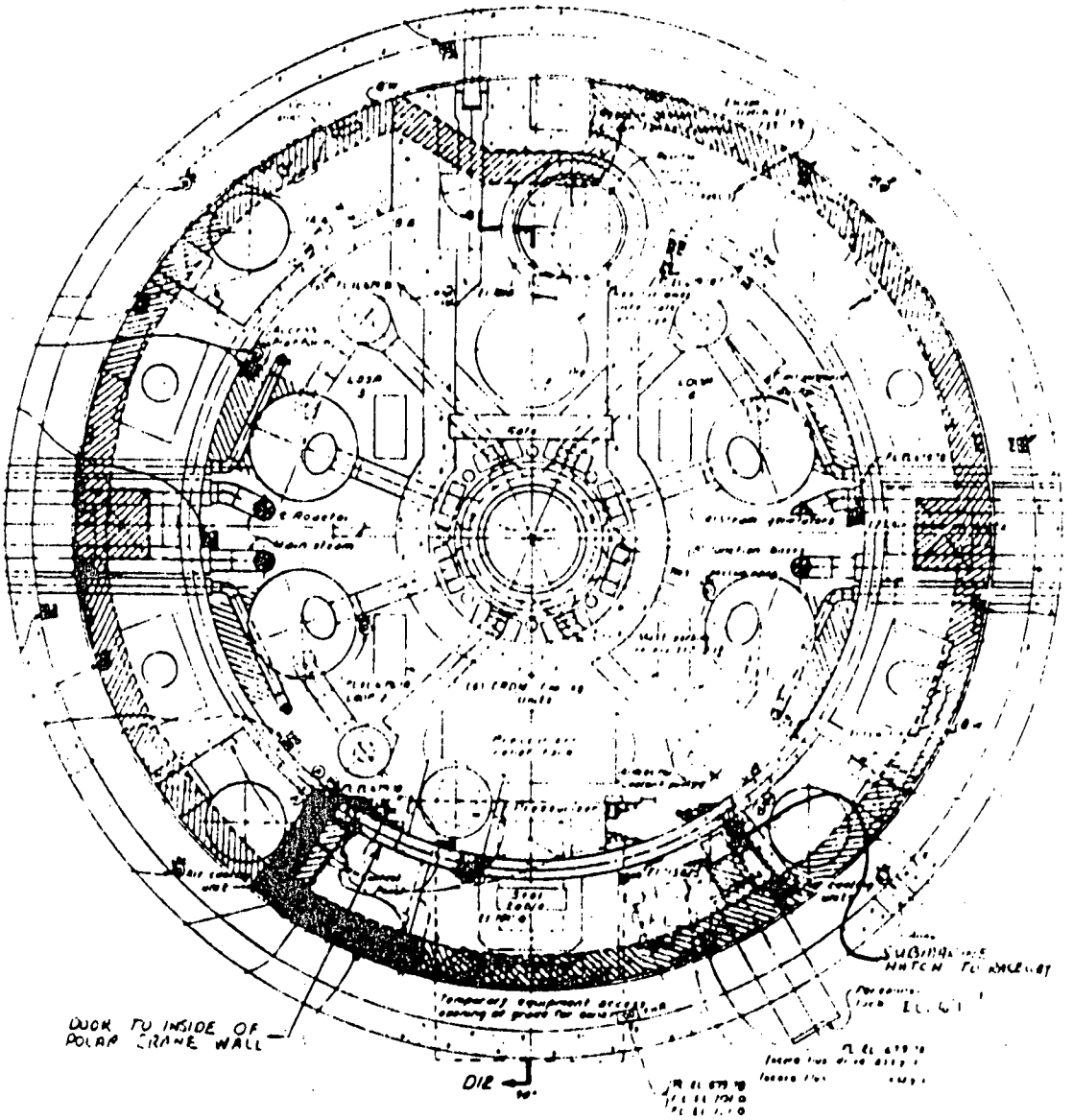
FIGURE 5



PIECE OF THIMBLE TUBE AND TYPICAL SWAGELOK FITTING

NOTE: THE FITTING INVOLVED IN THIS INCIDENT CONTAINED A GYROLOK FERRULE ASSEMBLY CONSISTING OF TWO PIECES (SEE FIGURES 1, 2, 5, AND 6 OF ATTACHMENT 1 FOR MORE DETAIL)

FIGURE 6



PLAN-LOWER COMPARTMENT
REACTOR CONTAINMENT

FIGURE 7A

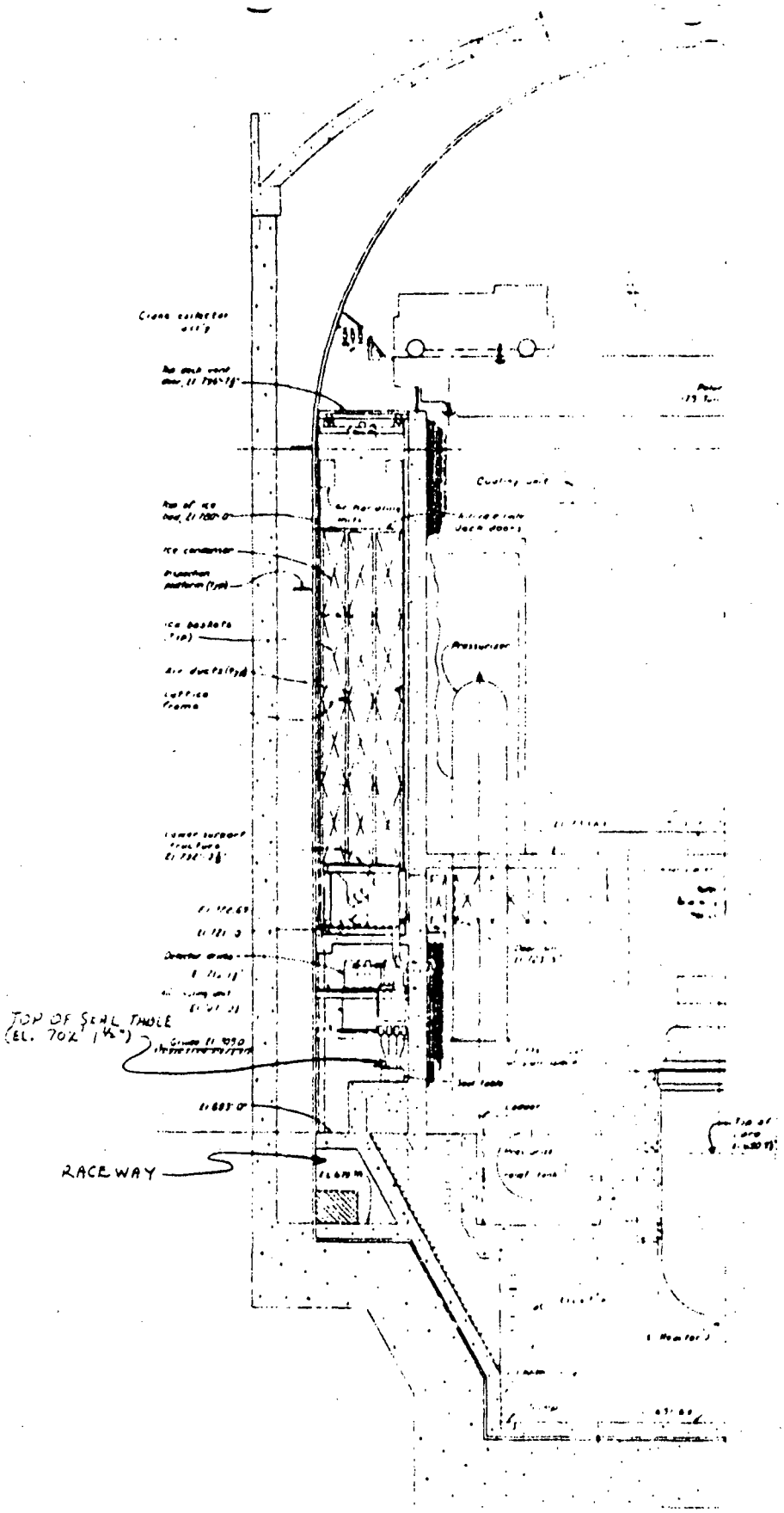
VIEW FROM INSIDE
THE REACTOR BUILDING
CONTAINMENT

HANDWHEELS TO OPEN AND
CLOSE INNER AND OUTER
DOORS

INSIDE AIRLOCK
IS EQUIPPED WITH
A TELEPHONE

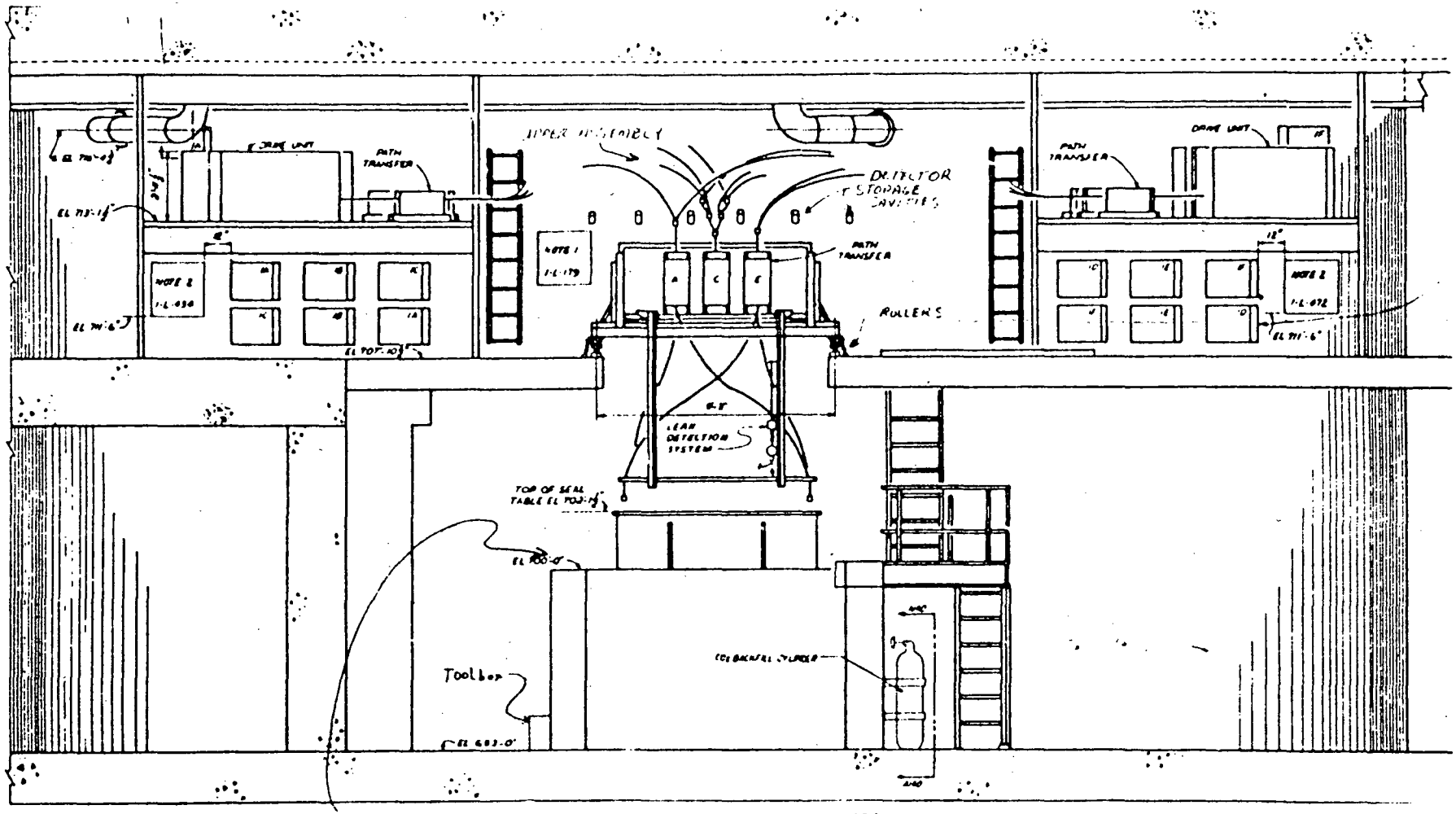
WATTS BAR NUCLEAR PLANT PERSONNEL AIRLOCK
DOOR (SIMILAR TO THE SQN EL 690' AIRLOCK)

FIGURE 7B



ELEVATION DRAWING OF THE REACTOR BUILDING

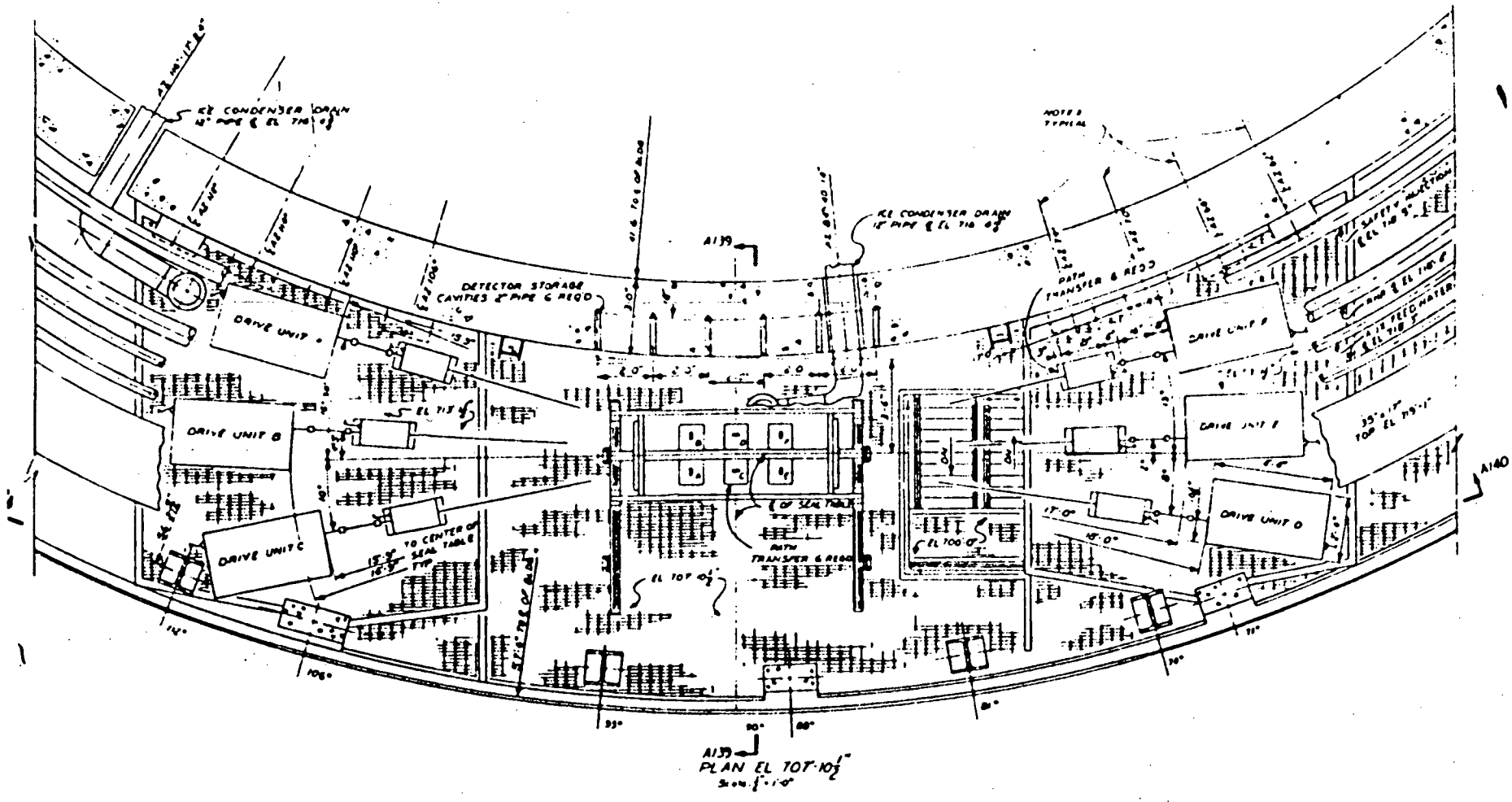
FIGURE 8



NOTE: THE SEAL TABLE PLATFORM AREA IS EQUIPPED WITH A HANDRAIL WHICH IS NOT SHOWN IN THIS DRAWING

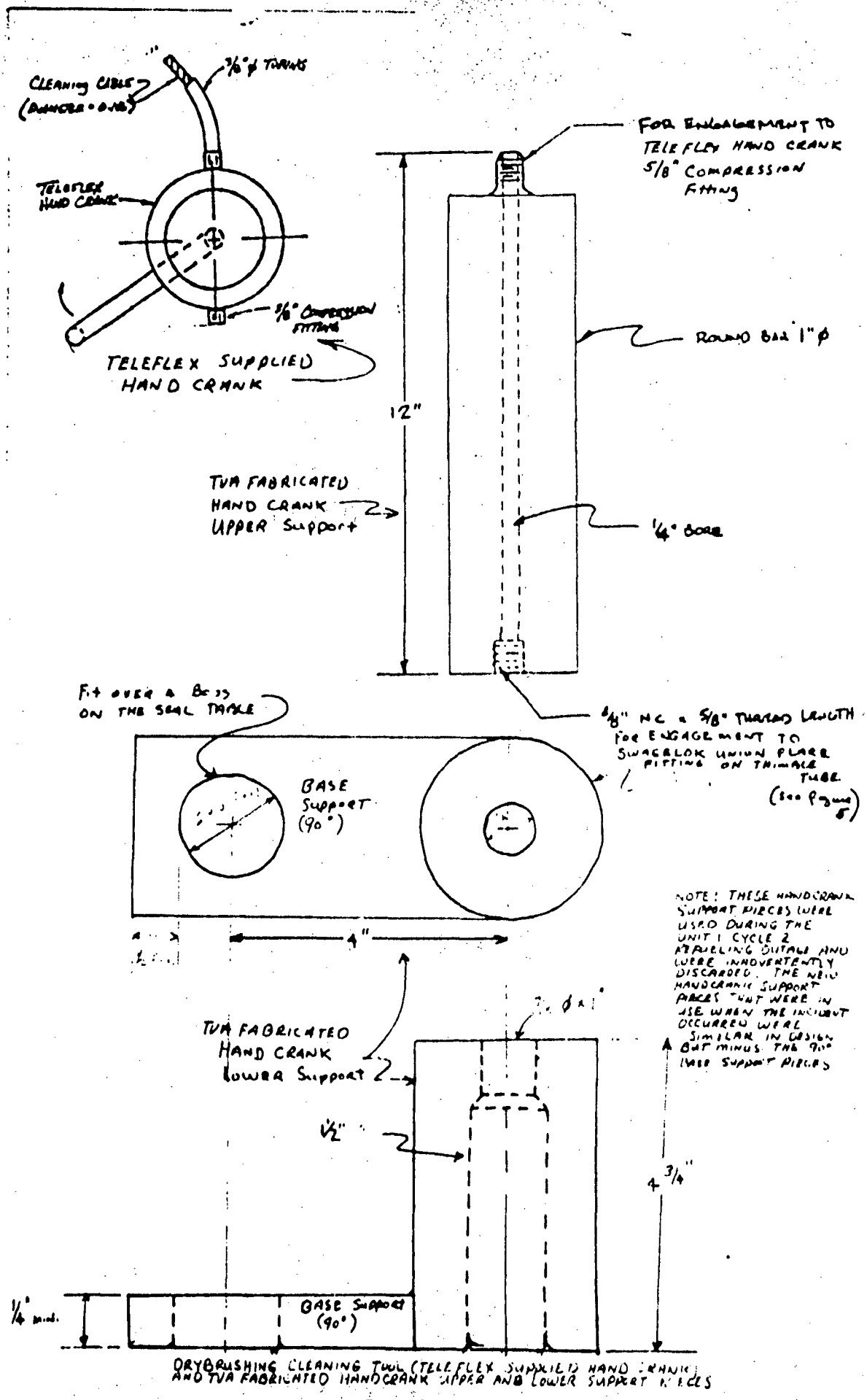
LOCATION OF THE INCORE INSTRUMENTATION SYSTEM EQUIPMENT IN THE INSTRUMENT ROOM

FIGURE 9



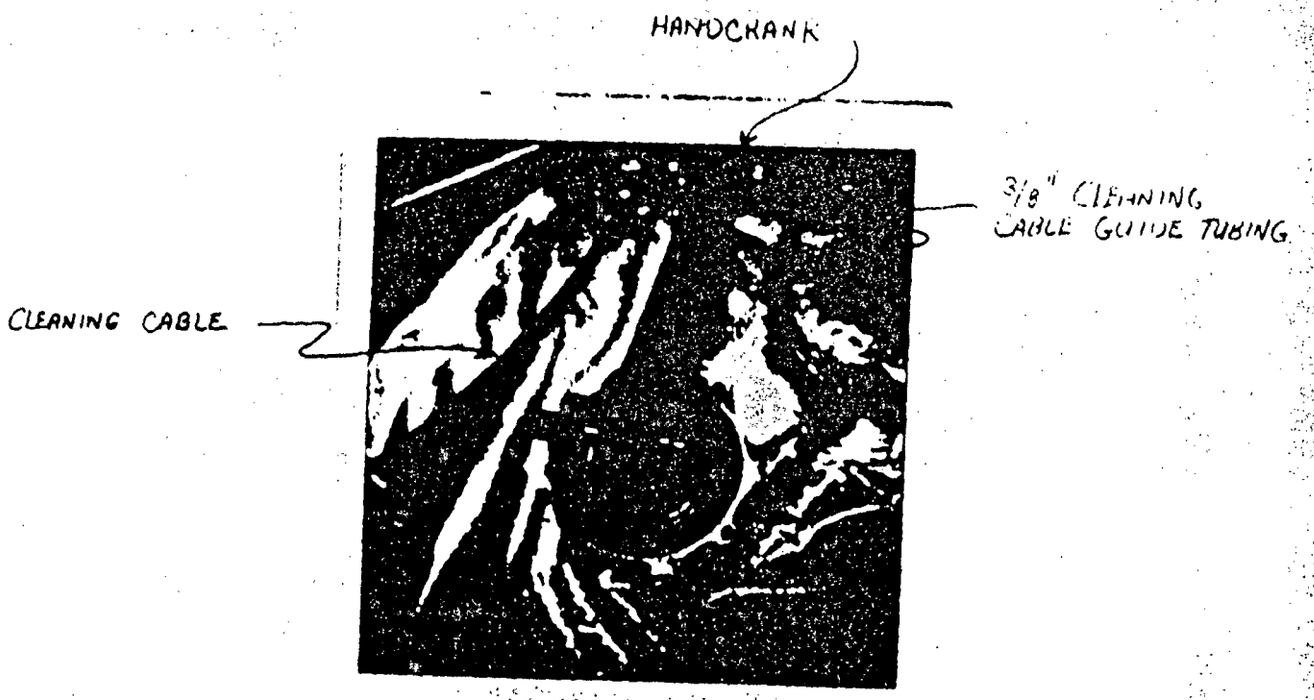
TOP VIEW OF THE INCORE INSTRUMENTATION EQUIPMENT IN THE INSTRUMENT ROOM

FIGURE 10



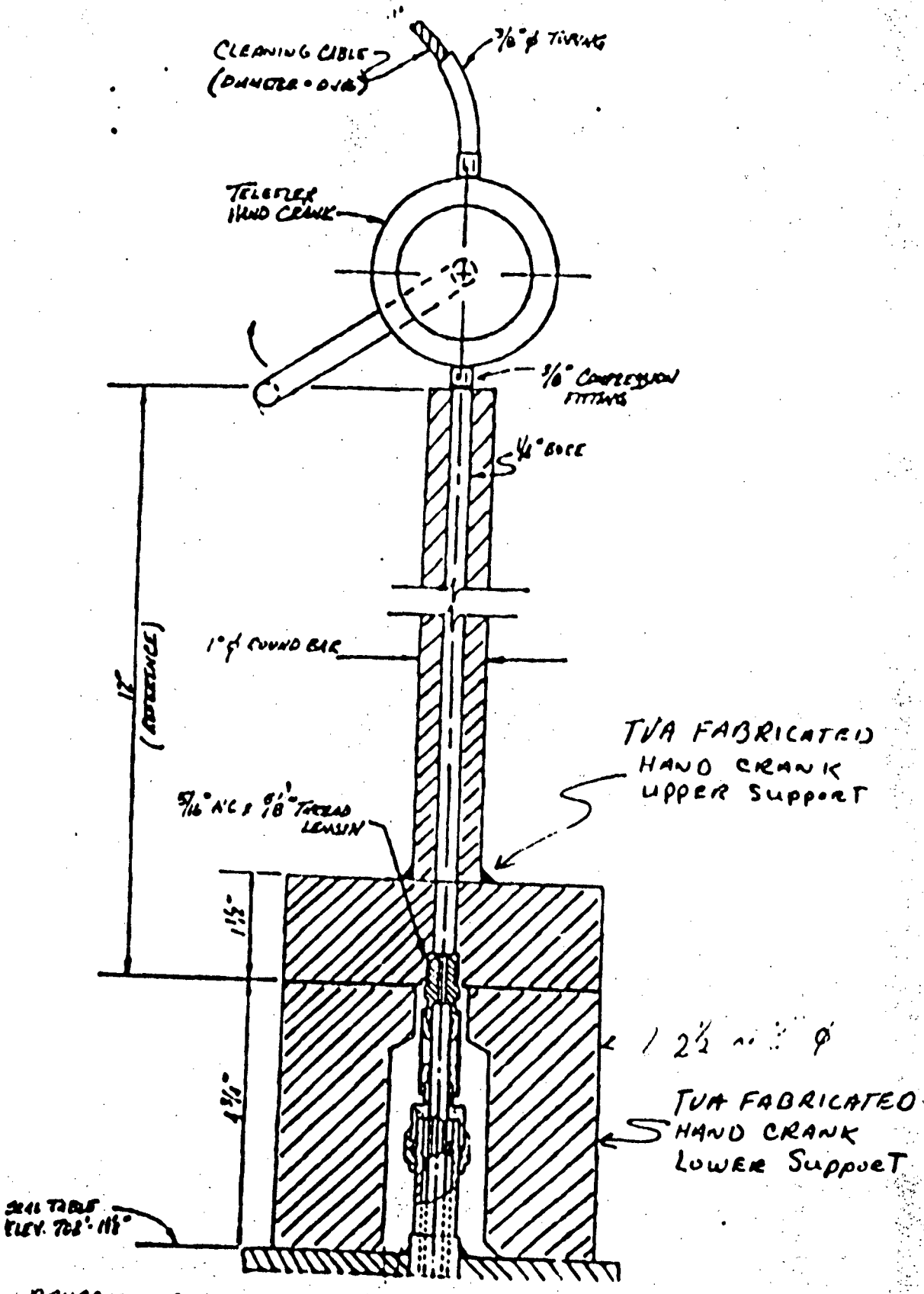
DRYBRUSHING CLEANING TOOL (TELEFLEX SUPPLIED HAND CRANK) AND TVA FABRICATED HANDCRANK UPPER AND LOWER SUPPORT PIECES

FIGURE 11A



TELEFLEX SUPPLIED HANDCRANK
(DISASSEMBLED)

FIGURE 11B



DRYBRUSHING CLEANING TOOL (TELEFLEX SUPPLIED HAND CRANK) AND TVA FABRICATED HANDCRANK UPPER AND LOWER SUPPORT PIECES IN USE WHEN THE THIMBLE TUBE WAS CLEANED

FIGURE 12 A

CLEANING TOOL UPPER BASE SUPPORT (WITH PART OF THIMBLE TUBE D-12 STILL ATTACHED (AFTER THE INCIDENT)

FIGURE 12B

(A) - EVENING SHIFT COORDINATOR - DIRECTING CLEANING ACTIVITIES

(B) - MECHANICAL GENERAL FOREMAN - OBSERVER, WOULD ASSUME THE ROLE OF DIRECTING CLEANING OPERATION IF THE EVENING SHIFT COORDINATOR LEFT THE JOB DUE TO RADIATION DOSE CONSIDERATIONS

(C) - STEAMFITTER FOREMAN - COUNTING NUMBER OF REVOLUTIONS ON HANDCRANK

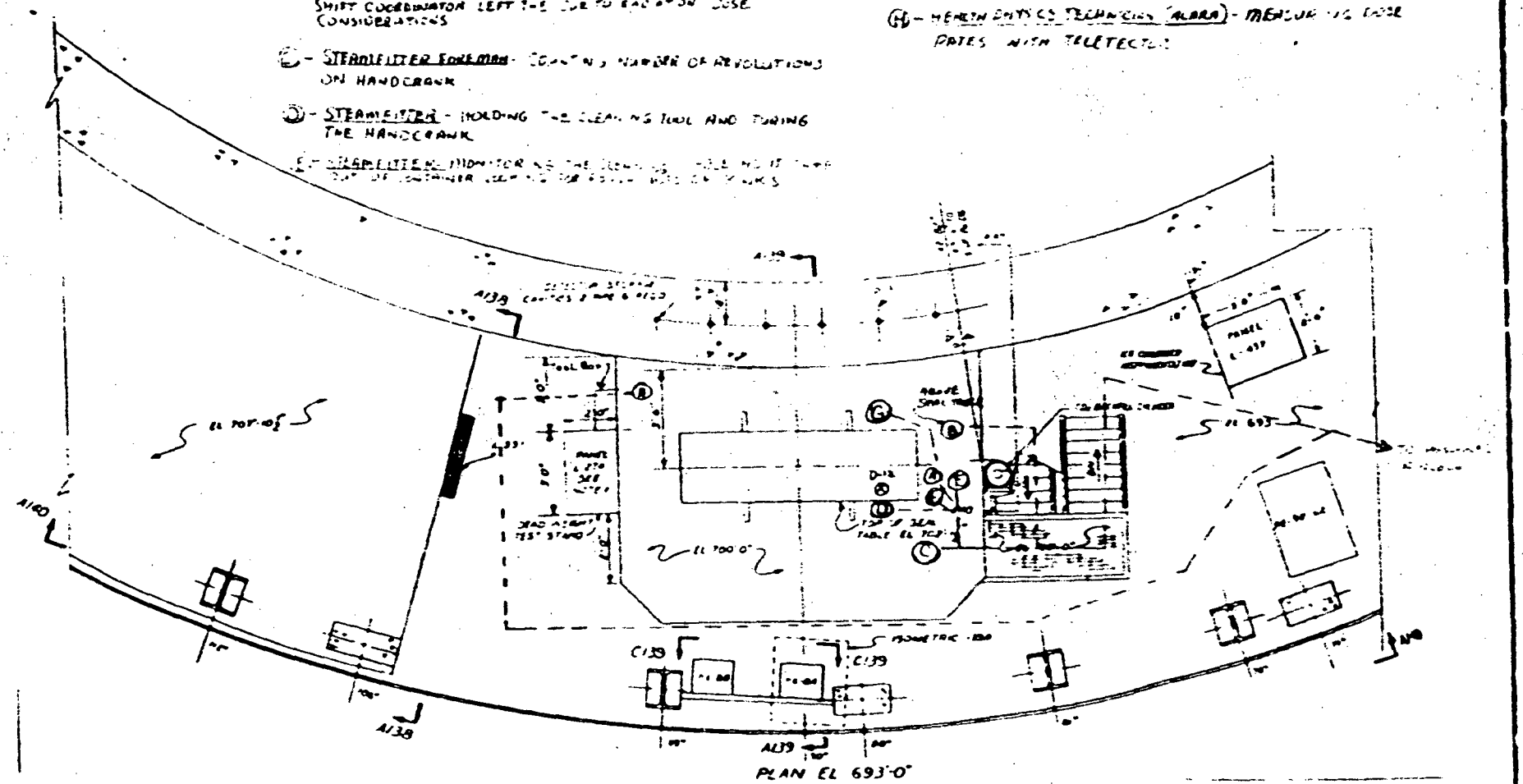
(D) - STEAMFITTER - HOLDING THE CLEANING TOOL AND TURNING THE HANDCRANK

(E) - STEAMFITTER - MONITOR AS THE CLEANING TOOL NOZZLE TOOK OUT OF CONTAINER [illegible]

(F) - STEAMFITTER - FEEDING HOSE INTO THE GUIDE TUBE ON THE CLEANING TOOL

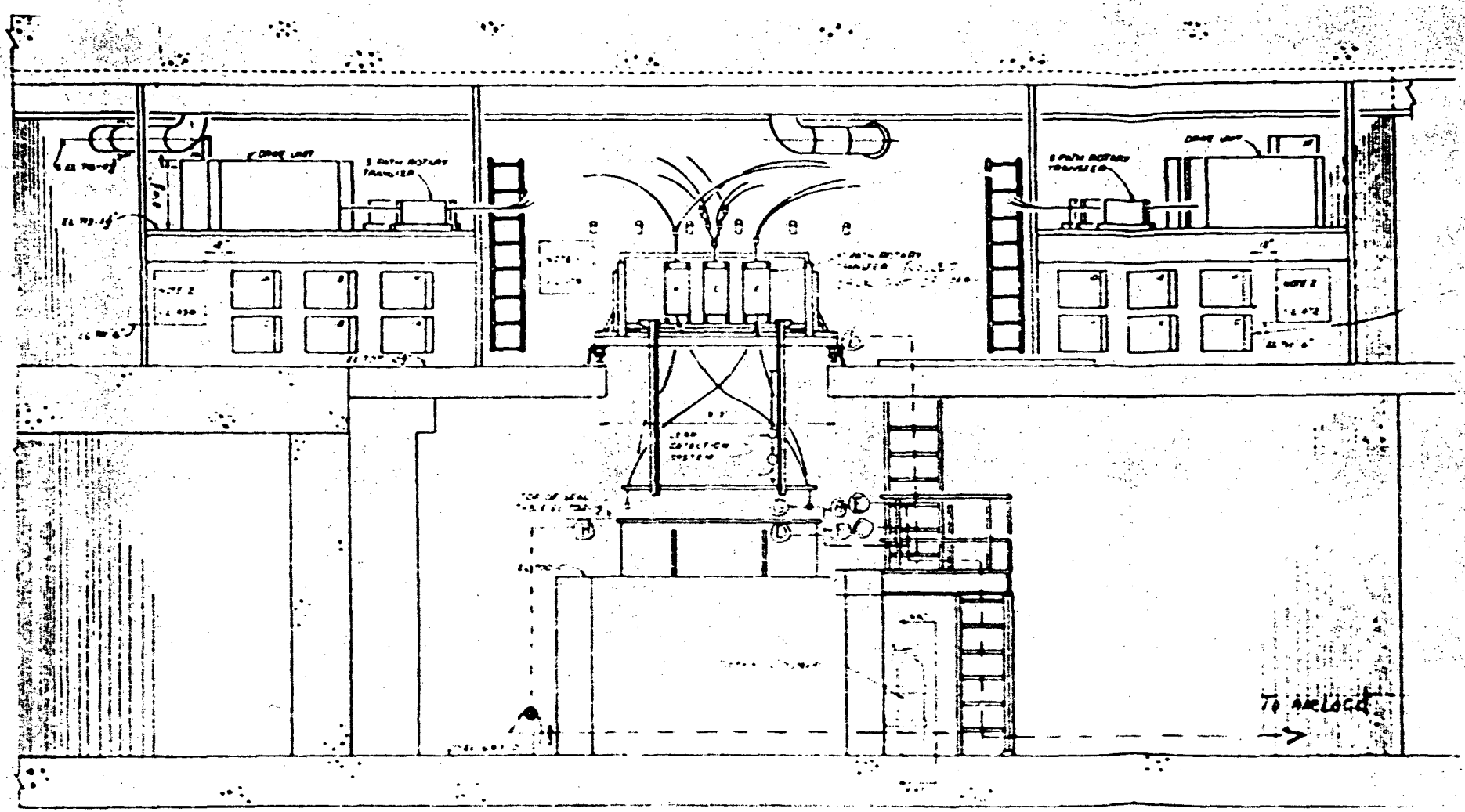
(G) - HEALTH PHYSICS TECHNICIAN - MEASURING DOSE RATES

(H) - HEALTH PHYSICS TECHNICIAN (ALARA) - MEASURING DOSE RATES WITH TELETECTOR



LOCATION OF WORKERS WHEN THE INCIDENT OCCURRED AND EGRESS ROUTES OUT OF THE INSTRUMENT ROOM

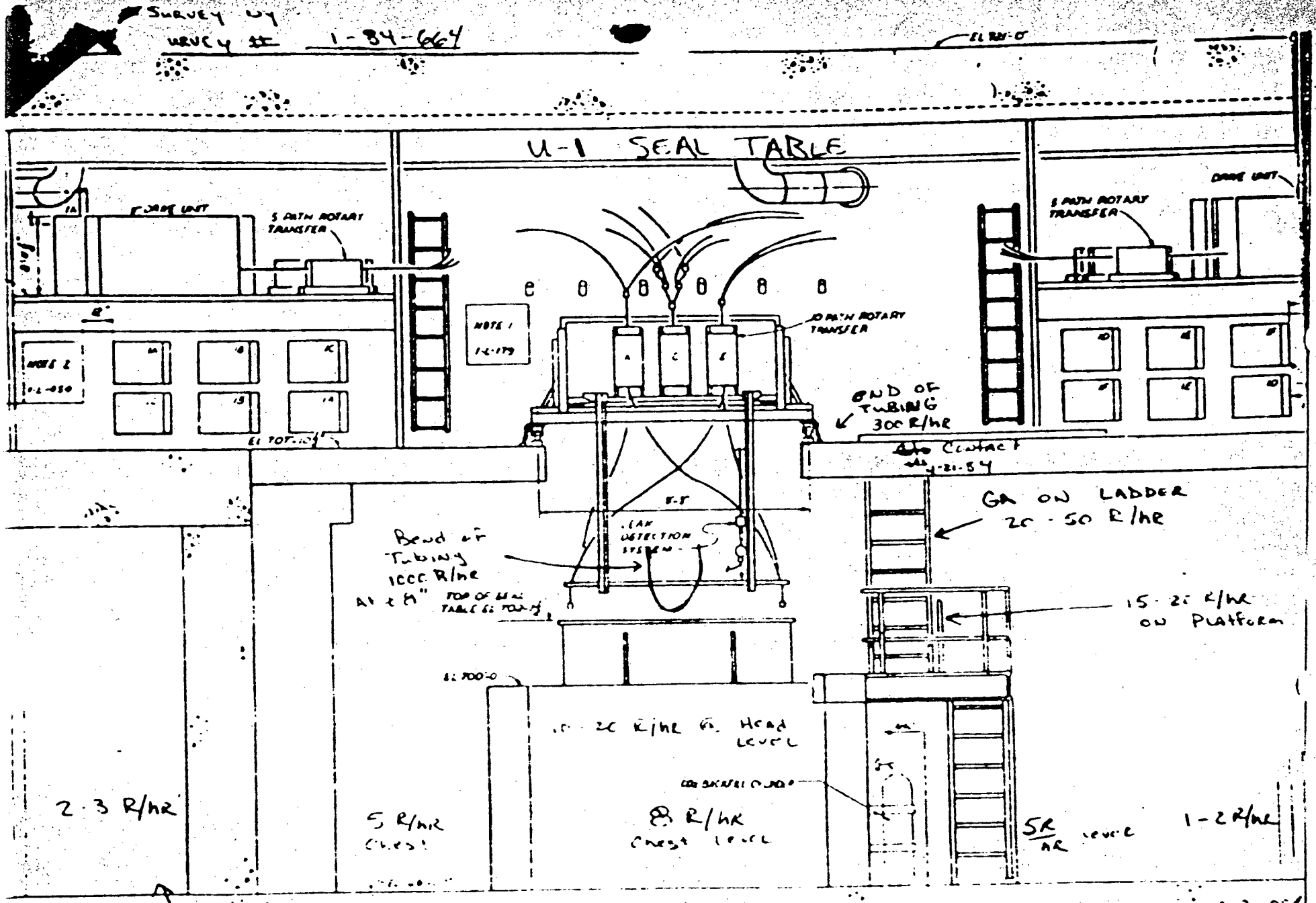
FIGURE 13A



NOTES (B) FLOWED IN FELL OVER
HOUSEKEEPING, HIT TOOLBOX AND EAR
TO THE AIRLOCK

LOCATION OF AIRLOCK AND HOUSEKEEPING

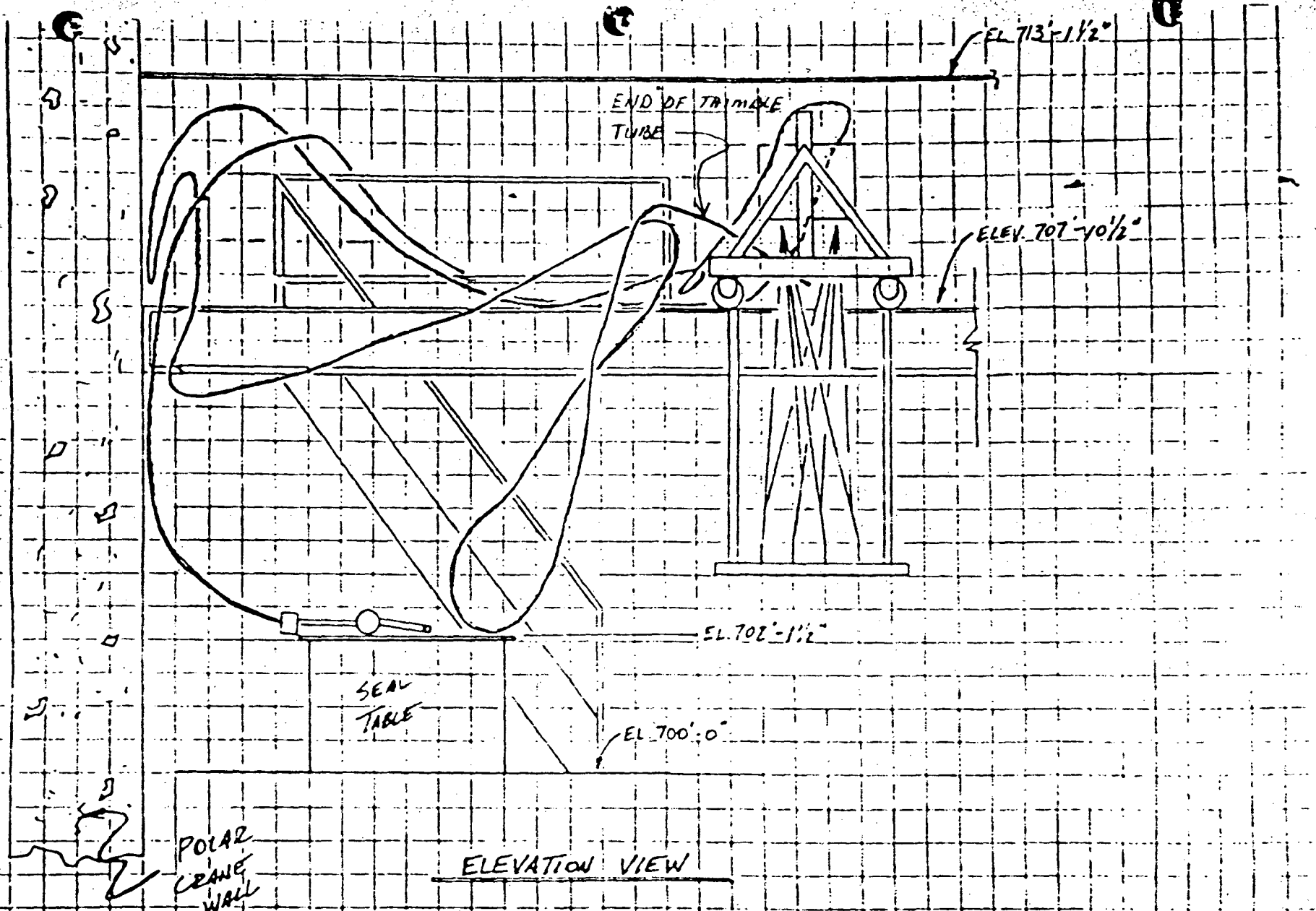
FIGURE 33



1) SMEAR TAKEN ON 693' FLOOR ≈ 60 mR/hr

RADIATION DOSE RATES TAKEN DURING INITIAL ENTRY TO THE INSTRUMENT ROOM

FIGURE 14



ELEVATION VIEW

(LOOKING TOWARD AIRLOCK)

REPRODUCED DURING CONSTRUCTION TRIMALE TUBE

FIGURE 15

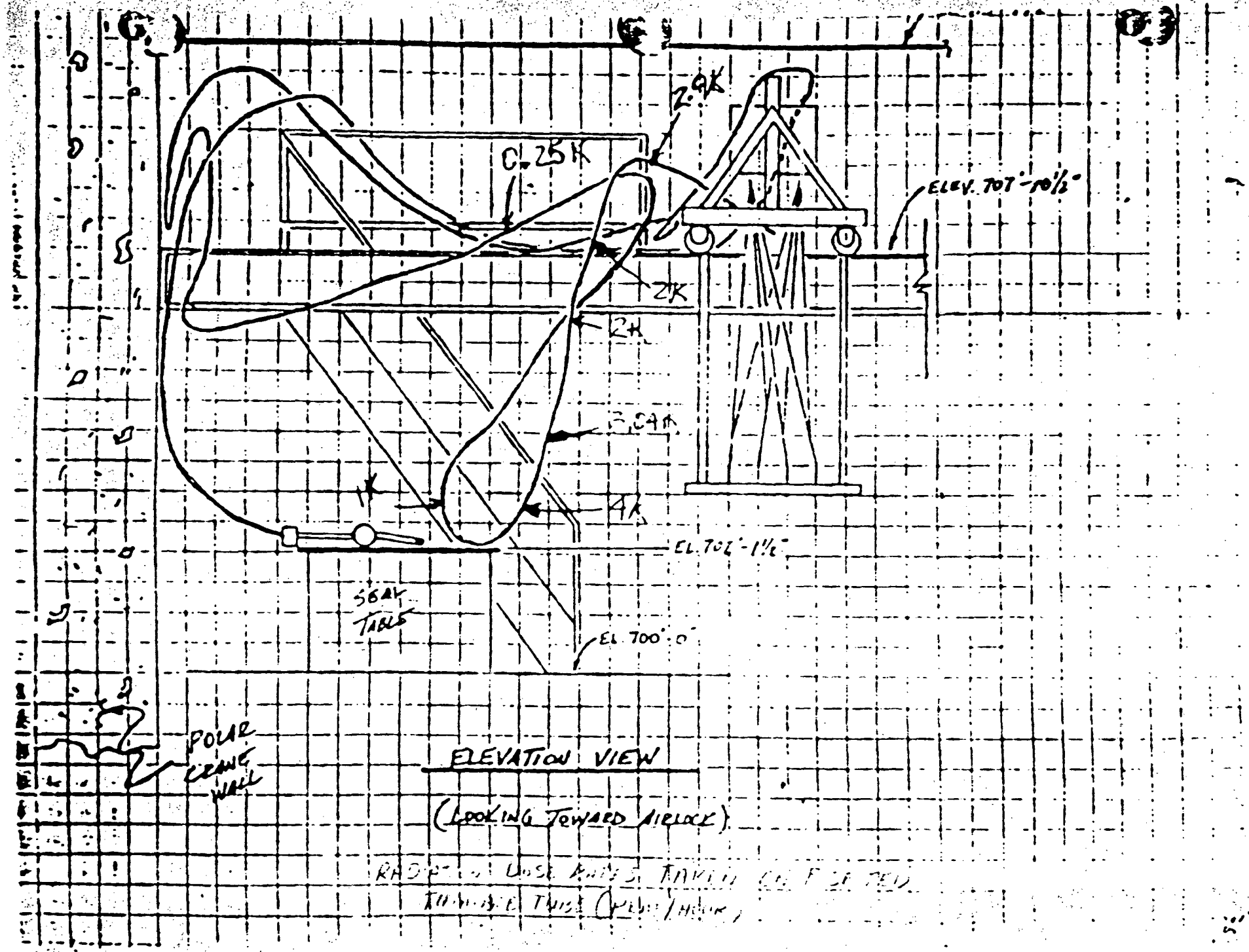

G
G
G
2.0K
0.25K
ELEV. 707'-10½"
2K
2K
3.04K
4K
EL. 702'-1½"
SEAT TABLE
EL. 700'-0"
POLAR CRANE WALL
ELEVATION VIEW
(LOOKING TOWARD AIRLOCK)

FIGURE 16

ATTACHMENT 1

TROJAN NUCLEAR PLANT

FLUX THIMBLE TUBE CLEANOUT AT FULL POWER

Gary Lair

INCIDENT:

On February 1, 1979 during a routine monthly flux map at 100% power (3411 MWT), block thimbles were discovered at 37 of 58 thimble locations. The blockages were at the bend minima.

HISTORY:

From the Trojan startup in December 1975 until the end of Cycle 1 in March 1978, fifty-three full core flux maps and numerous quarter-core maps had been taken with evidence of only one blocked thimble. Little or no neulube had been used.

The plant was shutdown from March 1978 until January 1979 for refueling/technical specifications and licensing intervention regarding seismic integrity. During the refueling, the tubes were evacuated, flooded with carbon dioxide, and capped off.

During the prolonged outage the flux mapping system was exercised every six weeks.

Between the start of the second cycle in January and the February blockage, nine full core flux maps and several quarter-core maps were taken with no significant problems encountered.

WORK PREPARATIONS:

Arrangements were made with Teleflex, the flux mapping system vendor to be on-site to assist in the brushing operation (using a 22-caliber rifle-cleaning, brass brush machined down to 20-caliber and welded to a dummy detector cable with a helical drive unit). Since Beaver Valley had also done a brushing at power, they were contacted to obtain general information. A ten-foot long, 1/2-inch rigid conduit was obtained to facilitate transfer of the wire brush between thimble locations. A funnel was made to facilitate brush entry into thimble.

Radiation control procedures were dev

WORK OPERATION:

The flux mapping system moveable "bird cage" was disconnected and rolled out of the way.

The maintenance man with hand-operated helical drive, positioned himself above the seal table on the upper stationary mounting frame.

WORK OPERATION (Contd.):

He drove the brush through the rigid conduit into each thimble location for brushing.

A Radiation Control Technician used a vacuum cleaner to suck up airborne activity produced when the cable and brush were withdrawn.

The area radiation monitor alarmed when brush emerged from thimble into rigid conduit.

RADIATION CONTROL:

Contact radiation levels at brush: [illegible] average
170R/hr maximum

Prime activation product was copper in brush.
(NOTE: Brush during shutdown cleaning 450 mr on contact)

Contact radiation level on cable: 5R/hr

A vacuum cleaner was used to collect airborne particulate from brush and cable as they were withdrawn from the thimble.

Airborne levels 1.5 MPC were measured when vacuum not held close to source.

Eventually levels were held to 0.3 MPC when improved vacuum cleaner suction maintained.

All personnel wore respirators.

Personnel Exposures

Disassembly	85 m-mr gamma, 10 m-mr neutron
Brushing	2100 m-mr gamma, 74 m-mr neutron highest man - 685 mr, average man - 166 mr
Reassembly	73 m-mr gamma, 17 m-mr neutron
Total Evolution	2267 m-mr gamma, 101 m-mr neutron

(Note total dose for brush and flush at shutdown was 260 mrem.)

SUGGESTIONS:

1. Consider utilizing a brush which does not contain copper if possible. Teleflex recommended bronze, and said stainless-steel is too hard for soft tube. Brush must be brazed on, not just screwed on.

2. Use airfed hood respirators rather than masks for personnel comfort.

3. Use a 12-foot rigid conduit rather than the 10-foot conduit if enough overhead space is available.

4. Consider a motorized helical drive, but be aware of kink potential on hitting obstruction.

5. Provide a support platform for helical drive operator above seal table.

6. Consider routine brushing and flush at refueling shutdowns.

7. Inspect and replace excessively rusting drive cables even if detector still good.

8. During prolonged outage, withdraw detectors back past safety limit switch into heated and shielded drive housing.

9. Use no neulube.

10. Exercise system monthly.

ATTACHMENT 2
INPO ENTRY

RECEIVE MESSAGES>ALL SUBJECT "THIMBLE TUBE", "SEAL TABLE"

TEXT "THIMBLE TUBE", "SEAL TABLE" END

ON 514 HALL (PSE&G/SALM) 03-MAY-83 10:59
SUBJECT: INCORE THIMBLE TUBE BLOCKAGE
SALEM UNITS: 4 LOOP WESTINGHOUSE PWRS

TO ALL OPERATING PLANTS:

SALEM UNITS HAVE ENCOUNTERED PROBLEMS WITH THE INCORE DETECTOR SYSTEM OVER THE YEARS. ONE RECURRING PROBLEM IS THE BLOCKAGE OF THE "THIMBLE TUBES" WHICH ARE THE ACCESS PATH FOR THE MINIATURE DETECTORS TO REACH THE REACTOR CORE. BLOCKAGES TEND TO BUILD UP IN THESE TUBES AT THE POINT WHERE THEY ENTER THE REACTOR VESSEL. AT THIS AREA THE TUBES GO FROM A RELATIVELY COOL TEMPERATURE (~ 100 DEGREES FAHRENHEIT) TO REACTOR COOLANT SYSTEM TEMPERATURES (~ 550 DEGREES FAHRENHEIT). THESE BLOCKAGES BLOCK THE DETECTOR/DRIVE CABLE ASSEMBLIES FROM ENTERING THE CORE REGION. LIKE SALEM, MANY WESTINGHOUSE PLANTS HAVE BEEN IN A CONDITION WHERE THEY COULD NOT MEET THE TECHNICAL SPECIFICATION REQUIREMENT FOR 75% OF THE THIMBLES USEABLE.

TO DISCOVER THE SOURCE OF THESE BLOCKAGES SALEM PERSONNEL RECENTLY REMOVED TWO THIMBLE TUBES FROM UNIT 2 THAT WERE KNOWN TO BE BLOCKED. SEVERAL 3 FOOT LONG SAMPLES OF THESE TUBES WERE OBTAINED CONTAINING THE BLOCKAGE. TECHNIQUES WERE USED TO ENSURE THAT NO WATER ENTERED THE TUBES. SALEM STATION IS PRESENTLY RECEIVING PROPOSALS FOR ANALYSIS OF THESE TUBE SECTIONS. ONCE THE ANALYSIS OF THESE SAMPLES IS RECEIVED WE WILL MAKE THE RESULTS KNOWN VIA NOTEPAD, HOPEFULLY DURING THE SUMMER OF 1983.

ALSO, THESE BLOCKAGES HAVE BEEN SUCCESSFULLY REMOVED AT SALEM WITH THE UNIT AT FULL POWER. BY PROBING THE THIMBLE TUBES WITH A TEST CABLE (NO DETECTOR) THE BLOCKAGES CAN BE KNOCKED LOOSE AND GROUND UP. THIS IS DONE MANUALLY FROM INSIDE THE CONTAINMENT NEAR THE SEAL TABLE. WE REMOVE THE INPUT TUBE FROM A 10-PATH TRANSFER DEVICE AND ATTACH A TELEFLEX HAND DRIVE WITH A TEST CABLE LOADED INTO IT. WE DRIVE THE CABLE TO THE AREA OF THE BLOCKAGE AND "PUSH" IT OUT OF THE WAY. CARE MUST BE TAKEN NOT TO DRIVE THE CABLE INTO THE CORE REGION AS IT WILL ACTIVATE THE CABLE VERY QUICKLY (ABOUT 100 R/HR WHEN RETURNED). WE MEASURE THE CABLE INSERTED LENGTH BY COUNTING THE TURNS ON THE MANUAL DRIVE HAND CRANK (1 TURN PER FOOT OF CABLE). WE DRIVE IT UNTIL WE REACH A DISTANCE THAT IS SIX FEET FROM THE CORE. AFTER RETRACTION THE 10 PATH CAN BE ROTATED TO THE NEXT PATH OF INTEREST AND THE PROCESS REPEATED. THIS IS EASY FOR US SINCE OUR 10 PATH DEVICES ARE LOCATED IN AN AREA OF LESS THAN 1 MR/HR AT FULL POWER.

FOR FURTHER INFORMATION CONTACT JEFF JACKSON, SALEM OPERATIONS, AT (609) 339-4472.

INFORMATION CONTACT:

TVA 64 (08-9-65)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

GNS '841227 050

TO : J. P. Darling, Manager of Nuclear Power, 1750 CST2-C

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

DATE : DEC 27 1984

SUBJECT: WATTS BAR NUCLEAR PLANT (WBN) - OPERATIONAL READINESS REVIEW PHASE III - NUCLEAR SAFETY REVIEW STAFF (NSRS) REPORT NO. R-84-15-WBN

From August 6 to September 7, 1984, NSRS conducted the third in a series of reviews at WBN to determine the operational readiness of the facility. The series of reviews will continue until fuel is loaded in the reactor.

This particular review was conducted during the performance of the mini-hot functional testing program and focused on related Operations, Preoperational Test, Maintenance, and Engineering Sections activities. Adequacy of and adherence to procedures were stressed. Selected portions of the health physics program and actions taken in response to recommendations made in the first two operational readiness reports were also evaluated.

Eight recommendations were made in this report requiring WBN attention. NSRS requests a written response to these items by January 31, 1985. If there are any questions concerning this report, please contact G. G. Brantley or M. S. Kidd at extensions 4815-K or 7637-K respectively.

H N Culver

H. N. Culver

GGB:BJN
Attachment
cc (Attachment):
C. W. Crawford, 670 CST2-C
H. G. Parris, 500A CST2-C
MEDS, W5B63 C-K

NSRS FILE

TABLE OF CONTENTS

GNS '841227 051

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS REPORT NO. R-84-15-WBN

SUBJECT: OPERATIONAL READINESS REVIEW - PHASE III

DATES OF REVIEW: AUGUST 6 - SEPTEMBER 7, 1984

TEAM LEADER: M S Kidd for R. W. TRAVIS — DATE 12-24-84

REVIEWERS: M S Kidd for G. G. BRANTLEY — DATE 12-24-84

M. D. WINGO — DATE 12-21-84

M S Kidd for C. M. KEY — DATE 12-24-84

APPROVED BY: M S Kidd M. S. KIDD — DATE 12-24-84

I. BACKGROUND

This is the third of a minimum of four reviews that will be performed by the Nuclear Safety Review Staff (NSRS) of activities at Watts Bar Nuclear Plant (WBN) to evaluate the operational readiness of that facility. NSRS report Nos. R-84-02-WBN and R-84-05-WBN were the first two in the series. Table I is an updated outline of the NSRS Operational Readiness Review Plan.

II. SCOPE

The activities to be reviewed are generally controlled by the Office of Nuclear Power (NUC PR). Each review is conducted in sufficient detail to facilitate the formulation of an NSRS opinion as to the status of the specific area being reviewed. When the series of reviews has been completed, the status of the specific review areas will be evaluated and the operational readiness of the facility determined. This particular review focused on preoperational testing, maintenance activities, conduct of licensed operations, reactor safety and criticality controls, chemistry control, and health physics. Also, review of actions taken in response to recommendations made in the first two operational readiness reports was performed. This review was performed during the mini-hot functional testing and the NSRS used the tests to serve as a framework to review other activities since these tests simulated actual operating conditions as nearly as possible without using nuclear heat.

III. MANAGEMENT SUMMARY

Within the scope of this review the six areas evaluated were considered adequate with some exceptions. The most significant of these exceptions are summarized below while the specific conclusions and recommendations relating to these program areas are contained in section V of this report.

Preoperational Testing

There appeared to be a philosophical difference when regulatory guides, industry standards, and the N-OQAM were compared with the actual testing experience. Upper tier documents indicated that the testing should be a functional final system checkout while in practice the systems were not as complete and ready for testing as might be expected. Improvement in the conduct of the testing program for unit 2 at WBN is recommended.

Conduct of Licensed Activities

Configuration controls and independent verification of system status were considered to be in need of improvement. Shift and relief turnover of some Operations Section personnel were not conducted in accordance with established requirements. While NSRS recognizes that unit 1 is not licensed for operation, improvement in these areas is recommended in accordance with good operating practices.

Health Physics

The current plant organization does not provide the Health Physics Section independence from operational pressures and the organizational authority to deal directly with all aspects of the plant health physics program. NSRS recommends that the plant organization be revised to have the health physics supervisor report directly to the plant manager.

Previously Identified Open Items

Twenty-eight items requiring WBN attention were made in NSRS reports Nos. R-84-02-WBN and R-84-05-WBN, issued in April and June of 1984. Of these 28 items 16 have been closed and the remaining items remain open pending further review by the NSRS or further action by WBN. Of those items remaining open the NSRS considers R-84-05-WBN-08, R-84-05-WBN-15, R-84-05-WBN-17, and R-84-WBN-24 to be the most significant. These open items involve high density fuel storage racks attenuation testing, material inspection during construction/modification/ maintenance activities, the two-year review cycle for procedures and instructions, and interface reviews after unit 1 fuel loading.

IV. STATUS OF PREVIOUSLY IDENTIFIED OPEN ITEMS

A. R-84-02-WBN-01, Noncompliance with TVA Commitments and NUC PR Requirements for GET Training

In a plant-level document, NUC PR had exempted the plant superintendent and the assistant plant superintendents (the wording did not reflect the most recent reorganization) from certain General Employee Training (GET) courses required by ANSI 18.1. The NSRS recommended that this exception either be removed or a formal exception be obtained from established requirements. NUC PR and NSRS still are in disagreement on this concern. This item will remain open pending further discussion. See section VI.A.1 for details.

B. R-84-02-WBN-02, Expansion of the FQE Survey No. 3QT(a)

NSRS recommended that the FQE Survey No. 3QT(a) checklist be expanded in scope. It was also recommended that the survey be put into compliance with the plant-level document (AI-10.1) regarding an acceptable timeframe for initial training. Based on revision of the survey checklist this item is closed. See section VI.A.2 for details.

C. R-84-02-WBN-03, Problems with Scheduling and Recordkeeping Associated with the Health Physics and Security Bypass Examinations (GET 2.35 and 3.15)

The NSRS had found a large number of personnel delinquent in the bypass exam for health physics and security. Most of the delinquencies were caused by people taking the training again instead of the bypass exams because the failure rate of the bypass exams

was too high. These people should not have been listed as delinquent. This was only a recordkeeping error and NUC PR took proper corrective action. The other delinquencies were caused by too few classes being scheduled. The schedule was changed in accord with NSRS recommendations. This item is closed. See section VI.A.3 for details.

D. R-84-02-WBN-04, Enhanced Employee Awareness of TVA's Policy on Expression of Staff Views and Preferred Methodology for Reporting Nuclear Safety Concerns

The NSRS concern was that TVA employees interviewed did not know that they were encouraged by TVA management and policy to take nuclear safety concerns to the NSRS before taking them to the NRC. NUC PR responded that no corrective action was required as the information is given to employees during General Employee Training (GET). However, the GET courses contained no reference to NSRS. The NSRS contends that a GET course should include the stated TVA management policy concerning the NSRS and its role in employee concerns. This item remains open. See section VI.A.4 for details.

E. R-84-05-WBN-01, Definition of Responsibilities and Authorities for Administration of the STA Program

AI-2.16 had been revised to clearly establish the authority and duties for administering the STA program, as recommended by NSRS. This item is closed. See section VI.A.5 for details.

F. R-84-05-WBN-02, Station STA Training

The station Shift Technical Advisor (STA) training had not been completed. STAs were undergoing station training with September 1 as the projected completion date. This item remains open until the training is completed and the NSRS reviews the training records. See section VI.A.6 for details.

G. R-84-05-WBN-03, Annual STA Retraining

Formal records to indicate the status of retraining had been added to STA training records as recommended. This item is closed. See section VI.A.7 for details.

H. R-84-05-WBN-04, Certification of WBN STAs

Certification records had been placed in STA training files. This item is closed. See section VI.A.8 for details.

I. R-84-05-WBN-05, STA Plant Familiarization Walkthrough

The Engineering Section Instruction Letter was revised to upgrade the walkthrough portion of the STA program. This item is closed. See section VI.A.9 for details.

J. R-84-05-WBN-06, Divergence from the Intent of the STA Program to Provide a STA Corps Independent of Commercial Operations

AI-2.16 indicates that the primary responsibility of the STA on shift is the performance of STA duties. Administratively the NSRS concern has been satisfied. The STA program performance will be evaluated during the startup of unit 1. This item is closed. See section VI.A.10 for details.

K. R-84-05-WBN-07, Two-Party Verification For Fuel Loading

TI-1 has been revised to delineate "separate and independent parties" and the NSRS concern is satisfied. This item is closed. See section VI.A.11 for details.

L. R-84-05-WBN-08, High Density Fuel Storage Racks Attentenuation Testing

EN DES has yet to respond to the requested justification of the 15-percent sample size. This item remains open. See section VI.A.12 for details.

M. R-84-05-WBN-09, Surveillance Requirements for Changing Modes of Operation

Subsequent review and discussions with the WBN staff on their response indicated that the NSRS concern was adequately addressed by the present system. This item is closed. See Section VI.A.13 for details.

N. R-84-05-WBN-10, Workplan Quality Assurance (QA) Requirements

The NSRS had discovered discrepancies between Engineering Change Notices (ECNs) and workplans to implement the ECN. The site response to correct these discrepancies appeared adequate. Review of a limited number of workplans issued since June 25, 1984 did not reveal any discrepancies where the ECN was marked "QA" and the workplan was marked to indicate that "QA" did not apply. This item will remain open until more workplans are initiated and can be reviewed for compliance with the corrective action. See section VI.A.14 for details.

O. R-84-05-WBN-11, Workplan Functional Tests

The NSRS did not believe that adequate functional tests were being performed after workplan completion. The site response did not indicate any corrective action to be performed. Furthermore, a review of approximately ten workplans by the NSRS did not indicate any additional problems. This item will remain open pending a more detailed review of additional workplans to ensure that they contain instructions for functional tests (when required) or references to approved instruction. See section VI.A.15 for details.

P. R-84-05-WBN-12, Supplemental Information Added to Workplans

The NSRS review had discovered workplans with information added in ink without a date or initials. Review of workplans indicated that supplemental information added to workplans was being initiated by the responsible engineer and was being dated. This item will remain open until a more thorough review of additional workplans can be performed to ensure continued compliance. See section VI.A.16 for details.

Q. R-84-05-WBN-13, Plant Modifications Made by Use of a Maintenance Request (MR)

The NSRS reviewed an MR which appeared to be installing a plant modification, a practice which is not allowed. However, the MR in question was issued to install a temporary alteration in accordance with AI-2.15. The temporary condition was then made permanent by the issuance of an ECN. This item is closed. See sections VI.A.17 and VI.B.2 for details.

R. R-84-05-WBN-14, Inspector Certification Records

Inspection certification records were either not onsite or were incomplete. The site response indicated that an interactive computer system would be obtained to improve the certification documentation process. The response appeared adequate. This item will remain open until implementation of corrective action can be reviewed. See section VI.A.18 for details.

S. R-84-05-WBN-15, Material Inspection

The NSRS did not believe that inspection activities for plant modifications were equivalent to the plant construction inspections. The response to allow a cognizant individual to verify material in lieu of a QC holdpoint is unacceptable. In addition, the use of surveys by FQE to verify installation of material does not meet the requirements cited in NSRS report R-84-05-WBN. This item will remain open. See section VI.A.19 for details.

T. R-84-05-WBN-16, Records

The NSRS reviewer found that there was a problem with record retrieval. An As-Constructed Drawing Task Force is scheduled to review this problem in December 1984. Any recommendations made by the As-Constructed Drawing Task Force will be evaluated upon completion of the task force's review for identification and retrievability of CONST workplan records. Also, the site, with the aid of OQA, has implemented a record retrieval procedure. This item will remain open pending NSRS review of the procedural implementation. See section VI.A.20 for details.

U. R-84-05-WBN-17, Two-Year Review Cycle for Procedures and Instructions

The NSRS did not believe that the WBN procedure review program was adequate in that a successfully documented performance of an instruction was considered to be an instruction review. NUC PR stated they believed it met the intent of the review requirements but that once the QA reorganization was completed a response would be made by QA. NSRS believes that the NUC PR position is in contradiction to regulatory and corporate document requirements. This item will remain open until AI-3.1 is revised to comply with the requirements to upper-tier documents. See section VI.A.21 for details.

V. R-84-05-WBN-18, Field Quality Engineering (FQE) Review of Procedures and Instructions

A quality assurance program instruction (AI-3.1) had been written by FQE. The required FQE review of this instruction was performed by the same individual who had written it. The NUC PR response was basically correct. The NSRS position is that in the future FQE procedures and instructions should have the documented review performed by someone other than the author. This item is closed. See section VI.A.22 for details.

W. R-84-05-WBN-19, Operator Response to Critical Alarms Before Licensing

The NSRS did not believe that operator response to an alarm indicating water in the spent fuel pit was adequate. The operators were instructed on response to alarms. The NSRS is satisfied with this response to the concern. This item is closed. See section VI.A.23 for details.

X. R-84-05-WBN-20, Interface Study Report

The NSRS recommended that a new interface study be conducted. The plant disagreed. The NSRS was aware that the Interface Study Report was not the controlling document and described the controlling documents for interface control in its report. The NSRS found enough problems in the interface control to believe another study should be performed. The NSRS, however, considers the response to other unit interface control concerns at the plant to be adequate to consider this item also closed. See section VI.A.24 for details.

Y. R-84-05-WBN-21, Interface Hold Orders

The NSRS agreed with the statement made in the response. However, this statement did not address the NSRS concern. The NSRS noted several instances of test procedures not installing interface points that were recommended in the interface study report. Valves were closed but no interface control was used. Since the

NSRS review these points had been put under the interface control. The written response was inadequate but the actions were adequate. This item is closed. See section VI.A.25 for details.

Z. R-84-05-WBN-22, Marked-up Drawings for Interface Points

The NSRS had twice requested the shift engineer (SE) to find the interface drawings that were required to have been submitted to him. The SE searched his office and could not find them. From the response, the drawings were in the SE office and the problem appeared to be with the awareness of the SE and not with the preoperational section. The drawings are now in a well-marked book and at least two SEs are aware of the interface drawings. This item is closed. See section VI.A.26 for details.

AA. R-84-05-WBN-23, Interface Points in Unit 2 Reactor Protection Cabinets

NSRS recommended unit 2 reactor protection cabinet be transferred to NUC PR as was also recommended in the Interface Study Report. The plant responded that they had been. Upon further review, the NSRS discovered that the unit 2 SSPS cabinets had not been transferred. Some wiring inside the cabinets had been, but NUC PR did not have control of these cabinets. The transfer was made the week of August 20, 1984. This item is closed. See section VI.A.27 for details.

BB. R-84-05-WBN-24, Interface Review After Unit 1 Fuel Loading

The NSRS agreed that controls had been put in action to ensure that interface points will be properly established for the unit 1 fuel loading. However, the NSRS concern was with keeping them in place for the timeframe between unit 1 and unit 2 fuel loading (approximately two years). Due to the importance of the interface system to plant and personnel safety the NSRS continues to recommend that physical reviews of interface points be formally scheduled and accomplished on a periodic basis (at least every six months) after unit 1 fuel loading. This item remains open. See section VI.A.28 for details.

V. CONCLUSIONS AND RECOMMENDATIONS - NEW REVIEW AREAS

A. R-84-15-WBN-01, Preoperational Testing

Conclusion

The preoperational testing was being conducted by qualified personnel in accordance with established test instructions. Test instructions were adequate and administrative controls were being properly applied, but there were more deficiencies, changes to procedures, and inoperable equipment than would be expected with the optimum program as described in the upper tier documents

governing the testing program. A possible root cause that promotes these type of problems is the philosophy of testing as soon as possible any part of a system that can be tested even if much simulation is required. See section VI.B.1 for details.

Recommendation

To improve the quality and efficiency of testing activities NUC PR should assure that systems are essentially completed, checked out, and tested by OC prior to transfer. System operability and test instruction adequacy should be verified before conducting the formal testing activity. These actions should enhance compliance with the intent of upper tier governing documents.

B. Maintenance Activities

1. Instrument Maintenance

Conclusion

Implementation of the Instrumentation Section surveillance program was checked during this review by observing the performance of Surveillance Instructions (SIs). The activities associated with the conduct of the surveillance program appeared adequate. NSRS was concerned that there were many instrument maintenance surveillance instructions that must be corrected and performed before fuel load but site management was aware of the problem and was taking appropriate action.

2. Mechanical Maintenance

R-84-15-WBN-02, Storage of Maintenace Requests (MRs)

NSRS reviewed the MR process in the Mechanical Maintenance Section for compliance to procedural requirements. It appeared that MRs were being handled in accordance with those requirements. However, the NSRS reviewer did identify an item of concern involving MRs awaiting final review.

Conclusion

Completed MRs (awaiting final review) were being kept at the field quality engineering (FQE) reviewer's desk for up to three days without any precautions being taken to prevent possible damage to the documents or their loss. See section VI.B.2 for details.

Recommendation

MRs (and other quality-related documents) should be stored in a suitable environment to prevent possible damage or loss while awaiting final FQE review.

C. Conduct of Licensed Operations

1. R-84-15-WBN-03, Configuration Control and Independent Verification by Operations Section

Conclusion

The configuration control and independent verification procedures were adequate to maintain the required status of the plant systems. However, the execution of those procedures was lax and faulty. The actual status of some systems was not recorded and verified by the required checklists and signoffs. See sections VI.B.3.a and VI.B.3.b for details.

Recommendation

NSRS recommends that OSL-2A be reviewed with all operations staff in conjunction with the performance of SOIs, GOIs, and other plant procedures to maintain 100 percent control of system alignments.

NSRS recommends that AI-2.19 be reviewed with all operations staff to stress the requirements and importance of independent verifications of required equipment.

2. R-84-15-WBN-04, Shift and Relief Turnover of Operations Section

Conclusion

The shift and relief turnover procedure, AI-2.10, was adequate to maintain the transfer and flow of information between working shifts. The execution of this procedure was adequate between the AUOs and UOs. The turnover between the ASEs and SEs was inconsistent with requirements and needed improvements to comply with AI-2.10. See section VI.B.3.b for details.

Recommendation

NSRS recommends that AI-2.10 be reviewed with all operations staff to emphasize the requirements and importance of shift and relief turnover.

D. Reactor Safety and Criticality Controls

1. Fuel Handling Operations

Conclusion

Fuel handling and training were being accomplished in accordance with TI-2, "SNM Control and Accountability System." Preoperational testing was being used as a training function. Plant activities in this area appeared adequate. See section VI.B.4.a for details.

2. Reactor Safety Controls

Conclusion

The area of reactor criticality control reviewed for this report, involving a modification to the reactor protection system, was adequate. See section VI.B.4.b for details.

E. Chemistry Control

Conclusion

The implementation of the chemistry control program observed during the heatup phase to 250°F was adequate. Procedures and instructions were in place and were being followed by Chemical Unit personnel. Personnel training on Surveillance Instructions was continuing. See section VI.B.5 for details.

F. Health Physics

1. Operational Quality Assurance Branch (OQAB) Activites

Conclusion

The OQAB had performed an operational readiness review of the WBN radiological protection program at the request of WBN management. That review was effective in identifying problems in the radiation protection program in the area of health physics instrumentation and equipment, health physics staffing, and administrative controls. The plant health physics staff was actively addressing the problems identified by the OQAB review team. See section VI.B.6.a for details.

2. R-84-15-WBN-05, Field Quality Engineering (FQE) Activities

Conclusion

FQE checklists for surveillance of health physics activities had not been prepared, and surveillance of health physics activities had been minimal. Appropriate corrective actions had been taken by the health physics staff in response to Deviation Reports (DRs) written by FQE. See section VI.B.6.b for details.

Recommendation

FQE surveillance checklists should be prepared and surveillance scheduled during the fuel loading and startup phases of unit 1 to assure that the radiation protection functions are being performed in compliance with established program requirements and to determine the quality of that performance.

3. R-84-15-WBN-06, Health Physics Organization

Conclusion

The plant organization does not provide the plant Health Physics Section independence from line operational pressures and organizational flexibility to deal directly with all aspects of the plant health physics program. The reporting chain of the Health Physics Section supervisor is through the Operational and Engineering superintendent to the plant manager. See section VI.B.6.c for details.

Recommendation

NSRS recommends that the plant organization be revised to establish the reporting chain of the Health Physics Section supervisor directly to the plant manager.

4. Health Physics Section Staffing

The Operation Unit of the Health Physics Section was expected to be adequately staffed with well qualified personnel by the end of November 1984 to support the startup and operation of unit 1. However, as earlier identified by OQAB, the Technical Unit was still not staffed to perform the functions planned for that unit and could not support the startup and operation of unit 1. The plant staff was taking appropriate actions to make personnel selections to fill these positions. See sections VI.B.6.c and VI.B.6.d for details.

5. R-84-15-WBN-07, Health Physics Program Administrative Controls

Conclusion

Administrative controls had been provided to control radiation protection activities addressed by section 6 of the draft WBN Technical Specifications. The additional detailed procedures required to instruct the health physics staff in implementation of the health physics program were issued or would be issued in the immediate future. The Special Work Permit/Radiation Work Permit (SWP/RWP) program had only recently been significantly revised. See section VI.B.6.e for details.

Recommendation

As the RWP system is new and is the primary administrative system for controlling personnel exposure to radioactive materials and radiation, awareness seminars for the RWP program should be provided to the plant staff prior to the startup of unit 1.

6. Health Physics Instrumentation, Equipment, and Facilities

Conclusions

The WBN portable survey instrumentation and air samplers met the requirements as specified in the FSAR and the program for control of the instrumentation was adequate. Some other required equipment and facilities had not arrived onsite. The health physics staff was aware of these inadequacies and was expediting procurement of the equipment and final construction of the facilities. The required equipment not yet received that would significantly impact the health physics program are the TLD processing equipment and the C-zone supplies. Although not planned, TLD services could be provided by the Radiological Hygiene Staff (RHS) and C-zone supplies could be borrowed from other NUC PR facilities to support startup of unit 1. See section VI.B.6.f for details.

7. R-84-15-WBN-08, Health Physics Section Personnel Stopwork Responsibility and Authority

Conclusion

The health physics section personnel do not have sufficient authority to terminate an activity involving imminent danger conditions or situations. RCI-1 indicates that termination of an activity will be accomplished through the plant manager or his designated representative. See section VI.B.6.g for details.

Recommendation

The stopwork responsibility and authority statements in RCI-1 for imminent danger conditions should be revised to specify that health physics personnel have the responsibility and authority to stop work or order an area evacuated when, in their judgment, the radiation protection conditions warrant such an action and such actions are consistent with plant safety. It should be clear that only the Plant Manager, Health Physics Section supervisor, or their designated representatives on backshifts can overrule a stopwork action initiated by health physics personnel.

8. FSAR Description of the WBN Health Physics Program

Conclusion

The WBN FSAR did not accurately depict the planned WBN health physics program. The plant health physics staff initiated actions to review the respective sections of the FSAR and submit revisions as necessary before the end of the NSRS review. The NSRS will review the respective sections

of the revised FSAR at a later date to determine if it accurately depicts the implemented WBN health physics program. See section VI.B.6.h for details.

VI. DETAILS

A. Previously Identified Open Items

1. R-84-02-WBN-01, Noncompliance With TVA Commitments and NUC PR Requirements for GET Training

TVA is committed to Regulatory Guide (RG) 1.8, "Personnel Selections and Training," and ANSI 18.1, "Selection and Training of Nuclear Plant Personnel," through the TVA Topical Report. ANSI 18.1 states that all persons regularly employed in the nuclear power plant shall be GET trained. WBN AI-10.1 exempts plant superintendents and assistant plant superintendents from all initial training and retraining on GET (except GET 2 and 3). NSRS recommended that the exemption should be removed from AI-10.1 to be in full compliance with TVA commitments and NUC PR requirements or formal exemption to the commitments and requirements should be obtained.

WBN had not revised AI-10.1 to be in full compliance with TVA commitments nor had they requested formal exemption from established requirements. The WBN staff feels that they meet the intent of ANSI-18.1 in that the plant manager, assistant plant manager, and superintendents are, by virtue of their positions, knowledgeable in the areas of concern. The WBN staff feels that whether this knowledge is obtained through formal training courses or otherwise is not relevant. While this may be true, TVA has not taken formal exception to the requirements committed to by TVA for GET and is therefore subject to violation of commitments to the NRC in the event appointed plant managers or assistants do not receive the required training. This item remains open until the exemption is removed from AI-10.1 or formal exception is taken in all applicable documents.

2. R-84-02-WBN-02, Expansion of the FQE Survey No. 3QT(a)

The NSRS found that the FQE survey was not representative of GET training status, did not survey status of retraining, and the specified timeframe was inconsistent with AI-10.1. NSRS recommended that the survey checklist be expanded to better represent the overall status of compliance with AI-10.1. WBN had expanded the FQE survey to better represent the overall status of compliance with AI-10.1. A formal response from NUC PR denied part of the finding but from further NSRS review it was determined that the recommended changes had taken place after the intital NSRS review and before the writer of the response had reviewed the area of concern. This item is closed.

3. R-84-02-WBN-03, Problems With Scheduling and Recordkeeping Associated With the Health Physics and Security Bypass Examinations (GET 2.35 and 3.15)

There were indications that all six plant sections reviewed by NSRS were delinquent for the bypass examinations for health physics and security. The root cause appeared to be scheduling and recordkeeping. NSRS recommended that sufficient bypass examinations should be scheduled and/or the methodology for updating the "Train Report" should be adjusted to give credit for GET 2.35 and 3.15 when personnel take retraining courses in lieu of the bypass examinations.

The NSRS recommendations had been implemented as GET 2.35 and 3.15 were scheduled every Thursday and Friday of each week for the timeframe of July 2-September 28, 1984. Additionally, when retraining is successfully completed, the due dates for the bypass exams for health physics and security are now automatically updated. The formal NUC PR response did not address the concern raised by NSRS. This was possibly caused by the NUC PR review being made after corrective action had occurred. However, the plant actions were adequate for both NSRS recommendations. This item is closed.

4. R-84-02-WBN-04, Enhanced Employee Awareness of TVA's Policy on Expression of Staff Views and Preferred Methodology for Reporting Nuclear Safety Concerns

Employees selected at random and interviewed by NSRS were generally unaware of the TVA policy for expression of staff views and the preferred method for reporting nuclear safety concerns as defined in TVA Code II, Expression of Staff Views. NSRS recommended that training and retraining in the form of GET should be provided to all WBN employees to enhance their awareness of TVA's policy for expression of staff views and to ensure that they are aware of the preferred method for reporting nuclear safety concerns.

The plant response indicated that no corrective action is necessary as the information is given to employees verbally in GET-2.1 (HP) and GET-4 (QA and QC) classes as part of taped script with slide show. Slides 43 and 47 of the GET-4 presentation did address reporting adverse plant conditions to employee supervisors and the direct access to NRC but contained no reference to an internal system at the plant or to the NSRS. The instructor notes for GET-2.1 discussed this issue in a like manner. However, the employee responsibilities for reporting nuclear safety concerns along with the TVA preferred reporting methodology as defined in TVA Code II were not adequately addressed in GET-2.1 and GET-4. As the reporting of concerns is covered in GET it should be covered properly. This item remains open until the policy and preferred procedures defined in TVA Code II have been

adequately addressed in GET training and the procedures defining the process at the site are adequate.

5. R-84-05-WBN-01, Definition of Responsibilities and Authorities for Administration of the STA Program

The responsibilities and authority for administration of the STA program were not defined in any formal plant document other than the MAS goals of the Engineering Section and Reactor Engineering Unit supervisors. NSRS recommended that the authority and duties for administering the STA program be clearly established in AI-2.16, "Shift Technical Advisors."

AI-2.16 had been revised to clearly establish the authority and duties for administering the STA program as recommended by NSRS. This item is closed.

6. R-84-05-WBN-02, Station STA Training

None of the STAs had completed the Results Section Training, RST-26 "Station Shift Technical Advisor Training," at the time of the original review of this area. NSRS recommended that RST-26 training be completed prior to assignment of the STAs to shift duties for the first time.

RST-26 training had not been completed, although STAs were receiving the required training at the time of this NSRS review. The STAs were scheduled to complete the training by September 1, 1984. This item remains open until the training is completed and the NSRS reviews the training records.

7. R-84-05-WBN-03, Annual STA Retraining

At the time of the first review, all STAs were reported to be up to date with annual retraining requirements. However, no formal plant training records were available in the plant files that documented the up-to-date status. NSRS recommended that the formal plant training records should be maintained current to indicate the accurate status of the STA retraining.

Formal records (TVA 1453) to indicate the status of the STA retraining had been added to the STA training records. This item is closed.

8. R-84-05-WBN-04, Certification of WBN STAs

Certification records for only 6 of the 11 qualified plant STAs were in the plant training records at the time of the original review. NSRS recommended that certification records for all qualified STAs be added to the formal plant training records.

Certification records had been placed in the training records of the 11 qualified STAs. This item is closed.

9. R-84-05-WBN-05, STA Plant Familiarization Walkthroughs

The walkthrough portion of the STA training program had been conducted by AUOs. This was determined appropriate for certain portions of the walkthrough program but not appropriate for other portions. NSRS recommended that the STA walkthrough program be upgraded to require that SROs or qualified STAs be required to conduct walkthroughs for certain portions of the STA training program.

The follow-up review of this item revealed that ENSL-R4 had been revised to upgrade the walkthrough portion of the STA program as recommended. This item is closed.

10. R-84-05-WBN-06, Divergence from the STA Program to Provide a STA Corps Independent of Commercial Operation

The WBN STA program had diverged from the original NRC intent that a corps of trained and experienced STAs be available to provide independent operational and accident assessments. The original intent was that the STA would be independent from duties associated with commercial concerns for operation of the plant. All STAs at WBN perform the STA functions as a collateral assignment while having other assignments associated with commercial operation. NSRS recommended that the STAs assigned to shift coverage should be removed from any other duties other than those associated with that function.

The plant staff disagreed with the NSRS conclusion based upon the fact that recent NRC draft proposed rulemaking would replace the STA with a position that provides engineering expertise by requiring, on each shift, a person with primary management authority for integrated facility operations and engineering assessment expertise as well as plant operation knowledge and experience. They felt that the present NRC-proposed position did not perpetuate the idea of independence from commercial concerns.

NSRS agrees with the plant staff's position as it relates to the present NRC direction. However, it is clear that the STA assignment is a collateral assignment and that there is a possibility that the STA's performance in that capacity could be affected by conflicts with his primary work supervisor because of his unavailability to perform those functions he is normally responsible for when he is not assigned to STA shiftwork. As a result of these conflicts the performance of STA duties could be adversely affected.

AI-2.16 indicates that the STA primary responsibility while on shift is the performance of the STA duties and answering to the shift engineer. Therefore, administratively the NSRS concern has been satisfied. NSRS will monitor the performance of the STA program during the startup of unit 1. This item is closed.

11. R-84-05-WBN-07, Two-Party Verification for Fuel Loading

The follow-up review of this item with the Reactor Engineering Unit (REU) revealed that TI-1, "SNM Control and Accountability," had been revised to reflect the requirements of the N-OQAM for "Separate and independent parties" for verifying fuel transfers. The new revision defined the duties and responsibilities between REU and FQE. This item is closed.

12. R-84-05-WBN-08, High Density Fuel Storage Racks (HDFSR) Attentuation Testing

No additional information had been received to evaluate the technical justification for the 15 percent sampling rate of the attentuation test as requested in the earlier report. EN DES was requested to supply this information to the Site Director, but no response had been transmitted to either him or the NSRS. This item remains open..

13. R-84-05-WBN-09, Surveillance Requirements for Changing Modes of Operation

Information obtained during the follow-up review of this item with the FQE and Operations staff on the verification of surveillance requirements for changing modes of operation satisfies the NSRS concerns. The planning and scheduling section will comply with the requirements of changing modes by delineating the required SI performances by a schedule to be supplied to the Operations staff. This item is closed.

14. R-84-05-WBN-10, Workplan Quality Assurance (QA) Requirements

This finding dealt with a discrepancy between engineering change notices (ECNs) and workplans. During the review of workplans, ECN cover sheets were observed that were marked "yes" to "QA applies." However, the workplans were marked to indicate that QA did not apply. The NSRS recommended that these differences be resolved. The WBN site response indicated that all workplans would be reviewed for this discrepancy and would be corrected beginning June 25, 1984. The response appeared adequate. The NSRS reviewer examined a limited number of workplans initiated since June 25, 1984 and did not observe any further examples of problems in this area. However, this item will remain open until additional workplans are issued and can be reviewed for compliance with the corrective action.

15. R-84-05-WBN-11, Workplan Functional Tests

Section 5.2.1c.3g of AI-8.5 cited details of functional tests that the cognizant engineer should include, as applicable, in the instruction portion of the workplan. During a random review of workplans, it was observed that workplans

requiring functional tests did not appear to have sufficient details as required by the procedure for testing. The WBN response indicated that no corrective action was required. A review of approximately ten more workplans did not reveal any recurrences of this problem. This item will remain open pending a thorough review of workplans to ensure detailed instructions or references to approved instructions for functional tests are written in the workplan.

16. R-84-05-WBN-12, Supplemental Information Added to Workplans

This finding dealt with the concern that supplemental information had been added to workplans. There was no identification of the person who added the information nor was it dated to indicate when the information had been recorded. Since these additions to workplans were not dated, it could not be determined if the comments were added during the initial review cycle or later. The site's response to review all workplans to ensure that additions were identifiable and dated appeared adequate. Review of a small number of more recent workplans indicated that supplemental information was being initialed and dated by the responsible engineer. However, this item will remain open to allow a thorough review of workplans to ensure continued compliance.

17. R-84-05-WBN-13, Plant Modifications Made by Use of a Maintenance Request (MR)

This concern dealt with the site possibly doing modification work on an MR. Review of the information in the site's response indicated that information was acceptable and accurate.

MR 224689 had been issued to install a temporary alteration (TACF 1-84-11-271). This action was in compliance with AI-2.15, "Temporary Alterations." The temporary condition was then to be made permanent by the use of an ECN. No further examples of possible plant modifications using an MR were discovered. This item is closed.

18. R-84-05-WBN-14, Inspector Certification Records

This finding dealt with the fact that inspector certification records were not kept at the site. The document used to provide inspector certification was a monthly printout received from Power Operations Training Center (POTC). Problems existed with the printout. SQN had previously been cited by OQA with a deviation for not possessing a current list of certified inspection personnel. The problem also appeared to exist at WBN. The site responded to indicate an interactive computer system would be obtained which should improve the certification documentation process. This response appears adequate. This item will remain open until implementation of the corrective action can be reviewed.

19. R-84-05-WBN-15, Material Inspection

AI-8.5 stated that "the originator of the workplan shall ensure that the design, construction, installation, inspection, and testing of modifications meet quality assurance standards at least equal to those of the original CONST installation requirements." One requirement cited from the OEDC Quality Assurance Manual for ASME Section III Nuclear Power Plant Components (NCM) was that "all items shall be identified during manufacture and/or installation to facilitate control and maintenance of records." In CONST identification of items is accomplished by quality control (QC) inspectors' routine and required inspections of all items during fabrication and installation. At WBN, NUC PR allowed the user of the materials to verify correct identity before installation. The user of the material was determined to be the responsible craft. This practice appeared to violate the NCM requirement and Criterion X of 10CFR50 Appendix B. The following was the site response:

> N-OQAM, Part II, Section 5.3, Attachment 1, Paragraph I.H, allows a cognizant individual to verify material in lieu of a holdpoint; therefore, QC holdpoints have not been established for Section III installations. FQE had established an activity survey checklist in March 1984 which physically verifies the installation of material. This is performed on QA Levels I and II material which includes ASME Section III materials. The first performance of this survey (WB-AS-84-83) was March 21, 1984, prior to the NSRS review. Subsequent surveys will physically verify installation of materials, including Section III material, by revising the checklist to specifically reference ASME Section III material.

The following paragraphs detail the NSRS position that the response is inadequate.

(a) N-OQAM, Part II, Section 5.3, Attachment 1, paragraph I.H does allow a cognizant individual to verify material. However, paragraph 3.2 of this same procedure states, "A QC inspection program based on inspection by peers or cognizant engineers shall not be acceptable." These two statements are conflicting. However, the difference between a cognizant individual and cognizant engineer is not clear. It appears to NSRS that the requirement stated by paragraph 3.2 is the proper method to be utilized for a QC inspection program.

(b) From an interview with the FQE supervisor, it appeared that the survey, WB-AS-84-83, was not performed on a scheduled interval, but only a random basis. Even

having FQE do a scheduled survey of material traceability would not satisfy the NCM requirement that all items shall be identified during installation. Also review of the completed survey indicated that material issue only was observed and page 5, item IV, of the survey documents that FQE did not physically verify installation of materials.

The site response is unacceptable since it does not appear to implement the minimum requirements outlined in report R-84-05-WBN. This item will remain open.

20. R-84-05-WBN-16, Records

Workplan activity may be performed by CONST or NUC PR. If the activity is accomplished by CONST, then all the inspection documents are stored in the CONST records vault. Review of CONST inspection records indicated that documents may possibly not be readily identifiable and retrievable. The response submitted by NUC PR assigned the task of reviewing workplan records to the As-Constructed Drawing Task Force. The task force will evaluate the identification and retrievability of CONST workplan records and make recommendations as appropriate. Also, OQA is working with WBN in the preparation of a retrieval instruction. The implementation of that procedure will be reviewed during the next review. This item will remain open until NSRS determines the implementation of the procedure and adequacy and implementation of recommendations made by the task force.

21. R-84-05-WBN-17, Two-Year Review Cycle for Procedures and Instructions

The NSRS recommended that the plant level document for controlling the review of procedures and instructions, AI-3.1, "Plant Instructions - Control and Use," be put into compliance with the upper tier controlling documents. NUC PR responded that they disagreed with this item. They also stated that the requirement was being discussed with Quality Assurance and that the discussion could be resolved after the reorganization is complete.

The NSRS still believes this is a valid concern. AI-3.1 is not in agreement with NRC Regulatory Guide 1.33, "Quality Assurance Program Requirements," ANSI N18.7-1976/ANS3.2, "Administrative Control and Quality Assurance for the Operational Phase of Nuclear Power Plants," nor N-OQAM, Part II, Section 1.1, "Document Control." AI-3.1 allows a successfully documented procedural performance to substitute for the required two-year review and this is in direct contradiction to all upper tier documents. This item will remain open until AI-3.1 has been revised to comply with the respective requirements of R.G. 1.33, ANSI-N18.7-1976/ANS-3.2, and the N-OQAM and until a two-year review requirement independent of successfully documented procedure performances has been implemented.

22. R-84-05-WBN-18, Field Quality Engineering (FQE) Review of Procedures and Instructions

The NSRS recommended that in the future, FQE procedures and instructions that implement quality assurance requirements and that are written by FQE should have the documented review performed by someone other than the original author. NUC PR stated in their response that the OQAM (Part III, Section 1.1, April 11, 1984) does not require an independent review for OQAM, DPM and ID-QAP implementation and that the independent review referred to in paragraph 4.4.3.1-b of the OQAM is the PORC review.

NUC PR is correct in its response that what is required is a review and concurrence by FQE to assure that plant instructions correctly implement the division quality assurance program. However, that is not the issue. The NSRS does not believe that a review and concurrence of a procedure by the person who wrote it initially and subsequently revised it is appropriate. It violates the basic concept of quality control, that is, verification of an activity by a person other than the one who performed it.

The NSRS position is that in the future, FQE procedures and instructions should have the documented review performed by someone other than the author. This item is closed.

23. R-84-05-WBN-19, Operator Response to Critical Alarms Before Licensing

The NSRS recommended that Operations personnel should be made more aware of the potential problems associated with ignoring alarms initiated during construction and testing phase of plant life. The incident which brought up this item was a water level alarm in the spent fuel pit to which Operations did not respond until the covers were removed from the pit for a surveillance instruction to be performed on a radiation monitor. NUC PR replied that a letter had been sent to all Operations personnel emphasizing the point that certain alarms must be responded to even before fuel loading. Based on this letter, this item is closed.

24. R-84-05-WBN-20, Interface Study Report

The NSRS recommended that the report be updated or a new study conducted, that the preoperational test director and engineer be trained in the interface program, that the interface coordinator be more active in the interface program, and that the interface log be reviewed against the preoperational tests.

The NUC PR response stated that "the 1980 Unit Interface Study was performed to be a study and was not intended to be the controlling document for interface." It went on to

describe all the controlling documents. These controlling documents were also described in the NSRS report. The response then described the periodic training and the test that test engineers must pass. This had also been described in NSRS Report No. R-84-02-WBN and had been noted by the NSRS as a commendable response to NRC findings.

The NSRS made the recommendation because of the problem that it had identified in reviewing the interface program. The official response did not address several of the recommendations that had in fact been followed by the interface coordinator. He had become more active in his role, the log had been compared with the test procedures, and interface points had been installed in several preoperational test procedures by the use of change sheets. The written response stated that:

> The periodic reviews of the interface points by the interface coordinator show that the interface points are correctly established at this time.

In fact, the NSRS found this to be the case on its follow-up review but not so during the initial review. This item is closed based upon the site activities, not upon the formal reply.

25. R-84-05-WBN-21, Interface Hold Orders

The NSRS recommended that test procedures should be reviewed for instances where hold orders should be applied to control interface points. During the review the NSRS had noted that the Interface Study Report had listed valves to be closed for interface points. These valves had been closed in the test procedure but were not shown as interface points.

NUC PR responded that: "For all interface points that had been established at the time of the NSRS audit, hold orders or TACFs had been installed in accordance with AI-1.6." This statement was true but it did not address the NSRS recommendation. The NSRS report stated that valves were closed which should have had interface controls on them but they had not been properly identified in the test procedure and controlled as interface points.

When the NSRS reviewed this area again, it was determined that change sheets had been written for preoperational test procedures to establish these already closed valves as interface points by using the interface hold order as the control mechanism.

The NSRS felt that the written response did not address the issue raised but that activity at the plant was adequate. This item is closed.

26. R-84-05-WBN-22, Marked-up Drawings for Interface Points

During the R-84-05-WBN review the NSRS attempted to verify that marked-up drawings for interface control points were given to the shift engineer as required by AI-6.1. The NSRS reviewer looked through the TACF and Hold Order (HO) Log Book in the SE's office on April 5, 1984, and found only one drawing. On a second trip to the SE's office on April 6 the SE was asked to find the drawings. He said that the one drawing was the only drawing available. On a third trip to the SE's office, also on April 6, the SE performed a thorough search of the office and could not locate any other drawings that had been supplied by preoperational test engineers to show interface points. The NSRS recommended that marked-up drawings for each set of interface points be submitted to the SE.

NUC PR responded that: "Apparently, the shift engineer only showed the NSRS inspector one print on which an interface point had just been established." Also, the response stated that the book that holds these prints had been labeled more clearly and a review had been made of drawings to ensure that all interface points were marked.

During the NSRS follow-up, two shift engineers were asked for the interface TACFs and HOs and both of them knew without being asked that a set of drawings went with them and where these drawings were located. The book with these drawings was distinctively labeled and was on the top slot in a drawing rack next to the SE desk. Three minor deficiencies were noted in the drawings and reported to the interface coordinator for correction. This item is closed.

27. R-84-05-WBN-23, Interface Points in Unit 2 Reactor Protection Cabinets

The NSRS recommended that the Solid-State Protection System (SSPS) output cabinets for unit 2 be transferred to NUC PR. This was recommended in the Interface Study Report because wire lifts and jumpers had been installed in the output cabinets of this system. NUC PR responded that these cabinets had been previously transferred as a direct result of the interface study. During the NSRS follow-up, it was determined that the unit 2 SSPS cabinets had not been transferred to NUC PR but that some of the internal wiring had been. The reason for transferring the cabinets was to allow them to be controlled by NUC PR. During this latter review, the SSPS cabinets for unit 2 were transferred to NUC PR. This item is closed.

28. R-84-05-WBN-24, Interface Review After Unit 1 Fuel Loading

Since there would be two years between fuel loading for unit 1 and unit 2, the NSRS recommended that periodic physi-

cal reviews of interface control points be made during this two-year period. At the time of the NSRS review upper plant management was in agreement with NSRS on this item even though there was no regulatory requirement for periodic reviews of interfaces. The NSRS stated in the details of the report that administrative controls for installing and controlling interface points seemed adequate if these controls were implemented as described. Also from the NSRS report:

> The interface program coordinator in the Preoperational Test Section stated that he had committed to plant management that a walk through of the physical interface control points would be conducted two weeks prior to fuel loading.

This was noted in the NSRS report and accepted as a good idea.

NUC PR responded to the recommendation that:

> An activity has been added to the project schedule for all interface points to be reverified prior to fuel loading for unit 1. This 100-percent verification, coupled with the normal controls placed on all hold orders and TACFs as shown in the plant administrative instructions, is felt to be adequate to ensure that the unit interface points have been properly established and are in place for unit 1 fuel loading.

NSRS agrees that the planned NUC PR actions are appropriate and should ensure that a proper interface is installed for fuel loading of unit 1. However, the interface will become more important to personnel and plant safety after unit 1 is operational. Unit 1 operation can adversely affect unit 2 construction and testing activities, and unit 2 activities can adversely affect unit 1 operation. The NSRS concern was with keeping the interface in place after unit 1 is operational. Due to the importance of the interface system to plant and personnel safety the NSRS continues to recommend that physical reviews of interface control points be formally scheduled and accomplished on a periodic basis (at least every six months) after unit 1 fuel loading and subsequent operations. This item remains open.

B. New Review Areas

1. Preoperational Testing

For background information, a description of the preoperational testing program and its controlling documents will be outlined.

NRC Regulatory Guide 1.68 (R.G. 1.68), "Initial Test Programs for Water-Cooled Nuclear Power Plants," Revision 7, 1978, is the controlling document for the preoperational test program.

R.G. 1.68 states:

> Preoperational testing, as used in this guide, consists of those tests conducted following completion of construction and construction-related inspections and tests, but prior to fuel loading, to demonstrate, to the extent practical, the capability of structures, systems, and components to meet performance requirements to satisfy design criteria.

Appendix A, "Initial Test Program" to R.G. 1.68 states:

> To ensure valid test results, the properational tests should not proceed until the construction of the system has been essentially completed.

ANSI N18.7-1976/ANS 3.2, "Administrative Controls and Quality Assurance for Operational Phase of Nuclear Power Plants" states:

> The preoperational testing program shall demonstrate, as nearly as can be practically simulated, the overall integrated operation of plant systems at rated conditions, including simultaneous operation of auxiliary systems.

N-OQAM, Part II, Section 4.1, paragraph 6.4 states:

> Each system or subsystem should be tentatively transferred in sufficient time to permit a period of pretest checkout before the formal test. The power plant operations section should operate the system in as many modes as possible during the pretest checkout period. Such operation shall be coordinated with the NUC PR test director. Where possible, this operation should include practice runs of the system utilizing the approved test instruction to identify weaknesses in the instruction.

Paragraph 6.5 of the same document states:

> The NUC PR test director, with the advice of the CONST test representative, shall recommend conduct of the test when he is assured that the installation status of the system is adequate for conduct of required testing activities.

Area Plan 1104.01, "Test Staff Program Manual - Preoperational Test Program," and AI-6.5, "Procedure for Initial Operation, Testing, and Transfer of Equipment and Auxiliaries" are implementing procedures and are generally in agreement with higher tier documents.

AI-6.5, section 3.10, defines hot functional testing as follows:

> For purposes of this instruction, hot functional testing is defined as beginning when the CONST Project Manager and the NUC PR Plant Manager authorize, by signature, the conduct of the preoperational test involving the initial reactor coolant system heatup to operating temperature.

In the hot functional test the primary system is brought up to operating temperature using heat from the operation of the reactor coolant pumps and pressurizer heaters but not from nuclear reaction.

In August 1983 the WBN hot functional tests (HFT) were completed. Because of deficiencies encountered during the testing it was decided that a second hot functional test, to be called the mini-hot functional test (MHFT), should be performed. The MHFT was begun in August 1984. The NSRS reviewed the MHFT and observed all plant sections as they operated within the context of simulated plant operation.

At the time of the first HFT it was known that several systems were incomplete but it was thought that these items could be tested during initial startup, which is a common practice. These systems include the Steam Generator Blowdown System, the Auxiliary Feedwater System, and some incore thermocouples among others.

For the mini-hot functional test, the NSRS observed portions of the following preoperational tests being performed:

a. W-1.1, <u>Heatup for Hot Functional Test</u>

   This test forms the framework for conduct of several other preoperational tests as well as being a test in itself. It is the controlling document for bringing the plant from ambient conditions up to no load operating temperature and pressure (557°F and 2235 psi respectively). The test had been performed in its entirety for the first HFT. For the MHFT a lengthy change sheet was written to incorporate only sections of the test that had to be repeated. This change sheet was used for test conduct.

   The NSRS noted no problems in the conduct of W-1.1. Everything was handled as required by controlling

documents. The test procedure, the test log, change sheets and test exceptions and deficiencies were reviewed and found satisfactory.

b. W-1.2, Hot Functional Testing

Hot functional testing was a continuation of TVA-1.1, "Heatup for Hot Functional Testing." The same engineers were conducting W-1.2 as were conducting W-1.1 and their performance continued to be acceptable.

c. W-1.7, RCS Thermal Expansion and TVA-23B, Thermal Expansion of Piping System (Feedwater Piping)

These preoperational (preop) tests were observed by NSRS at the 150°F plateau. The tests appeared to be performed in accordance with the test instructions. In addition to preoperational test engineers, the test director was assisted by an Engineering Design (EN DES) representative onsite while the preoperational tests were being conducted. It appeared to NSRS that the presence of an onsite EN DES representative enhanced the performance of the tests. Steamfitter craftsmen were assigned to collect data readings. The document used by the steamfitters to record data identified: (1) the temperature at which the readings were taken, (2) the support checked, (3) movement readings, and (4) time data recorded and data recorder. With the exception of one steamfitter, the craftsmen assigned to this task were cognizant of their duties. After a discussion among the craftsmen during lunch of the first day of testing concerning the method used to take the measurements, the test director was asked to explain the method. It was then determined that the one craftsman had misread all the data points taken by him up to that time. Before this person was allowed to continue collecting data, he was reinstructed by the test director on how to properly take movement readings. After reinstruction, the craftsman repeated all the measurements that had been taken prior to that time.

During review of the test instructions, the results of the thermal expansion test performed as a part of the previous hot functional test were reviewed. Review of the results by NSRS revealed a minor item of concern. Deficiencies from the previous thermal expansion test had been properly documented and submitted to EN DES for resolution. EN DES response indicated that some hangers would be modified. However, the reply memorandum did not identify the engineering change notice (ECN) that would control the modifications. In this instance the EN DES person who handled the deficiencies was the onsite representative during the second thermal

expansion test and was able to communicate this information to the test director. It appeared to NSRS that this needed information should have been included in the EN DES reply memorandum instead of relying on verbal means as a proper method of communicating information. The EN DES representative was apprised of this condition from conversation between the NSRS reviewer and the test director; however, no proposed corrective action was decided upon.

d. TVA-1, Shield Building Inleakage Tests, Emergency Gas Treatment System Functional Tests

The portion of this test dealing with shield building inleakage was observed. Administrative controls for the test were being implemented properly, and the engineers conducting the test appeared competent. This test had been conducted earlier in the year and after much effort to seal leaks, an acceptable rate of inleakage was achieved. Prior to the actual conduct of this test during this review, several days were spent again attempting to seal leaks before an acceptable inleakage rate could be achieved. The NSRS reviewer watched while several leaks were sealed. These leaks were either caused by deterioration of materials, accidental damage, or willful damage. The test engineers stated that they believed several leaks around piping boots were obviously damage-induced but how the damage occurred could not be determined. The NSRS reviewer agreed with the engineers since there were clean breaks and cuts. While monitoring this test performance, the general condition of equipment inside the protected area was observed. Several problems were noted:

(1) There was a pigtail with broken flexible conduit on annulus vacuum fan 1A.

(2) The boots around the containment penetrations for main steam and feedwater had leaks, some of which appeared to be caused by people climbing and one which the preoperational test engineer said appeared to be a cut by a sharp object.

(3) In the north steam valve room, the conduit to a valve motor was broken. The valve number was not determined.

(4) An instrument sensing line for a flow transmitter associated with annulus vacuum fans was badly bent.

These items were noted in a short time of observation and could mostly be attributed to personnel working in

the area to complete construction or to perform maintenance. NUC PR realized this problem existed and was implementing a program to control the number of workers in the protected area.

e. TVA-22, Auxiliary Feedwater System

Portions of this test were observed by the NSRS while the reactor coolant system was at 557°F. The test director appeared competent, the procedure was adequate, and all administrative controls were being implemented. FQE personnel were monitoring this test and appeared thorough in their work. The NSRS did note that deficiency number (DN) 195 was written against the test during the activities being monitored. Most of the deficiencies were equipment and system failures. Also, there were 86 change sheets associated with this test procedure. Many of these changes were to accommodate plant conditions that could not be predicted by the test director before the test conduct began. (A level IV violation had been written against this test by NRC during the first HFT, but the conduct of the test this time did not appear to violate any procedural requirements.)

f. TVA28, Sampling System

The sampling system test attempts to prove that certain important systems can have samples taken for analysis during all operational conditions. The NSRS monitored the test conduct. The test engineers appeared competent, the procedure adequate, and administrative requirements handled properly. The test itself could not be performed at the time of the observations because of leaking isolation and bypass valves for sampling. MRs were written to correct the deficiencies in the valves.

g. TVA-29, Steam Generator Blowdown System

The NSRS attempted to review activities associated with this system, but equipment problems prevented the test from being performed. This system was required for secondary side chemistry control and had been used for this during the first HFT and was being used for this purpose during the MHFT, but the system was not in a testable condition during the MHFT because instrumentation was not operating properly. The Instrument Maintenance Section calibrated and loop-checked the instrumentation and it was in proper functioning order; but it was later discovered that instrument sensing lines were incorrectly routed, slopes were incorrect, there were some wires incorrectly terminated, and air was in the lines. Since the NSRS review, a problem had developed with the system's pumps. The problem was still under investigation by NUC PR.

From the observations made by the NSRS during the MHFT it was concluded that the MHFT testing was being conducted by qualified test directors in accordance with established test instructions. However, it appeared that optimum quality and efficiency had not been achieved during testing activities as evident by the number of deficiencies, change sheets, and inoperative equipment associated with TVA-22 and -29. The conduct of TVA-22 and -29 did not appear to comply with the intent of the respective regulatory guides, ANSI standards, and the N-OQAM in that there was a marked difference between the upper tier guidance of "pretest checkouts," "practice runs . . . to identify weaknesses in the instructions" and "construction tests . . . satisfactorily completed" and the actual field testing experience of 195 deficiencies and 86 required changes to the test instruction for TVA-22 and the inoperable equipment associated with TVA-29. TVA-22 was only one example of a test with many deficiencies and change sheets.

A possible root cause that promotes the type of problems encountered with TVA-22 and -29 is a philosophy of testing as soon as possible any part of a system that can be tested even if much simulation by means of wire lifts and jumpers is required. This philosophy encourages the transfer of systems before construction is essentially complete. The potential for this problem was identified in NSRS report No. R-81-28-WBN, Mini-Management Review, conducted from November 16 through December 4, 1981, wherein it was noted:

> There is potential for problems caused by systems being transferred before completion.

And later:

> The scope of work being conducted under these circumstances opens the way for a potential loss of control of the work function especially as it involves quality-related activities. CONST has continued to transfer systems to NUC PR with hundreds of open items. The systems are transferred in this configuration to meet the present schedule. If the schedule is unrealistic, this method of meeting the schedule may increase the potential for the performance of non-quality work . . .

At that time there were 8000 items on the Outstanding Work Item List. These were on systems that had been transferred to NUC PR but the work was to be completed by CONST.

To improve the quality and efficiency of preoperational and noncritical systems (ICS) testing, NUC PR should assure that systems are essentially completed, checked out, and tested by construction prior to transfer. System operability and test construction adequacy should be verified before conducting the formal testing activities. These actions should enhance compliance with the intent of the upper tier documents governing these activities and result in a better quality testing program.

2. Maintenance Activities

During this review the control of the mechanical maintenance activities by the MR system was evaluated to determine the degree of compliance with established requirements. Additionally instrument maintenance activities were observed to evaluate performance of Surveillance Instructions, to determine the status of the preparation of required Surveillance Instructions, and to assess the qualifications of instrument maintenance personnel. The results of the NSRS activities in these areas are detailed as follows:

a. Mechanical Maintenance

AI-9.2, "Maintenance Program," was the administrative instruction that established the method and responsibility for initiating, planning, scheduling, performing, tracking, and documenting maintenance at WBN. This instruction applied to all maintenance work, including preventive maintenance.

Paragraph 5.3 of this administrative instruction gives a description for the information needed on an MR.

Approximately 90 MRs assigned to the Mechanical Maintenance Section were reviewed for compliance to procedural requirements. It appeared that MRs were being handled in accordance with stated requirements.

Paragraph 5.1 and item 14 of paragraph 5.3 of AI-9.2 stated that modifications could not be made on maintenance requests. During the NSRS examination of MRs, one maintenance request (A-400922) reviewed had work instructions that appeared to be a modification. The instructions required temporary cooling water supply and drain lines be installed on "B" auxiliary feedwater pump inboard and outboard bearing housings. The NSRS reviewer talked with the MR originator concerning this particular maintenance request. The MR originator contacted the responsible engineer and learned that TACF 1-84-125-3 had been written to control this tempo-

rary alteration (TA). A check of the SE logbook revealed that the TA was valid for this work. However, the TA number had not been recorded on the maintenance request causing continuity of information to be lacking. It appears to NSRS that when maintenance requests are used to install a temporary alteration then the TA should be recorded on the MR. This would allow cross reference between the documents.

Item 30 of paragraph 5.3 (AI-9.2) required that field quality engineering (FQE) perform a review of "entries for CSSC corrective maintenance in a timely manner to ensure the format and contents are in compliance with plant quality assurance requirements." NSRS observed that MRs that had been completed with the exception of the FQE review were being kept at the reviewer's desk for up to three days without any measures being provided to prevent possible damage to the documents or their loss. These documents are one-of-a-kind records which contain needed information. An interview with the FQE supervisor revealed that their document review responsibilities would probably increase in the future and thus the time to review them. If present conditions continued, the result would be more records being retained at the FQE unit without protection from possible damage or loss. It is the NSRS conclusion that these records should be placed in a fire-rated storage cabinet while awaiting this final review. As a minimum, this would guard against records being lost or destroyed, which would require a recreation of those records. This may be a generic problem which could apply to all QA records.

b. Instrument Maintenance

During this review an NSRS team member observed the conduct of Surveillance Instruction SI-3.1.12II, "Pressurizer Pressure Protection Set II." The SI was being conducted using a temporary change to an instruction. A problem was encountered with a transmitter and an MR was written to recalibrate the transmitter. Everything associated with the conduct of the SI appeared adequate. In an interview with management it was determined that the SIs were being performed with temporary changes or on an MR as unofficial performances to get problems corrected. The MHFT was being used as a framework for performance of the SI.

From a presentation by plant management to the NRC on August 16, 1984, it was determined that there was a total of 546 SIs for WBN. Of these, 110 were not yet written and approved by PORC, with about one-half of these being Instrument Maintenance SIs, of which all but 6 were drafted and in the review process. At the

same presentation it was stated that there would be 1271 total instructions at WBN with 126 left to run.

At that time the NSRS was concerned that there were many SIs to be performed and only two months to perform them before the scheduled October 11 fuel loading date. The fuel loading schedule slippage has reduced some of the pressure for completion of the SIs and thus the NSRS concern.

The NSRS interviewed approximately 25 personnel in the Instrument Maintenance Section. These people included instrument mechanics, senior instrument mechanic foremen, general foremen, engineers, and other management level personnel. These people all appeared competent. Several changes in management personnel in the last six months were noted by the NSRS. In reviewing the individual's experience, all management personnel met the requirements of ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel," which requires a high school diploma and four years experience in the craft to be supervised.

3. Conduct of Licensed Operations

a. Configuration Control and Independent Verification by Operations Section

During the performance of the MFHT the Operations crew shift activities were observed by NSRS. The observations included the performance of System Operating Instructions (SOIs), General Operating Instructions (GOIs), and shift/relief turnover between operations staff. These plant procedures were reviewed to the Standard Practice Manual and the Administrative Instruction Manual for compliance regarding the authority and duties for performing activities affecting the safety functions of structures, systems, and components.

Three SOIs were reviewed and their implementation observed. The SOIs were SOI, 62.1, "CVCS - Charging and Letdown;" SOI 68.2, "Reactor Coolant Pumps With Appendix A;" and SOI 74.1, "Residual Heat Removal System." Also, the implementation of GOI-1, "Plant Start-up From Cold Shutdown to Hot Stand-By" was observed.

Deviations from the SOIs were necessary due to the requirements of the MHFT program. The systems were in an alignment that would not necessarily be found during normal operation. The configurations were controlled by the SOI valve checklist and the configuration control log. The position of any valve could be found on

the valve checklist or log. The log provided the control for deviating from the valve checklist for a test condition or unusual alignment. The instruction to provide guidance for maintaining system status was OSL-A2, "Maintaining Cognizance of Operational Status."

The MHFT also required deviations from the GOIs to perform tests. While the SOIs supported the GOIs, the unit assistant shift engineer (ASE) and unit operator (UO) would maintain entries in the log books to indicate the current conditions and system status that differed from the GOIs.

A licensed senior reactor operator (SRO) was assigned to the MHFT to coordinate the testing with the MHFT preoperational test director in the control room. This SRO maintained the configuration control log and the completed valve checklists which were kept in the main control room.

The NSRS team performed a survey of the valve positions on the Residual Heat Removal system (RHR) for verification against the valve checklist and the configuration control log. One valve was found out of position when the "as-found" position was compared with the valve checklist and configuration control log.

The SRO acknowledged the valve position could be improperly noted by the paperwork, but not in an improper position for the current alignment. The SRO did know the required position versus what was recorded, and the paperwork was corrected.

Subsequent to the valve position survey, a survey of independent verifications was made on the RHR, CVCS, and RCS by checking the completed paperwork. Three two-party verifications had incomplete signoffs. AI-2.19, "Independent Verification," states in section 5.8:

> Independent verification is the determination by two separate individuals that a function has been accomplished as required. It is the policy at Watts Bar that these two individuals may verify the action at the same time, that is, may travel about the plant together or at different times. It shall be stressed that when traveling together each individual must verify the action. For example, if an independent verification must be made that a manual valve is open and the valve is in remote location that requires climbing a ladder, both individuals must climb the ladder to verify the valve is open. One

person going up and calling down to the other that the valve is open is unacceptable.

A discussion of these discrepancies with the FQE supervisor revealed that an inhouse survey by FQE was in progress. The FQE survey was being performed to identify any configuration control and independent verification problems. As a result of the FQE surveys two corrective action reports (CAR) were issued. WB-CAR-84-38 addressed the failure of the Operations Section to perform independent verification per AI-2.19. WB-CAR-84-39 addressed the failure of the Operations section to maintain configuration control. The FQE surveys revealed an average of 6.4 percent errors in configuration control and independent verification entries for the systems sampled. The errors involved wrong valve positions and incomplete verifications. The SROs took immediate action to correct the discrepancies identified to them. The Operations supervisor was planning to have meetings with the staff to stress the necessity of maintaining complete configuration control and documenting independent verifications to comply with the FQE corrective action requirements.

b. Shift and Relief Turnover of Operations Section

The shift and relief turnover of the Operations staff was observed on several occasions during the MHFT. The assistant unit operator (AUO) turnover was good with the exchange of technical information and current system status being made. The oncoming shift AUO appeared to receive a total update from the face-to-face turnover from the offgoing shift AUO. The unit operator (UO) turnover was good with the review of the current status and logs of prior events. However, some of the ASE and SE turnovers were not adequate to meet the requirements of AI-2.10. Some ASEs and SEs were not present for the oncoming shift turnover, so in some cases the only turnover was in the logs.

AI-2.10, "Shift and Relief Turnover," defines the manner and minimum information required to pass between oncoming and offgoing shift personnel.

The observed inadequate shift turnovers were discussed with the Operations Section supervisor and assistant supervisor. The supervisor said that AI-2.10 requirements would be discussed with all Operation staff assigned to shift work and would be complied with prior to fuel loading.

4. Reactor Safety and Criticality Controls

a. Fuel Handling Operations

No substantial fuel handling had occurred since the NSRS phase II review. A complete fuel inventory and audit was being performed during the last review by the NSRS (phase III review). All activities observed were conducted according to Technical Instruction TI-2, "SNM Control and Accountability System," with no observed discrepancies.

Preoperational tests were conducted on the fuel handling tools and fixtures necessary for core loading operations during the current NSRS review. The preoperational test W6.1, "Fuel Handling Tools and Fixtures" was reviewed and found thorough, detailed, and complete as a preoperational test. The test actually utilized new fuel to verify operability of all tools and fixtures. Several Operations personnel were involved in this test for familiarization with the equipment.

Selection and training of fuel handling operation crews was to have begun shortly after the NSRS review was completed.

b. Reactor Safety Controls

The Phase II review identified NRC Information Notice 83-18, "Failures of the Undervoltage Reactor Trip System Breakers," which required a modification to the reactor protection system (RPS). The required modification had been made per Work Plan 4336.

Electrical Maintenance Instructions, Surveillance Instructions, and Operating Instructions were reviewed to verify the incorporation of the modification into procedures necessary for fuel loading. The modification met the requirements of Information Notice 83-18 and should satisfy the NRC's concerns.

5. Chemistry Control

Prior to and during the heatup phase to 250°F of the hot functional test, NSRS observed the WBN chemical unit activities while analyzing and adjusting the primary and secondary chemistry parameters to within specifications listed in preoperational test procedure W1.1 and TI-16, "Plant System's Sampling and Chemical Criteria." The analyses were conducted in accordance with established technical instructions and results recorded on official log sheets for each system. Parameters such as dissolved oxygen in the primary system and Pb, copper, and dissolved oxygen in the secondary system were initially out of specifications, which is not

uncommon during system heatup from shutdown conditions. The heatup was restrained until various parameters were adjusted and the plant chemical unit took appropriate corrective actions to adjust the specified chemical concentrations to within specifications.

During the performance of preoperational test W1.1 to the 250°F plateau, NSRS observed that there was an apparent breakdown in the communications between the test director and the chemical laboratory personnel which caused some delays in the heatup process. It appeared that the shift chemical laboratory personnel were not being informed of the progress of the test and were thus not prepared for the addition of chemicals and the preop test required analyses when various plateaus were reached. This observation was discussed with the Chemical Unit and Preoperational Test Unit supervisors. Those supervisors indicated that the information interface between the two units would be improved.

It was noted during the review that the Chemical Unit supervisor had initiated a formal program to ensure that all of the radiochemical laboratory analysts were trained on all Surveillance Instructions that they would be required to perform. The status of this training was being tracked by the supervisor via a matrix which clearly illustrated the training progress of each analyst. This practice should enhance the quality of the Chemical Unit surveillance program.

6. Health Physics

The WBN health physics program was assessed by NSRS to determine its readiness for fuel loading, initial criticality, operation, and an unplanned outage. The assessment consisted of discussions with site, plant, NUC PR Central Office (NCO), and Operational Quality Assurance Branch (OQAB) personnel along with review of regulatory, TVA corporate, NUC PR, and WBN documents. The following areas relating to the WBN health physics program were assessed:

- OQAB Activities
- WBN FQE Activities
- WBN Health Physics Organization
- WBN Health Physics Qualifications and Staffing
- Health Physics Program Administrative Controls
- Health Physics Instrumentation, Equipment, and Facilities
- Health Physics Section Personnel Stopwork Responsibility and Authority
- FSAR Description of the WBN Health Physics Program

The results of the assessment are detailed as follows:

a. OQAB Activities

The WBN plant management had requested the OQAB to perform an operational readiness review of the plant's radiological protection, radwaste control, and radiological emergency planning programs. The review was performed the week of June 18, 1984, by a three-member team consisting of personnel with professional health physics experience from OQAB, NUC PR, and the Radiological Hygiene Staff (RHS). The findings and recommendations from that review were reported to WBN on July 19, 1984 (see reference VII.DD). A follow-up review by OQAB was not scheduled.

Utilizing existing offsite TVA resources to determine the operational readiness of the WBN radiological protection, radwaste control, and radiological emergency planning programs represents a progressive management attitude and the respective WBN management should be commended for their initiative. Additionally it should be noted that OQAB was responsive to the request.

The review performed by OQAB indicated that some problems existed in the areas of health physics instrumentation and equipment, health physics staffing, and the status of related radiological protection procedure preparation. As a follow-up OQAB review was not scheduled, NSRS evaluated the status of actions taken concerning selected findings and recommendations from that review. The status of those actions will be discussed in respective sections of this report.

b. WBN FQE Activities

NSRS interviewed FQE management personnel to determine the extent of that plant section's surveillance activities in the program area of health physics. In addition, corrective action report status logs for 1983 and 1984 were reviewed to determine if any problems had been identified in the area of health physics via the Corrective Action Report (CAR) and Deficiency Report (DR) systems. The results of the interviews with FQE management personnel and the review of the CAR and DR status logs are detailed below:

(1) FQE Surveillance of WBN Health Physics Program Activities

Management personnel in FQE reported that checklists for surveillance of health physics activities had not been prepared as the significant implementation of the health physics program begins at fuel loading and startup. The FQE

surveillance of health physics activities had therefore been minimal.

FQE surveillance checklists should be promptly prepared and surveillance scheduled during the fuel loading and startup phases of unit 1 to assure that the radiation protection functions are being performed in compliance with established program requirements and to determine the quality of that performance.

(2) Problems Identified In the Area of Health Physics Via the Corrective Action Reporting System (CARs and DRs)

No CARs had been issued for corrective action assigned to the WBN Health Physics Section during the timeframe of 1983-June 1984. Two DRs had been issued requiring action in the areas of housekeeping and QA records. Acceptable corrective actions had been taken and those DRs were closed. Several (13) DRs had been issued in April 1984 identifying Radiation Protection Area Plan procedures that had not been implemented in plant instructions. Health Physics Section management personnel reported that the respective plant instructions had been prepared and were issued or were in the review and approval cycle at the time of the NSRS review, and the DRs would be closed in the near future.

c. WBN Health Physics Organization

NSRS discussed the planned health physics organization and staffing with the Site Services Manager and the Plant Manager's staff. The results of those discussions are detailed below:

(1) Facility Organization

Figure 1 of this report depicts the planned site and plant organizations as specified in Figure 6.2-2 of the August 7, 1984 draft WBN Technical Specifications.

(a) Site Director's Staff

It was planned to add a staff health physicist answering through the Site Services Manager to the Site Director. This position had not been filled at the time of the review nor had the range of activities to be performed and the methodology of interface with the plant and offsite organizations been

defined. Site management indicated that the staff health physicist position would be filled and the scope of activities along with the methodology of interface would be defined after a decision had been made concerning which health physics functions would remain with the NCO and which functions would be transferred to the Site Director's staff. That determination was underway and was expected to be completed by October 1, 1984.

The addition of a health physicist to the Site Director's staff should enhance the WBN health physics program. This position should be filled and the scope of activities along with the methodology of interface with the plant and offsite organizations defined before startup of unit 1.

(b) Plant Health Physics Staff

The current reporting chain of the Health Physics Supervisor is depicted by Figure 1 of this report. The Health Physics Supervisor reports through the Operations and Engineering Superintendent to the Plant Manager.

Reg. Guide 8.8 and NUREG-0731 state respectively:

> The Radiation Protection Manager (RPM) onsite has a safety function and responsibility to both employees and management that can be best filled if the individual is independent of station operations, maintenance, or technical support, whose prime responsibility is continuity or improvement of station operability.

and

> The reporting of the functional areas of radiation protection, quality assurance, and training should assure independence from operating pressures.

After the TMI accident in 1979 the NRC conducted a health physics appraisal program to evaluate the adequacy and effectiveness of radiation protection programs at the nuclear power plants in operation at that time. The

results of that appraisal were reported in NUREG-0855 issued in March 1982. In that NUREG the NRC reported that significant weaknesses in the area of radiation protection organization and management were identified at approximately a third of the facilities inspected. One of the significant weaknesses involved "lack of management support." The NRC stated:

> The lack of management support of radiation protection programs was reflected in several ways. At some facilities the Radiation Protection Manager's (RPM) reporting chain was such that the RPM must compete with others within the same group to bring radiological problems and concerns before the station manager.

Additionally, the NRC stated:

> At some facilities, the quality of radiation protection was found to be significantly less where the RPM was not reporting directly to the station manager. It was noted in these organizations that health physics was more of a routine service organization than a radiation protection support function, integrated into the fabric of all plant operations. It was noted that personnel within these organizations generally lacked incentive and a depth of technical knowledge.

The plant organization does not provide the plant Health Physics Section independence from line operational pressures. On the one hand the Operations and Engineering Superintendent is responsible for operations and engineering (technical support) activities which affect generation availability. On the other hand, he is responsible for health physics activities which by necessity may delay the operation and engineering process and can result in decreased generation availability. The Health Physics Section Supervisor may have to compete with the Operations and Engineering Group Supervisors, both of whom are higher grade levels. In the absence of the Operations and Engineering Superinten-

dent the Health Physics Section Supervisor may be required to report to either the Operations or Engineering Group Supervisors if one of those supervisors temporarily assumes the Superintendent's responsibilities.

A significant amount of the Health Physics Section's functions will be associated with maintenance and modification activities. The present organization does not promote a radiation protection support function integrated into the fabric of all plant operations which includes maintenance and modification activities as well as engineering and operations activities. The Health Physics Section should have the same relationship with maintenance and modification personnel that it has with operations and engineering personnel. At the time of the NSRS review there was no indication of conflicts or problems. It is recognized that the present organization could actually enhance the health physics program by providing increased upper plant management availability to health physics and increased interface with the Operations and Engineering personnel to work out problems encountered during plant operation and routine testing activities. However, increased interface with operations and engineering personnel may be at the expense of proper interface with maintenance and modifications personnel.

To minimize the potential for conflicts and program degradation and to organizationally provide the health physics supervisor the flexibility to deal effectively and directly with all aspects of the health physics program the NSRS recommends that the current plant organization be revised to establish the reporting of the Health Physics Section supervisor directly to the plant manager. This aspect of the organization is considered consistent with current regulatory policy.

d. Health Physics Qualification and Staffing

The WBN health physics staffing status at the time of the NSRS review is depicted in figure 2. The OQAB review team had evaluated the qualifications and adequacy of the health physics staff to meet the minimum staffing requirements and found that the Health Physics Section personnel met the qualification requirements of

ANSI N18.1-1971 and was adequately staffed with the following exceptions:

- There was an insufficient number of HP technicians onsite to adequately respond to an ALERT.

- The Health Physics Technical Unit was inadequately staffed to effectively implement their planned activities during unit operations. These activities include:

  - Operation of the respirator fitting test equipment.

  - ALARA and health physics dose tracking.

  - Operation of the body counting systems.

  - Dosimetry issuance and Thermo Luminescent Dosimeter (TLD) processing.

The OQAB review team recommended that the plant staff fill the dosimetry technician and data entry operator positions with permanent or temporary personnel prior to startup or the establishment of large radiological controlled regulated areas. They recommended that if the estimated reporting dates for individuals filling these positions was within two months of startup, the plant staff should obtain temporary personnel to allow for adequate training.

The NSRS evaluated the current adequacy of the plant health physics staff to support the startup of unit 1. The results of the NSRS evaluation are detailed below:

- The health physics staff had 17 SE-5 level HP technicians onsite and job offers had been made to an additional 8 candidates for that position.

- There were 8 SE-4 level HP technician trainees onsite with an additional 17 technician trainees due onsite November 12, 1984. The majority of the SE-4 level technician trainees had participated or are participating in on-the-job training at SQN (some for up to one year).

- The Technical Unit staffing had not changed significantly since the OQAB review and that unit was still not adequately staffed to support the startup, testing, and off-normal events. For the primary inadequacies, vacant position announcements had been issued, some had closed, and selections had been made. Job offers for the dosimetry technicians were scheduled to be made by August 14,

1984. A NUC PR training class for dosimetry technicians was scheduled to start on September 10, 1984.

- The Outage Support Unit had not been staffed. The staffing status of this unit should not adversely affect the startup of unit 1.

By the end of November 1984 the number and experience level of the WBN HP technicians and trainees in the Operational Unit should provide adequate staff to handle that unit's normal and expected off-normal health physics support for fuel loading, initial startup testing, and minor forced outages during the startup process of unit 1. It is probable that WBN will have the largest number of well qualified technicians of any TVA nuclear facility during initial startup and this should enhance the facility health physics program.

The staffing of the Technical Unit was still inadequate to support the startup and operation of unit 1. The plant staff was taking appropriate actions to fill the vacancies in that unit.

e. Health Physics Program Administrative Controls

The OQAB review team reviewed the radiation protection procedures for general content to determine their readiness for plant startup. Their review included those minimum procedures required by the draft WBN Technical Specifications and the additional detailed procedures required to instruct the health physics staff in the implementation of the health physics program. The OQAB found that neither the minimum required or additional detailed procedures to instruct the health physics staff in the implementation of the health physics program were adequate for startup.

The OQAB review team recommended that the required procedures be revised and/or issued by July 15, 1984 or at least two months before fuel loading to give the plant staff ample time to learn the procedures before startup.

The NSRS discussed the status of each procedure addressed in the OQAB review report with the Health Physics Section Supervisor. From these discussions the NSRS determined the following:

- With only a few exceptions the existing procedures had been revised and new procedures written as recommended by OQAB. Even though some of the procedures had been issued others were still in

the review and approval cycle or in reproduction for issue. Health physics management expected that those procedures in the review and approval cycle would be approved and issued in the immediate future.

Through discussions with the plant health physics management and observation of existing procedures NSRS concluded that administrative controls had been provided to control those activities relating to the radiation protection program addressed in section 6 of the draft WBN Technical Specifications. These include the following:

- Control of the Health Physics Section overtime.
- Surveillance of radioactive sources.
- Health physics technician training, retraining, and replacement training.
- Inplant radiation monitoring for airborne iodine concentrations in vital areas under accident conditions.
- Access control of high radiation areas.
- Reg. Guide 1.33 required procedures.

The additional detailed procedures required to instruct the health physics staff in the implementation of the health physics program were in place or were in the review and approval cycle and were expected to be in place in the near future. It was noted that the primary administrative system for controlling personnel exposure to radioactive materials and radiation (Radiation Work Permit) had recently been revised at WBN. The significant revision places more responsibility for radiological safety on the job foreman and workers. Due to the importance of this system of controls and the significance of the revision, NSRS recommended that awareness seminars for the new RWP program be provided to the plant staff prior to the startup of unit 1.

f. Health Physics Instrumentation, Equipment, and Facilities

(1) OQAB Review Team Findings and Recommendations

The OQAB had evaluated the adequacy of the Health Physics Section's instrumentation and equipment for fuel loading. The OQAB found that an adequate supply of portable survey instruments, laboratory equipment, and respiratory protection equipment

was available for use. However, they also found that some necessary equipment and supplies (C-zone clothing, TLD processing equipment, and portable monitors) were not onsite. The following is a summary of their findings and recommendations:

The OQAB recommended that emphasis should be placed on expediting the procurement of those necessary items that were not yet onsite. The NSRS determined the status of those items and concluded the following:

- TLD processing equipment had not yet arrived from the vendor.
- Portal monitors had not arrived from the vendor. One monitor had been shipped by the vendor the week of the NSRS review and the other monitor was being packaged for shipment in the immediate future.
- C-zone supplies (including C-zone clothing) had been ordered but had not been received.
- The TLD issue area was being constructed and should be completed by fuel loading of Unit 1.

Procurement and completion of necessary equipment and facilities were being expedited by the plant staff.

(2) <u>NSRS Inspection of Health Physics Instrumentation, Equipment, and Facilities</u>

NSRS inspected the portable instrumentation inventory and other health physics equipment and facilities and reviewed respective program controls. The results of the inspection and review are as follows:

(a) <u>Portable Health Physics Survey Instrumentation and Air Sampler Inventory and Control</u>

Section 12.5.2 of the WBN FSAR states, "The portable health physics survey instrumentation will be equivalent to the instrumentation described in Regulatory Guide 8.8.C.4." Section C.4.b of Regulatory Guide 8.8 states that, "Portable instruments needed for measuring dose rates and radiation characteristics will include:, (1) Low-range (nominally 0 to 5 R per hour) ion chambers or G-M rate meters; (2) High range (0.1 to at least

500 R per hour) ion chambers; (3) Alpha scintillation or proportional count rate meters; (4) Neutron dose equivalent rate meters; (5) Air samplers for short-term use with particulate filters and iodine collection devices (such as activated charcoal cartridges); and (6) Air monitors with continuous readout features."

The inventory and characteristics of the WBN instrumentation and portable air samplers are tabulated in tables 2 and 3 of this report.

NSRS determined that the inventory and characteristics of the portable health physics survey instruments and air samplers met the requirements of section C.4.b of Reg. Guide 8.8 and section 12.5.2 of the FSAR. The normal inventory was adequate to perform radiation surveys to support fuel loading, startup, operation, and off-normal occurrences.

Controls had been established in HPSILs for periodic calibration, functional checks, use, storage, and segregation of defective or out-of-calibration instruments.

Portable instruments for emergencies were stored in the backup health physics laboratory and in the health physics monitoring van. Implementing procedures for the Radiological Emergency Plan required that in the event the health physics laboratory has to be evacuated, the technicians will remove portable survey instruments, air samplers, and supplies as they leave the laboratory and transport this equipment to a designated area. NSRS interviewed a health physics shift supervisor and two technicians to determine if they were aware of the requirement. Those interviewed were well aware of the requirement.

Air flow rate measuring devices for air sampling were integral components of air sampling equipment. The WBN health physics staff planned to calibrate the air flow rate measuring devices at WBN using established procedures (HP TSIL-25) and calibration equipment certified to the National Bureau of Standards. However, certification documents for the calibration equipment were not available at the plant site. This certification should be acquired and maintained by the plant staff.

(b) Other Health Physics Facilities, Equipment, and Controls

The NSRS inspected and discussed other health physics facilities and equipment. The facilities and equipment appeared to be adequate with the following exceptions:

- Health Physics Laboratory Counters

  The health physics laboratory was equipped with four counters to process smears and air samples. At the time of the NSRS review three out of the four counters were out of service because of breakdowns. One of the out-of-service counters was still in warranty and was being repaired by the vendor while two of the out-of-service counters were being repaired by the plant Instrument Maintenance Section. As these counters are essential for supporting normal and off-normal plant operation, service contracts and/or training should be provided to maintenance personnel to provide for preventative maintenance and prompt and effective repair of inoperative equipment.

- Health Physics Laboratory Fume Hood

  The fume hood in the health physics laboratory was not operational and had not been checked for proper air flow velocities and patterns. The fume hood is located in the immediate vicinity of the laboratory counters and has no instrumentation or indicator to alert personnel as to its operational status. Fume hoods are provided to facilitate processing samples that are potentially highly contaminated. The location of the fume hood (adjacent to the laboratory counters) and the lack of indication of its operational status make use of this hood for processing highly contaminated samples questionable. If the plant staff elects to use this fume hood for processing radioactive or contaminated samples, the laboratory counters should be relocated (for background and contamination considerations) and the fume hood provided with some sort of device that would alert the technicians to the operating status prior to use.

- Laundry Monitors

  The laundry monitors were onsite but had not been installed and calibrated. Additionally, the laundry workers had not been trained on use of the monitors. Installation, calibration, and necessary training should be accomplished before the startup of unit 1.

- Respirator Storage, Issue, and Repair Facility

  This facility located adjacent to the health physics laboratory in the service building was not fully constructed and had not been turned over to the health physics staff. This facility should be completed and equipped prior to startup of unit 1.

- High Radiation Area Lock Tumblers

  Section 6.12.2 of the draft Technical Specifications require that radiation levels greater than 1000 mR/hour at 45 Cm from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the shift foreman on duty and/or the Health Physics Supervisor. Special tumblers for the locks to be used for controlling access to high radiation areas had been ordered but had not yet arrived on site.

- Gamma Ray Spectrometers for Analyzing Particulate, Iodine, and Noble Gas Samples

  The plant health physics staff was not provided with gamma ray detectors for the purposes of qualitatively and quantitatively analyzing air samples for radioactive particulate, iodine, and noble gas concentrations. The health physics staff must rely on the plant chemical unit to provide these services. Discussions with chemical unit personnel indicated that the chemical unit spectrometers had been calibrated for the health physics sample geometries and

calculator programs had been developed to process data from the analyses. A methodology for establishing the analysis priority for the health physics samples had been established in the Radiological Emergency Plan Implementing Procedures (REP IP). However, the chemical unit was not aware that they would be required to analyze several samples (projected to be 20-40) on a daily basis. Historically at BFN and SQN there have been problems with priority setting between health physics and chemical unit samples. Any confusion concerning priority setting for both normal and off-normal samples should be resolved prior to the startup of unit 1.

In general the instrumentation, equipment, and facilities were still not adequate to support startup of unit 1 primarily in the area of dosimetry processing equipment and contamination zone supplies. However, the dosimetry services could be provided by the RHS in Muscle Shoals and contamination zone supplies could be borrowed from other NUC PR facilities. The plant staff is taking actions to assure that the necessary equipment or services and supplies will be provided in sufficient time to allow for implementation of the planned programs and to allow for refinement of those programs prior to the startup of unit 1.

g. <u>Health Physics Section Personnel Stopwork Responsibility and Authority</u>

Section V.B of RCI-1, "Radiological Hygiene Program" states that: "When imminent danger or major violations of the Radiological Control Instructions are encountered, the Health Physics Supervisor (or his designated representative, either the Shift Supervisor or the Assistant Health Physics Supervisor) has the responsibility and authority to take the necessary corrective action including termination of an activity through the plant manager or his designated representative.

Imminent danger is defined as a condition or situation where there is a reasonable certainty that immediately, or within a short period of time, the condition or work operation will cause death or serious physical harm to any employee or person exposed to the particular hazard."

These statements indicate that health physics personnel must obtain the approval of the Plant Manager or his designated representative before terminating an activi-

ty when imminent danger or major violations of RCIs are encountered. This would be inconsistent with corrective actions necessary to mitigate an imminent danger condition or situation as it could take some time to make the necessary contacts with the Plant Manager or his designated representative.

The stopwork responsibility and authority statements in RCI-1 for imminent danger conditions should be clarified to specify that health physics personnel have the responsibility and authority to stop work or order an area evacuated when in their judgement the radiation protection conditions warrant such an action and such actions are consistent with plant safety. It should be clear that only the Plant Manager, Health Physics Section Supervisor, or their designated representatives (Shift engineer or Health Physics Shift Supervisor) on backshifts can overrule a stopwork action initiated by health physics personnel.

Plant management indicated that they would evaluate the NSRS concern in this area.

h. FSAR Description of the WBN Health Physics Program

Some sections of the FSAR did not accurately depict the planned WBN health physics program. Examples of the inaccuracies are as follows:

- Section 13.3.4.4.1 of the FSAR states that portal monitors will be located in the gatehouse. It was planned to locate high sensitivity portal monitors at the exits of the "power block." It was not planned to install portal monitors in the gatehouse even though those have been procured and were onsite.

- Section 12.5.1, "Organization," makes reference to the Radiological Hygiene Branch (RHB) and its responsibilities and activities. The RHB no longer exists and its responsibilities and activities have been divided between the new RHS, NUC PR NCO, and the WBN plant staff.

- Section 12.5.2, "Equipment, Instrumentation, and Facilities," contains inaccuracies as follows:

  - The health physics technician base of operations and communications is described as the laboratory located at the boundry between the office and service buildings when in reality it is in the new laboratory in the service building.

- The WBN plans and program for processing TLDs are not addressed.

° Section 12.5.3, Procedures," references the "SWP" system which has been discontinued and replaced by the "RWP" system which is significantly different.

NSRS recommended to plant health physics management that respective sections of the FSAR should be reviewed to determine inaccuracies and updated to accurately reflect the planned WBN health physics program and activities prior to unit 1 startup. The plant health physics staff initiated an effort to accomplish this recommendation prior to completion of the NSRS review.

In summary the administrative controls for health physics activities were in place or would be in place in the immediate future. The personnel complement (by November) and experience level of the Operation Unit should enhance the health physics program. The personnel complement of the Technical Unit was still inadequate to provide those services planned for that unit. However, unplanned support could be obtained from the RHS for TLD dosimeter processing and equipment such as C-zone supplies could be acquired from other NUC PR facilities. The WBN health physics management was aware of these conditions before the NSRS review and was making progress toward obtaining the required personnel, equipment, and facilities. The progress (particularly personnel staffing) has been slowed somewhat by the recent TVA reorganization.

VII. LIST OF DOCUMENTS REVIEWED

A. SOI-74.1, "Residual Heat Removal System Unit 1 or 2," R7

B. SOI-68.2, "Reactor Coolant Pumps Unit 1 or 2," R4
App. A, "RCP Local Inspection Checklist," August 2, 1984

C. SOI-62.1, "CVCS - Charging and Letdown Unit 1 or 2," R4
Valve Checklist, July 11, 1984

D. AI-2.10, "Shift and Relief Turnover," R7

E. AI-2.19, "Independent Verification," R1

F. OSL-A2, "Maintaining Cognizance of Operational Status"

G. GOI-1, "Plant Startup from Cold Shutdown to Hot Standby," R5

H. SI-3.1.12 II, "Pressurizer Pressure Protection Set II," R2

I. NRC Regulatory Guide 1.68, "Initial Test Program for Water-Cooled Nuclear Power Plants," R7

J. Preoperational Test Section Instruction Letters

K. Instrument Maintenance Section Instruction Letters

L. ANSI N18.7-1976/ANS-3.2, "Administrative Controls and Qualtiy Assurance for Operational Phase Nuclear Power Plants"

M. N-OQAM, Part II, Section 4.1, "Preoperational Test Program," April 11, 1983

N. Preoperational Test Instructions

a.	W-1.1	Heatup for Hot Functional Test, R0
b.	W-1.2	Hot Functional Testing, R0
c.	W-1.7	RCS Thermal Expansion, R0
d.	W-6.1	Fuel Handling Tools and Fixtures
e.	TVA-1	Shield Building Inleakage Tests, Emergency Gas Treatment System Functional Tests, R0
f.	TVA-22	Auxiliary Feedwater System
g.	TVA-23B	Thermal Expansion of Piping System (Feedwater Piping), R0
h.	TVA-28	Sampling Program, R0
i.	TVA-29	Steam Generator Blowdown System, R0

O. Area Plan 1104.01, "Test Staff Program Manual - Preoperational Test Program," June 11, 1984

P. AI-6.5, "Procedure for Initial Operation, Testing, and Transfer of Equipment and Auxiliaries," R5

Q. N-OQAM, Part II, Section 2.1, "Plant Maintenance," July 18, 1984

R. N-OQAM, Part II, Section 5.3, "Maintenance and Modification Inspection Program," July 30, 1984

S. AI-2.15, "Temporary Alteration," R6,

T. AI-4.1, "Quality Assurance Records," R7,

U. AI-9.2, "Maintenance Program," R11,

V. WBNP-QCI-1.25, "Control of As-Constructed Drawings," R7 (Addendum 1), January 25, 1984

W. Engineering Change Notices (ECNs) 4533 and 4558

X. Temporary Alteration Control Form (TACF) 1-84-11-271 and 1-84-125-3

Y. Activity Survey WBN-AS-84-83, "Issue of Material and Traceability," March 20, 1984

Z. Work Packages (WPs) 4453, 4440, 4547, 4519, 4025, 3271, 4065, 4126, 4437, 4523, 4301, 2683, 3126, 2578, 3906, 4124, and 3915

AA. Maintenance Requests - Several

BB. NRC Regulatory Guide 1.33, "Quality Assurance Program Requirements," R2

CC. ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel"

DD. Memorandum from R. L. Moore to W. T. Cottle, "Watts Bar Readiness Review," July 19, 1984 (OQA 840719 703)

EE. AI-2.16, "Shift Technical Advisors," R4

FF. TVA Topical Report, TVA TR75-1A, R7

VIII. PERSONNEL CONTACTED

J. R. Anderson	Health Physics Shift Supervisor
R. A. Beck **	Health Physics Supervisor, WBNP
H. B. Bounds **	Plant Superintendent (Maintenance), WBNP
S. B. Billings	Radiochemical Laboratory Analyst, WBNP
S. R. Bradley	Health Physics Shift Supervisor, WBNP
J. L. Brown	Radiochemical Laboratory Analyst, WBNP
J. K. Bryant	Preoperational Test Engineer, WBNP
D. R. Bucci	Nuclear Engineer, BFNP
W. L. Byrd *	Test Section Supervisor
C. R. Cook	Senior Reactor Operator
W. T. Cottle	Site Director, WBNP
W. S. Delk *	Acting Engineering Section Supervisor, WBNP
J. E. Engelhardt **	Compliance, WBNP
E. R. Ennis **	Plant Manager, WBNP
T. O. Frizzell	OQAB
E. O. Gambill	Senior Reactor Operator
J. E. Gibbs	Site Services Manager
R. R. Gibbs	Acting Reactor Engineering Unit Supervisor, WBNP
F. K. Hecker	Chemistry Unit Supervisor, WBNP
G. R. Hendricks	Reactor Engineer
B. J. Hensley	Preoperational Test Engineer
T. F. Huth	Reactor Engineer
T. E. Kendrick	Safety Aide
M. E. King	Chemical Engineer, WBNP
R. J. Kitts	Health Physicist, NCO
H. F. Koehler	Preoperational Test Engineer
J. L. Lee	Instrument Engineer
D. L. Lester	Test Section Group Leader, WBNP
J. A. McLean	Health Physics Technical Unit Supervisor, WBNP
B. Z. Mears	Preoperational Test Engineer
E. S. Murphy	Mechanical Maintenance Engineer
M. E. Murray	Chemical Engineer, WBNP
R. B. Neal	Power Stores
T. L. Newman	OPS Refuel Floor Supervisor
R. Norman **	Operations Supervisor, WBNP
J. W. Olson	Mechanical Maintenance Engineer
D. Ormsby *	Licensing

L. E. Ottinger **	Instrument Maintenance Engineering Supervisor, WBNP
C. C. Parker	Preoperational Test Engineer
H. L. Pope	FQE Supervisor
W. V. Rusbridge	Senior Reactor Operator
D. G. Sanders	Mechanical Engineer
R. C. Sauer **	Compliance, WBNP
M. E. Selewski	Preoperational Test Engineer
D. P. Shaffer	EN DES Preoperational Test
R. H. Smith	Assistant Test Section Supervisor
F. J. Spivey, Jr. **	Acting Health Physics Operational Unit Supervisor, WBNP
W. D. Stevens	Senior Reactor Operator
W. M. Stone	Preoperational Test Engineer
R. E. Swatzell	Chemical Engineer, WBNP
A. S. Thomas	Instrument Shop General Foreman
G. V. Tippens	Quality Assurance
E. B. Turnbill	Sheetmetal Foreman
D. W. Wilson	Design Services Manager
J. S. Woods	Instrument Maintenance Section Supervisor
R. E. Yarborough, Jr. *	Operations Section Assistant Supervisor

* Attended Entrance Meeting
** Attended Exit Meeting

TABLE I

WATTS BAR
OPERATIONAL READINESS REVIEW

Phase	Review Area
I 2/13/84 - 2/17/84 (Completed - Report R-84-02-WBN Issued)	1. General Employee Training 2. Employee Awareness of Regulatory and TVA Requirements and Policies Relating to Nuclear Safety Issues and Expression of Staff Views 3. Preoperational Testing (Partial)
II 3/26/84 - 4/6/84 (Completed - Report R-84-05-WBN Issued)	1. Organization 2. Qualifications of Personnel in Key Management Positions 3. Shift Technical Advisors (STA) Program 4. Control of Licensed Activities 5. Plant Procedures (Partial) 6. Unit Interface Control 7. Reactor Safety and Criticality Control (Partial) 8. Modifications and Outage Control
III 8/6/84 - 8/24/84 9/4/84 - 9/7/84 9/24/84 - 9/26/84 (Completed - Report R-84-15-WBN)	1. Mini-Hot Functional Test - Operations Section, Preoperational Test Section, and Chemical Unit personnel activities were reviewed during this time. Adequacy of and adherence to instructions and procedures were stressed. 2. Maintenance Activities 3. Conduct of Licensed Activities 4. Reactor Safety and Criticality Control 5. Health Physics
IV At Scheduled Fuel Load	Initial Fuel Load

Note 1. Plant staffing and organization will be further evaluated during subsequent reviews due to changes caused by the reorganization.

Note 2. Regulatory compliance is a part of all reviews.

TABLE 2

WBN PORTABLE HEALTH PHYSICS SURVEY
INSTRUMENT INVENTORY

Instrument ID	Detector Type	Type Radiation Monitored	Range	Inventory
1. Ludlum Model 3	G-M	beta, gamma	0.1-200 mR/hr	6
2. Ludlum Model 5-5	G-M	gamma	0.1-2000 mR/hr	17
3. Ludlum Model 300-10	G-M	gamma	1-10,000 mR/hr	15
4. Eberline Model E530N	G-M	gamma	0-20 R/hr	20
5. Eberline Teletector Model 6112	G-M	beta, gamma	0-1000 R/hr	25
6. Eberline Model RO-2A	Ion Chamber	beta, gamma x-ray	0-50 R/hr	45
7. Eberline Model RO-7	Ion Chamber	gamma	1 mR/hr-20,000 R/hr	2
8. Eberline Model PAC-4S	Scintillation	alpha	0 - 2x 10^6 Counts per min.	5
9. Eberline Model PNR-4	BF_3 proportional	neutron	0 - 5000 mRem/hr	6

TABLE 3

WBN PORTABLE HEALTH PHYSICS AIR SAMPLERS

Sampler ID	Type Samples Obtained	Inventory
1. Radeco Model H-809V, H-809V-II	Gases, particulate, radioiodine	8
2. HI-Q Model CF900V	Particulate, radioiodine	35
3. Eberline Model RAS-1	Particulate, radioiodine	17

FIGURE 1

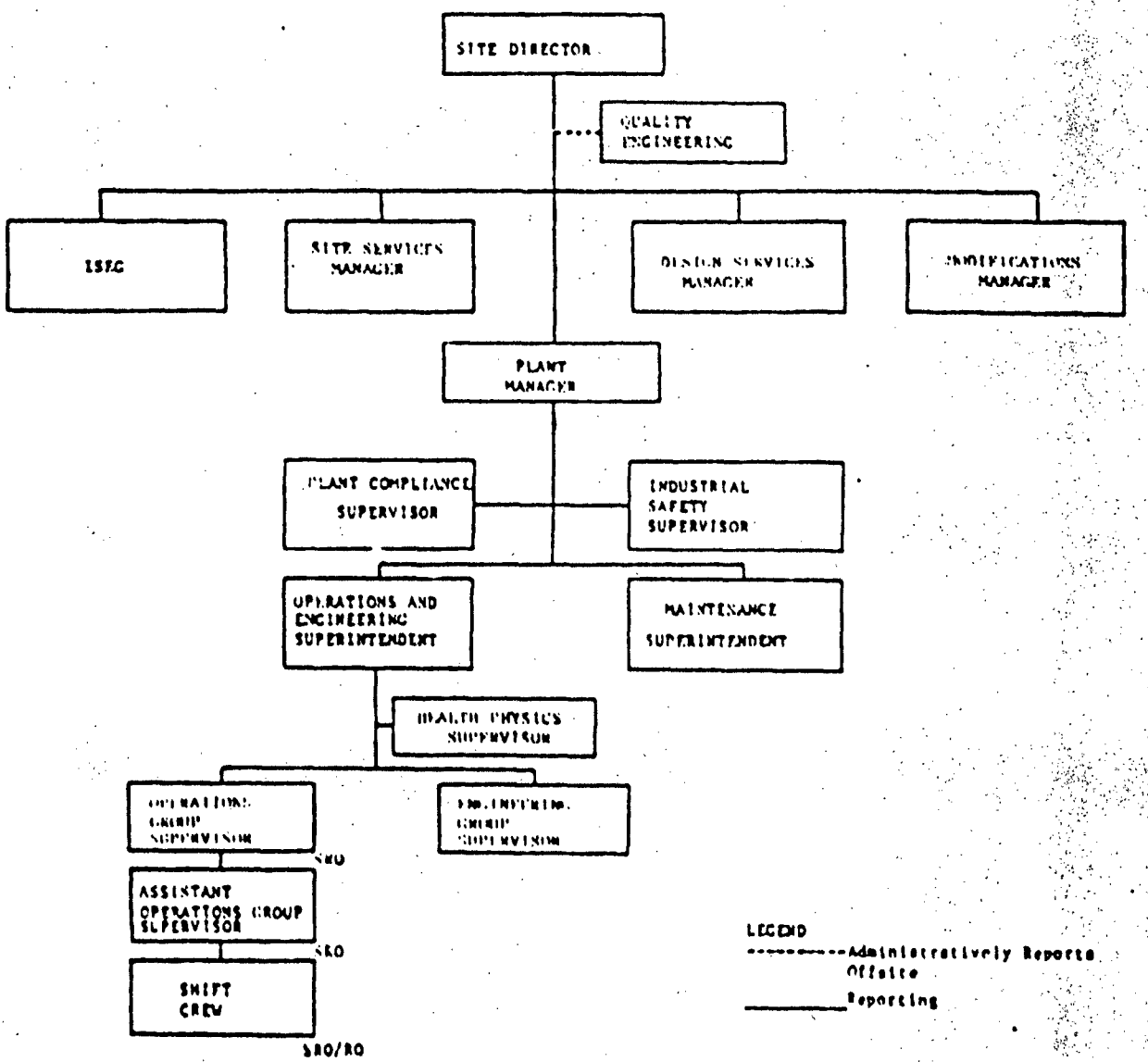
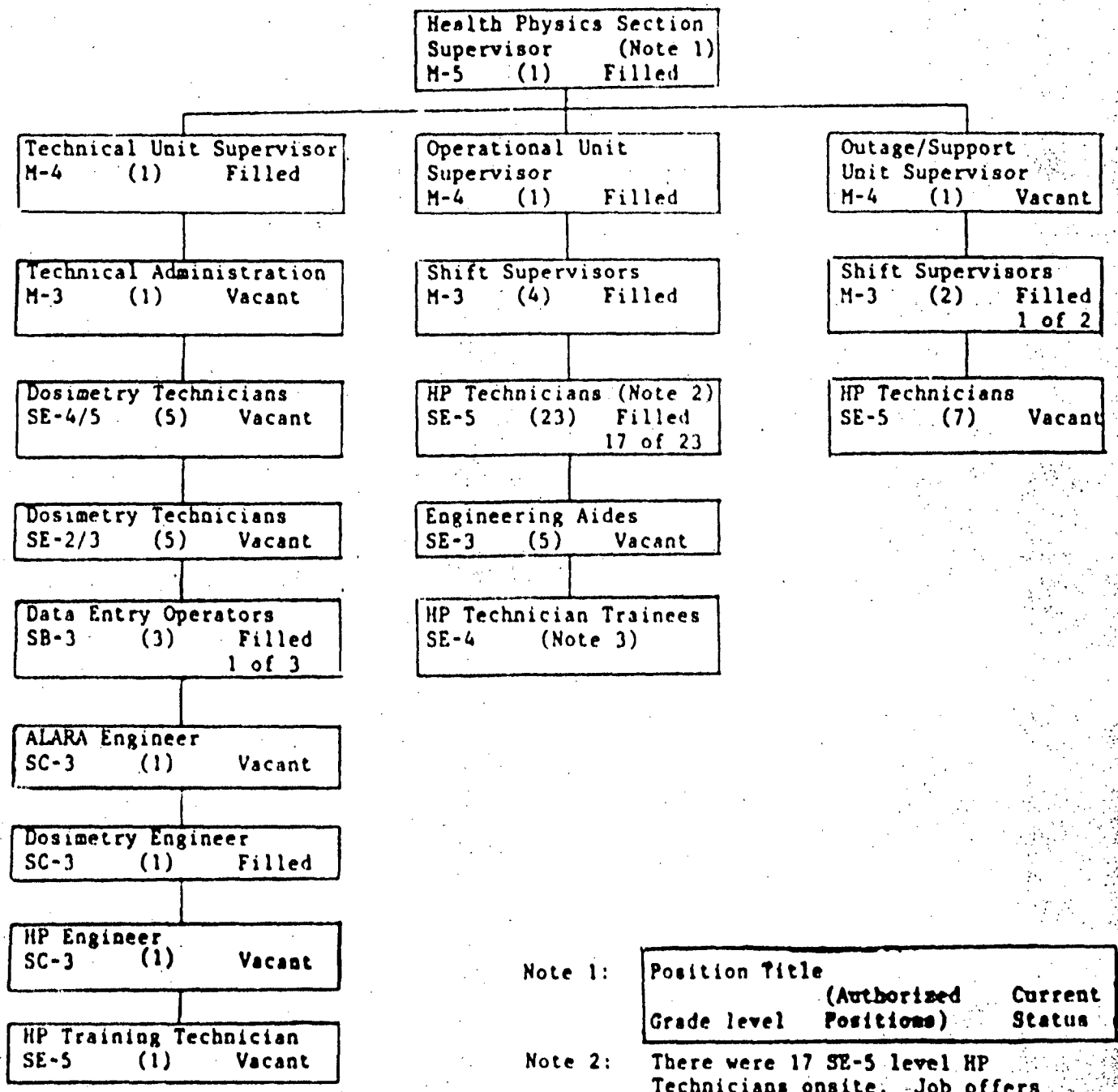


Figure 6.2-2
FACILITY ORGANIZATION

AUG 7 1984

FIGURE 2

WBN HEALTH PHYSICS SECTION ORGANIZATION AND STAFFING LEVELS



Note 1:

Position Title Grade level	(Authorized Positions)	Current Status

Note 2: There were 17 SE-5 level HP Technicians onsite. Job offers had been made to 8 more.

Note 3: There were 8 HP Technician Trainees (SE-4) onsite. Due on site 11/12/84 are an additional 17 (these trainees were at SQN for on-the-job training).

GNS 840627 052

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

INVESTIGATION

NSRS REPORT NO. I-84-16-BFN

SUBJECT: NSRS INVESTIGATION OF CONCERNS RELATED TO FAILURES OF THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM AT BROWNS FERRY NUCLEAR PLANT

DATES OF INVESTIGATION: MARCH 29, 1984 - MAY 10, 1984

INVESTIGATOR: P R Washer — P. R. WASHER — 6-15-84 DATE

APPROVED BY: James F. Murdock — J. F. MURDOCK — 6/15/84 DATE

MEDS, W5B63 C-K

TABLE OF CONTENTS

I. SUMMARY

The Nuclear Safety Review Staff (NSRS) was made aware of concerns, with numerous High Pressure Coolant Injection (HPCI) system failures and whether the system could meet its required safety function, by Stephan Mindel of the Office of Quality Assurance (OQA). With the many failures that have occurred, the concern is not only with reliability but also with degradation of the piping and supports to the extent that they may not perform in a safe manner if and when they are needed. The investigation was conducted between March 29, 1984 and May 10, 1984. The NSRS team concluded that the employee concern was valid and plant safety, in the event of a LOCA at Browns Ferry (BFN), was in jeopardy. NSRS recommendations to correct this condition are contained in the report.

II. SCOPE

This investigation was performed to address concerns regarding numerous failures of the HPCI system at BFN; possible damage to the piping, which would require NDE work for assessment; and the delays in correcting deficiencies so that the system can meet its required safety function. The concern centered on the fact that this is a safety system necessary for high pressure core cooling in case of a LOCA. The investigation was conducted by interviewing personnel, reviewing documentation, and by personal observation of the system itself.

III. FACTS

A. Background

On March 26, 1984 NSRS was made aware of an employee concern related to the HPCI system at BFN. The concern was brought to NSRS by Stephan Mindel of OQA and Dan Fisher of EN DES. The concern related to numerous failures of the system and whether or not it was still in a condition to perform its safety function. Both felt that some NDE work should be done on the piping in the vicinity of where restraints R-23 and R-24 had failed on unit 2. Their principal concern was that this is an important safety system and the pace of fixing the system has been very slow. The purpose of the employee concern was to elevate this problem to a high priority status.

B. Information Obtained from Interviews

On March 29, 1984 Stephan Mindel, Leonard Blankner, and Lillard Blevins of OQA were contacted by NSRS concerning this subject. Mr. Mindel felt there was a potential for damage of the piping at the lugs where the broken supports were attached. He also thought that someone should determine what motions the pipe had been through and the resulting effects. He stated that the system has had a history of broken parts during testing and use. Mr. Mindel felt that a review should be made of the overall maintenance history of the system including the break of the HPCI

turbine pedestal in 1980. Mr. Blankner stated that during his working time at BFN in the 70's there were problems with the FCW 73-45 valve allowing back leakage of feedwater at approximately 370° F, which causes steam in the line. When the HPCI system was initiated, this caused a water hammer in the pipe. He felt this could be a major part of the cause of breaking the pipe restraints. Mr. Mindel stated that review of the history of maintenance on the pipe supports would show that they had been bent before. He felt the struts were seeing a periodic overload condition which could create a fatigue failure in the piping. He also felt very strongly that some NDE work should be done on the pipe in the area of the lugs in order to qualify it for future use.

On April 6, 1984, Dan Fisher of the EN DES Boiling Water Reactor Project was contacted by NSRS concerning this subject. Mr. Fisher got involved in this problem in January 1984. Since that time there have been two support failures on this system. The first one occurred in January 1984 and the second one in March 1984. Mr. Fisher has reviewed in depth the failure that occurred in March 1984 to support R-23 in unit 2. This failure resulted in the base plate being pulled out of the wall. Upon review of the support, it was obvious that the support had been altered or repaired at a previous time. Mr. Fisher stated that the R-23 and R-24 supports were the only rigid supports in the system. The rest of the supports are dead weight hangers. Both of these supports have been repaired as damaged in the past. Mr. Fisher stated that the configuration of the lugs is such that high stress is transferred into the pipe welds and the pipe itself. He strongly feels that NDE work should be done in this area of the welds. From discussion with the support manufacturer, Bergen-Patterson, they have determined that the critical buckling load for the support is approximately equal to 33.5 kips. Since the support broke, the load had to be much higher than 33.5 kips and the load was transferred into the weld and the pipe. Mr. Fisher has copies of numerous maintenance requests or work orders to do repairs on these supports in the past. He feels this is a recurrent problem that has done some damage to the HPCI piping by now. He does not know whether a walkdown has been done on units 1 and 3 to evaluate them. However, he feels the same conditions exist for units 1 and 3 as do for unit 2.

On April 10, 1984 Ray Cole of the OQA Operations group was contacted by telephone at Bellefonte Nuclear Plant. He had been the OQA representative at BFN following the HPCI system until approximately one month before. Mr. Cole indicated that the HPCI system had given problems since initial startup of unit 1 in 1973. He issued numerous OQA documented complaints about the system and a HPCI committee was established in 1980 to review the problems. Mr. Cole said there were problems related to an electronic versus mechanical governor package along with an inherent design problem in gland-seal condenser and a problem with ruptured exhaust diaphragm. The big concern has been with needing this safety system and it tripping out and not being

available. Mr. Cole stated that there had been a problem with corrosion in the EGM system since moisture got into the box. A study on humidity in the area was started in February 1984, which showed there were humidity excursions in the area and the reason had not been determined. Mr. Cole also stated that recent tests at Vermont Yankee, Special Test 82-11, demonstrated a control modification on the HPCI turbine which improved the HPCI system quick start transient. This should reduce the transient loads on the piping system. He stated that this same test was recently conducted on unit 2 at BFN.

In discussions with Mr. Cole at BFN on April 17, he stated that history records are not very good prior to 1979. He also said the old system of Trouble Reports (TRs) was not effective since they were filed by TR number and not by system. As a result, it would be very difficult to find all TRs on the HPCI system. Since February 1983 the Maintenance Requests (MR) system has been in effect, which provides a report on a system-by-system basis. As an example of the ineffectiveness of the TR system, he said the TRs showed a changeout of a part on a turbine was made three times, while Power Stores showed ten parts were sold out.

Mr. Cole also stated that part of the problem is with attitude. The HPCI system must be recognized as a safety system that is maintained in a standby condition. Operations people must be convinced to follow procedures even though they may be skeptical of the chances of success. By following procedures in a step-by-step manner, failures can be analyzed if they occur.

On April 17, 1984, the NSRS investigator arrived at BFN for interview of onsite people regarding the HPCI concern. The first person interviewed was Randy Widick, the Mechanical Systems Engineer for the HPCI system. The discussion with him included the Special Test 82-11 results along with interviews of individuals from the operations group, valve test group, and the hanger group. The final part of this trip included a personal review of the piping system and restraints, which included photographs of the piping, hangers, and the two repaired restraints, R-23 and R-24. Mr. Widick said the 82-11 test had apparently taken out the early spike in discharge pressure. Copies of curves were provided which showed a reduction in the discharge pressure after the modification. Mr. Widick stated that in this procedure the governor is closed down so that RPMs pick up gradually and eliminate the pressure spike. He stated that this spike in discharge pressure may have caused the high forces in the piping and the restraints. This could also have caused the problem with rupturing of the gland seal condenser head gasket. Mr. Widick stated that the test on unit 2 was not done with all system and restraints repaired, since there had not been a complete walkdown of the piping system prior to the test. He recommended that this walkdown, along with any repairs needed, be completed prior to the next test.

Mr. Widick stated that they have not had the know-how to handle the complicated control problems. Part of this has been a result

of rapid turnover of electrical and mechanical people, eliminating the advantage of experience. He also stated that this Terry turbine system being used infrequently in conjunction with the complicated controls makes it very unreliable. He also stated that the operators are afraid of this system and very cautious about testing it. He has noticed marked differences in performance with different shifts. The operators are afraid to bring the turbine up to high RPMs fast. This probably complicates the problem since it is not designed to operate below 2000 RPMs. Both GE and Terry turbine personnel have stated that it should be brought above 2000 RPMs and kept there.

Mr. Widick thinks the operational method using Special Test 82-11 should be tried on all three units with close checks of system and restraints before and after the test. This would show whether the reduced pressure spikes would solve the problem of broken restraints.

On April 18, 1984, Jim Traglia of the Onsite Hanger Group was interviewed regarding the 79-14 inspection program along with his observations concerning the R-23 and R-24 restraint failures. Mr. Traglia stated that the first 79-14 inspection was done in 1980. They only made notes on missing welds, missing nuts, bent parts, etc. They did not establish a failure history. The problems that were found were put into three categories by EN DES. The three categories were: (1) repair in 30 days, (2) repair at next outage, and (3) repair when stress analysis is complete. There is no requirement for a follow-up inspection under the 79-14 program.

Mr. Traglia gave the following information concerning restraint R-23. In 1980 inspection, notes were made of members being turned in the wrong direction. From then to March 1984, when the embedded plate was pulled out of the wall, the W members and angles had been bent by injections. These bent members were removed and new members installed after the anchorage pullout in March 1984. He gave the following information concerning R-24. In 1979 the strut and welds were repaired. In 1981 a 10° bend in the strut was noted. In January 1984 the strut was broken. The support had obviously been receiving periodic high loads.

Mr. Traglia stated that he did not believe there were enough supports on the pipe. In some areas the length of unsupported pipes goes up to 40 feet. He thinks there should be more deadweight supports and restraints on the pipe. This would apply to all three units since the same type failures have been observed on units 1 and 3 also.

During the personal inspection of the HPCI piping, the observation was made that there are only two restraint-type supports on the pipe. These are the two that have failed. All the other supports are deadweight hangers. The measured distance from R-24 to a deadweight hanger was 20 feet in either direction.

Mr. Traglia stated that there is no way of knowing whether the supports are like they were when inspected, due to the massive modifications to other systems in the area. As a result, we cannot do an evaluation of the cause of the support failures. If we could inspect before and after injections, then we could do an analysis of the cause of the problems.

The next discussion on April 18 was with Ron Shadrick of the Field Services Valve Group. He gave information concerning the testing of the FCV 73-45 valve, the one that earlier was believed to have been leaking steam into the line. Mr. Shadrick stated that they have not had any problems with leakage of the valves since they were rebuilt about three years ago. That was the first time they had been rebuilt since the plant was constructed. The three units will have all FCV 73-45 valves changed out due to a different reason. The disc has been sticking in the closed position. The new valves will have a softer seat. His concluding statement was that there is no leakage problem with the FCV 73-45 valves.

The final interview at the BFN site on April 18 was with Tommy Jordan, Supervisor of Operations. Mr. Jordan stated that while opening the tell-tale drain valve, 73-551, to determine whether FCV-73-45 is leaking, that they had not observed a problem with units 1 and 2. At one time they did have a problem with unit 3, but that has been fixed. He does not believe steam leakage back through FCV-73-45 is the reason for hammer on the discharge line and breakage of supports. He agrees with Mr. Widwick that low RPM speeds on the turbine, before the 8211 change, was causing the problem.

A subsequent discussion was held with Dan Fisher of EN DES on May 10, 1984 regarding stress analysis of the HPCI piping. Mr. Fisher said there has been no stress analysis done on unit 2 HPCI piping. There has been a 79-14 type analysis for seismic events on unit 3. A similar type analysis will be done on unit 2. As far as he knows, Mr. Fisher stated there has been no transient analysis done on the system by either GE or TVA. He confirmed that R-23 and R-24 were the only restraints on the pipe. He said any discussions about adding additional restraints should be with Ron Cook in CEB, who is in charge of the analysis.

Ron Cook of EN DES was contacted on May 10 concerning the restraints and analysis. Mr. Cook said that no documented 79-14 type analysis had been done for any of the three units. He said the only analysis that had been done was the original design basis and he did not think any transient loads from dynamic fluid effects were included. He referred me to Jim Kincaid of EN DES to get background information on dynamic analysis.

On May 10, 1984, Mr. Kincaid stated that the HPCI piping has not been analyzed for transients and the seismic analysis may not be adequate at this time. He also said that the 79-14 analysis has not been documented for the three units. Mr. Kincaid believes we

could not design supports to hold the surge load if we had to analyze for transients using the old startup procedure. He agrees that we probably need to include the operational changes, tested by the 82-11 test, to prevent pressure spikes and then analyze for realistic loads and determine support requirements on that basis. He stated that we do not know the condition of R-23 before the last 82-11 test was run. He also stated that he agrees we should proceed with the FCR BF-DCR No. P-2040 to move the EGM controls.

IV. CONCLUSIONS AND RECOMMENDATIONS

From interviews of people from the various TVA organizations, it is obvious that the HPCI system has incurred many failures and has been very unreliable from startup of unit 1 in 1973. The failures have not been limited to one specific area. The failures have occurred in the components, the control mechanisms, and in structural supports. For a system that is essential for high-pressure cooling of the core in the event of a LOCA, 11 years is too long to repair-as-broken instead of providing a permanent fix. This system must be raised on the priority list and be recognized for its importance to public safety.

The system has been operated in such a manner as to cause fear of testing by the operations group, since they know something will break when testing. The control system in the EGM boxes has malfunctioned continuously due to poor environment. The hanger group along with EN DES personnel agree the piping is inadequately supported to take the loads induced by the fluid flow. Of all the people interviewed, no one stated that the system as-designed and as-built is adequate to meet its safety function.

Due to the importance to safety of this system, a complete fix to the system so that it will perform its required safety function must be pursued immediately. Eleven years is too long to rely on not needing a system rather than having it ready if and when it is needed. From the interviews and information gathered, the following recommendations are to be implemented to make this an operable system:

I-84-16-BFN-01 - The HPCI system should be recognized by all as a system that is necessary to maintain the plant in a safe condition. It should be viewed with the same importance as any system necessary to keep the plant operating.

I-84-16-BFN-02 - NDE examination of the welds and HPCI piping in the vicinity of the lugs for the R-23 and R-24 restraints should be conducted to insure that the piping is still qualified to meet its safety function. Further NDE examination of the restraints themselves is needed.

I-84-16-BFN-03 - The Special Test 82-11 should be implemented for units 1, 2, and 3. In conjunction with this operational change, the piping and supports should be inspected before and after each injection. This would allow evaluation of the cause of problems if any subsequent failures should occur.

I-84-16-BFN-04 - Design Change Request BF-DCR No. P-2040, to relocate the EGM governor control box to one of the walls in the HPCI room, should be implemented immediately. This should eliminate many of the control problems by locating the controls in a better environment.

I-84-16-BFN-05 - After implementaiton of Special Test 82-11, a transient analysis should be conducted on the HPCI system piping. From the transient analysis, the maximum loads from the fluid flow should be determined. In conjunction with this, pipe supports should be designed to restrain the system. If the analysis and design show that additional restraints and/or dead load supports are required, then these new supports should be installed.

ATTACHMENTS

1. Mechanical Maintenance Instruction 99 for HPCI Pipe Support No. 24 on Browns Ferry Unit 2

2. Mechanical Maintenance Instruction 99 for HPCI Pipe Support No. 23 on Browns Ferry Unit 2

3. Metallurgical Report on Browns Ferry Nuclear Plant Unit 2, Failure Evaluation of HPCI Restraint R-24

4. Design Change Request, BF-DCR No. P-2040

5. Curves showing operating conditions before and after implementation of Special Test 82-11 on Browns Ferry Unit 2

TVA 64 (01-8-81)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

001 '85 0312 050

TO : J. P. Darling, Manager of Nuclear Power, 1750 CST2-C

FROM : K. W. Whitt, Director of Nuclear Safety Review Staff, 249A HBB-K

DATE : MAR 12 1985

(1)

SUBJECT: NUCLEAR SAFETY REVIEW STAFF (NSRS) REPORT R-84-17-NPS - REVIEW OF PROCUREMENT PRACTICES AND PROCEDURES FOR OPERATING NUCLEAR POWER PLANTS

Attached for your use is a copy of the subject report. The NUC PR procedures for procurement were judged to be cumbersome, and the problems experienced by NUC PR in procuring materials in a timely manner were for the most part problems created by NUC PR and not outside support organizations.

It is important to understand that this review covered the procurement system in effect prior to the NUC PR transfer of NCO procurement activities to the sites. Consequently, changes to NUC PR's procurement process instituted since that time are not reflected in this report and may or may not adequately address recommendations contained in this report. In either event, your implementation plans and timeframes for completion are requested by April 26, 1985, with quarterly follow-up reports until the items are closed.

The cooperation of your staff and all other organizational groups involved in this review was appreciated. Questions regarding the content of this report should be directed to R. D. Smith at extension 4813 in Knoxville.

Original Signed By
K. W. Whitt

K. W. Whitt

RDS:BJN
Attachment
cc (Attachment):
R. W. Cantrell, W11A9 C-K
C. W. Crawford, 670 CST2-C
A. T. Mullins, 546 CST2-C
J. L. Williams, 1000 CUBB-C
RIMS, SL26 C-K
H. G. Parris, 500A CST2-C

NSRS FILE

Q01 '85 0312 051

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

REPORT NO. R-84-17-NPS

SUBJECT: REVIEW OF PROCUREMENT PRACTICES AND PROCEDURES FOR OPERATING NUCLEAR POWER PLANTS

DATES OF REVIEW: JUNE 11, 1984 - DECEMBER 5, 1984

TEAM LEADER: Richard D. Smith — R. D. SMITH — 3/8/85 DATE

TEAM MEMBER: Jon T. Muecke — J. T. MUECKE — 3/8/85 DATE

APPROVED BY: M S Kidd — M. S. KIDD — 3-12-85 DATE

RIMS, SL26 C-K

TABLE OF CONTENTS

I. BACKGROUND

Problems associated with the timely receipts of procured materials have been the subject of the Division of Nuclear Power (NUC PR) discussions on numerous occasions, and as a result of the Regulatory Performance Improvement Program (RPIP) at Browns Ferry Nuclear Plant (BFN), NSRS support was solicited in the form of a review. The review was conducted to examine and evaluate the procurement process for nuclear plants and determine the reasons for time delays and problems.

II. SCOPE

The procurement activities associated with TVA's nuclear power program were divided into two phases, operating plants and plants under construction. This review covered only the operating plants and centered on quality level I and II items and services. Those services or items manufactured by TVA were reviewed for only BFN.

As a part of the NUC PR reorganization effort to move personnel from the Nuclear Power Central Office (NCO) and to solve procurement problems identified by a joint Division of Purchasing (PURCH), NUC PR and Power Stores task force, NCO procurement activities were transferred to the sites. Interorganizational communications and working relationships were stated to have been developed to attack problems and streamline operations. NSRS did not include the evaluation of the reorganization within this review, but did evaluate the task force findings based upon the findings of this review.

III. MANAGEMENT SUMMARY

During the past several months NSRS has been reviewing the procurement process of materials and services for TVA's operating nuclear plants. The review began on June 11, 1984, and continued until the final closeout in Chattanooga on December 5, 1984. As a part of the review, closeouts were held at the completion of the onsite review at BFN, SQN, NCO, and PURCH. Throughout this review, people within NUC PR, Power Stores, PURCH, and the Office of Engineering (OE) were very helpful, cooperative, and in many cases candid. Virtually everyone interviewed considered procurement to be a major problem, and to a large extent the problem was the "other guy." Interviews were conducted with a number of dedicated people trying hard to do their job as they saw it, but frustrated because the system, regulations, QA, etc., were perceived to be working against them. Each group within the procurement chain had real problems and had several examples they were willing to share.

The problems experienced by NUC PR in procuring materials in a timely manner were for the most part problems created by NUC PR. In broad terms, there were five categories within which identified deficiencies could be placed.

A. General Unfamiliarity With Procurement Cycle

Personnel associated with each step of the procurement cycle were aware of what they were supposed to do or what they perceived to

be their responsibility; however, they were not aware of the role, function, or problems of others within the procurement cycle. No one was found, of the more than 90 people interviewed, that knew the entire system. Unrealistic expectations were placed upon the procurement system by originators of procurements. Ordered material was requested to be onsite generally within 90 days when, based upon this review of procurements, 6 months to 1 year would be more realistic. No one knew how long it would take to purchase materials but it was generally accepted that they would not be there when needed.

That lack of knowledge of the procurement system and associated problems produced frustration along the procurement chain. At the sites the procurement cycle and regulations were viewed at all organizational levels as a burden and designed to make the procurement process more difficult. The system and regulations were viewed as roadblocks telling the sites why they could not purchase something versus how to purchase something, and were also designed to purchase something (low bid) the site did not want over what it did want. As a result, the sites were putting more effort into using the system shortcuts through the overuse of emergency purchases and field purchases rather than learning the system for normal procurements and how to work within it.

There was no procurement training identified at the sites for personnel within the procurement cycle. For the most part personnel were introduced to the rigors of procurement by being handed a copy of the site procurement procedure (e.g., SQA 45), which was over 300 pages long, and told to read it. The procurement of items appeared to be viewed by site personnel as a required undesirable job as if it were part of an initiation.

B. <u>Excessive and/or Ineffective Review of Purchase Requests and Requisitions</u>

Typically 17 approval signatures and initials, some by the same people required to sign both the purchase request and purchase requisition, were required for a site-originated procurement. The value added to those documents beyond what the originator, quality assurance, and authorizing official contributed was, for the most part, minimal. In a very small number of procurements that were considered more complicated, the NCO provided valuable input. Considering the timeframe to prepare, approve, and transmit a procurement requisition from the sites to vendors for bids, the sites typically took one to four days, PURCH about three days, and the NCO weeks to months. The value added by the NCO, which was primarily editorial in nature, could not support the continued time delay by the NCO in the procurement cycle; consequently, the removal of the NCO from the review cycle and transfer of the affected NCO personnel to the sites was viewed by NSRS as a positive action provided the NCO problems and delays were not transferred with them.

It appeared to NSRS that the entire procurement system, with all its reviews, was predicated upon the concept of safety in numbers, i.e., the more people involved in reviewing, the better the product. In actuality what NSRS found was procurement documents being changed for no apparent good reason other than a perceived need to demonstrate a degree of usefulness by each successive reviewer.

All procurements generated by the sites, both QA and non-QA, were reviewed by the site Field Quality Engineering Group (FQE). For the most part there was one individual performing that review at each site. Those procurements included direct charges, IQTs, field purchases, transfers, and Material Management System (MAMS) reorders. For example, at BFN during May and June 1984 there were 1051 procurement actions or about 26 procurements per day that required FQE review and approval. The effectiveness of the review on that number of procurements by one individual is questionable, and the effectiveness of the review of QA procurements could be enhanced if the review of non-QA procurements by FQE were performed only on a sampling basis.

C. Ineffective Use of Available Procurement Systems

IQT contracts are supposed to be time savers in that once the IQT has been reviewed and approved, Requests for Delivery (RDs) against the IQT can be issued directly to the vendor without the review and approval process required for new procurements. NUC PR's procurement procedures negated any time savings afforded by an IQT because they required the review and approval of each RD as if it were a new procurement. There was no mechanism to identify large use items, such as steel, as potential candidates for IQT contracts.

MAMS is a computerized system to maintain an established supply of inventoried stock items throughout Power, and has the potential of being a very powerful tool. The maximum inventory level and minimum reorder point for some materials were inadequate, and the sites established the practice of hoarding items such as mops and plastic bottles to compensate. The sites viewed the established maximum/minimum levels as arbitrary and an effort to reduce stock inventories. In actuality the established maximum/minimum levels were neither, and the site problems can be attributed to poor communication between the site and the Materials Management Services Staff (MMSS), who administered MAMS.

Although MAMS had the capability of reordering QA items automatically, when initiated by Power Stores, this capability could not be utilized due to the reorder program not being approved as a QA system. As such, unauthorized changes to MAMS information on specifications, etc., could occur. Therefore, efforts were underway to write a QA program for MAMS. In addition, MAMS had the capability of combining like orders from different sites for non-QA material, but according to MMSS personnel was constrained by Office of the General Counsel (OGC) requirements such that it

could not be used. MAMS was also disadvantaged by not having a complete usage history of inventory items. Inventory items could be supplemented by field purchases and direct charge purchases which never became a part of a usage history.

D. Apparent Lack of Planning

NSRS did not specifically look at work planning and its associated impact upon the procurement process during the review. It was covered in an NSRS review of outage controls (see NSRS Report R-84-27-SQN/BFN). It was evident, however, from the conspicuous absence of the discussion of a planning or scheduling phase during interviews that whatever work planning was occurring, it had little positive effect upon procurement. That observation was supported by the identified fact that engineers at the plants were scheduling modifications without having the needed material onsite, with unrealistic expectations on delivery dates, and were using a large number of emergency purchases. Engineers were relying upon their ability to find the needed material somewhere within the TVA system when ordered material had not arrived onsite. The review did not attempt to determine how many jobs required cancellation or rescheduling due to material shortages. Contributing to the problem of planning work was the fact that no one interviewed really knew how long it took to procure an item. It is understood by NSRS that there is no one timeframe applicable to all items procured. Examples were found by and identified to NSRS of procurements that ranged from a few days to over three years and still waiting. A reasonable estimate should be established for routine procurements based upon past procurements, be it six months or one year, for use in planning and scheduling.

E. Quality Assurance

The quality requirements for items procured was a portion of this review. The Operational Quality Assurance Manual (OQAM) was reviewed with regard to procurement and found to be rather cumbersome and conflicting in some cases. The main problems identified were the intermingling of 10CFR Part 21 requirements with quality assurance requirements and the use of commercial grade items as basic components.

The quality level I and II designation is used for basic components and 10CFR21 applicability was determined for all procurements with those QA level designations. In the determination of Part 21 applicability, Part 21 could be determined not applicable because the item being procured was a commercial grade item. If it were a commercial grade item then the quality requirements could be significantly reduced to allow the procurement from an unapproved vendor and receipt inspection by an inspector not qualified to ANSI N45.2.6. The OQAM, Part III, Section 2.1, Appendix F, form for determining Part 21 applicability was deficient and was being misused in that if an item was identified as commercial grade no determination was required of its effect upon

the safety function of a CSSC component or system. Many QA level I and II, Part 21 N/A, procurements of commercial grade items were seen. All procurements, however, in the QA level I category required TVA-approved vendors and quality documentation. For those with a QA level II designation, which is almost equally important from a safety standpoint as a QA level I item, most required no QA documentation. Procurement with a QA level designation and no QA documentation or manufacturing requirements results in an implied level of quality that just may not be there; also, it results in purchased equipment whose quality characteristics are not known.

The use of commercial grade items as basic components is allowed by the NRC. In using such an item as a basic component TVA assumes the sole responsibility of assuring that that item will perform as required when required, including an accident situation. Currently TVA has no receipt inspection program for commercial grade items that includes testing or some other mechanism, such as vendor audit, that can make that assurance.

Considering the five basic categories of problems enumerated above and other findings identified elsewhere within this report, a comparison was made with the findings of the NUC PR Procurement Problems Task Force Report. With regard to the work of the task force and their findings, NSRS believes it represents a good work effort. Based upon the findings of this review, NSRS can support many of their recommendations that are directed toward changing the system, such as:

- Establish a planning group
- Improve PURCH/site communications
- Eliminate unnecessary procurement cycle steps
- Better utilize automated systems

NSRS understood that many of these recommendations were being implemented, but did not review the extent of the implementation. Other task force recommendations, however, appeared to be directed toward correcting the system as is or developing the ability to place blame within the present system with which NSRS does not agree.

In the details of this report additional problems are identified in the areas of approving vendor services, documentation inadequacies with internal TVA transfers, TVA-fabricated equipment, receipt inspection program, and materials with a limited shelf life. As negative as the findings may be, NSRS wants to emphasize that the findings are not for the most part people problems but are system problems. People did not have the procedures or training to perform the task more efficiently.

An NSRS suggested solution to the problems found during this review is contained in Attachment 1.

IV. CONCLUSIONS AND RECOMMENDATIONS

A. R-84-17-NPS-01, The Procurement System is Too Cumbersome and Not Well Known by the Users

Conclusion

The biggest problem found with the procurement system used by NUC PR was its wasteful and cumbersome nature. Procurements were overloaded with redundant reviews producing little value added in most cases and causing unnecessary time delays up to months. Virtually anyone could initiate a procurement action with little or no training. No one was found in the procurement process that knew the process much beyond their sphere of involvement. That resulted in unrealistic expectations being placed upon the system by the originator with regard to delivery time, and in perpetuation by others in the process who did not correct the problems or expectations. One of the more cumbersome and redundant review processes occurred within the NCO, and the removal of that review process on October 1, 1984, with the transfer of people to the sites, will help streamline the process provided the Central Office problems were not also transferred to the sites. To correct the problems with the system, drastic introspective management analysis and action are required (see sections V.B.1, .2, .3, .4b; V.D; and V.H).

Recommendations

R-84-17-NPS-01A
The Procurement Problem Task Force recommendation to eliminate all unnecessary steps in the procurement cycle with the goal of placing very few, if any, steps between the requisitioner and the purchasing agent should be given the highest priority.

R-84-17-NPS-01B
A formalized, documented training program covering the entire procurement process should be developed and required for all personnel within the procurement cycle from the originator (requisitioner) through the purchasing agent.

R-84-17-NPS-01C
A realistic timeframe(s) should be established for routine non-special order procurements, based upon past experience, to cover the time required from procurement origination through receipt of the material onsite. A mechanism should be included in the procurement system to periodically evaluate and adjust that timeframe as necessary, as well as communicate the timeframe to involved personnel (planners, procurers, etc.).

R-84-17-NPS-01D
Material availability and procurement timeframes should be included in all maintenance and modification planning activities. (NOTE: This recommendation is predicated upon information that NUC PR is developing a maintenance and modification planning and

scheduling function at each site. Also see NSRS Report R-84-27-SQN/BFN on outage control.)

B. R-84-17-NPS-02, Lack of Approval of Onsite Vendor Services at SQN

Conclusion

The OQAM, Part III, Section 2.1, paragraph 10 requires and identifies three acceptable methods for evaluating and accepting the work performed onsite by vendors. Contrary to that requirement SQN received services on three separate occasions and could not provide, after repeated requests, objective evidence that the service had been evaluated and accepted in accordance with the OQAM requirement. It is therefore concluded that OQAM, Part III, Section 2.1, paragraph 10 is not being implemented at SQN. (See section V.B.4.a.)

Recommendation

SQN should develop and implement a program that satisfies the requirement and intent of OQAM, Part III, Section 2.1, paragraph 10.

C. R-84-17-NPS-03, Excessive Review of Requests for Deliveries (RDs) on IQT Contracts

Conclusion

NUC PR was reviewing and approving RDs with the same rigor as the IQT contract, against which the RDs were written. That constituted a redundant effort costing 20 days or more delay in receipt of the commodity or service. (See section V.B.4.b.)

Recommendation

NUC PR should streamline its procedure for the review and approval of RDs, with no change of contract involved, to be in line with the requirements of the TVA Procurement Manual.

D. R-84-17-NPS-04, Insufficient Documentation for Transferred Material

Conclusion

ID-QAP 4.3 requires the original contract to be reviewed by the site receiving the transferred material for technical and QA requirements. No objective evidence could be found substantiating compliance. Sites requesting material to be transferred to them by another TVA organization or location did not specifically identify documentation requirements or require a copy of the original contract the material was purchased under. Therefore, the receiving site had a limited basis for accepting material during the receipt inspection process. The site assumed that all

applicable documentation had been sent by the transferring organization without knowing exactly what the original specifications (technical/QA) were. (See sections V.B.4.c and V.B.5.)

Recommendation

NUC PR should implement the requirements specified in ID-QAP 4.3 regarding transferred material. A copy of the original contract should be in the possession of and used by the receiving site during receipt inspection, and QC documentation required with the transfer should be specifically identified.

E. R-84-17-NPS-05, Cable Assemblies at BFN with Assigned QA Level I Designations Fabricated by TVA from QA Level II Parts with No Mechanism to Upgrade QA Classification

Conclusion

Cable assemblies manufactured by TVA were improperly classified QA level I items. The assemblies were manufactured from parts with a lesser QA level II designation and no mechanism was found that was capable of upgrading the QA level designation. (See section V.B.4.d.)

Recommendation

BFN should take whatever steps are necessary to assure that the cable assemblies, identified in section V.B.4.d, in stock, in use, and fabricated in the future satisfy the technical and QA specifications required.

F. R-84-17-NPS-06, BFN Power Stores Receipt Inspected Material Not Trained to Inspect

Conclusion

Power Stores receipt inspectors are not trained to receive material with Certificates of Compliance or Certificates of Conformance (COC), Certified Mill Test Reports (CMTR), or other similar QC documentation. On at least two separate occasions BFN Power Stores personnel receipt inspected and accepted material with CMTRs. One CMTR was for different material than specified in the contract and was not nonconformed. The other CMTR was for similar material substituted by the vendor but no TVA approval of the substitution was found. While the OQAM, Part III, Section 2.2 does not prohibit Power Stores personnel from receipt inspecting material with QC documentation, they should not be allowed to receipt inspect shipments with QC documentation they have not been trained to interpret. (See section V.B.5.)

Recommendation

NUC PR should revise the OQAM to prohibit receipt inspection of material with QC documentation by Power Stores and that BFN evaluate and take corrective action as necessary for the items identified in section V.B.5.

G. R-84-17-NPS-07, Material With Limited Shelf Life Not Reordered In a Timely Manner

Conclusion

The OQAM and DPM system of procedures required the periodic inspection of material with limited shelf life at one half the shelf life. Through DPM revisions that OQAM requirement was deleted. Prior to deletion of the requirement, periodic inspection was being performed (before the shelf life expired) at BFN but not at SQN. BFN required (BF 16.4) reordering shelf life material with a three-month lead time, but SQN had no requirement. Neither BFN nor SQN were reordering material with sufficient lead time to have new material in place before the existing material shelf life expired. Considering the latest industry philosophy regarding shelf life material, as contained in ANSI/ASME NQA-1-1979, the deletion of inspection requirements and reordering of items with insufficient lead times to assume an adequate supply of fresh material is considered inappropriate. (See sections V.B.6 and V.B.2.)

Recommendation

NUC PR should revise the OQAM to establish programs to inspect and reorder shelf life material to assure an adequate supply of fresh material. Also, the current three-month reorder lead time specified in DPM N77A2 should be reevaluated and adjusted as necessary.

H. R-84-17-NPS-08, Materials Management System (MAMS) Under Utilized

Conclusion

The MAMS system was being under utilized in that its capability to track inventory items usage and to reorder inventory items automatically was not being used. Considerable manpower was being expended to perform those functions manually and the MAMS system was not receiving all sources of inventory item usage. One deterrent to a more complete utilization of MAMS was the fact that its program did not have any quality assurance control to prevent unauthorized changes to specifications or other QC information. Efforts were reported to be underway to prepare a quality control feature for MAMS which NSRS highly endorses. (See section V.C.1.)

Recommendation

NUC PR, Power Stores, and the Materials Management Services Section should jointly increase efforts to utilize the MAMS in the most effective and efficient manner possible.

I. R-84-17-NPS-09, 10CFR21 Requirements Incorrectly Linked to NUC PR QA Requirements

Conclusion

Reporting of defects and liability requirements imposed upon vendors by 10CFR21 were incorrectly linked in the OQAM to NUC PR quality levels in that if 10CFR21 was determined not applicable then manufacturing quality and receipt inspection requirements were automatically reduced. In addition, the OQAM, Part III, Section 2.1, Appendix F, attachment 1, form for determining Part 21 applicability was incorrect in that if an item was determined to be commercial grade then its affect upon or use within a Critical System Structure or Component (CSSC) could incorrectly be ignored. NUC PR agreed with NSRS that the form should be corrected. (See section V.G.)

Recommendation

The OQAM and NUC PR procedures should be revised to remove influences of 10CFR21 applicability upon the determination of required quality levels for items and services, and training in the requirements and limitations of 10CFR21 should be provided to all personnel in the procurement cycle. It is further recommended that the OQAM, Part III, Section 2.1, Appendix F, attachment 1, be corrected as soon as possible and separated from the general OQAM revision so that all questions on the form are answered whether or not 10CFR21 is applicable to the item or service.

J. R-84-17-NPS-10, Commercial Grade Items with QA Level I and II Designations

Conclusion

Commercial grade items were being purchased with little or no QA requirements or from vendors or manufacturers without TVA-approved QA programs and classified as QA level I or II. That practice was contradictory to the purpose of having QA level I and II items with considerable QC documentation attesting to its suitability for fulfilling an intended function. (See section V.G.)

Recommendation

Items purchased with no QA requirement or requirements for material certifications (COC, CMTR, etc.) and/or from vendors or manufacturers without TVA-approved QA programs should not be purchased with a QA level I or II designation.

K. R-84-17-NPS-11, Quality Verification for Commercial Grade Items

Conclusion

The use of commercial grade items as basic components places the responsibility for assuring that the item will function as intended under all conditions solely upon TVA. The QA program within TVA, at the time of this review, was not capable of providing that assurance because it did not include a receipt inspection program which included testing of the item or comparable mechanisms such as an audit of the vendor's QA program for commercial grade items. (See sections V.E and V.G.)

Recommendation

NUC PR should establish a receipt inspection program which includes testing or comparable mechanisms, such as audit of vendor's QA program, verification of certificate of conformance, etc., for replacement commercial grade items that will be dedicated as basic components or parts thereof, that would provide documented assurance that the item will function as intended when necessary including accident conditions.

L. R-84-17-NPS-12, Receipt Inspection of QA Level I and II Items by FQE

Conclusion

Considering the changes recommended by NSRS in this report with regard to the procurement of quality level I and II material and commercial grade items to be dedicated as basic components, the division of receipt inspections between FQE and Power Stores, in effect during this review, will be inadequate. (See sections V.B.5 and V.G.)

Recommendation

All items procured as QA level I and II and commercial grade items to be dedicated as basic components should be receipt inspected by FQE or others qualified to ANSI N45.2.6.

V. DETAILS

The procurement process was evaluated by review of pertinent NRC regulations, consensus standards TVA was committed to, TVA policy documents, and various levels of procedures. As part of this evaluation the procedural flow of selected procurements was followed from the originator through PURCH. NSRS attempted to identify all points of origination for procurements for operating nuclear plants and included the site, OE, NCO, vendor-supplied items with services, and internal TVA transfers between sites and organizations. More than 100 procurement requisitions selected at random from BFN, SQN, OE, and the NCO were reviewed. Of those reviewed, 45 were selected for further study as being representative of basic types of procurements in the

electrical, mechanical, and structural areas. Those procurements included items, services, and internal transfers procured as either direct charge, Indefinite Quality Term (IQT), or Material Management System (MAMS) stock reorders. Of those 45 representative procurements, 21 were followed through the entire review and approval cycle to PURCH. Of the 21 procurement actions, 12 were classified as emergencies with the remaining 9 being normal procurements.

During the course of selecting and following the procurement actions over 90 people at all organizational levels were interviewed with regard to their function in the procurement process and problems associated with their function. At PURCH, in addition to their function within the TVA procurement system, discussions were held on U.S. Government procurement regulations and their impact upon TVA.

A. Upper-Tier Documents

Throughout this review an evaluation was made which compared the regulatory requirements contained within the Code of Federal Regulations and Regulatory Guides committed to in the TVA Topical Report against TVA implementing procedures which included the Operational Quality Assurance Manual (OQAM), Division of Nuclear Power Area Program Manuals (DPMs), TVA Procurement Manual, Office of Engineering Engineering Procedures (EPs), and applicable plant procedures. A detailed listing of the procedures reviewed is contained in section VII, "Documents Reviewed." Except as noted elsewhere in this report TVA's procurement programs in the areas reviewed were in compliance with regulatory requirements and were implemented in accordance with program procedures.

B. Plants

Activities at the plants associated with the procurement and ultimate receipt and storage of items and services were reviewed. Unless otherwise specified the findings are applicable to both SQN and BFN.

1. Preparation and Review Cycle for Procurements

At the sites procurement activities were confined primarily to five groups or functional areas of responsibility. Each is discussed as follows:

a. Originator

Once it was determined that an item or service was required for operation of the nuclear plants, some individual had to initiate the procurement action and that individual was identified as the originator. As identified in the OQAM and found in practice, the originator could be any "cognizant engineer, supervisor, or responsible designee." The originator was responsible for preparing the purchase request and specifying thereon all technical and quality require-

ments associated with the procurement. Specifically, as assigned in the OQAM, the originator should:

- be familiar with the functions of the system the procurement is associated with,
- be familiar with the system's importance to safety, and
- be familiar with the compliance, technical, and quality requirements of the system.

With regard to the above requirements, and in procurements evaluated during this review, all originators occupied positions where the qualifications to fill those positions satisfy the requirements to procure items and services specified above. In addition, the originator was required by the OQAM to specify completely and accurately on the purchase request as applicable the following:

- technical description of the procurement
- component or system of use
- applicable regulatory code
- QA level
- design basis
- other manufacturing requirements
- identify required tests, inspections, and examinations
- list documentation requirements
- specify special handling, packaging, or storage requirements
- determine the original EN DES procurement QA requirements
- evaluate 10CFR21 applicability
- identify special receipt inspection requirements
- specify the date the procurement is wanted

If the originator prepared the purchase request completely and accurately as required, no additional review of the document would be necessary. In practice, contrary to established procedures, the originator was not expected to complete the purchase request

accurately and completely for all items above. For example, the manufacturer was expected to provide information on special handling, packaging and storage requirements. The final QA requirements and Part 21 applicability were specified by the plant FQE staff and NCO QA Group, and the final technical requirements were specified by the plant specifications engineer and NCO QE Group.

NSRS expected to find a training program for originators in place and functioning. None was found nor required. What NSRS found instead was a description by plant supervision of self training. Originators were given a copy of the plant procurement procedure(s) and expected to learn on their own. As a result NSRS found that most routine purchase requests prepared by the originator were changed at the site by FQE and Materials personnel. Those changes ranged from significant (wrong QA level or technical specification identified) to editorial.

While the originator was required by the OQAM and plant procedures to specify everything required in all procurements, they were not given the training necessary to accomplish this.

b. Specifications Engineer

Once the originator completed the purchase request, it was sent to the specifications engineer for review of the technical specifications and coordination with FQE, other maintenance and engineering staffs, administrative staff, Power Stores, and Plant Manager for their input and/or approval. Basically, the specifications engineer was to assure the purchase request was complete and accurate. Both BFN and SQN handled this function somewhat differently.

At BFN there were positions of specifications engineer for both Operations and Field Services that were staffed with engineers. For the Field Services Group, the specifications engineer assumed the function of the originator and filled out the purchase request. For the Operations Group the specifications engineer reviewed the purchase request as completed by the originator.

At SQN the specifications engineer's function was fulfilled by a materials officer from the Materials Unit, who did not have an engineering background. An attempt to review technical specifications was described, but the function primarily fulfilled by the materials officer was one of expediting procurements.

Materials Units at both SQN and BFN were the principal points of contact for technical and quality assurance changes made in the NCO. The NCO Quality Engineering Branch (QE) prepared a form letter to the file with the name of the originator and FQE persons concurring by telephone with the NCO changes. It reportedly was the materials officer's responsibiltiy to assure that input by the originator or FQE was obtained before the changes were approved. On procurements reviewed by NSRS, a variety of names were listed for the originator concurrence on the QE form letters and most of the time the individual was other than the originator of the purchase request. Some approvals were by technical personnel and others were by the materials officer. In all cases the plant FQE engineer approving the purchase requisition concurred in the NCO changes. The materials officers are not technically qualified to approve either technical or QA changes on their own. Contrary to obtaining the originator's approval for changes, a materials officer concurred in the NCO proposed changes for the originator on four specific instances (PRs 942988, 951133, 951028, 951134). With regard to PR 951134, correspondence was found in the procurement file which showed the originator was aware and agreed with the change before the materials officer approval was given; however, in the other three cases no similar correspondence was found. With regard to PR 951028, the NCO added a technical requirement in which the materials officer concurred and to which the vendor took exception. The plant wanted zinc-coated sheet-metal, not oiled. The NCO added not chemically treated. The vendor quoted chemically treated no oil, and the site materials officer approved the exception. The site materials officer should not be concurring in technical changes.

c. Field Quality Engineering

The FQE was also required to review quality level I and II purchase requests for completeness and accuracy and approve them. At both SQN and BFN one individual in each FQE group was assigned that responsibility. NSRS found that in addition to the procurements of quality level I and II materials and services, which this review centered on, that engineer also reviewed all QA level III and IV procurements as well as non-QA procurements. Virtually everything procured by the sites was approved and/or reviewed to assure the proper quality level and requirements had been placed upon the item being procured. To put perspective on the magnitude of that effort, BFN Power Stores provided a compilation of procurements for May and June 1984 showing 1051 procurements, but did not show how many were quality level I or II. That number of procurements

would result in an average of about 26 per day requiring FQE approval and/or review.

The FQE review of purchase requests was described in the OQAM as including both technical and QA requirements. Depending upon the item or service being procured, FQE described their review as including a comparison of the technical and QA requirements with plant drawings and previous procurements of the same item. Although NSRS did not physically observe this type of FQE review in process, the mechanics involved in performing it could be time consuming. Considering the number of purchase requests reviewed, the completeness of each review is of concern. FQE described the number of procurements found with QA levels lower than required as very few. The routine review of non-QA procurements by FQE is, therefore, considered too time consuming by NSRS for the benefit received, and the time expended could be more effectively utilized on QA procurements. A periodic review of a sampling of non-QA purchases should be sufficient to detect program deficiencies.

d. Plant Superintendent

Upon completion of the review and approval cycle of the purchase request, the purchase request was sent to the plant superintendent for authorization.

e. Power Stores

The purchase request with all approvals and authorization was sent to Power Stores for determination of the appropriate method of obtaining the requested item(s) and preparation of the associated documentation. Methods available to Power Stores included direct charge procurement using a purchase requisition, request for delivery under an existing IQT contract, field purchase order to purchase items of less than $300 from local suppliers, or a transfer requisition (TR) used to transfer items from one TVA site to another. Once the method was selected, Power Stores would transcribe the approved purchase request writeup verbatim on the appropriate form and add on all the purchase request attachments which could include QA and technical specification requirements. Procurement forms prepared by Power Stores were defined as QA documents in OQAM, Part III, Section 4.1, whereas purchase requests were viewed as worksheets and not QA documents. The official QA document requires the signature of FQE and the plant superintendent. Consequently both FQE and the plant superintendent were signing the same procurement action twice.

The purchase request as described to NSRS was not considered a QA document because it was not always prepared in indelible ink and could fade with time. The OQAM, Part III, Section 4.1, required QA records to be prepared in ink or typed. In conflict with that philosophy, NSRS observed attachments to purchase requests (e.g., 10CFR21 applicability form) that were prepared partially in pencil that were classified as QA documents.

In addition to signing the procurement form for record purposes, FQE was required to review for accuracy the documents prepared and attachments included by Power Stores. As FQE was the last to see the purchase request prior to transcription by Power Stores, their review of the finished product was editorial in nature. FQE, therefore, was required by the OQAM to duplicate its own work in reviewing and approving all QA level I and II purchases twice.

Like FQE, the Plant Manager was required to duplicate his work in authorizing both the purchase request and purchase requisition.

With all required signatures obtained, Power Stores then transmitted the procurement package to Chattanooga for additional review and approval prior to going to Purchasing.

2. Functioning of Plant Procurement System

Overall, the procurement system used at the plants contained redundancy and was predicated on the concept that additional review will promote a better product. As a general rule, a procurement from the identification of need through transmittal of the completed procurement package to Chattanooga contained 10 signatures or initials signifying review and approval of the procurement. Generally that preparation process was not considered excessive--taking 4 days for normal and emergency direct charges and 1 day for an emergency RD under an IQT contract. More complicated procurements were seen that required 1 to 1-1/2 months to prepare. Of the 10 signatures or initials in the approval process, only 2--the originator and FQE--were identified as having any substantive technical or quality contribution regarding the specifications or requirements of the procurement.

Personnel interviewed were generally aware of their function in the TVA procurement process but generally unaware of the function and responsibility of other sections in that process. With the possible exception of Power Stores, knowledge of Federal procurement regulations was lacking among those in the procurement process. For example, no one at

the plants or elsewhere within TVA knew how long it normally took to procure something. They knew it took too long and seldom arrived when needed. The originators would allow 90 days, in establishing a date wanted, from the time a purchase request was initiated for routine nonemergency purchases of common items (nuts, bolts, steel, etc.), until those items were expected onsite. That date was virtually ignored throughout the procurement process. As a result, the purchase requisition rarely arrived at Purchasing with sufficient time remaining, until the date wanted was passed, to advertise for quotations, let alone time to review the quotations, award the contract, manufacture, and deliver the item. Of the requisitions reviewed by NSRS, it typically took 6 months to 1 year to receive material onsite and the sites were only allowing 90 days. With sufficient training in and knowledge of the procurement process, personnel within the procurement cycle could establish more realistic timeframes in order to receive needed materials.

Probably the most frequent complaint about the procurement process expressed by the sites was the material was not there when needed. That complaint results in large part from the unrealistic expectation placed by the sites upon the procurement system. The site routinely wants rapid results, 90 days or less, and the system can't handle it.

3. Planning

Planning of work for maintenance and outage modifications was not a formal part of the review; however, the obtaining of needed materials to perform work was reviewed. As identified in section V.B.2 above, ordered material often arrived at the site after it was needed. There was no one factor producing that condition but several factors beginning with the originator and including all steps through receipt inspection of the material onsite. This review found, however, no evidence onsite that material availability was factored into the work planning process. For further information on planning see NSRS Report R-84-27-SQN/BFN (GNS 841220 052).

For example, in ECN modification work, site personnel explained that a complete ECN package was not received from EN DES. Portions of the ECN package would arrive in stages over some time period. When the sites had what they believed to be a sufficient amount of the ECN package, work would be scheduled and materials ordered. Engineers at the site in charge of the modification work and ordering material openly stated that the material ordered would probably not arrive before the ECN work started. Considering that only 90 days was allowed for procurements that sometimes took 6 months to 1 year to get, that expressed concern was well founded. No effort was found, however, to include a more realistic timeframe in the site planning process.

What resulted, when the material did not arrive, was an exercise in resourcefulness by the site engineers, which they appeared to be very good at. The engineers had to find the material they needed through borrowing it from another engineer onsite, finding it at another site and having it transferred, or initiating an emergency purchase. Effective as the engineers may be, that effort in resourcefulness is time consuming and wasteful. With more effective planning a significant portion of the time wasted on obtaining materials could be eliminated.

4. Special Methods of Obtaining Material and Services

Much of the emphasis within this report is generic in nature and applicable to all types of procurements. The sites have numerous methods available for obtaining material and services with direct charge contract being the most common. This review examined not only direct charge contracts but other methods as well, and this section will focus on the less common methods and their associated strengths and weaknesses.

a. Service Contracts

Often the services of consultants or workers with specialized expertise is required. Like materials being received at a site requires a receipt inspection, the receipt of a service at the site also requires an evaluation and acceptance. Acceptance of a service is identified in the OQAM, and three acceptance methods are listed in Part III, Section 2.1, paragraph 10.0.

On two separate occasions, SQN obtained the service of Furmanite, Inc., to stop leaks which could cause a shutdown of the plant. Those services were requested by purchase requisitions 959104 and 955163 which specified the vendor shall comply with the technical requirements of IQT contract 82P38-925403. On one occasion a Gulf and Western Service representative was requested, under purchase requisition 940060, to perform work at SQN. In all three requisitions it was specified that work was to be performed and documented under TVA procedures and QA program. On four separate occasions SQN mechanical maintenance and compliance personnel were asked for the documentation contained in the work packages or elsewhere, for those three contracts, which satisfied the acceptance of service requirement of the OQAM specified above. NSRS did not receive any such documentation or an explanation of how the OQAM was satisfied and must therefore conclude that it does not exist.

b. Indefinite Quantity Term (IQT) Contracts

An IQT contract can be a powerful tool when procuring the same item or service on a routine, repetitive basis. When such an item or service is identified, an IQT contract can be prepared following the same preparation, review, and approval procedures as if it were a direct charge contract. IQTs are advertised, sent out for bids, bids reviewed, and contract awarded no differently than a direct charge. The difference is, or is supposed to be, that when items or services are required under the IQT, a request for delivery (RD) is prepared and sent to the vendor bypassing the review and approval process. That procedure was described as being followed by EN DES, but not by NUC PR. In each of the NUC PR RDs reviewed by NSRS, the RDs went through the same review process as the original IQT, therefore eliminating any savings of manpower or time gained by having the IQT. Arguments were presented that the IQT was not an actual contract, the RD was, and therefore, had to go through the review and approval process to satisfy QA documentation requirements. It was also argued that some times not everything purchased on an RD was covered by the original IQT.

With regard to the first argument, the IQT contract was retained as a QA record as were the RDs. The RDs specified what was wanted and that the terms and conditions of the specified IQT were applicable. Each RD was reviewed onsite by FQE and an authorizing official. In NSRS's opinion that should be sufficient to satisfy QA requirements and the RD should be sent directly to the manufacturer.

With regard to the second argument, NSRS views that as a completely separate issue. If an item or service is to be procured that was not initially contracted for and a change of contract was required, then it should go through the review and approval process as a new procurement.

The additional review of all RDs by the central office only resulted in additional verbiage added to the RD which was already contained in the IQT. This was considered redundant, unnecessary, and resulting in needless time delays of 20 days or more. The elimination of the central office review by NUC PR should eliminate the problem provided that in transferring the central office positions to the sites the problem was not transferred as well. That transfer was not evaluated as a part of this review.

In the process of evaluating procurement at the sites, a number of direct charge procurements for steel were

observed at each site. It would appear that an IQT contract would be beneficial for those.

A problem with IQTs developed and was apparently solved during this review. It was determined by OGC that RDs for greater than $10,000 would require advertising in the Commerce Business Daily as if it were a new purchase. OGC later determined according to NUC PR personnel that a periodic generic advertisement would be sufficient to satisfy Federal procurement regulations.

c. Transfers

Although the transfer of material within TVA is not a procurement in a true sense of the word, transfers do provide another commonly used mechanism of introducing new materials to a site. As such, transfers were reviewed as if they were procurements from outside vendors.

Many items have become available for transfer due to the cancelled units and are shipped to all nuclear sites from HTN. Other transfers occur when one operating plant or construction site has unused material which can meet an emergency need at another site.

> NOTE: Transfer of electrical cable was not pursued by the review team at this time. Both BFN and SQN personnel identified documentation problems occuring with cable which resulted in a large number of nonconformances being written. Basic problems as told to NSRS stemmed from lack of coordination between the Office of Engineering (Electrical Engineering Branch) and the site and erroneous documentation accompanying cable transferred from HTN. Most participants were already aware of the problem and appeared to be working on a solution. The implementation of the solution will be subject to review in Phase II of the NSRS procurement review.

Form TVA 4139, "Request for Shipment of Material," is used as the means to transfer items between divisions (NUC PR and CONST). The form includes descriptive information of the item, Part 21 applicability, FQE signoff, and other miscellaneous signatures. ID-QAP 4.3, "Transfer of Items," states that the following steps are to occur in an interdivisional transfer:

(1) Requesting organization establishes a source of available items.

(2) Requesting organization prepares request for transfer.

(3) Requesting organization reviews original contract for technical and QA requirements.

(4) Copies of all appropriate records are transferred on the requested item.

(5) Source organization transfers materials to requesting organization.

No objective evidence of a technical review occurring utilizing the original contract could be identified. In actuality, the original contract (or copy) was not even requested of the source organization. Site procedures and transfer requisitions were not specific enough in stating what documentation was to be sent. The statement generally found on all transfer requisitions was, "all applicable documentation" to be included. The employee performing the receipt inspection cannot discern if all appropriate documentation has been received if it is not known what the contract required. The decision as to what was applicable documentation was the responsibility of Power Stores personnel transferring and receiving the material. Quality level material with COCs and CMTRs were among the items Power Stores was allowed but not trained to receipt inspect.

Another concern identified by NSRS involved the significant number of nonconformances written against material shipped from cancelled units, usually due to the absence of the original receipt inspection report or a disagreement between the material shipped and the original receipt inspection report. In some cases materials have been transferred that are similar to items requested but technically not the same and useless to the requesting organizations. Site employees expressed concern in utilizing HTN as a source of material. Basically, they had no assurance that the material received on a transfer would be what was originally requested. Although the HTN shipping process wasn't reviewed, NSRS identified enough nonconformed material to substantiate the concern. Controlling of the HTN warehouse will be transferred from Construction to Power Stores in the future and Power Stores will establish it as a distribution center, similar to the present Power Stores distribution center located in Chattanooga. The documentation problem on transferred materials was not limited to HTN. Inadequate documentation similar to that found from HTN occurred from other plant sites transferring materials.

d. TVA Fabricated Equipment

Another mechanism to introduce quality equipment into the plants was for TVA to manufacture the part from stock material. This review had planned to include work performed by the Power Service Shop for the nuclear plants as if the shop were another vendor. This was eliminated from the scope of the review after the BFN review segment because of time constraints.

Before the TVA-fabricated equipment was removed from the review, operator console cables and control festoons for the BFN refueling platform and jib crane were identified to NSRS as being fabricated by BFN electrical maintenance personnel. Completed control cables and festoons were classified as QA level I items and stocked within the Power Stores warehouse. Documentation was obtained on the manufacturing process and materials used to fabricate those items. It was determined that each cable and festoon was manufactured from QA level II parts and nothing could be found that showed how the assembling of QA Level II parts produced a QA level I finished product. When identified to BFN management, they assured NSRS the matter would be corrected.

5. Receipt Inspection Program

The NSRS review of the receipt inspection program was limited to a review of receipt inspection reports and associated documentation. Actual receipt inspections being performed were not observed. The review effort consisted of selecting requisitions at random from the Power Stores files and verifying that all documents requested in the contract had been received. The proper group performing receipt inspection as required by the QA level assigned to the item was noted, i.e., Power Stores personnel only or Power Stores assisted by FQE inspectors. Selected personnel from Power Stores and FQE were interviewed to verify their understanding of receipt inspection procedures. Those interviewed appeared knowledgeable of the NUC PR requirements and site procedures.

The receipt inspection program at the nuclear plant sites was directly linked to the quality level assigned to the material being received. As such, material with higher quality levels was inspected by FQE inspectors and material with lower or no quality level was inspected by Power Stores clerks. The types of inspection performed in those groups had varying degrees of difficulty associated with them, and therefore, the inspector training was significantly different between FQE and Power Stores personnel.

FQE inspectors according to the OQAM perform receipt inspection of all QA level I, level II substituted items (items substituted by the vendor as being equivalent to those asked for), level II and ECN items to which 10CFR21 applies and which are shipped from the vendor directly to NUC PR. The FQE inspectors receive formal training and certification through the Power Training Center that meets requirements established in Regulatory Guide (RG) 1.58 (which endorses ANSI 45.2.6).

Certified Power Stores clerks according to the OQAM perform inspections of QA level II non-10CFR21 items, QA level III, QA level IV, and ECN material transferred to NUC PR from CONST regardless of the QA level. The Power Stores receiving clerks must be certified by the plant QA supervisors. To become certified, 550 hours of on-the-job training must be completed and an examination passed with a score of 70 percent or better. Recertification was required at intervals not to exceed 18 months. Power Stores personnel were also delegated (by the Topical Report) the responsibility of inspecting commercial grade items.

The separation of FQE versus Power Stores performing receipt inspection occurred at QA Level II and was determined by the applicability of 10CFR21, "Reporting of Defects and Noncompliances," to the item procured.

The documentation associated with the QA level II items can vary from certificate of conformance (COC) and certified material test reports (CMTR) to packing slips. During the course of the review, no consistency could be established for documentation required of QA level II items. For example, QA level II items with Part 21 applicability could require certificates of conformance provided by the manufacturer and/or certified materials test reports. Contracts for QA level II, Part 21 not applicable, could also have the previous same requirements and/or a packing slip.

If a packing slip is the only documentation requested on a requisition, the inspection is essentially a number check and the "on-the-job" Power Stores training is acceptable to perform the inspection. However, the appropriateness of Power Stores personnel performing document reviews (such as COCs and CMTRs) that they have not been trained to perform against material received is questionable and of concern to NSRS. Examples of Power Stores receipt inspecting CMTRs on QA level II (Part 21 N/A) material were identified at BFN. Material requested was to be either ASTM A336 or A479, type 316 stainless steel. The CMTR received with the material specified results for ASTM A276, type 316 stainless steel. The items weren't nonconformed. The difference in material was being evaluated by BFN PQA Staff at the conclusion of the review.

Two examples of materials being received without having all associated documentation were identified at SQN. Those items had been received on a transfer from HTN. They did not have the original CONST receiving reports and had not been nonconformed by SQN (reference V.B.4.c). The CONST reports were later obtained by Power Stores personnel.

A basis for 10CFR21 nonapplicability, in the NUC PR structure, was the determination of an item to be "commercial grade." A commercial grade item is considered to be an industry manufactured standard product with sufficient use history in non-nuclear applications to justify its use in a nuclear application. When Part 21 was determined not applicable, Power Stores receipt inspected the item.

ANSI 45.2.6, "Qualifications of Inspection, Examination, and Testing Personnel for Nuclear Power Plants," defines inspection as:

> A phase of quality control which by means of examination, observation, or measurement determines the conformance of materials, supplies, parts, components, appurtenances, systems, processes or structures to predetermined quality requirements.

The personnel who perform those inspections are required to also meet the established standards of inspectors as specified in ANSI 45.2.6.

If an item has no predetermined quality, then an inspection as defined above wouldn't apply, and the Power Stores clerks performing receipt inspection wouldn't have to be trained to ANSI 45.2.6 requirements.

The inspection and treatment of a commercial grade item takes on new meaning and becomes quite important if the item is subsequently used as a basic component. The Code of Federal Regulations in Part 21, "Reporting of Defects and Noncompliance," states that a commercial grade item can be designated for use as a basic component through "dedication." The dedication process is basically TVA accepting responsibility for the quality and performance of the dedicated commercial item. When TVA accepts that responsibility, there should be some documented assurance of the quality of the item.

That assurance could be established through means such as the following:

(1) audit of the supplier
(2) testing of the item
(3) verification of certificate of conformance
(4) maintaining records documenting supplier history

The previously identified concern of Power Stores personnel performing receipt inspection on materials and related documentation with minimal training becomes more important when realizing that commercial grade materials can be dedicated as basic components.

In many cases, the Power Stores clerk is the only one who has evaluated the quality of an item before use. The importance of the inspection has gone beyond the basic inventory of items received versus items ordered and should be performed by FQE inspectors who have been trained in the review of documentation related to procured items.

However, it also should be emphasized that the FQE inspector receiving the certificate of conformance is basically looking for a signature. The review team did not pursue the validity of COCs but did observe the following:

(1) No testing is routinely performed on material to verify material properties as stated by the manufacturers. (FQE formal training doesn't include testing methods.)

(2) Not all vendors who provide COCs with commercial grade material have been audited by TVA. Therefore, the value of the COC would be in question.

(3) Supplier and product history has not been maintained on materials received and used onsite. Therefore, no documented bases exist to substantiate the acceptability of an item on that basis.

As a result, for true commercial grade items supplied by a vendor without an approved QA program and with no supporting QA documentation, the only assuring activity remaining for TVA is testing of the item. No program was identified that tested items upon receipt. As identified in section V.G there are no TVA controls over the manufacturing process or materials for true commercial grade items that could assure that the materials of construction and operability of an item is acceptable for its intended purpose or its environment of operation.

NUC PR personnel stated a functional test of equipment was required when repaired or replaced. That test should provide suitable evidence that the item functions properly under normal operating conditions. That test will not, however, provide any assurance or demonstrate an ability of that item's functionability with respect to time or environmental conditions present during an accident. As such that functional test is unsuitable for assuring a commercial grade item used as a basic component will function as required during accident conditions.

6. Storage and Reorder of Shelf Life Items

The complete storage program for all procured materials was not reviewed at this time due to the amount of work already performed in this area by other TVA organizations. Some inadequacies and noncompliance to DPM requirements have already been identified through audits performed by OQAB of the storage program (reference audit report BF-8400-03). BFN and SQN are currently planning and building larger storage facilities which will reduce some inadequate storage conditions. Follow-up activity related to the OQAB audit should include a review of the entire storage function on all procured materials to verify compliance to DPM N82A17, "Equipment and Materials Storage Requirements for Nuclear Power Stores."

The NSRS review team limited the storage portion of the review to an area not emphasized in previous audits, but one that presented problems in the accessibility of materials. A basic problem identified by site employees was the unavailability of routine inventory items, with limited shelf life, when needed for maintenance. Examples told to the review team at both SQN and BFN involved rubber products such as O-rings and gaskets and chemicals reaching their expiration date with no suitable replacements available in stock. A limited review in the storage area was performed to address the specific problem of shelf life items. The storage requirements reviewed include the OQAM Part III, Sections 2.1 and 2.2, DPM N82A17, DPM N77A2, BF16.4 and SQA45. Discussions were held with Power Stores representatives who have the responsibility of performing the inspection on shelf life items and the subsequent reordering of materials. Records were also reviewed for completeness and accuracy of previous inspections performed.

The N-OQAM addresses the inspection of shelf life items in both Part III, Section 2.1, "Procurement of Materials, Components, Spare Parts, and Services," and in Part III, Section 2.2, "Receipt and Inspection, Handling, and Storage of Materials, Components, and Spare Parts."

Part III, Section 2.1, paragraph 3.2.3.4, "Limited (or Shelf) Life Material," states that "For additional guidance in NUC PR's policies with regard to limited shelf life or natural aging life refer to DPM No. N77A2, 'Storage and Shelf Life Considerations for Materials with Natural Aging Life.' This document covers requirements for procurement, receipt inspection, periodic inspection, and disposition."

While DPM N77A2 previously contained shelf life requirements its revision log under the entry of March 21, 1983, stated that the "Revision removes requirements for periodic inspection of materials with limited shelf life." It did, however, contain a requirement to reorder shelf life material

at least three months prior to their expiration date. As stated in section V.B.2, three months may be an inadequate lead time.

Part III, Section 2.2, paragraph 4.3, "Inspections," stated that "Inspections shall be performed and documented on a periodic basis to ensure the integrity of the item and its container is being maintained . . . specific inspection requirements for equipment and material are delineated in DPM N82A17." That DPM, which did not cover all items with a shelf life, was revised on September 7, 1984 removing the inspection criteria.

Consequently, whatever inspection process was intended by the OQAM reference to lower tier document requirements was lost with the revision of both DPMs.

The Power Stores personnel at SQN and BFN had different procedures for inspecting and reordering material with a shelf life. Materials were being inspected at SQN near the expiration date and then reordered. BFN was performing an inspection when material reached about one-half its specified shelf life but assigned a low priority to the reordering of those materials.

SQN Power Stores personnel did state that a shelf life item inspection program would be initiated in the near future, but no specified date was identified. The program described would provide an inspection at six months prior to the expiration date.

The BFN site procedure BF16.4, "Materials, Components, and Spare Parts Receipt, Handling, Storage, Issuing, Return to Store Room, and Transfer," was reviewed. BF16.4 referenced incorrectly DPM N77A2 for the storage and inspection requirements of shelf life items. BF16.4, section 4.8, was consistent with DPM N77A2 and required the reorder of shelf life material at least three months before the expiration date.

In order to verify compliance with the BF16.4 inspection requirement, three months of computer printouts (May 1984 through July 1984) were reviewed that listed all shelf life stock items due to expire during the month. A checkmark (✓) had been placed by each item by Power Stores personnel verifying that an inspection had been performed. That type of documentation did not meet the requirement of OQAM Part III, Section 2.2, paragraph 4.3, which stated that a form similar to attachment 4 of that OQAM section should be used for inspections.

At BFN five items were selected from the June 1984 computer listing to evaluate the shelf life inspection process. Of those, two items were judged by BFN to be in acceptable

condition and their shelf life did not expire for another year. Therefore, no reorder was required. In contrast, three of the items were due to expire within three months and had not been ordered, i.e., a purchase request had not been written.

It was emphasized by Power Stores that those small quantity items were not being reordered until a larger quantity order could be made. Certain constraints regarding the minimum dollar value of orders were imposed by PURCH and manufacturers. While the size of an order may be relevant, materials should also be ordered in a timely manner. The consolidation of orders to make "quantity orders" was the responsibility of MMSS. (See V.C.1 for details.)

When material exceeded its specified shelf life, which appeared to be a common occurrence, specific approval by PORC was required at each site to use the outdated item. More recent industry philosophy regarding materials with shelf life is contained within ANSI/ASME NQA-1-1979, Supplement 8S-1. That standard requires that shelf life items be identified and controlled to preclude the use of items exceeding their shelf life. As such it appears inappropriate for TVA to remove requirements regarding shelf life inspections, and to continue the practice of reordering material as the shelf life expires or without sufficient lead time to assure a supply of fresh material.

C. Power Stores

1. BFN and SQN Power Stores

The initial review time in Power Stores was spent gaining an understanding of the basic mechanics of site procurements (forms used, terminology, coordination required, time delays, etc.) and reviewing procurement files. Power Stores maintains a file on each procurement, which includes all available information relating to the specific procurement (request, requisition, receipt inspection report, etc.). Those files became a main source of information for the review team and, with a few minor exceptions, were essentially complete records. Various personnel were interviewed to ascertain their understanding of the total procurement system and to identify their specific responsibilities and problems within the procurement system. Those interviewed appeared conscientious in the performance of their understood responsibilities and demonstrated a willingness to assist the review team in locating documents relating to specific procurements. Areas reviewed included the automated reordering of stock items, the utilization of the MAMS database, the shelf life item inspection program, the Power Stores receipt inspection program, associated training, and handling of records. (The receipt inspection program and storage of shelf life items were previously discussed in sections V.B.5 and V.B.6 respectively.)

Power Stores personnel were responsible for typing requisitions and coordinating all the required signatures. They also helped locate needed materials within TVA by utilizing available information on the MAMS system and assisted in coordinating transfers of materials between divisions and other storerooms. Reorders of stock items were also initiated by Power Stores.

The utilization of the MAMS database onsite was controlled by Power Stores. Basic information on stock items was available through that system which functioned on a Reorder Point/Reorder Quantity (ROP/ROQ) concept. In principle a maximum (MAX) inventory level was established for each item to support plant needs without excessive inventory. A minimum level or ROP was also established which allowed sufficient time to order and receive replacements without exhausting the inventory. When the stock level reached the ROP (MIN) amount, an order could be placed for the ROQ to bring the inventory back to the MAX level. The MAMS system also contained data as to the date and amount of an item withdrawn at that time, i.e., a usage history.

The development, maintenance, and changes to the MAMS system with inputs from Power Stores and NUC PR are the responsibility of the Materials Management Services Staff (MMSS). The MAX-MIN levels have been evaluated by the MMSS in an attempt to better utilize stock inventories, either increasing or decreasing as necessary. A basic problem faced by site personnel was caused when MMSS reduced stock levels based on incomplete information. Procurements through emergency and field purchases on a specific item were not included in the MAMS usage history and therefore were not included in the evaluation. Both users of MAMS and MMSS personnel offered explanations of why and how stock shortages of certain items occurred. NSRS decided that ascertaining the validity of the explanations would not be fruitful, as Power Stores and MMSS were well aware of the problem and appeared to be cooperating in establishing meaningful usage histories to base stock reductions and increases on. Unfortunately, the originators at SQN and BFN felt they were being hampered in their work by not having basic materials available when needed. They considered MMSS the problem because MAX levels weren't high enough. A problem, resulting from the shortage of materials in stock and the MAX level being too low, was identified at both BFN and SQN and involved the hoarding of materials.

In an attempt to ensure adequate supplies when needed, user organizations would "buy out" certain items as they arrived in the stockroom, thus forcing the reorder of that item. Power Stores personnel stated that on some specific items, no matter what the MAX level was, they could never keep material in stock. Those items varied from mops and cleaning supplies to plastic bottles and electrical equipment.

Power Stores is responsible for issuing material when requested and not for questioning the usage of material.

The hoarding of materials demonstrated the frustration level experienced by maintenance and modification personnel and their lack of confidence in the procurement system. The hoarding problem was discussed with Power Stores personnel at the central office. Plans were described for better utilization of the Power Stores Distribution Center in Chattanooga as a source of heavy use stock items. Plans also included the establishment of the HTN warehouse as a Power Stores Distribution Center.

Contained within the MAMS system was the capability of MMSS in Chattanooga to monitor stock levels at all Power Stores locations and to automatically reorder material as the reorder point was reached. MAMS also had the capability of combining orders of like material, but MMSS personnel stated they were prohibited from using that feature by OGC. The reasoning behind that prohibition was not pursued.

While the MAMS system has an automatic reorder feature, it was not being utilized because the plant FQE was required by the OQAM to review and approve all procurements (both QA and non-QA). Consequently, all inventory reorders were prepared by hand and the combination of like orders by MMSS was performed by hand. Rationale for not using the automatic reorder system was that the MAMS system was not a QA system and changes to the information within MAMS (material specifications, QA level, etc.) could be made without QA knowledge or approval. Information was provided NSRS which explained efforts underway to develop a procedure acceptable to QA which would protect the MAMS system from unauthorized QA changes. NSRS highly endorses that effort. Upon completion of that feature the MAMS system should be usable to a larger extent, thus eliminating the considerable manpower requirements currently required to manually reorder inventoried material.

2. Power Stores Distribution Center

The review of the distribution center in Chattanooga and the Investment Recovery Program (IRP) warehouse at HTN was limited to discussions with Power Stores personnel.

Power Stores currently has a distribution center warehouse in Chattanooga. At the time of this review over 100 items were being stocked there. The basic concept of that center was to provide a warehouse of inventoried items that the plants stockrooms could draw from. Described plans included maintaining a 6-month supply of items, thereby allowing the plants to reduce their inventory and associated storage requirements. In concept that idea appears functionally sound but will require the cooperation of all concerned to

work within the system. During this review, Power Stores was having difficulty maintaining a stock of mops, for whenever a delivery was made to BFN to replenish their inventory, plant personnel would "buy out" the mops and hoard them. That process created a real shortage within the Power Stores system based upon a perceived shortage by the users.

The distribution center did not have a QA program, but Power Stores personnel stated that one was being developed. At the time of this review the only QA material stored at the center consisted of welding rod and dye penetrant. As Power Stores personnel were not ANSI N45.2.6 trained receipt inspectors, any quality material received at the center required an FQE inspector to go from SQN to the center to perform the receipt inspection.

The IRP associated with TVA's canceled nuclear plants provided a vast supply of materials to the remaining nuclear plants. Power Stores was in the process of taking control of the HTN IRP warehouse operation. It was described as containing material with an acquisition cost of approximately 100 million dollars including approximately 33,000 valves. Approximately 40 percent of that material had QA documentation sufficient to support use in a QA system and the remaining 60 percent was suitable for non-QA systems or fossil plants. Like the Chattanooga distribution center, Power Stores described plans to keep the HTN facility as a distribution center for large items. The HTN facility was also having QA and preventive maintenance procedures prepared.

D. Central Office

The central office portion of the procurement review primarily involved the following groups: Nuclear Central Office Quality Assurance Branch (NCO QEB), NCO Materials Management Section (MMS), Central Power Stores, and the Materials Management Services Staff of Operations Support.

The OQAM (Part III, Section 2.1) was the reference used to define the responsibilities that each of the above groups had in the procurement cycle. Flowcharts which correspond to the OQAM-defined responsibilities were found in DPM N72A14. Requisitions for QA level I and II (Part 21 applicable) materials and services were reviewed by various groups in the NCO. Power Stores and MMSS basically reviewed requisitions for inventory items. Implementation of these documents was evaluated with only a few exceptions to compliance identified.

It should be noted that many of the NCO procurement responsibilities and associated personnel had been transferred to the plant sites on October 1, 1984, and the organization reviewed by NSRS was the one in place prior to October 1, 1984. As the functions

and responsibilities no longer exist at the NCO, an individual breakdown of each organization and associated problems will not be presented but an overall summary is provided.

Procurements of QA level I and II 10CFR Part 21 applicable materials and services were circulated for review and approval throughout the NCO groups identified above. The only group with any visable impact upon a procurement package was the NCO QEB. Other groups provided signatures of approval or acknowledgement or were within the distribution cycle due to the mandates of organizational communications. The value added to the procurement documents by QEB on the 21 procurements followed from the sites through the NCO was minimal. For the most part QEB changes were editorial rather than substantive (e.g., changing the verbiage specifying 10CFR Part 21 was applicable). Technical review of procurements were also being performed by QEB. Both the OQAM and DPM N72A14 specified it was to be performed by the NCO technical branches when required. NSRS found that the technical branches who were previously performing most of the technical reviews were no longer doing so and it was being performed by an SC-2 mechanical engineer in QEB.

The responsibilities of the MMS were essentially clerical. They were to "coordinate central office NUC PR procurement communications among the nuclear plants, Power Stores, and the NCO." (OQAM Part III, Section 2.1, 2.2.1) They also performed a review of requisitions for "administrative correctness and completeness." (OQAM, PART III, Section 2.1, paragraph 3.1.2.10) The review was similar to others performed by the site and not considered necessary by NSRS. The MMS served as a paper coordinator that moved QA levels I and II requisitions between Power Stores, NCO QEB, and the technical branches. Files had also been maintained on specific requisitions, but these files were not evaluated for completeness because they were being transferred to the plant sites. The MMS also interfaced with other procurement groups on IQT contracts. The IQT tracking of available funds and administration of IQT contracts were functions still performed by the MMS after the October 1984 reorganization.

Findings regarding length of time to prepare, review, and approve procurements within NUC PR can be summarized as follows:

1. Normal direct charge procurements took 4 days to prepare and approve at the sites and 2 months to review and approve in the NCO.

2. Emergency direct charge procurements took 4 days to prepare and approve at the sites and 7 days to 1 month (15 days average) for the NCO to review and approve.

3. Emergency Requests for Delivery took 1 day to prepare and approve onsite and 8 to 14 days for the NCO to review and approve.

4. More complicated nonroutine procurement took 1.5 months for the site to prepare and approve and 7 months for the NCO to review and approve.

Even though NSRS found that the NCO provided little assistance on most procurements, one procurement of services to decontaminate and repair a Westinghouse CCP motor for SQN was reviewed where considerable NCO help and input was provided; however, considering all the procurement documents reviewed, the value added by the NCO could not support the continuation of several weeks or months delay between preparation at the site and transmitting the procurement package to vendors for bids. The NCO was not providing a service the plants could not provide for themselves with proper training. NSRS supports the NUC PR decision to eliminate the NCO from the procurement review cycle provided the function was not just transferred unchanged to the site.

E. Office of Engineering

The Office of Engineering (OE, formerly Engineering Design) was reviewed from the standpoint of their involvement in the procurement process for operating plants. Their involvement primarily consisted of design work on modifications. As a general rule if a modification involved the procurement of engineered items (valves, pumps, etc.) OE would procure those items. NUC PR would procure any remaining stock type items (steel, pipe, conduit, etc.). A part of the modification package consisted of a Bill of Materials which listed all the materials needed for the modification and identified by procurement contract number those purchased by OE. A problem expressed by NUC PR, but not pursued as a part of this review, was that the Bill of Materials did not necessarily arrive onsite in time for NUC PR to know what materials to buy. As a result, modifications were sometimes started not knowing if all the required materials were available.

Inconsistencies between OE and NUC PR terminologies and procedures were identified which could present problems. One such inconsistency involved the QA level assigned procured material. Within OE material was either QA material or not and if it were QA material 10CFR Part 21 was applicable to the vendor. NUC PR, on the other hand, had four different levels of quality within the QA materials it purchased and non-QA material. Within the four QA levels two had optional 10CFR Part 21 applicability. Consequently what was designed and constructed as either a QA system or a non-QA system was being maintained and modified using six different QA classifications and no QA. This is not to imply either is more or less correct, but to point out an inconsistency within TVA of doing work that really should not be there.

Another problem was identified in that the nomenclature used to define the various design classifications for piping systems were different for each plant and no official definition for the classifications could be found. An engineer within OE provided a list he developed for his own use. Engineers at the plants have

a problem knowing what a piping classification means on a modification drawing because the plant engineers don't classify their systems the same way as OE. That problem results in the plant engineer having to communicate with OE for an interpretation before material is bought so the appropriate material specification can be placed upon the item procured.

Regarding OE procurements, one good practice was identified in that Requests for Delivery on an IQT contract could be issued directly from OE without going through the laborious review and approval process employed by NUC PR. Another practice, which will be discussed further in section V.G, of questionable validity was identified. Where a large component was assembled from commercial grade parts (parts not requiring an ANSI N45.2 QA program over manufacturing) and qualified to an 1E environment, OE continues to procure replacement parts as commercial grade and assumes the component maintained its 1E classification.

No areas for improvement specific to OE were identified in the limited areas reviewed.

F. Purchasing

The review time spent in PURCH involved gaining an understanding of the laws pertaining to Federal procurements, identifying the purchasing agents' (PA) responsibilities and their specific problems within the procurement cycle, and tracking specific requisitions through the bid process and award of contract. Specific internal PURCH procedures were not reviewed or evaluated due to time constraints. The PAs appeared conscientous and professional in the performance of their responsibilities and demonstrated a willingness to help in improving the procurement system. They consistently expressed concern over the excessive use of emergency purchases and unrealistic "want" dates and how these affect TVA credibility with vendors. The PAs also stressed the need to be technically accurate on all specifications found in requisitions. In the PA's opinion, too many specification problems were being identified by vendors and not within the TVA review cycle.

One review area involved obtaining a general understanding of the legal constraints placed on Federal procurements. Many were identified including low bid and EEO and small business requirements. Many of these requirements, including their impact upon the procurement process such as time delays, were unknown to site technical and NCO personnel. A relatively new constraint, Public Law 98-72 and the associated requirement that procurements of $10,000 and over be advertised in the Commerce Business Daily prior to the bid process were known by site personnel and presented more consternation than any of the others discussed during this review. That law allows all interested vendors the equal opportunity to bid on an item and delays bid opening up to 45 days. Unfortunately, due to the great number of items required to be advertised in the Commerce Business Daily by all Federal

agencies, a 3- to 30-day waiting period resulted at the Department of Commerce before the ad was placed. That presented an additional significant time delay in an already lengthy process. Although many people informed the review team of the 45-day advertising requirement, there was no awareness of the waiting period delay by the site originators. Had they been aware of the additional delay there was no reason to expect that that time delay would be factored into the ordering lead times because no other time delay had been factored in either by the originator or anyone else in the procurement process.

A consistent problem identified by the PAs was the amount of time taken in the resolution of problems identified on a requisition and exceptions taken by vendors when submitting bids. The PA did not communicate directly with the originator. In fact, the PA typically did not know who the originator was. The signature of the originator was not on the requisition. Therefore, the PA had to rely on someone else (possibly from Power Stores or the Materials Unit onsite or NCO) to coordinate resolution of problems identified with the requisition after the bids were received. The agents varied as to the method used in the coordination process although all were aware of the resultant time delay.

Another problem identified by the PAs was the time delay involved in getting bids approved by the NCO QA Staff. The "review" performed by NCO QA (when no exceptions to the contract are taken) consisted of stating which of the low bidders were on a list of vendors with a TVA-approved QA program. The process of PURCH sending the bids for review was extremely cumbersome and time consuming. PURCH sent the bid to Management Services, who sent it to Materials Management, who sent it to QA. After approval the process was reversed. If an exception was involved, QA would send the exception to the site and coordinate approval between FQE, Materials Unit, and originator as necessary.

The time delay resulting from the memorandums and paperwork generated in stating which of the lowest bidders had a TVA-approved program was considered excessive and unnecessary by NSRS. In many cases the PA had worked consistently with a particular commodity and was knowledgeable of the approved vendors. To eliminate time delays and excess written communication, it would appear prudent to establish guidelines to allow PAs the task of selecting the lowest bidder from the approved vendors list. That responsibility would be applicable for cases only in which no exception was taken by the vendor.

The PURCH portion of the review occurred a few days prior to the October 1, 1984 transfer of NCO procurement responsibility to the sites. The PAs had limited or no information concerning the changes which would affect the procurement cycle. Although some time is required in a transition stage to incorporate changes, the review team considered this symptomatic of what appeared to be limited communication occurring between NUC PR and PURCH.

G. Quality Assurance

The NRC regulations and TVA procedure recognize that basic components can have varying degrees of quality placed upon them depending upon their importance to safety. The OQAM establishes four QA levels (level I, II, III, and IV) to which items or services for CSSC may be assigned. Guidelines in assigning levels are listed in paragraph 3.2.5.2 and are identical to those listed in ANSI N45.2-1971. These are interpreted by NSRS to range from items requiring considerable QA activities to those requiring little or no QA, i.e., commercial grade items of standard design which have proven successful for many years. In reviewing the QA levels in OQAM, Part III, Section 2.1, paragraph 3.2.5.2, it is found that each apply to CSSC with QA level I basically applying to, among other things, ASME Code material and items procured to a standard unique to the nuclear industry and decreasing in safety importance to QA level IV with no safety-related function.

Reviewing the definitions contained within 10CFR21, 10CFR50, and associated appendices, regulatory guides, and the OQAM, it was clear that TVA has equated the following terms:

1. Basic component.
2. Critical systems, structures and components (CSSC).
3. Structures, systems and components important to safety.
4. Safety-related structures, systems and components.

Those definitions being equivalent are used throughout the OQAM in a variety of contexts and introduce conflict and confusion.

A contradiction is introduced in the description of QA levels III and IV. In the OQAM both levels III and IV are described as being for CSSC items, but elsewhere the OQAM specifies that levels III and IV are not for basic components.

The use of commercial grade items in association with QA levels presented confusion and contradiction. Commercial grade items were allowed to be purchased by the OQAM, Part III, Section 2.1, paragraph 3.2.5.2 with QA levels I through IV. However, OQAM, Part III, Section 2.1, paragraphs 4.3.1.7 and 4.5, excluded level II as an option for purchasing commercial grade items. In addition the OQAM, Part III, Section 2.1, Appendix F, paragraph 2.2, stated commercial grade items were not basic components.

Items procured with quality level I and II designations require considerable documented quality control unless procured as commercial grade. A commercial grade quality level I or II procurement could be from vendors with an unapproved QA program, require no documented quality assurance, and receipt inspected by Power Stores personnel. Although allowed by the OQAM most but not all procurements of QA level I and II commercial grade items seen by NSRS were required to be from vendors with N45.2 approved programs.

As the QA requirements are all essentially the same for commercial grade items, NSRS believes there is a fallacy in trying to pigeonhole purchased commercial grade items into a variety of QA levels. The origin of this fallacy appeared to stem from the application of 10CFR21 to items procured to either QA level I or II. TVA, in the OQAM, stated that the determination of Part 21 applicability applied only to QA level I and II procurements. Determination of Part 21 applicability was contained within OQAM, Part III, Section 2.1, Appendix F. In order for Part 21 to be determined not applicable, the item being purchased must have been a commercial grade item, must not have been a complete basic component, or several other criteria. Appendix F, Attachment 1, was a form, "Determination of Part 21 Applicability," which when completed became a QA document. The first question asked was "is the item 'commercial grade' (yes or no). . ." If the answer was yes, Part 21 was not applicable and any remaining questions remained unanswered, such as, could its failure cause a basic component not to perform its required safety function. NUC PR QA personnel agreed this was a problem.

Considering whether or not a commercial grade item could affect the ability of a basic component to perform its safety function was addressed by the NRC when Part 21 was developed. In its first publication of Part 21 as a proposed rule on March 3, 1975, the wording was such that Part 21 could be considered applicable to off-the-shelf or catalog items. In response to inquiries and public meetings, NRC amended Part 21 on October 19, 1978, and recognized that commercial grade items could be purchased without the Part 21 requirement to report defects and the associated liabilities for not reporting them. This recognized that commercial grade items could be purchased for use as a basic component and Part 21 would become applicable after "dedication" of the part as a basic component. Based on discussions with TVA Office of the General Counsel (OGC), this dedication means to put into use and at that time Part 21 reporting requirements becomes the responsibility of TVA. Consequently, the NRC has allowed the use of items with a variety of QA levels including commercial grade as basic components. However, the use of commercial grade items with Part 21 not applicable does not eliminate the need for some level of quality, rather it shifts the burden of assuring quality and the continued ability of that item to perform its safety function from a joint manufacturer/TVA responsibility to TVA's sole responsibility. That is, if TVA procures a commercial grade item for use as a basic component, it must either assure quality during the manufacturing or through receipt inspection, testing, or other means. For a true commercial grade item that is purchased off the shelf by part number with no documented quality, the only avenue available to TVA to assure quality is through receipt inspection and testing.

In OQAM, Part III, Section 2.1, Appendix F (2.3.1) the statement is made, "Specific components, systems, and structures listed on the CSSC list are basic components by definition unless procured as commercial grade." Therein lies the fallacy. A basic com-

ponent remains a basic component whether or not it is replaced with a pedigreed item or commercial grade item.

Part 21 specifies [21.3(a)(4)] that "a commercial grade item is not a part of a basic component until after dedication." It does not state that a basic component ceases to be a basic component if supplied as a commercial grade item. A commercial grade item can be used as a basic component once dedicated, and it can be used where its failure could cause a basic component not to perform its required safety function. All the Part 21 applicability means for a commercial grade item is if TVA finds it defective at some point in time, TVA must report the defect to NRC just as the vendor or TVA would have to do if a defect were found on an item where TVA imposed Part 21 upon a manufacturer.

In determining Part 21 applicability one criterion for judging Part 21 not applicable is by identifying the item as a commercial grade item. Most Part 21 not applicable determinations seen during this review were because the item was identified as being commercial grade. That determination has resulted in what NSRS concludes as a misapplication of the definition of commercial grade. One example is offered in support of that conclusion:

- Requisition number 951134 from SQN was written to procure sheetmetal for ECN 2768. The metal was to be manufactured to ASTM specifications and required the manufacturer, through Appendix E Attachment 8, to have a quality assurance program that met the requirements of ANSI N45.2-1971. The items being procured were classified as commercial grade and assigned a QA level I Part 21 not applicable.

Purchasing that material to an ASTM Standard and requiring an N45.2 QA program is certainly more restrictive and prescriptive than purchasing an item to a catalog number. It therefore should not be classified as commercial grade. Part 21 may still not be applicable, but for different reasons such as it would not adversely affect the performance of a safety function.

It appears that a situation occurred where material was being procured not for a basic component but for an application that still required QA level I attention. As the Appendix F, Part 21, applicability form first questioned whether or not it is commercial grade, it appeared that personnel completing the form were taking the easy way of determining Part 21 not applicable by calling it commercial grade thereby avoiding the evaluation of other significant qualifying factors.

It could be argued that it makes no difference if Part 21 is declared not applicable by either calling the item commercial grade or by deciding it is not a basic component. The argument breaks down, however, when, as stated previously, it is recognized that the manufacturing of commercial grade items requires no approved QA program or FQE receipt inspection while other

non-CSSC items may require considerable QA with approved programs. An example of a determined nonbasic component purchase with conflicting Part 21 determinations is as follows:

- Requisition number 343910 from SQN was written to procure reactor coolant pump seal parts. Specific information was provided on the Westinghouse pump and the parts were required to be manufactured in accordance with the original requirements of the Westinghouse E-Specification 677355. The plant FQE staff originally classified the procurement as QA level II, Part 21 applicable, but NCO revised the procurement to be QA level II, Part 21 not applicable because they were not on the CSSC list and would not affect the safety function of a basic component. Considering that the specification used for the seals was developed by the manufacturer (Westinghouse) of the main coolant pump, it could well be a specification unique to the nuclear industry. As such, the procurement should probably have been identified as a basic component and designated Part 21 applicable. When Westinghouse supplied the parts they included the Part 21 applicability.

In other cases the use of a QA level I or II for a commercial grade item may be inappropriate and imply a level of quality that is just not present. For example:

- Requisition number 932925 from BFN was written to procure a selector switch for ECN L2115. It was ordered by the manufacturer part number, it was assigned a QA level II, and it required a packing slip for documentation.

As it was purchased from a supply house with no unique QA requirement or nuclear standards, it is considered by NSRS to truly be a commercial grade item. As such the QA level II is considered artificially high implying quality that may not be factual. That switch was to be used in a panel that was qualified, along with that model switch, by TVA to 1E requirements. It therefore clearly falls into the category of commercial grade items dedicated as a basic component.

OE has made a decision regarding commercial grade 1E equipment. Basically stated, if a component is assembled with commercial grade items (such as a motor control center) and that component is physically tested to 1E requirements and is qualified, then the qualification of that component will remain if replacement parts are the same (same stock number) or equal.

Even though commercial grade items are allowed to be used as basic components, in doing so TVA assumes added responsibilities which it is not adequately fulfilling. Recognizing, for example, that TVA had qualified, at some point in time, a commercial grade item for use as a basic component or a part of a basic component, TVA currently has no program to assure that future replacements

will in fact be exactly the same as the one qualified, and therefore maintain that qualification. If a true commercial grade item is purchased by part number from a manufacturer or supplier, the manufacturer or supplier is not required to have an approved QA program, and TVA only receipt inspects the item to assure that the part number is correct. There was no testing or inspection by TVA identified that would assure that the item would perform as required or that detrimental changes to the item occurred or did not occur. NUC PR does perform a functional test of newly installed equipment which should provide some assurance that it will perform during routine operations. That test, however, will not provide any assurance that the item will perform as required under accident conditions. The manufacturers of commercial grade items are under no obligation or authority to identify changes. NSRS was informed that changes generally are accompanied by a part number change by the manufacturer. That, however, is by convention rather than by requirement. Additionally, what would constitute a change would probably differ from manufacturer to manufacturer, and a change as subtle as using a different lubricant (which could have a very detrimental effect under accident conditions) would probably not be considered a change by any manufacturer.

OE personnel interviewed stated that some manufacturers will not sell commercial grade items to a nuclear plant. OE personnel stated that if certain manufacturers received an order for a commercial grade part and knew it was to go, e.g., to SQN, they would automatically provide the QA documentation on the item, delay shipment about six months while assembling the documentation, and would charge ten times the amount they would charge for the same item if it were commercial grade. OE stated no value was added to the part, it was not manufactured any differently than the commercial grade item, and TVA already had the item so if a defect were found TVA would receive its 10CFR Part 21 notification on the previous or original orders. To avoid what OE considered exorbitant pricing, an ordering procedure was devised when ordering parts from certain manufacturers where the Power Stores Distribution Center was the recipient of the commercial grade item. Specific instructions were provided to PURCH on the Purchase Requisition not to mention 10CFR Part 21, IE qualification, or nuclear plant. At the time the Purchase Requisition was prepared, a Transfer Requisition was prepared for the use of the Power Stores Distribution Center when the item was received. That Transfer Requisition changed the classification of the commercial grade item to a QA item and directed shipment to the appropriate nuclear plant.

That procedure had been reviewed and approved by both OGC and OQA. Discussions with Division of Quality Assurance, Procurement Evaluation Branch, personnel revealed that the manufacturer in question did, according to OQA audits, have different production runs and QA requirements for items going to nuclear plants; therefore, it appears that some value was added to the commercial grade item for the increased fee.

This entire question was not pursued any further as a part of this review. NSRS has serious reservations regarding this practice and reserves final judgement until it can be evaluated further. Until that time it would be considered prudent on the parts of OE and NUC PR PEB to evaluate this practice on their own.

With the conflicts, confusion, and fallacy described above, the situation has developed where the QA level system is being further divided within the levels I and II, through the use of Part 21 applicability, to accommodate commercial grade items. In doing so an artificial QA level is implied for a commercial grade item (i.e., commercial grade item purchased with no QA and assigned a QA level of I or II), or items appropriately purchased with a QA level and requirements are called commercial grade. It is considered more appropriate and less subject to errors if the commercial grade items are recognized for what they are, either QA level IV or non-QA, and procurement of QA level I and II commercial grade items should be prohibited. In addition, all QA level I and II items regardless of the Part 21 applicability should be receipt inspected by FQE. Further, the quality requirements associated with an item adequately performing or affecting a safety function need to be separated from the Part 21 commercial grade determination which has nothing to do with quality. Whether the quality assuring activities for an item's ability to perform a function is jointly shared by the manufacturer and TVA or solely by TVA, is irrelevant to the required quality activities.

There is a basic philosophical problem with the QA program for items purchased as basic components versus items purchased as commercial grade but dedicated as a basic component. TVA's procurement QA program for basic components is based upon adding additional TVA quality assurance activities where there is quality assurance to begin with in the manufacturing process and have no quality where there is no verifiable quality in the manufacturing process.

To fulfill its responsibility when using commercial grade items as basic components, TVA will have to develop some mechanism to qualify replacement commercial grade items such as a receipt inspection and testing program that is more stringent than what is currently in place for QA items requiring FQE receipt inspection. (For additional suggestions and information on receipt inspection see section V.B.5.)

With regard to the QA program associated with procurement, the OQAM was found cumbersome and sometimes contradictory, 10CFR21 applicability was being used incorrectly as a determinant in establishing quality levels, and the Appendix F, Attachment 1 form, for 10CFR21 applicability was inappropriate and being misused. In addition, commercial grade items were being given implied quality by assigning a quality level to them, and TVA had no mechanism to assure a commercial grade item used as a basic component would function when needed during accident condition.

H. NUC PR Procurement Problem Task Force

The review team interviewed two of the three-member NUC PR Procurement Problem Task Force to gain an understanding of the perceived problems within the procurement system. The task force report and recommendations were issued subsequently on August 10, 1984 in a report from Eric Kvaven to Jim Darling (L00 840810 294). That report was reviewed by NSRS considering all material assimilated during the procurement review. The major recommendations identified in the Management Summary of the task force report were to:

(1) Establish an adequate planning group at the plant.

(2) Implement status tracking systems.

(3) Set goals for turnaround time for each review/approval cycle step.

(4) Improve and add adequate resources for expediting efforts.

(5) Improve communication between PURCH and the site.

(6) Eliminate all unnecessary steps in the procurement cycle with the goal of placing very few, if any, steps between the requisitioner and the purchasing agent.

(7) Improve the inventory stock out problem.

(8) Better utilize the automated systems.

(9) Develop improved QA procedures and training.

(10) Redefine QA responsibilities for procurement.

The following observations were made concerning the proposed recommendations:

(a) Items 1 through 5 above have a basic emphasis of incorporating more people into the procurement cycle by adding various expeditors, trackers, and designated contacts for PURCH and OE interface. The basic premise is to eliminate the delays. It should be emphasized though that time delays at the site could not be substantiated by the review team. The only consistent time delays involved procurements which traveled through the Central Office. Those time delays stemmed from the amount of handling a requisition received traveling between the Materials Management Unit, NCO QA, Power Stores, MMSS, and PURCH. The Central Office QA review was eliminated in October with all reviews now performed at each site. Adding more resources to the cycle to perform the recommended functions will not eliminate a basic inherent problem of too many people already in the procurement cycle.

(b) Although item 6 recommends the elimination of all unnecessary steps in the procurement cycle, the steps are not readily identified in the report. It appears that the extensive tracking proposed would be established to follow a cycle similar to what presently exists. The tracking would apparently start with the procurement request and be maintained until the item is received, set aside, and finally used. Some tracking may be appropriate and effective, but the emphasis appears to be to find the people who are not performing their job properly. Instead of developing a method to track all the reviews, more emphasis is needed in simplifying the present system, i.e., identifying the reviews not needed and better utilization or elimination of resource people presently available within the system.

(c) Necessary action on item 7 was observed during the review. To alleviate the stock out problem, more emphasis was being placed on having accurate usage history available. MMSS and Power Stores were coordinating that effort. An additional Task Force report recommendation to alleviate the stock out problem was the increased usage of IQT contracts. NSRS observed, under the current NUC PR system, no benefit in using IQTs to reduce time delays due to the RD on an IQT being treated as a new contract, i.e., going through the same review cycle every time an RD is to be used. A definite benefit can be realized if NUC PR uses the IQTs as intended and prescribed in the Procurement Manual. Another improvement can be made if site Power Stores order parts as inventories become low and not save them up for a big order.

(d) Item 8 is highly supported, and establishing a uniform database with QA control could enable the use of the automated system for reordering of all inventory items both QA and non-QA. This would be an effective method to eliminate the unnecessary site review performed on an item each time it is reordered. Current emphasis by MMS, Power Stores, and NUC PR should remain on QA program development.

(e) Although items 9 and 10 appear to be directed toward QA, the report substantiates the need to train all personnel in the procurement chain and to revise and standardize all procedures. NSRS fully agrees with this recommendation.

The Task Force report identifies some real problem areas in the procurement cycle and makes many valid recommendations. Immediate emphasis should be placed on the more simplified solutions like eliminating unnecessary steps that could provide significant improvements in the present system An NSRS suggested solution to the problems with procurement s presented in Attachment 1.

VI. PERSONNEL CONTACTED

A. Browns Ferry Nuclear Plant

R. E. Burns	Group Head, Instrument Maintenance
J. A. Coffey	Site Director
R. Cole	OQAB
T. D. Cosby	Head, Electrical Maintenance Group
J. A. Dement	Supervisor, Materials Unit
H. L. Johnson	Quality Assurance Engineer
G. T. Jones	Plant Manager
H. C. Le	Chemical Engineer
R. E. Mabry	Materials Officer, Power Stores
D. C. Mims	Head, Engineering Group
J. R. Nebrig	Supervisor, Modifications Section
J. C. Owen	Materials Officer, Power Stores
W. J. Percle	Supervisor, Electrical Section
J. R. Pittman	Assistant Plant Manager, Maintenance
R. D. Pittman	Assistant Supervisor, Power Stores
S. W. Solley	Electrical Engineer, Electrical Maintenance
W. C. Thomison	Supervisor, Engineering
V. M. Vargas	Specifications Engineer, Support Services
C. G. Wages	Head, Mechanical Maintenance
B. H. Weeks	Supervisor, Power Stores
T. F. Ziegler	Branch Chief, Site Services
W. P. Zimmerman	Materials Officer, Power Stores

B. Sequoyah Nuclear Plant

L. D. Alexander	Supervisor, Mechanical Modification
R. E. Alsup	Section Supervisor, Compliance Staff
C. E. Brannon	Supervisor, Power Stores
T. L. Burke	Mechanical Engineer
S. Butler	Acting Supervisor, Field Quality Engineering
M. L. Campbell	Mechanical Engineer, Mechanical Engineering Unit
S. B. Campbell	Materials Officer, Power Stores
M. L. Crane	Unit Supervisor, Materials Unit, Field Services
N. D. Ebel	Materials Clerk, Power Stores
R. W. Fortenberry	Supervisor, Engineering
L. J. Freeland	Material Officer, Power Stores
J. R. Hamilton	Supervisor, Field Quality Engineering
P. H. Hitchcock	Mechanical Engineer
B. A. Kimsey	Electrical Engineer
D. L. Love	Supervisor, Mechanical Maintenance Engineering Section
G. W. Petty	Material Officer
J. Robinson	Group Head, Modifications
J. R. Staley	Supervisor, Power Stores
C. R. Stutz	Quality Assurance Engineer, FQE
S. W. Vickory	Material Officer, Materials Unit
P. R. Wallace	Plant Manager

C. Central Office

D. A. Carter	Material Officer, Materials Management
C. R. Favreau	Mechanical Engineer
R. D. Hicks	Materials Officer, Materials Management
J. Hood	Supervisor, Nuclear Power Stores
E. A. Jewell	Assistant Supervisor, Power Stores
M. D. Kelley	Chemical Engineer
J. E. Law	Branch Chief, Quality Systems
F. H. Lewis	Supervisor, Quality Assurance
J. W. Mabee	Head, External Supplier Evaluation Group
E. W. Mansfield	Supervisor, Power Stores
L. Moerland	Supervisor, Materials Management
R. J. Mullin	Director, Division of Quality Assurance
D. C. Nowading	Branch Chief, Materials Management Services Staff
G. Odell	Supervisor, Management Services Staff
R. C. Parker	Assistant Director, Division of Quality Assurance
D. R. Parks	Supervisor, Materials Analysis
J. H. Pratt	Material Officer, Materials Management
K. R. Ramsey	Quality Assurance Engineer
E. K. Sliger	Supervisor, Radiological Emergency Preparedness Group
J. R. Watson	Quality Assurance Engineer
G. B. Workman	Materials Officer, Materials Management

D. Purchasing

L. N. Arms	Purchasing Agent, Nuclear Fuels Section
D. A. Blackwell	Supervisor, Mechanical Plant Equipment and Special Projects Sections
C. R. Dobson	Supervisor, Nuclear Fuels Section
F. W. Hannah	Purchasing Agent, Open Market Electrical Section
J. E. Henegar	Purchasing Agent, Components Unit
D. E. Henry	Quality Assurance Engineer
E. C. Kidder	Supervisor, Quality Assurance
F. A. Lowe	Assistant Director, PURCH
W. H. Mary	Purchasing Agent, Open Market Electrical Section
D. Owen	Administrative Assistant, Equipment Procurement Branch
G. S. Owensby	Administrative Assistant, Materials Procurement Branch
S. W. Palmer	Purchasing Agent, Open Market Construction Section
G. D. Settles	Supervisor, Open Market Construction Section
D. A. Smith	Purchasing Agent, Nuclear Fuels Section
J. M. Smith	Purchasing Agent, Open Market Electrical Section
R. H. Sunderland	Chief, Procurement Support Staff

E. Office of Engineering

G. F. Grant	Electrical Engineer
L. A. Johnson	Mechanical Engineer, Facility Support and Services
A. W. Lewis	Nuclear Engineer, NEB, Design Support
T. S. Orr	Nuclear Engineer, NEB, Design Support
J. L. Purkey	Supervisor, Facility Support and Services
A. C. Robertson	Supervisor, Mechanical Section 2
F. B. Rosenzweig	Supervisor, Cable and Miscellaneous Equipment
E. R. Taylor	Mechanical Engineer, NEB, Special Support
W. C. Wylie	Electrical Engineer

F. Office of the General Counsel

W. W. LaRoche	Attorney

VII. DOCUMENTS REVIEWED

A. Requisitions (Each requisition listed includes all associated documentation.)

171173	835401	938284	943842	951129
290205	925526-2	938298	945707	951133
322026	930478	940060	946728	951134
334177	931892	940163	947434	951148
341210	931941	940185	947454	951819
343910	932925	940222	950455	955163
355456	932973	940276	950509	956101
832041	932992	942925	951019	959025
833678	935570	942962	951023	959104
834705	935718	942988	951028	959122
834706				

B. Regulations Standards and Upper-Tier Documents

10CFR21, "Reporting of Defects and Noncompliance"

10CFR50, "Domestic Licensing of Production and Utilization Facilities"

Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)"

Regulatory Guide 1.29, "Seismic Design Classification"

Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operational)"

Regulatory Guide 1.38, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants"

Regulatory Guide 1.58, "Qualification of Nuclear Plant Inspection, Examination, and Testing Personnel"

Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants"

Regulatory Guide 1.123, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants"

NUREG-0302, "Remarks Presented (Questions/Answers Discussed) at Public Regional Meetings to Discuss Regulations (10CFR Part 21) for Reporting of Defects and Noncompliance"

NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment"

IE Bulletin No. 79-01B, "Environmental Qualification of Class 1E Equipment"

IE Information Notice No. 83-79, "Apparently Improper Use of Commercial Grade Components in Safety-Related Systems"

IEEE Standard 323A-1975, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations"

IEEE Standard 308-1980, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations"

IEEE Standard 336-1980, "IEEE Standard Installation, Inspection, and Testing Requirements for Class 1E Instrumentation and Electric Equipment at Nuclear Power Generating Stations"

IEEE Standard 467-1980, "IEEE Standard Quality Assurance Program Requirements for the Design and Manufacture of Class 1E Instrumentation and Electric Equipment for Nuclear Power Generating Stations"

IEEE Standard 494-1975, "IEEE Standard Method for Identification of Documents Related to Class 1E Equipment and Systems for Nuclear Power Generating Stations"

IEEE Standard 627-1980, "IEEE Standard for Design Qualification of Safety Systems Equipment Used in Nuclear Power Generating Stations"

IEEE Standard 344-1975, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations"

ANSI N45.2-1971, "Quality Assurance Program Requirements for Nuclear Power Plants"

ANSI/ASME N45.2.2-1978, "Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants"

ANSI/ASME N45.2.6-1978, "Qualifications of Inspection, Examination, and Testing Personnel for Nuclear Power Plants"

ANSI/ASME N45.2.9-1979, "Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants"

ANSI N45.2.13-1976, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants"

ANSI 18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants"

ANSI/ASME NQA-1-1979, "Quality Assurance Program Requirements for Nuclear Power Plants"

Topical Report TVA-TR75-1, Revision C, "Quality Assurance Program Description for Design Construction and Operation"

Operation Quality Assurance Manual, December 31, 1983

Procurement Manual, June 11, 1984

ID-QAP 4.3, "Transfer of Items," September 21, 1983

B. Division of Nuclear Power Procedure Manual

N72A14, "Procurement and Materials Support Program," March 30, 1983

N77A2, "Storage and Shelf Life Consideration for Materials with Natural Aging Life," March 21, 1983

1200R05 (N77A14), "10CFR21 Evaluation and Reporting Requirements," October 26, 1983

1601.02 (N73015B), "Nuclear Plant Reliability Data System (NPRDS), Maintenance History and Class 1E Tracking and Trending - Reporting Failures of Components and Systems," December 9, 1983

TS 01.00.08.14.03 (N83M2), "Environmental Qualification of All Safety-Related Electrical Equipment (IE Bulletin 79-01B and NUREG-0588)," July 25, 1984

N76A4, "Standards and Substitutions Guides for Quality Assurance Level III Items," June 9, 1982

N76A10, "Purchase Specifications for CSSC Metallic Materials, Wire, and Cable Used Inside Primary Containment, Welding, and Brazing Materials, Valve Parts, and Pump Parts," January 20, 1984

N82A17, "Equipment and Material Storage Requriements for Nuclear Power Stores," July 29, 1983, and September 7, 1984

- Division of Nuclear Power Quality Control Inspector Training Manual, September 30, 1982

C. Browns Ferry Nuclear Plant Procedures

BF 16.2, "Procurement," June 5, 1984

BF 16.3, "Quality Control of Material Components, Spare Parts, and Services," July 20, 1982

BF 16.4, "Material Components and Spare Parts Receipt, Handling, Storage, Issuing, Return to Storeroom, and Transfer," June 5, 1984

BF 16.9, "Procurement, Shipment and Receipt of Services and Material Involving Power Service Shop," July 5, 1983

BF 6.10, "TVA Fabricated Parts Used in CSSC," June 29, 1982

BF EMI 41, "Electrical Maintenance Instruction 41, Refueling Platform and Jib Crane Checkout," Revision in effect on June 12, 1979

D. Sequoyah Nuclear Plant Procedures

SQA45, "Quality Control of Material and Parts and Services," May 30, 1984

AI-11, "Receipt Inspection, Nonconforming Items, QA Level/Description Changes and Substitutions," May 21, 1984

E. Office of Engineering Procedures

G-28, "Construction of Piping Systems for Boiling Water Reactor Nuclear Power Plants," December 30, 1982

SS-E18.11.04, "Quality Assurance Requirement for Electrical or Mechanical Equipment Requiring Seismic Category 1 (L) Qualification," June 19, 1984

SS-E18.10.01, "Environmental Qualification Requirements for Safety-Related Electrical Equipment," August 29, 1984

SS-E18.11.02, "Quality Assurance Requirements for Safety-Related Electrical and Mechanical (Non-ASME Section III) Equipment," June 19, 1984

SS-E18.11.01, "Quality Assurance Requirements for ASME Code, Section III Control Valves," May 26, 1983

SS-E18.11.03, "Quality Assurance Requirements (Certificate of Conformance)," June 19, 1984

IDP-N 51.05, "Preparation and Processing of Preliminary Design Change Requests (P-DCR)," May 9, 1983

IDP-N 51.03, "Modification Implementation and Control Modification Tracking Program," January 11, 1984

IDP-N 54.01, "NUC PR Procurement Request to EN DES," May 9, 1983

IDP-N 50.02, "Division of Nuclear Power and Engineering Design - Interfaces," March 1, 1984

ID-QAP 2.4, "Control of Modifications," March 30, 1984

MEB-EP 23.5.5, "Specification of Quality Assurance Program Requirements for MEB Requisitions," May 27, 1983

F. Correspondence and Reports

Memorandum from F. W. Chandler to J. A. Raulston dated August 13, 1984, "Browns Ferry Nuclear Plants Units 1, 2, and 3 - Report to NRC Resulting From NRC Meeting of May 24, 1984, on TVA's Environmental Qualification Program," including attachments (EEB 840813 938)

Test Report 17503-1 from Wyle Laboratories dated January 6, 1984, "Control Equipment Consisting of Therminal Blocks, Pushbutton Operators, Selector Switches and Contact Blocks and Various Cables"

Letter from Richard T. Trosclair to N. M. Sprouse dated April 10, 1972,"Contract 71C-3-54360, 70C-61-54041 and 69C5-64545"

Test Report 3359B dated November 19, 1971, Inland Testing Laboratories, Inc., "Report of Test on (20) Switching Components for Cutler Hammer"

Memorandum from Eric Kvaven to James P. Darling dated August 10, 1984, "Report of the Task Force for Studying and Solving Procurement Problems in the Office of Nuclear Power (NUC PR)" (L00 840810 294)

Letter from R. H. Ector to George J. O'Dell dated April 6, 1984, "Brief Procurement Process Analysis of Field Services Branch Purchase Requisitions Originated June 1 to December 31, 1982" (L12 840405 813)

Report CH-8200-14 dated February 25, 1983, "Environmental Qualification Program," by Office of Power Quality Assurance and Audit Staff (OQA 830225 703)

Memorandum from H. J. Green to R. L. Lumpkin, Jr., dated May 10, 1983, "Quality Program Audit Report No. CH-8200-14" (OQA 830511 700)

Report BF-8400-03, dated March 1, 1984, "Browns Ferry Nuclear Plant Procurement/Storage Activities" (OQA 840220 701)

Memorandum from M. N. Sprouse to H. J. Green dated November 6, 1981, "Procurement of Materials, Components, Spare Parts, and Services, N-OQAM Part III, Section 2.1, dated April 17, 1981" (EEB 811110 912)

Memorandum from G. A. Keller to Electrical Engineering Files dated August 23, 1983, "All Nuclear Plants - Development of a Comprehensive Environmental Qualification Program (CEQP)"

Memorandum from T. G. Campbell to Those listed dated September 14, 1983, "Procurement of Class 1E Euipment - All Nuclear Plants - Field Services" (L23 830815 884)

Memorandum from T. E. Spink to OQAB Files dated October 19, 1983, "Meeting of Special Evaluation Team - Environmental Qualification of Safety-Related Equipment with Duke Power"

Memorandum from T. E. Spink to OQAB Files dated October 19, 1983, "Meeting of Special Evaluation Team - Environmental Qualification of Safety-Related Equipment with Commonwealth Edison Company (CECO)"

Memorandum from H. J. Green to James P. Darling, et al, dated July 25, 1983, "Browns Ferry and Sequoyah Nuclear Safety Review Boards' Annual Report for 1982"

Memorandum from R. L. Lumpkin, Jr., to F. A. Szczespanski dated August 16, 1983, "Browns Ferry and Sequoyah Nuclear Safety Review Boards' Annual Report for 1982" (OQA 830816 700)

Memorandum from H. J. Green to James P. Darling, et al, September 22, 1983, "Browns Ferry and Sequoyah Nuclear Safety Review Boards' (NSRB) Annual Report for 1982" (L69 830916 800)

Memorandum from P. R. Wallace to Those listed dated February 15, 1984, "Browns Ferry, Sequoyah, and Watts Bar Nuclear Plants - Maintenance Activity of Class 1E Equipment" (DES 840209 002)

Memorandum from Harold R. Denton, NRC, to William J. Dircks, NRC, dated January 21, 1983, "Equipment Qualification Program Plan"

Form TVA 45 from George J. Odell to Dick Smith dated December 10, 1984, "Materials Management Section Training Sessions"

Memorandum from James L. Williams, Jr., to Those listed dated June 4, 1984, "Promoting a Better Understanding of TVA's Purchasing Function"

Memorandum from Herbert S. Sanger, Jr., to W. F. Willis dated October 20, 1983, "Public Law No. 98-72 - Mandatory 45-Day Waiting Period - Procurement and Personal Services Contracts"

Memorandum from W. F. Willis to Those listed dated October 31, 1983, "Public Law No. 98-72 - Mandatory 45-Day Waiting Period ACT - Application to Procurement and Personal Services Contracts"

Memorandum from James L. Williams, Jr., to Heads of Offices and Divisions dated November 3, 1983, "Public Law No. 98-72 - Mandatory 45-Day Waiting Period - Procurement Contracts"

Memorandum from Herbert S. Sanger, Jr., to Those listed dated November 8, 1983, "Public Law No. 98-72 - Mandatory 45-Day Waiting Period Act - Extent of TVA's Voluntary Compliance for Power Program Contracts"

Memorandum from H. M. Crine, Jr., to Those listed dated December 12, 1983, "Public Law No. 98-72 - Mandatory 45-Day Waiting Period Act - Extent of TVA's Voluntary Compliance for Power Program Contracts"

Memorandum from W. F. Willis to Those listed dated February 3, 1984, "Public Law 98-72 - Mandatory 45-Day Waiting Period - Application to Procurement and Personal Services Contracts"

Memorandum from J. A. Coffey to Those listed dated February 29, 1984, "Public Law 98-72 - Mandatory 45-Day Waiting Period - Application to Division of Nuclear Power Procurements" (L25 840301 460)

Memorandum from J. G. Holmes, Jr., to H. S. Sanger, Jr., dated March 15, 1984, "Public Law No. 98-72 - Mandatory 45-Day Waiting Period Act - Extent of TVA's Voluntary Compliance for Power Program Contracts" (L00 840319 474)

ATTACHMENT 1

SUGGESTED SOLUTION TO PROCUREMENT PROBLEM

NSRS offers an approach to solving NUC PR procurement problems starting with the basic procurement function. An attitude change should occur whereby the procurement of items is considered for what it is--a very important function. Procurement within TVA is not simple and requires a level of expertise and knowledge not inherent in any position currently at the plants. The knowledge and experience must be taught and learned. Presently, the time delays and inadequacies are associated to a large extent with individuals learning on their own how to procure things. NSRS contends that procurement of items should be elevated in stature and importance to a professional level.

To make the concept work, NUC PR should change its practice that everyone can and should be able to procure materials to one where a dedicated and trained staff provides all procurement services. People need to know how to procure things before they are faced with the task. With proper training, a significant number of learning errors could be eliminated and the quality of the procurement process, both from a materials standpoint as well as a time delay standpoint, could be improved. A training program on the entire procurement process to include TVA's procedures, quality requirements, purchasing requirements, and Federal procurement requirements should be developed and provided to personnel performing a procurement function. Satisfactory completion of that training should be a requirement before an individual is allowed to procure anything.

An extension to the training requirement could be the establishment of a group whose responsibility is the procurement of materials. In that concept, engineers requiring items or services would go to the procurement group and specify what was needed. That group staffed with the necessary expertise would, in turn, prepare the necessary procurement documents, define the material specifications, quality requirements, and provide a completed procurement package ready for the approving official, be it the Plant Manager or the Board of Directors. That staff would be responsible for assuring that the procurements were correct and require no further review or approval with the exception of the authorizing official(s) and interface directly with PURCH. Power Stores personnel and their ordering of stock items would not be included in this staff but would work closely with them on procurements of stocked quality level materials.

GNS '840705 054

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS REPORT NO. R-84-19-WBN

SUBJECT: NSRS ASSESSMENT OF THE RESULTS OF THE BLACK AND VEATCH INDEPENDENT DESIGN REVIEW OF THE WATTS BAR NUCLEAR PLANT AUXILIARY FEEDWATER SYSTEM

DATES OF REVIEW: JANUARY 10 - JUNE 15, 1984

REVIEWERS: TECHNICAL ANALYSIS AND REQUIREMENTS GROUP

APPROVED BY: James F. Murdock
JAMES F. MURDOCK

DATE

TABLE OF CONTENTS

I. PURPOSE AND SCOPE

The NSRS performed an assessment of the results of the Black and Veatch (B&V) Independent Design Review of the Watts Bar Nuclear Plant Auxiliary Feedwater System to determine if we could support the policy committee report regarding the B&V review and to document areas where NSRS considered additional action was needed. The adequacy of the immediate corrective actions and the degree of "lessons learned" to other Watts Bar plant features and to the TVA nuclear program in general were assessed.

The Black and Veatch initial and supplemental reports, the activities of the TVA program team and cognizant line organizations, the activities of the TVA management policy committee and its report, the activities of the policy committee task force and its report, discussions with the TVA line engineers and managers, and onsite verification form the scope and basis of the NSRS assessment. Since the full applicability of the Black and Veatch findings to all the TVA nuclear facilities has not been determined by TVA, NSRS will perform followup assessments where appropriate.

II. EXECUTIVE SUMMARY

From January through June 1984, the NSRS performed a review and analysis of the results of the Black and Veatch (B&V) Independent Design Review of the Watts Bar Nuclear Plant (WBN) Unit 1 Auxiliary Feedwater (AFW) System and the TVA activities in response to the B&V findings. The NSRS review covered the B&V initial and supplemental reports, the activities of the TVA program team and the cognizant line organizations, the activities of the TVA management policy committee, and the activities of the task force of the policy committee.

The impetus of the NSRS effort was to assess:

1. the quality and appropriateness of the B&V review and the selection of the AFW system as the representative system;

2. the technical adequacy and consistency of the TVA responses to the individual B&V findings;

3. the determination of causes and generic applicability of findings and categories of findings to other WBN unit 1 systems and to other TVA plants; and

4. the analyses and evaluations performed to determine the safety implications of the findings or categories of findings had the B&V activity not taken place.

The following conclusions were reached from the NSRS assessments;

1. The AFW system was a good choice for a multidiscipline representative review. The B&V review was generally complete in depth and technically competent. A weakness in the B&V review was the lack of detailed examination of the consequences of findings by onsite verification. Since the B&V review was based upon the FSAR of record in 1982 and further design changes or FSAR amendments have been and will be implemented, the degree of conformance with current regulatory positions could not be determined. This could lead to continuing direction from NRC as the plant begins operation, particularly from the I&E Office.

2. For the greater part, NSRS agrees with the resolution of specific findings. The notable exceptions are in the electrical discipline (cable tray fill and treatment of protective devices) and in the structural discipline (embediments and attendent attachments) Followup discussion with EN DES, both in Knoxville and at the site, have led to resolution of the NSRS concerns in the structural area assuming the current (post-B&V study) EN DES practices are formalized in the proper procedures. In addition to the specific technical disagreements, NSRS found the definition of the safety impacts of the findings to be inconsistent with the basic engineering and safety reasons for having the features in place. A second weakness was that the record of resolution of the findings was not uniformly and completely documented.

3. NSRS substantially agrees with the task force grouping of findings into categories by way of determination of causes and the determination of generic applicability. In some cases more than one root cause could have been assigned a finding. Thus, the judgement of the reviewer as to the more important factors could be questioned; however, these differences are considered of low consequence.

4. The identification and correction of deviations or questionable conditions is a very important result of an effort such as the B&V review. Since a perfect plant or system is not likely to be found, the determination of the effect on the plant performance had the deficiencies not been discovered is deemed the most valuable aspect of the B&V review program implemented by TVA. The NSRS assessments support the conclusions of the policy committee that there is no direct indication that any affected structure, system, or component would not have performed its safety function. The safety evaluations performed by EN DES also support the conclusion that the safety functions would be performed. However, it is the assessment of NSRS that some of the identified deficiencies could lead to indeterminant conditions or conditions adverse to quality and to safety which reduce the margin of safety. Further actions were and are required to assure the margin of safety committed in the FSAR are met.

Additional observations drawn from the results of the B&W review are:

1. Considering the degree of completion of Watts Bar unit 1, the number of deviations found by B&V which would probably not have been discovered otherwise was fairly high. This underscores the value of independent design reviews by parties outside the TVA system. The findings would have been much easier to correct or avoid had the review been conducted earlier in the design/construction process. For lessons learned value, the Bellefonte project should consider this assurance tool.

2. The nature of the deviations indicates a need to substantially upgrade the configuration management processes and personnel training programs in TVA.

III. RECOMMENDATIONS

A. R-84-19-WBN-1 (Category 3)

All controlled documents should be clearly identified for all plants. The purposes and uses for each of the documents should be delineated. Information contained in documents designated to be controlled should be assessed for contribution to the intended purpose and use. Superfluous information should be deleted and discrepancies in documents with overlapping information should be corrected. Establishing a verified "as built," rigorously documented, should be assigned a very high priority.

B. R-84-19-WBN-2 (Category 9)

Procedure EN DES EP4.03 should be revised as has been verbally committed to reflect that visual examinations supported by field calculations are the basis for documenting acceptability of changes to or additional attachments to embedded plates in the field.

C. R-84-19-WBN-3 (Cagegory 9)

Consideration should be given to additional sampling for multiple attachments to imbedded plates made prior to February 1983 or an evaluation of the consequences of failure of 1 to 1½ percent of the supports in view of the overstressed anchor in one of the 69 plates already sampled.

D. R-84-19-WBN-4 (Category 20)

The methods and procedures for determining the proper values, physically setting, and verifying time delay relays settings should be reevaluated and indicated changes should be expeditiously made TVA-wide.

E. R-84-19-WBN-5 (Category 34)

See recommendation for Category 3.

F. R-84-19-WBN-6 (Category 35)

The instantaneous trip breakers should be verified to be set in agreement with the intent of the National Electric Code. The documentation for the design as well as for the testing and operations of the equipment should reflect the proper values. The program should be implemented TVA wide.

G. R-84-19-WBN-7 (Category 36)

Criteria should be developed for field use to control actual cable tray fill levels and to provide a basis for QC inspection. A feedback system should be included from the construction forces pulling cable to the designers routing cable to avoid the overfill problems to date. Although the problems at WBN 1 may be beyond fixing in many instances, expeditious action should be taken to upgrade the system for WBN 2 and Bellefonte.

IV. DETAILS

A. Background

From September 1982 through February 1984, an independent review of the Watts Bar Nuclear Plant (WBN) auxiliary feedwater system was performed by Black and Veatch (B&V) to determine the conformance of the system to commitments docketed in the FSAR. A TVA program team provided responses and additional information to the B&V reviewers to resolve questions and define corrective actions for confirmed deviations.

In a separate activity, a policy committee and a task force, both composed of senior TVA staff, evaluated the B&V findings for significance to other WBN unit 1 and 2 systems. The findings were evaluated for root cause and sorted into groups of similar nature. Where deemed appropriate, safety evaluations were performed to determine the consequences to the plant had the B&V review not been performed and the deviations gone undetected.

The B&V review findings were published initially in April 1983 and supplemented in February 1984. The results of the TVA task force efforts are documented in their March 1984 report to the policy committee chairman. The policy committee efforts are documented in their summary report to the EN DES Nuclear Engineering Branch.

The NSRS was involved throughout the process by being represented on the policy committee, by participating in the continuing reviews of the B&V findings and the TVA responses and by performing evaluations of the task force activities and report. The task force grouped the B&V findings into categories by determin-

ing the causes and generic applicability. The NSRS has evaluated each of these categories and basically agrees with the groupings of the findings. In the following section the NSRS evaluation of the categories is discussed. Any recommended follow-up action resulting from these evaluations is set forth in section III.

B. Discussion

1. General Observations

From an overall perspective, the B&V review showed that TVA did an acceptable design job in meeting the first order design requirements. Although there were a number of instances where the licensing commitments and licensing bases were not satisfied, further evaluations showed no cases where the ability to safely shutdown the plant was defeated. The deficiencies for the most part were failures to provide the additional margins of assurance committed in the FSAR. The basic causes for deficiencies involved lack of or poor training, failure to follow procedures, poor understanding of the commitments and lack of clear procedural definitions of commitments. In some instances the commitments were not rigorously met because they were viewed as enhancements as opposed to firm requirements.

Safety evaluations were performed by EN DES of the categories of findings where the licensing bases were not met. In all cases it was found that the ability to shutdown the plant had been maintained. The impacts of the reduced margin on overall plant safety and the effects of failure to implement the criteria for protective devices for plant equipment were not assessed.

2. Task Force Category 3

Category 3 contained 25 B&V findings where logic/control drawings did not agree with the electrical drawings. The identified cause for the category was failure to implement design review procedures as required by engineering procedure EP 4.25. The task force concluded the problems were generic to logic, control, schematic and connection diagrams throughout WBN units 1 and 2. The review was extended to three additional systems where similar problems were found. It was determined that corrective action was required for both past and future work.

The line organization has issued ECNs and FCRs to correct identified errors in hardware wiring and training was conducted in the I&C section of Sequoyah/Watts Bar Project (SWP) for EP 4.25. The drawings will be stamped to restrict the use to the intended function. No further reviews of other systems is planned to determine if other systems have

the same problems, in spite of the widespread problems identified in the four systems that were reviewed. These problems included instances where as many as 13 wires shown on one drawing were installed on the wrong terminals (FCR E-3508, system EA). Finding 1805 identified a crosstie between normal and emergency 125 V dc systems. Schematic 45W603-46-6R4 was different in many significant respects from logic diagram 47W611-3-4R2.

NSRS agrees with the TVA line actions to the point of correcting known wiring errors. We do not agree that their corrective action for past and future work is adequate. Since the problems have been demonstrated to be common in the four systems reviewed, it is reasonable to assume the deficiencies are institutional and all the plant systems should be reviewed and deficiencies corrected. Further to knowingly allow discrepancies to continue to exist in overlapping documents and depend upon a note to control document usage is very poor practice. Although the precise reason for the wiring errors cannot be ascertained, having conflicting information on overlapping documents cannot be helpful in precluding such errors. Further, during operations, personnel such as ROs, SROs, maintenance engineers and crafts rely upon such drawings as logic diagrams and schematics to perform their jobs. Allowing discrepant information on any controlled document places too great a burden on the administrative control systems to preclude use for a wrong purpose and further errors attendent to this practice can be expected including further wiring errors and misoperating equipment. The correct way of handling drawing errors is specified in EP 1.26 "Nonconformances Reporting and Handling by EN DES." It may be necessary to have more training in procedures at the management levels since these actions are being prescribed by management. NSRS recommends that all controlled documents should be clearly identified for all plants. The purposes and uses for each of the documents should be delineated. Information contained in the documents designated to be controlled should be assessed for contribution to the intended purpose and use. Superfluous information should be deleted and discrepancies in documents with overlapping information should be corrected. Establishing a verified "as built," rigorously documented, should be assigned a very high priority.

3. Category 4

Category 4 contained 12 B&V findings which the task force described as failure to design/maintain design records for the AFW system as specifically described in the FSAR. The task force identified cause was that TVA personnel were not aware of the FSAR statements. When the design changed, the FSAR was not uniformly amended to reflect the new designs. The problem was deemed generic to both WBN units and required corrective action for past and future work.

A special engineering procedure (SEP 83-05) was written to verify the accuracy of the WBN FSAR. Additionally, EP 2.01 was revised (revision 5) to upgrade the procedure for processing FSAR changes and EP 4.02 has been revised to require that engineering change notices (ECNs) describe FSAR changes needed as a result of the design change. NSRS reviewed the SEP to assess its completeness. Little guidance is offered to the reviewers of the FSAR as to the depth or method of review. Further the B&V review found deficiencies in the TVA response to IE Circular 81-13 and IE Bulletin 80-20. The SEP 83-05 review was restricted to FSAR sections; questions, responses to IE bulletins, NRC generic letters, etc., were not included in the review. The SEP review may not have corrected the deficiencies in the remaining commitments.

NSRS reviewed a sample of the proposed FSAR revisions and found a number of inconsistencies. The problem was discussed with OQA; a program was instituted by OQA to address the NSRS concerns.

4. Task Force Category 5

Category 5 had 10 findings where procurement forms and flow diagrams specified different requirements for various valves and qualification documentation was not tied to the design and procurement process. The task force concluded this category required corrective action for future work and for past work as appropriate.

The underlying problem for this cateogry was a breakdown in the ECN process. Although some of the problems were attributed to the inappropriate use of S1 ECNs, some breakdown in the ECN, squadchecking, and signature process occurred. Although the task force identified two EPs being changed or issued, the procedures in place at the time that these problems developed were adequate if rigorously implemented. The NSRS agrees with the TVA and task force actions for this category.

5. Task Force Cagegory 6

Category 6 contains 7 findings of discrepancies between documents (analysis results, load tables, isometric drawings, flow diagrams, etc.) used in the design of piping systems. The task force found this category required corrective action for both past and future work.

The NSRS agrees in general with the TVA and task force conclusions for this category. But even though individual areas of the design may indeed have random and unique errors, an overview of these areas indicates a generic problem of implementation of procedures, attention to detail and lack of a really independent review process.

7. Task Force Category 7

Category 7 has 17 findings of nonconforming conditions in construction of previously inspected and accepted pipe supports. This set of findings required some modification to future activities; other TVA actions in place prior to the B&V findings are expected to resolve any deficiencies in completed work.

All of these items were due to the pipe supports in the field being different from what was shown on the drawings. In many cases there were ECNs and FCRs pending when B&V did their study. This resulted in drawings being different than field conditions because CONST had not made the modifications yet. Also, the NRC Bulletin 79-14 program, "walk-down," under WBN-QCP-4.56 had not been implemented when B&V did their study. The discrepancies probably would have been been corrected by the 79-14 program. There is no safety concern after implementation of the 79-14 program, and correction of any deficiencies found, which is required prior to unit fuel loading. The pipe supports would have been inspected, and the ones with problems would have been corrected.

7. Task Force Category 9

Category 9 has 8 findings of failure to adequately control and evaluate embedded plate capacity when multiple attachments were made to the plate by construction. The task force concluded that corrective actions already identified and scheduled would have resolved the deficiencies and that some modification to planned corrective action for future work is needed.

NSRS substantially agrees with the task force and EN DES responses and actions for this category except in the area of embedded strip plates.

The initial NSRS review and discussions with cognizant EN DES designers concluded the findings relative to the strip plates would not have been corrected by actions already identified nor would the deficiencies have been corrected by the corrective action plan identified. There is no control system to identify and maintain records of as built loads on the plates. With this lack of record or system, there is no way of knowing whether plates are overloaded.

NSRS has two points in question on this subject which lead to a direct safety concern:

1. Although it was not discussed by B&V, the embedded plates have been analyzed with a "rigid" plate analysis as opposed to a "flexible" plate analysis. This can be

an unconservative analysis. Of 69 cases that were checked, an anchor on one plate was shown to be stressed beyond the allowable stress in the acceptance criteria and a stiffener was added as in documented in the corrective action NCR WBN-CEB-82-02. This is a generic issue since all plates are analyzed in the same manner. It should be recognized that NRC has not fully accepted the TVA assumptions in response to IE Bulletin 79-02.

2. Of much greater concern is the control of attachments to embedded plates. There is no bookkeeping system to keep track of the cumulative load on any individual embedded plate. Construction Specification N3C-928 was implemented in February 1983 in response to the B&V findings to restrict locations of attachments. However, there was still no system established to identify and control the cumulative load on each plate. This specification should have been more restrictive. On January 6, 1984 revision 2 of N3C928 was issued which allowed the EN DES representative onsite "by visual examination" to determine whether a detailed evaluation of the plate is required. The representative has no guidelines or acceptance criteria, but uses engineering judgement. NSRS has serious concerns with this reduction in requirements. Black & Veatch had signed off on this finding on December 30, 1983 on the basis of the February 1983 revision of N3C-928. TVA relaxed the requirements in the specification seven days later. The only way to ensure that plates do not fail is to do an analysis using the actual loads or to compare the actual loads to the loads used in the prior analysis and show that the revised loads are within the envelop of the analyzed case.

Based upon the second concern, NSRS visited WBN to review the visual examination process. The EN DES representatives were actually checking loads against the allowables, not just visually examining; however, the results were not being documented. After the NSRS visit, OQA issued deviation report C03-S-84-0089-D01 and EN DES designers have agreed verbally with NSRS and in response to OQA for the deviation to revise Appendix 4 to EN DES EP 4.03 to document the field calculations as the basis of approval. This would eliminate our concern on the cumulative loads for attachments made under N3C-928.

NSRS has a residual concern for all the multiple attachments made prior to February 1983. The sampling of 69 plates revealed one plate with an overstressed anchor requiring a stiffener. The EN DES cognizant designers should consider taking a larger sample to gain greater confidence that all the plates are adequate. If the additional sampling is not

done, consideration should be given to performing a safety evaluation of the supported members with a basis of 1 to 1½ percent support failures since 1 in 69 of the embedded plates sampled had an overstress condition when compared to the allowable.

8. Task Force Category 11

Category 11 has 2 findings of inadquate documentation of operational modes data used in the analyses of piping systems. These findings were classified as deviations from the licensing commitments and bases and required corrective action for both past and future work.

A sampling program of rigorously analyzed piping was instituted to provide assurance that no design problems remained. Initially 20 problems were evaluated and none required reanalysis. The sample problems represent approximately 10 percent of the total number of rigorously analyzed problems. Another 30 percent have been updated for other reasons and the proper operational mode data were included.

NSRS agrees with the TVA actions and conclusions for this category.

9. Task Force Category 12

Category 12 had one finding of failure by EN DES and CONST to properly implement and document the alternate analysis criteria for seismically supported piping. It was concluded that although there was a deviation from a licensing commitment, actions already being taken by TVA would have corrected the problem without reliance on the B&V study.

NSRS agrees with the TVA actions defined in EN DES SEP 8218 and SWP EP 43.21 dealing with alternate analysis problems and the task force conclusions.

10. Task Force Category 13

Category 13 had one finding in which termination information on documentation was in error and was not updated to reflect the actual configuration. The task force review concluded based on a sampling of 40 additional AFW termination records with no discrepancies that this finding was an isolated case and no further action is required.

NSRS supports this conclusion.

11. Task Force Category 14

Category 14 had 22 findings where various supports on the AFW system had not been modified, redesigned, or initially designed per revised analysis of ECN 2576. The task force concluded the findings were departures from licensing commitments and licensing bases. Corrective action was designated for both past and future work. The EN DES evaluation of the overallimplications of the discrepancies revealed that the problem was substantially isolated to the one ECN. A total of 5500 supports were reviewed--5000 in ECN 2576 and about 500 in ECN 3184 to support the conclusion. Although about 8 percent of the supports covered by ECN 2576 required some construction modification, only one support covered by ECN 3184 required construction modification which very strongly supports the conclusion that ECN 2576 was an isolated occurence albeit over an extended period of time.

An evaluation of the support deficiencies showed that the reserve stress in the pipe was not exceeded such that even through a support may have failed, the piping would not be overstressed. NSRS fully agrees with the task force and EN DES conclusions and corrective actions for this category.

12. Task Force Category 18

Category 18 had one finding where a technical note on a piping support drawing was found to be invalid for some applications. It was concluded that the finding condition was a deviation from a licensing commitment but the licensing basis was met. There was corrective action for future work; no modifications to existing support bolting was required.

NSRS agrees with the task force and EN DES evaluations and corrective action for this category.

13. Task Force Category 19

Category 19 had two findings where equipment could not be determined to be environmentally qualified to NUREG0588. These findings represented deviations from the licensing commitment; TVA already had a program in place which could have reasonably been expected to correct the problems.

NSRS agrees with the TVA conclusions for this category.

14. Task Force Category 20

Category 20 had five findings where, as stated by the task force, no procedure existed for documenting preoperational

testing determined time delay relay settings and the preoperational test scoping document did not identify or require documenting the settings. The task force classified the findings as deviations from the licensing commitments but the safety consequences were indeterminate. The task force documentation indicates the settings made prior to June 1983 were documented adequately by an interim memorandum (EEB 830614 439). The preoperation test scoping document and EN DES procedure SEP 83-11 have been written to require documentation of all the settings determined after June 1983.

The NSRS evaluations of this problem showed the scope to be greater than the task force addressed in this category since there appears to have been no effective control over time delay relays. Corrective actions for significant NCRs for these findings included procurement of new time delay relays to provide an adequate range. The existing relays would not allow setting the time called for on logic diagrams, hence the logic had not been properly implemented. This may be related to the lack of procedures governing logic diagrams (Category 3). The extent of the generic applicability review for this category is not clear to NSRS. The methods and procedures for determining the proper values, physically setting, and verifying time delay relay settings should be reevaluated and indicated changes should be expeditiously made TVA-wide.

15. Task Force Category 23

Category 23 has two findings related to the AFW turbine pump trip and throttle valve not being included on the active valve list and the valve schematic not including the required control room bypass and test indication nor automatic bypass of the open torque switch. It was concluded the discrepancies were deviations from both the licensing commitments and bases. Corrective action was required for past and future work. The evaluation for cause concluded the deficiency was an isolated error resulting from failure to include the valve on the active valve list. Including the valve on the active valve list, providing the automatic torque switch bypass and providing the control room indication of bypass and test of the thermal overload correct the deficiency and the licensing requirements are met. The EN DES safety evaluation concluded the nuclear safety of the plant would not have been reduced if the deficiency had not been corrected.

NSRS agrees the corrective action taken is acceptable and the requirements are satisfied.

16. Task Force Category 25

Category 25 has one finding of flange evaluations being omitted in some analysis calculations. The task force concluded that the licensing commitment had not been met but evaluation showed the licensing basis was met. The corrective action included a 100 percent review of all completed calculations to assure flange qualification. Since the deficiency was attributed to individual errors, the corrective action for future work is to more clearly define the requirements.

NSRS agrees with the EN DES corrective actions and conclusions.

17. Task Force Category 30

Category 30 has two findings of failure to satisfy design criteria for (1) monitoring operability and (2) providing adequate electrical protective devices for the motor-driven AFW pump lube oil pump. The task force concluded the licensing commitment and the licensing basis were not met. The evaluation for causes revealed inadequate training and poor or lack of communications with NUC PR and EN DES. In reviewing other equipment, only one additional instance of failure to provide electrical protection was found. Thus the deficiency was not widespread. The EN DES safety evaluation of the two findings concluded there would be no safety concern had the defects not been corrected.

NSRS agrees with the specific corrective actions for the identified problems for this category.

18. Task Force Category 31

Category 31 has two findings of editorial discrepancies in licensing documents. The findings did not represent compromises of the licensing basis. The low number of errors found in this category support the conclusion that no action beyond correcting the identified errors is warranted, particularly in light of the extensive efforts detailed in Category 4.

NSRS agrees with the task force conclusions for this category.

19. Task Force Category 32

Category 32 has nine findings of incompatible hanger drawings and piping isometrics. The errors were deemed to be caused by checking and design verification of documentation between branches not being done as required by procedures.

The corrective action was to train designers in the procedural requirements. The errors did not result in any identified safety concerns since much of the work was not complete and system walkdowns could be expected to identify any incorrectly placed or installed supports.

NSRS has one residual concern with the EN DES corrective action. Since the root cause was inadequate training in procedure requirements, a continuing or periodic training program would appear to be needed. One time training is not felt to be totally adequate. Further, the corrective action was through SWP-A11 R2, which appears to apply only to the Sequoyah/Watts Bar design projects not to TVA design projects in general.

20. Task Force Category 33

Category 33 has two findings of inadequate cable tagging. The two cited instances were the result of an oversight in one case (correct information, wrong color tag) and information being obscured on the tag due to wear and tear from rework in the other. No corrective action for past or future work was indicated since the frequency of occurence was low and walkthroughs are already designed to find and correct errors of this type. No safety concerns were expressed.

NSRS agrees with the task force conclusions for this category.

21. Task Force Category 34

Category 34 has 11 findings where "out of function" features of drawings were not in agreement with the latest design drawings showing the detailed design of the "out of function" features. The task force concluded that the "out of function" features do not impact the technical adequacy of drawings and are not used for design, construction or operation of the plant. No corrective action was deemed necessary.

NSRS agrees with the technical impact conclusions reached by the task force; our recommendation for Category 3 is equally valid for this category.

22. Task Force Category 35

Category 35 has one finding where instantaneous trip settings for motor-operated valve breakers were not in accordance with EN DES criteria and vendor recommendations. The task force concluded the licensing commitment and the licensing basis were not met and corrective action was required for both past and future work. The EN DES safety evaluation concluded:

> While these high settings were found to violate good design practice and could lead to a motor control center failure, the high trip settings would not prevent the safe operation or the safe shutdown of the plant.

The basic cause of the deficiencies was lack of training and knowledge of changed requirements and expedient decisions not to correct deficiencies when the requirements were known not to have been met.

NSRS has substantial concerns with the EN DES and task force resolution of this finding category. First, the safety evaluation tabulates 444 breakers out of 610 having settings greater than 1300 percent of the motor full load current. Of the 444 breakers, 385 were either reset or replaced and set. The remaining 59 breakers were neither replaced nor reset and are still apparently not in compliance with the commitment to the requirements of TVA Design Standard E.9.2.1 (now superceded by a non-mandatory Design Guide E2.3.5, issued November 10, 1983) which references requirements of the National Electric Code (NEC). No justification was or has been documented for not resetting or replacing the 59 breakers. This misapplication of the NEC requirements as implemented by Design Standard 9.2.1 leaves TVA in noncompliance with the practices of industry as reflected in the NEC for motor circuit protection. This in turn places WBN in noncompliance with the FSAR commitment although the FSAR does not directly commit to the NEC, the Design Standard clearly does and the Design Standard has not been met in all cases.

A second and higher level concern is the failure of the cognizant EN DES personnel to properly consider applicable parts of the NEC. The EN DES safety evaluation very selectively quotes section 430-52 of the NEC by quoting an exception ". . . the setting of instantaneous trip circuit breakers shall in no case exceed 1300 percent of the motor full load current." Other parts of section 430-52 which are equally applicable state: "The motor branch circuit short circuit and ground fault protective device shall be capable of carrying the starting current of the motor. A protective device having a rating or setting <u>not exceeding</u> the value calculated according to the values given in Table 430-152 shall be permitted." The <u>maximum</u> allowed setting in Table 430-152 is 700 percent of the full load current of the motor. The full wording of the exception quoted in the safety evaluation is "Where the setting specified in Table 430-152 is not sufficient for the starting current of the motor, the setting of an instantaneous trip circuit breaker shall be permitted to be increased but <u>shall in no case exceed 1300</u> percent of the motor full load current."

The expressed EN DES electrical design practice and philosophy are not in concert with present day nuclear design logic or common industrial practice. By NSRS reading, the stated EN DES positions do some injustice to the reasons for having protective devices of any sort. Clearly protective devices should be set as closely to normal operating conditions as possible while recognizing the full range of conditions including starting loads and avoiding nuisance trips. The NEC specifies this clear philosophy by using words such as "not exceeding" and "maximum" throughout. Table 430-152 of the NEC specifies 1300 percent to be the maximum exception. The NEC does not specify that all the breakers should or can be set at 700 percent or 1300 percent or any other given value.

The EN DES safety evaluation is incomplete in that the consequences of the pervasive nature of the deficiencies was not thoroughly considered. A worst case consequence was proposed which could lead to a fire which could disable a complete motor control center. It was further stated that the scenario, while possible, is so improbable as to be considered incredible. NSRS is concerned that broad conclusions have been reached with so narrow failure analysis and consequences determination being documented. The misapplication of the breakers exposes equipment to unnecessary challenge. These challenges can cause undetected failures which would not be seen during periodic testing. At the best, the deviations would have reduced safety margins even though single failure criteria may have been met; therefore, the deviations were significant to safety.

The instantaneous trip breakers should be verified to be set in agreement with the intent of the National Electric Code. The documentation for the design as well as for the testing and operations of the equipment should reflect the proper values. The program should be implemented TVA wide.

23. Task Force Category 36

Category 36 has one finding that the cable tray fill criteria are not assured of being met because of the less than conservative nominal values used for cable cross sectional areas in the cable routing program. After evaluation by designers, it was concluded that the licensing requirements had been met and no corrective actions are required for either past or future work.

NSRS does not agree that the licensing commitment has been met; it is not clear that the licensing basis has

been met. The WBN FSAR states that ". . .low-voltage power cable tray fill shall be limited to a maximum of 30 percent of the cross-sectional area of the tray, except when a single layer of cable is used. Cable tray fill for control and instrumentation cables shall be limited to a maximum fill of 60 percent of the cross-sectional area of the tray." The supporting EN DES documentation for the conclusion that the licensing requirements had been met was based upon considerations of dead weight, ampacity and heating value of combustibles in isulation and jacket materials. While NSRS agrees these are important considerations, there are others such as mechanical protection of the cables from missles or casual hazards.

The FSAR describes a fully automated computerized system to route cables and to control cable fill using the criteria stated above. There is not a variable to control for cables of the same guage but different diameter; there is no formal feedback procedure to alert the designer, when for vagaries of construction, that the tray is full physically before all the cables are installed as computer routed. Further, no acceptance criteria have been provided for either the installer or a QC inspector to use to consistently determine that a tray is physically full.

Although not a part of the findings in task force Category 36 additional conditions adverse to quality noted by NSRS during a field trip to WBN to observe the cables in cable trays were:

1. Excess cable coiled and hanging from edges of cable trays.

2. Excess cable coiled and lying on the floor where people have to walk to access areas of the plant.

3. No record of megger test results for cables.

EN DES should develop acceptance criteria to be used by construction forces as well as the QC inspectors which define fill in measurable terms to supplement the arithmetical computer methodology. The additional problems above must be resolved. Until these deficiencies are corrected, TVA can not adequately justify that the licensing requirements are satisfied in full. NSRS believes safety evaluations should be made of the conditions described prior to substantial plant operation.

24. Task Force Category 37

Category 37 has one finding where valve wiring circuits were designed such that the red and green indicating lights on the unit control board would light dimly upon malfunction of the PAuto contact of the Westinghouse W2 control switch on the unit control board. It was concluded the design did not satisfy either the licensing commitment or basis and corrective actions were taken. EN DES recognized the requirement; however, the failure was a random design error in conjunction with inadequate design verification. The circuits with W2 switches were reviewed and the deficiencies were corrected when found.

NSRS agrees with the EN DES corrective action taken.

25. Task Force Category 38

Category 38 has two findings of failure of the thermal overload bypass circuit designs to meet the requirements of RG 1.106 and IEEE 279-1971. The task force concluded the licensing basis had been met and no corrective action was required.

NSRS agrees with the EN DES and task force resolution for this category.

26. Task Force Category 39

Category 39 has one finding where the specific configuration of 6.9kV bundled cables in trays had not been tested for the effects of fire retardant coating on the ampacity of the cable. The task force concluded the licensing commitment had not been met but the basis had been satisfied. An evaluation of the condition was prepared as part of the policy committee activity and was presented to NRC for acceptance.

NSRS agrees with the conclusions and actions taken by EN DES and the task force for this category.

TVA 84 (08-9-65)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

GNS '841017 050

TO : W. R. Brown, Project Manager (Bellefonte), 102 ESTA-K

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

DATE : OCT 17 1984

SUBJECT: BELLEFONTE NUCLEAR PLANT - NSRS REPORT NO. R-84-22-BLN - DECAY HEAT REMOVAL SYSTEM

Attached is an NSRS report concerning the design of the decay heat removal system at Bellefonte. In general, the design of the system appears to be adequate to meet its intended safety functions. However, the report does contain two recommendations to address apparent deficiencies in the design of the low temperature reactor vessel overpressurization protection features. Please provide responses to these recommendations within 30 days of the date of this memorandum. If you have any questions, please contact Bruce Siefken at extension 6860.

H N Culver

H. N. Culver

BFS:BJN

Attachment

cc (Attachment):

R. W. Cantrell, W11A9 C-K

MEDS, W5B63 C-K

H. G. Parris, 500A CST2-C

E. G. Beasley, W12B21 C-K

C. W. Crawford, 670 CST2-C

NSRS FILE

GNS '841017 051

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

REVIEW

NSRS REPORT NO. R-84-22-BLN

SUBJECT: BELLEFONTE NUCLEAR PLANT - DECAY HEAT REMOVAL SYSTEM

REVIEWER: Bruce F. Siefken — 10-15-84
BRUCE F. SIEFKEN — DATE

APPROVED: James F. Murdock — 10/16/84
JAMES F. MURDOCK — DATE

MEDS, W5B63 C-K

I. SCOPE

This review deals with the decay heat removal system (DHRS) at Bellefonte Nuclear Plant (BLN) and was restricted to a functional review of the system. That is, the various functions and design requirements were identified, and the design of the DHRS was then compared to the identified criteria. Both external documents (e.g. regulatory guides, standard review plan, etc.) and internal documents (e.g. FSAR, design criteria, etc.) were examined to determine the functional requirements for the DHRS. The review was limited to mechanical and fluid aspects of the design. Electrical and instrumentation requirements were not reviewed in detail. A detailed check of the adequacy of the DHRS in mitigating a loss of coolant accident (LOCA) was also not included in the review scope. A detailed review of these areas will be made and documented in a separate report.

II. BACKGROUND

The BLN decay heat removal system is typical of B&W's standard design for 205 FA plants. The decay heat removal system at BLN performs a number of safety-related functions and some nonsafety-related functions. The safety-related functions include decay and sensible heat removal, supply low pressure injection, recirculation cooling, piggy-back cooling, spent fuel cooling, auxiliary pressurizer spray, reactor vessel overpressure protection, and long-term cooling. Nonsafety-related functions include filling and draining the refueling canal and reactor coolant purification during refueling. The DHRS is the safety-related method of cooling the reactor from 305°F to cold shutdown. It is NSRS's position that cold shutdown is the most stable and safest plant condition in off-normal conditions, such as after a failure which requires maintenance. Thus, it plays an important role in plant safety. The diversity of its functions results in many design requirements being placed on the system. The DHRS was chosen since it plays an important role in safe plant operation and in accident mitigation and embodies many plant interdependent features. In this report the conceptual design of the system was examined to verify that the design meets the design bases for the system.

III. CONCLUSIONS/RECOMMENDATIONS

The conceptual design of the mechanical portions of the DHRS is generally acceptable with the following potential concerns:

A. R-84-22-BLN-01, <u>Adequacy of Low Temperature Reactor Vessel Overpressurization Protection</u>

FSAR Figure 5.2.2-3 indicates that at refueling temperatures, the maximum allowable reactor coolant system (RCS) pressure is approximately 450 psig. However, the overpressurization calculations indicate a maximum RCS pressure of 725 psig at the DHR discharge to the cold leg. (Refer to FSAR section 5.4.7.2.1 and 5.2.2.11).

NSRS recommends that additional low temperature reactor vessel overpressurization protection measures be instituted and documented in the design basis for BLN.

B. R-84-22-BLN-02, Adequate Design Margin for DHR Isolation Valve Opening

FSAR Figure 5.2.2-3 indicates that the DHR suction isolation valves must be open when the reactor coolant temperature falls below 305°F to ensure adequate low temperature reactor vessel overpressure protection. Bellefonte General Operating Instruction BLGOI-1C also requires the operator to open the DHR isolation valves at 305°F. The DHR pump equipment specification, B&W document No. 08-1130000007-07 lists the maximum liquid temperature of the pump suction as 305°F. Thus it appears that there is no margin in the design for instrument error and operator action.

NSRS recommends that additional temperature margin be incorporated into the DHRS design and be documented in the DHRS design basis.

IV. DETAILS

The functions of the DHRS are defined in several documents, but the principal functions are contained in the design criteria document for the DHRS, reference A, and in the B&W system description for DHRS, reference J. Regulatory requirements for each of these functions were compiled and the design of the DHRS was compared to the compilation. Instrumentation adequacy was assessed by reviewing the available operating instructions and the limits of system operation. No attempt to perform a human factors engineering review was made.

The review concentrated on the fluid and mechanical aspects of the DHRS. Detailed electrical aspects of the instrument, control, and power circuitry for the system were not reviewed. The sections below summarize the results of the functional review.

A. Remove Decay Heat and Sensible Heat

The DHRS pumps 5000 gal/min of reactor water through the decay heat removal heat exchangers. Component cooling water system transfers heat from the DHR Hx to the essential raw cooling water (ERCW) system. The ERCW system then carries the heat to the ultimate heat sink. Thus, decay and sensible heat are removed from the RCS. The decay heat removal heat exchangers are sized to remove 4.1 x 10^7 Btu/hr which is adequate for normal operation. A bypass line around the heat exchangers and throttling valves provides the means of controlling the RCS temperature to within limits. (There are maximum cooldown rates and an absolute minimum temperature limit.) Both trains of DHR are needed to cool the RCS from 305°F to 140°F within 14 hours as specified in the system description. The use of only one train lengthens the cooldown time considerably to about 140 hours to reach 140°F.

However, the RCS temperature can be reduced to 212°F in about 9 hours using only one train of the DHRS. These longer times do not meet the recommendations in proposed R.G. 1.139 (i.e. 36 hours to cold shutdown after a single failure), but this is a rather arbitrary limit. There is little safety significance in not meeting these times with the DHRS since the RCS can be quickly depressurized to atmospheric pressure if needed.

During refueling the DHRS cools the reactor coolant system and the refueling canal. This is accomplished in the same manner as plant cooldown. A small flow from the canal does go through the purification system and the DHR can be aligned to circulate this flow.

B. Supply Low Pressure Injection

The DHR pumps also serve as the low pressure injection (LPI) pumps following a LOCA. They are required to deliver 5000 gal/min to the reactor vessel at a pressure of 100 psig. The pumps appear adequate for this task by developing a minimum of 385 feet tdh at 5000 gal/min. This leaves an allowance of about 70 psi pressure drop in the piping from the pump to the reactor vessel. Pumps are automatically started and valves automatically opened on an engineered safety features actuation system (ESFAS) signal (triggered by low reactor pressure, high containment pressure, or low steam generator pressure). Minimum flow is provided by a recirculation line from the DHR cooler outlet to the pump suction. This flow (125 gal/min) has been properly accounted for in sizing the pumps.

The LPI function of the DHRS is required to be single failure proof. The design appears to meet this requirement with two independent, full-capacity trains. The initiating LOCA could affect the availability of some portions of the DHRS, but the LPI function can still be satisfied with a single failure. Cavitating venturies are used to limit the flow between the two LPI injection points to accommodate situations where the initiating LOCA breaks an injection line and a single failure fails one LPI train. A detailed review of the LPI function was not undertaken but will be included in a review of LOCA mitigation at BLN.

C. Recirculation Cooling Post-LOCA

The DHR/LPI function automatically switches its suction from the BWST (borated water storage tank) to the RBES (reactor building emergency sump). Thus, reactor coolant is recirculated. Low level in the BWST triggers this switch by first opening the sump isolation valves and then closing the BWST suction valves (after the RBES valves are 90 percent open). The cross connection to the makeup pumps is also automatically opened. [The makeup pumps double as the high pressure injection (HPI) pumps.] This "piggy-back" mode ensures that flow continues through the DHR pumps if the reactor pressure is high.

There is adequate net positive suction head (NPSH) available for the HPI operating mode. R.G. 1.1 was used to calculate the NPSH available with the conservative assumption that the sump fluid is at saturation. Thus, only the elevation difference of the sump and the pump suction is assumed to contribute to NPSH. The sump is the more limiting case since the BWST is subcooled considerably.

D. Piggy-Back Cooling

The discharge of the DHR pumps is automatically aligned to the suction of the HPI pumps on a low BWST level signal. This ensures adequate flow through the DHR pumps to prevent pump damage in the event that the RCS pressure is above the DHR pump shutoff head. This is needed since the minimum flow line from the DHR pumps is isolated automatically on low BWST level to preclude pumping contaminated water from the sump to the BWST. The LPI discharge valves are left open so that as the RCS pressure drops the DHR pumps can deliver flow directly to the reactor vessel. When this flow (direct to vessel) is greater than 700 gal/min, the piggy-back flow can be safely stopped. There is sufficient flow information for the determination of this flow, but several different flows must be algebraically summed to obtain the information. This procedure is adequate, however, since it is not critical to the safe shutdown of the plant.

E. Spent Fuel Cooling

The DHRS can be used to provide cooling to the spent fuel pool as a backup to the spent fuel cooling system. One train of the DHRS is adequate to cool 1-1/3 cores of fuel in the pool, the design condition for the pool cooling systems. The reactor should be defueled before placing the DHRS into the spent fuel cooling mode since manual valves need to be aligned, which results in the DHRS being unavailable for reactor cooling or low pressure injection. The spent fuel cooling system (SFCS) consists of two trains, but for the case of 1-1/3 cores in the pool, both spent fuel cooling trains are needed. The DHRS flow to the fuel pool will require throttling since the DHRS normal flow is 5000 gal/min and SFCS flow is 1650 gal/min per pump. This throttling can be accomplished by the valve in the decay heat cooler outlet. Temperature can be controlled with the cooler bypass valve. Sufficient instrumentation exists to allow the operator to accomplish these actions.

F. Provide Auxiliary Pressurizer Spray

- If the reactor coolant pumps (RCP) are tripped for any reason, the normal source of pressurizer spray is lost (i.e. the RCP discharge). If the reactor is at a high pressure, the makeup pumps are used to supply high pressure spray through a length of

2-inch DHRS piping to the pressurizer spray line. At low pressure, the DHR pumps can be used to supply the spray flow. The spray connection is located at the decay heat cooler outlet which results in the head available for spray flow being the pressure drop across the DHR throttling valve and the pressure drop in the piping to the reactor vessel. The design calls for a maximum flow rate of 150 gal/min through this line. This value is verified during the preop test since the available head appears to be small and there are a number of valves in the 2-inch line. Throttling of the spray flow is done by two valves operable from the control room in a parallel flow arrangement. The throttling capability is necessary for matching the depressurization rate to the cooldown rate. The parallel flow paths also ensure that if one motor-operated valve fails to open, there still exists a flow path for auxiliary spray. The valves are powered from different trains of the class IE electrical system. Since most accidents are required to be mitigated without offsite power, these valves are needed for a timely depressurization of the primary system.

G. Low Temperature Reactor Vessel Overpressurization Protection

The decay heat removal system contains two relief valves, one in each drop line to the DHR pumps from the RCS, which provide overpressure protection from the RCS at low temperatures. Each valve is sized to provide 100 percent of the required relief flow for a variety of overpressure events. The overpressure events include energizing all pressurizer heaters, loss of all decay heat removal, and spurious start of the HPI pumps. The start of the HPI pumps was the most limiting event. The relief valves were originally placed in the design to provide overpressure protection for the DHRS. However, recent NRC concerns on the need for low temperature RCS overpressure protection has prompted the use of these same valves for RCS protection. The relief valves appear adequate for DHRS overpressure protection.

The use of these valves to protect the reactor vessel, however, has some problems. The relief valves are located downstream of the decay heat letdown isolation valves. Thus, there are two isolation valves between the RCS and the relief valves. However, since the valves provide low temperature pressure protection, there needs to be some method of isolating them from normal operating pressures. In order to prevent a possible brittle failure of the reactor vessel, overpressure protection must be provided. As the temperature of the reactor vessel decreases, the maximum allowable pressure in the vessel decreases. FSAR Figure 5.2.2-3 shows this relationship. At refueling temperatures, the maximum allowable RCS pressure is about 450 psig, while above 305°F the maximum allowable pressure is 2,500 psig. The problem with the design is that the decay heat isolation valves must be open before the low temperature pressure protection is needed, but after the RCS temperature is below the maximum DHRS suction temperature. FSAR Figure 5.2.2-3 shows that the DHRS suction lines must be open at 305°F after the reactor vessel

has sustained 32 effective full power years of operation since the pressurizer safety valves no longer provide reactor vessel overpressure protection. The decay heat pump specification states that the maximum suction temperature of the pump is 305°F. Thus, the maximum temperature and the minimum temperature for opening the DHR isolation valves are both 305°F which leaves no margin for instrument errors or operator action. When instrument errors are taken into account, there exists the possibility of exceeding the maximum DHR suction temperature which may damage the pumps or exceeding the pressure-temperature limits of the reactor vessel.

The calculations done for the reactor vessel overpressure protection are the same calculations done for the DHR overpressure protection. The problem is that the worst case assumptions for the DHR case are not the worst case for the reactor vessel. A plot of the allowable RCS temperature versus pressure (FSAR Figure 5.2.2-3) shows that at refueling temperature (140°F) the allowable RCS pressure is only about 450 psig (estimated). The DHR relief valves were sized to prevent the DHR suction pressure from exceeding 500 psig (FSAR section 5.4.7.2.1). The overpressurization calculations indicate a maximum RCS pressure of 725 psig at the DHR discharge cold leg connections. Thus, it appears that these relief valves may not provide adequate reactor vessel pressure protection unless additional measures are taken.

The possibility of low temperature repressurization event closing the DHR isolation was investigated. The DHR letdown isolation valves will automatically close above a pressure of 620 psig in the RCS. If one considers the single failure criteria in mitigating an overpressure event, then the closure of some of the decay heat isolation valves could render some of the relief valves unavailable. However, there does not appear to be a single failure that would render both relief valves unavailable, and one relief valve should be adequate per the FSAR analysis. Thus the isolation valves should remain open during a repressurization event.

H. Long-Term Cooling

The DHR may be required to operate for many weeks after a LOCA. There was a concern that boron may tend to concentrate in the reactor vessel during the long period of time and could crystalize out of solution as the reactor vessel cooled. This might possibly result in some cooling channels being blocked and fuel overheating (termed the "bone-china syndrome").

The DHR establishes a positive flow through the reactor vessel by opening the dump-to-sump valves. These lines run from the hot leg to the reactor building emergency sump (RBES). The DHR draws from the RBES and discharges into the reactor vessel. These lines each have redundant isolation valves to ensure that flow from the RCS can be terminated, if necessary, with a single

action failure. Furthermore, a total of four lines are used which ensures that a dump-to-sump flow can be established with an assumed single failure and the consequences of the initiating LOCA. Motive power for these isolation valves is normally removed during power operation. The operator manually initiates the dump-to-sump feature within 24 hours of a LOCA by establishing a flow path from at least one hot leg to the sump.

I. Nonsafety-Related DHRS Functions

The DHRS has two nonsafety-related functions, filling and draining of the refueling water canal and plant purification during refueling. The 5000 gal/min flow from one pump of the DHRS can be aligned to draw from the BWST and discharge into the canal. Handwheel-operated valves are used to align the discharge to the refueling canal. The DHR can also partially drain the refueling water canal. Handwheel-operated valves allow water to be drawn from the canal and discharged to the BWST. The water level can be lowered to the reactor vessel flange only. The spent fuel cooling system normally accomplishes these functions.

The DHR also can route water to the makeup and purification system during refueling. This allows water cleanup and purification for chemistry control. These functions appear to be adequately implemented.

V. DOCUMENTS REVIEWED/REFERENCES

A. Design Criteria for Decay Heat Removal System, N4-ND-D740, R0

B. Design Input Memorandum DIM-N4-ND-D740-1

C. Design Input Memorandum DIM-N4-ND-D740-2

D. Design Input Memorandum DIM-N4-ND-D740-3

E. Design Input Memorandum DIM-N4-ND-D740-4

F. BLN FSAR Section 5.4.7, "Decay Heat Removal System"

G. BLN FSAR Section 6.3, "Emergency Core Cooling System"

H. NRC Standard Review Plan, Section 5.4.7, "Residual Heat Removal System"

I. NRC Standard Review Plan, Section 6.3, "Emergency Core Cooling System"

J. B&W System Description for Decay Heat Removal System, 15-403600000-08

K. ANSI N18.21973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants"

L. NRC IE Information Notice 82-17, "PWR Low Temperature Overpressure Protection"

M. NRC IE Information Notice 82-45, "PWR Low Temperature Overpressure Protection"

N. BLN FSAR, Section 5.2.2, "Overpressurization Protection"

O. B&W Report BAW10074A, R1, "Multinode Analysis of Small Breaks for B&W's 205-Fuel Assembly Nuclear Plants with Internals Vent Valves"

P. NRC Branch Technical Position ICS3B, "Isolation of Low Pressure Systems from the High Pressure Reactor Coolant System"

Q. NRC Branch Technical Position RS5B-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures"

R. B&W Specification 08-1130000004-07, "Decay Heat Removal Pumps"

S. B&W Contract Specification 08-1024000003-02, "Heat Exchangers for Auxiliary System Service (Decay Heat Removal System)"

T. B&W Specification 08-1125000003-06, "Decay Heat Removal Pump Drives"

U. B&W Specification 08-1137000002-05, "Lube Oil Pump Motors for Decay Heat Removal Pumps"

V. B&W Operating Specification 64-1002746-00, "Decay Heat Removal System Operation"

W. B&W Technical Document 67-1003781-01, "Plant Limits and Precautions 05-1101"

X. B&W Letter D-4303, "Decay Heat Letdown Valve Interlock Setpoints," August 25, 1982

Y. B&W Letter L-575, "Suggested Responses - NRC Questions 440.9 and 440.10," January 10, 1983

Z. TVA Letter K-7267, "Decay Heat Letdown Interlock Setpoints," January 19, 1983

AA. B&W Letter P-1891, "Low Temperature Overpressure Protection - Alarm," January 24, 1983

BB. B&W Letter D-4554, "Decay Heat Letdown Interlock Set Points," April 13, 1983

CC. B&W Letter D-4600, "Low Temperature Overpressure Protection (LTOP) Requirements," May 19, 1983

DD. TVA Letter K-7729, "Low Temperature Overpressure Protection (LTOP) - Alarm Design Requirements," July 29, 1983

EE. B&W Letter P-2182, "Low Temperature Overpressure Protection," January 9, 1984

FF. B&W Letter L-576, "NRC Question 57(a) - DHR Isolation Valves," January 13, 1983

GG. Memorandum to H. J. Green and J. A. Raulston from L. M. Mills, "Bellefonte Nuclear Plant Units 1 and 2 - Instrumentation and Controls System Branch Review of BLNP FSAR," dated September 29, 1982 (A27 820929 029)

HH. Bellefonte Operating Instruction BLOI ND, Decay Heat Removal"

II. Bellefonte General Operating Instruction BLGOI-1A, "Unit Heatup from Cold Shutdown to Hot Standby and Reactor Startup"

JJ. Bellefonte General Operating Instruction BLGOI-1C, "Unit Cooldown from Hot Standby to Cold Shutdown"

TVA 55 (08-9-65)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

GNS 841017 052

TO : W. R. Brown, Project Manager (Bellefonte), 102 ESTA-K

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

DATE : OCT 17 1984

SUBJECT: BELLEFONTE NUCLEAR PLANT - MAIN STEAM SYSTEM DESIGN ADEQUACY FOR MITIGATION OF A STEAM LINE BREAK - NUCLEAR SAFETY REVIEW STAFF (NSRS) REPORT NO. R-84-25-BLN

Attached is an NSRS report concerning the adequacy of the design of the main steam system in mitigating steam line breaks. The report contains several recommendations for improving the design and/or improving the documentation of the design. NSRS is particularly concerned about apparent inconsistensies between the FSAR and the actual system design. Please provide responses to these recommendations within 30 days of the date of this memorandum. If you have any questions, please contact Bruce Siefken at extension 6860.

H. N. Culver

H. N. Culver

BFS:BJN
Attachment
cc (Attachment):
E. G. Beasley, W12B21 C-K
R. W. Cantrell, W11A9 C-K
MEDS, W5B63 C-K
H. G. Parris, 500A CST2-C
C. W. Crawford, 670 CST2-C

NSRS FILE

GNS '841017 053

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS REPORT NO. R-84-25-BLN

SUBJECT: REVIEW OF BELLEFONTE NUCLEAR PLANT'S MAIN STEAM SYSTEM DESIGN ADEQUACY FOR MITIGATION OF A STEAM LINE BREAK

REVIEWER: Bruce F. Siefken — B. F. SIEFKEN — 10-12-84 DATE

APPROVED BY: James F. Murdock — J. F. MURDOCK — 10/12/84 DATE

MEDS, W5B63 C-K

I. SCOPE

This review was restricted to a review of how the main steam system design meets the functional and licensing requirements necessary to mitigate a steam line break. Tornado-induced steam line breaks were included in the review scope. TVA internal design documents were used to define and describe the main steam system. This design was compared to regulatory requirements, licensing commitments made in the FSAR, and to interface requirements placed on the design by Babcock and Wilcox (B&W), the nuclear steam supply system (NSSS) vendor. Only safety-related or important-to-safety aspects of the main steam system were reviewed.

II. EXECUTIVE SUMMARY

The NSRS assessment of the mitigation of postulated steam breaks at BLN indicates that some design changes may be necessary in the area of preventing two-steam-generator blowdowns and in tornado protection for the main steam isolation valves (MSIVs). In particular, it appears that the present design does not meet the single failure criteria for some steam line breaks and for the design basis tornado.

Of particular concern to NSRS is the conclusion that the BLN FSAR does not accurately represent the design bases for the main steam system as is required by 10CFR50.34. The TVA internal criteria exempt the MSIVs from the single failure criteria. The FSAR, however, does not reflect this philosophy but commits to single failures after some steam line breaks. Also the MSIVs in one steam valve vault are not protected from tornado missiles. This results in the plant being vulnerable to tornadoes. Finally, there are several discrepancies between licensing commitments in the FSAR and the plant design. NSRS's overall assessment is that the present design of the main steam system may be deficient and/or that the design documentation requires improvements.

III. BACKGROUND

The design of pressurized water reactors (PWRs) has traditionally contained only one MSIV per steam line while boiling water reactors (BWRs) have had two MSIVs. In the early seventies the NRC attempted to require that Combustion Engineering (CE) place two MSIVs in each steam line as a part of CE's standard plant design. CE fought this attempt and the matter was resolved in licensing hearings. The conclusion of the hearing process was that one MSIV per steam line was judged as sufficient since the turbine stop valves (TSVs) close reliably and could be used as a backup to the MSIVs. However, it was recognized that there are several branch lines between the MSIVs and the TSVs which may not automatically isolate on a turbine trip. Therefore, the NRC required that these leakage paths be identified and quantified, and that the total leakage be shown to be acceptable. These results, however, were not documented as formal licensing requirements, but it was understood by the NRC and the utilities that these matters would be pursued only in the questions to the FSAR. In the midseventies this dichotomy was brought to NRC's upper management's attention by dissenting reviewers and resulted in considerable

internal discussion. The NRC published NUREG-0138, reference A, to establish a uniform basis for this issue and to document their justification of a single MSIV. The issue was resolved by allowing credit to be taken for valves downstream of MSIVs closing or remaining closed. These valves include the turbine stop valves, the moisture separator/reheater intercept valves, and the turbine bypass valves. The justification given for this position is based on an analysis of a main steam line break and whether the mitigating equipment needs to be seismically qualified. The staff concluded that the use of the nonsafety grade equipment as a backup to safety equipment was acceptable for a main steam line break at a PWR. This position was supported by a simplified analysis of a steam line break which showed that the consequences of a two-steam-generator blowdown were not as severe as those which result from a large LOCA. Therefore, less strict quality standards could be applied to the mitigation of a steam line break as allowed by Criterion I of Appendix A to 10CFR50. The position was further supported by the NRC staff's estimation that the probability of a main steam line break, an earthquake, and the failure of an MSIV to close was low; and thus the overall safety of the plant would not be strongly affected by the occurrence of the postulated scenario. The possibility of tornado-induced steam line breaks was not considered in NUREG-0138. The NRC has revised their Standard Review Plan, references B and C, to allow the use of nonsafety-grade equipment as a backup to safety-grade equipment in mitigating a main steam line break. Thus the licensing requirements have become more formalized as a result of the internal NRC dissent.

In reviewing FSAR submittals the NRC has been using the licensing position described above. As part of this review, the NRC has asked TVA to provide information concerning BLN flow paths between the MSIVs and the turbine stop valves (see FSAR question 430.67, reference D). This question requests a list of flow paths, flow rates, valves, and their method of closure in order for the NRC to evaluate how TVA complies with issue number 1 of NUREG-0138, the use of turbine stop valves as a backup to the MSIVs. The total flow rate from all the unisolated paths between the unaffected steam generator and the turbine stop valves has to be shown to be acceptable if the stop valves are to be used as a backup to the MSIVs after a steam line break. The NRC also required the Yellow Creek Nuclear Plant (YCN) to assume the failure of an MSIV as evidenced by their questions on the YCN PSAR (references E, F, and G). Pebble Springs Nuclear Plant also was required to postulate the failure of an MSIV. Thus it was NRC's pattern and practice to require that applicants postulate the failure of an MSIV.

IV. CONCLUSIONS/RECOMMENDATIONS

A. NSRS concludes that TVA has not met the requirements of 10CFR50.34 inasmuch as the design bases for the main steam system which have been implemented are not accurately described by the BLN FSAR. Specifically, TVA does not consider failure of the main steam isolation valves to be credible, as stated in the design criteria for the main steam system, since the valves have redundant closure signals. However, the BLN FSAR does consider

single failures of MSIVs. Therefore, the FSAR conflicts with TVA's internal design criteria.

R-84-25-BLN-01. NSRS recommends that TVA resolve the conflict between the FSAR and the design criteria and document the resolution in the design criteria or FSAR as appropriate.

B. NSRS concludes that the present main steam system design does not meet regulatory requirements in the following area. NUREG-0138, issue 1, allows credit for the turbine stop valves in preventing a two-steam-generator blowdown after a steam line break and the failure of an MSIV. However, the design arrangement at BLN is such that in the event of a break, as postulated in NUREG-0138, the closure of the turbine stop valves will not terminate the two steam generator blowdown since the steam lines are cross-connected by a 42-inch-diameter header upstream of the stop valves. Furthermore, the NRC in FSAR question 430.67 asked TVA to provide additional information in the FSAR concerning all flow paths between the MSIVs and the TSVs. In TVA's response to this request, the 42-inch-diameter crosstie header between the main steam lines was omitted. NSRS concludes that the answer to FSAR question 430.67 is not complete.

R-84-25-BLN-02. NSRS recommends that design modifications be made to provide redundancy in the isolation of steam lines as required by NUREG-0138 or that conservative analysis of a two-steam-generator blowdown event be completed which shows that the consequences are acceptable. Additionally NSRS recommends that the response to FSAR question 430.67 be amended to more accurately reflect the design of BLN.

C. NSRS concludes that the present design of the main steam system does not comply with commitments made in the BLN FSAR in the following areas:

1. Section 3.6.2.1.2.1.5 of the FSAR states that pipe breaks outside of the main steam valve vaults do not jeopardize equipment within the main steam vaults. However, report CEB77-10 states that break SM B-94 may damage the "A" steam generator MSIVs.

   R-84-25-BLN-03. NSRS recommends that the design of BLN be modified to prevent all steam line breaks outside of the valve vault from damaging an MSIV or demonstrate that the break does not damage the MSIVs.

2. Section 3.6.2.1.2.1.5 of the BLN FSAR states that the pipe break exclusion applies only to the 32-inch-diameter main steam piping and to the 20- and 22-inch-diameter main feedwater piping. However, CEB report CEB 77-10 does not postulate any breaks in any size piping within the main steam valve vaults.

R-84-25-BLN-04. NSRS recommends that the inconsistencies in BLN design be corrected and that the commitments in FSAR section 3.6.2.1.2.1.5 and CEB 76-13 be fully met.

3. FSAR section 3.5.2 states that if tornado missles damage the main steam lines and isolation valves in the unprotected "A" steam valve vault, that the plant can be cooled down as described in FSAR section 15.1.5 with single failure capability. However, if this described scenario occurred and an MSIV in the "B" main steam valve vault were postulated to fail open, then a two-steam-generator blowdown would occur which the turbine stop valves would not be able to terminate. The BLN FSAR states in section 10.3 that the design of the main steam system prevents the uncontrolled blowdown of two steam generators after a steam line break and a postulated single failure.

   R-84-25-BLN-05. NSRS recommends that the design of the main steam system or the main steam valve vaults be modified to comply with the requirements of FSAR sections 3.5.2, 10.3, and 15.1.5 or that an analysis of a two-steam-generator blowdown event be made which shows that the consequences are acceptable.

D. The maximum allowable steam flow rate from the unaffected steam generator after a steam line break needs to be documented in the design bases for the main steam system. NSRS was unable to find evidence of these considerations by the NSSS vendor in the design bases for the main steam system.

   R-84-25-BLN-06. NSRS recommends that the maximum allowable steam flow rate from the unaffected steam generator after a steam line break be established by the NSSS vendor based on the analysis of such steam line breaks and that this flow and its basis be documented in the design bases for BLN and be traceable to the NSSS vendor's analysis.

V. DETAILS

A. Main Steam System Description

The BLN main steam system connects the steam generators with the turbine generator. The design includes four main steam lines, two from each steam generator, one main steam isolation valve (MSIV) per line, a 42-inch header cross-connecting the steam lines, eight turbine stop and control valves, and several connections for a variety of purposes. The conceptual design of the system is depicted on the design criteria diagram, main and reheat steam system, reference H. The steam lines run from the steam generators into the main steam valve vaults, one vault for each steam generator. The valve vaults are designated "A" and "B" and contain the steam lines from the "A" and "B" steam generators respectively. Reference I shows the steam line routing from the steam generators to the valve vaults. Inside

each valve vault, there are several lines and valves in the main steam system including the following:

1. Eleven main steam safety valves.
2. One MSIV per steam line.
3. Steam supply line and isolation valves to the auxiliary feedwater turbine.
4. Modulating atmospheric dump valve and bypass.
5. MSIV bypass line and valve.
6. Line to startup and recirculation system and isolation valve.

The main feedwater lines also pass through the main steam valve vaults.

The main steam valve vaults are located on opposite sides of the containment structure on the 0° and 180° azimuths. The "A" vault is located on the 0° azimuth and the "B" vault lies on the 180° azimuth as shown in reference J. The "A" steam valve vault is physically a part of the auxiliary building. Three walks and the floor are adjacent to auxiliary building areas while the fourth wall adjoins the secondary containment structure. Only the roof of the "A" steam valve vault is an exterior surface. The "B" main steam vault has a common wall with the secondary containment structure. The other walls and the roof are exterior surfaces. The steam valve vaults are required to be designed to withstand the effects of a steam line or feedwater line break within the vault. This requirement is met by providing vent areas to relieve the resulting pressures after such a break. The "A" steam valve vault relief area consists of a large blowout area on the valve vault roof covered by a number of panels which blow out at a pressure of 0.5 psig. There is no tornado missile protection for these panels. The "B" steam valve vault has all of its walls designed for 30 psig (including the roof) and is tornado missile proof. However, one wall was designed with a safety factor of 1.5 while the rest of the structure used a value of 2.0. Thus it is felt that the weaker wall will relieve first before the remaining structure is damaged.

After exiting the valve vaults, the main steam lines are routed over the roofs of the auxiliary and control buildings enroute to the turbine building. Once inside the turbine building, the steam lines from each steam generator are joined by a common pipe and then these two pipes are cross-connected by a common 42-inch-diameter header. The four lines to the high pressure turbine originate at this common header with each line containing a turbine stop and a control valve. Various miscellaneous lines are located upstream of the turbine stop and control valves including:

1. An 8-inch line to the main feedwater turbines.
2. Two 12-inch lines to the moisture separator/reheaters.
3. A 20-inch line to the turbine overload stop and control valves.

4. A 6-inch connection to the auxiliary steam system.
5. A 30-inch line to the turbine bypass valves.

B. TVA Design Criteria for the Main Steam System

The design requirements which EN DES placed on the main steam system are contained in the detailed design criteria for main steam, reference K. The design criteria references a number of other documents which place requirements on the design. B&W's system description for secondary system, reference L, is the principal referenced document and establishes those interface requirements for the main steam necessary to achieve compatibility with the nuclear steam supply system (NSSS) design. The TVA design criteria document addresses the issue of a single MSIV. The design basis is that the MSIV will be considered exempt from the single failure criteria provided the valve design meets several requirements, namely:

1. The valve shall fail closed on loss of electrical power or air pressure.

2. Each MSIV shall be served by two separate sets of engineered safety features actuation system (ESFAS) digital actuation channels (i.e. "A" and "B" channels) arranged in a 1-of-2 logic to close the MSIV.

3. The design shall be capable of closing the MSIV after a single failure in an instrumentation channel or power supply.

The design criteria specify that the piping layout reserve space such that a second MSIV could be placed downstream of the existing MSIV and that these valves be shown as future on TVA design drawings in the FSAR. The design criteria also specifies that for all normally open valves downstream of the MSIVs which do not automatically close on a turbine trip, the flow shall not exceed 6 percent maximum steam flow. If this condition is not met, then the design criteria states that measures shall be taken to restrict the flow to the 6 percent limit.

The design criteria also specifies nonsafety-related arrangement requirements. These include the steam line header arrangement described in section II above and the various lines which emanate from the main steam header.

The system description for the secondary system contains some requirements which affect the system arrangement, including:

1. The steam line pressure drop from each steam generator to the turbine stop valves shall be such that the outlet pressures of the steam generators are equal within 5 psi under valve wide open (VWO) flow conditions.

2. A connection between the main steam lines from the two steam generators shall be supplied to provide a minimum cross flow from one steam line to the other of 30 percent of the total turbine VWO flow while one of the steam generators is isolated.

C. NRC Requirements

The NRC requirements which the design of the main steam system must meet are contained in a number of documents. The Standard Review Plan (SRP) serves as a guide for the review of licensing applications by the NRC and contains or references the basic regulatory requirements which various systems should meet. Section 10.3 of the SRP concerns the review of the main steam system and references NUREG-0138 for the acceptability of taking credit for valves downstream of the MSIV to limit blowdown of a second steam generator in the event of a steam line break upstream of an MSIV. NUREG-0138 was discussed in the Background section of this report. Section 15.1.5 of the SRP discusses the review of postulated steam line break accidents and requires consideration of single active failures which might result in more than one steam generator blowing down. This implements the requirements of Criterion 34 of Appendix A to 10CFR50 which requires that systems used to remove residual heat have suitable redundancy in isolation provisions to accomplish their safety functions assuming a single failure. Appendix A to 10CFR50 defines a single failure of a fluid system component as the failure of any active component which results in the loss of its capability to perform its intended safety function. Therefore NSRS concludes that the NRC requires that the failure of an MSIV to close be included in the analysis of steam line breaks.

D. BLN FSAR

The BLN FSAR discusses the design of the main steam system and steam line breaks in four locations--chapters 3, 6, 10, and 15. Section 3.6 discusses pipe break postulation and evaluation, section 6.2 discusses steam line breaks as related to the design of primary containment, section 10.3 discusses the design of the main steam system, and section 15.1 discusses the analysis of steam line breaks. The NRC has also asked several questions concerning steam line breaks at BLN.

Section 6.2.1.4 presents the analysis of steam line breaks inside containment and the predicted pressure and temperature profiles which result. Table 6.2.1-27 lists the steam line breaks which were analyzed for containment response and shows that four of the ten cases assume the failure of an MSIV.

Section 10.3.1 presents the design bases of the main steam system and delineates the major acceptance criteria for a main steam line break. This section states that for postulated steam line breaks, as described in FSAR section 3.6, with postulated single failures that uncontrolled flow from more than one steam generator and a reactivity transient beyond the postulated load

in FSAR chapter 15 are prevented. Section 3.6 describes the manner in which pipe breaks are postulated and how their effects are analyzed. This section defines the main steam system as a high energy system for which circumferential breaks in the piping are postulated and gives the criteria for selecting break locations. The FSAR excludes breaks of the main steam and feedwater piping in the main steam valve vaults which is larger than 20 inches in diameter. NRC Branch Technical Position MEB 3-1 (included as part of the Standard Review Plan) allows the exclusion of breaks in piping up to the outermost containment isolation valve. The FSAR also excludes the piping downstream of the isolation valve (MSIV) that is within the valve vault. This expanded exclusion area has not been questioned by the NRC to date. The break exclusion criteria of MEB 3-1 are to be applied to the additional piping run. TVA has sent the NRC a copy of report CEB 76-13, reference U, which documents TVA's plan for implementing APCSB 3-1 and MEB 3-1. Section 10.0 of this report states that provisions have been made to accept postulated breaks in 12-inch-or-less piping inside the valve rooms. Report CEB 77-10, reference S, has been sent to the NRC to amplify TVA's pipe break evaluation. This report, however, does not contain any postulated pipe breaks inside the valve vaults, even though there are several lines which are outside of the FSAR exclusion, i.e., smaller than 20 inches in diameter. MEB 3-1 criteria may have been applied to these smaller lines however. Further, this report indicates that one analyzed break, number SM B-94, which occurs at the "A" valve vault roof may cause possible damage to the MSIVs. The report indicates that this is an acceptable interaction based on the reasoning that although structural damage occurs which may damage the MSIVs, they are not needed to safely shut down the plant. This reasoning is based on the "B" steam generator MSIVs closing which violates the NRC's criteria to assume single failures that could lead to two-steam-generator blowdowns. Since the postulated break was considered to be adequately mitigated, no corrective actions were proposed by the report.

The NRC asked FSAR question 430.67 to obtain additional information regarding compliance to NUREG-0138, issue 1, on providing redundancy to the MSIVs. This question requested information concerning all flow paths that branch off the steam lines between the MSIVs and the TSVs, and that this information be documented in the FSAR. TVA's response is contained in Table 430.67-1. However, this table does not contain the 42-inch crosstie header which connects the steam lines from the two steam generators. At present, there are no valves in this crosstie. Thus, there is no redundant backup capability to the MSIVs. The NRC requires this redundancy in the design, but they do not require that the backup for the MSIVs be designed to safety-related standards. NUREG-0138 contains the justification for allowing nonsafety-related components as redundancy for the MSIVs. It is not a justification for the lack of redundancy.

E. Design of Other B&W Plants

A brief look at other nuclear plants with B&W-supplied reactors was undertaken to determine if the lack of steam line isolation redundancy is an industrywide problem. There were 14 B&W plants which were docketed with the NRC. Six of these plants were reviewed to determine if redundancy exists for isolating steam lines. The FSAR for these plants, section 10.3, was the basis for the review, and the results are summarized below.

1. Rancho Seco and Crystal River 3 - These plants are both very early B&W designs announced in 1967. They do not meet the requirements of NUREG-0138, but they were essentially built before this NUREG was published in 1976.

2. Three Mile Island 2 - This plant was docketed in 1968 by the NRC and has redundancy in its isolation capabilities since the steam lines are crosstied downstream of the turbine stop valves.

3. Davis Bessie - This plant has unidirectional MSIVs and non-return valves in the steam lines which provide redundancy in the isolation. The plant was docketed in 1968.

4. Midland - This plant was also docketed in 1968 and has two isolation valves per steam line. Only one valve is designated the MSIV, but both valves are shown in FSAR figure 10.3-1.

5. WPPS-1 - This plant was docketed in 1973, the same year BLN was docketed. This plant does not have redundant isolation provisions.

Thus, of the six plants reviewed, two were designed before the issuance of NUREG-0138, three meet or exceed the requirements for redundancy, and one plant does not have redundant steam line isolation capability.

F. Problem Areas

After reviewing the main steam design, its design criteria, the BLN FSAR, and regulatory guidance, NSRS has the following concerns:

1. TVA has not consistently documented the design of the main steam system to the NRC in the area of MSIV failures. The internal design policy is that the MSIVs are exempt from the single failure criteria since there are redundant signals and solenoids to close the valves. However, this position has not been provided to the NRC. The BLN FSAR contains several references to the failure of an MSIV after a steam line break. The TVA response to FSAR question 430.67 is misleading in that it implies that TVA is in compliance with NUREG-0138, R1. The postulated break SM B-94 of CEB77-10 violates the NUREG position in that the break damages the

"A" steam valve vault MSIVs, and they cannot be relied upon to close. A postulated single failure of an MSIV in the "B" valve vault would then result in a two-steam-generator blowdown. Furthermore, the closing of the turbine stop valves would not, in this instance, provide backup since the 42-inch steam header cross-connects the steam generators upstream of these valves. Thus the TVA design does not meet the requirements of NUREG-0138, while it does meet the internally established criteria. 10CFR50.34 requires that the design bases for the plant be included in the license application.

2. The provisions of BLN FSAR section 3.6.2.1.2.1.5 are not met by the present BLN design. These sections require that pipe breaks outside of the break-exclusion area (i.e. not in the steam valve vault) are not allowed to affect the operability of the MSIVs. However, CEB77-10 states for break SM B-94 that these valves may be damaged.

4. The BLN FSAR states that the pipe break exclusion applies to only the 30-inch main steam piping and to the 20- and 22-inch main feedwater piping within the steam valve vaults. TVA report CEB77-10, however, does not analyze breaks of any piping in the main steam valve vaults; and report CEB 76-13 states that measures have been taken to accept postulated failures of this piping.

5. In order to take credit for the closure of the turbine stop valves as a backup to the MSIVs, one must calculate the expected flow from all unisolated flow paths from the main steam lines. This flow must then be compared to the maximum allowable flow from an isolated steam generator after a main steam line break. The maximum allowable flow rate is usually determined by the NSSS vendor and should be used as input to the accident analysis of a steam line break. Allowances for a stuck open atmospheric dump valve or safety valve are also considered in establishing the maximum allowable flow. NSRS has been unsuccessful in finding this flow rate in the NSSS vendor design documentation.

6. The roof of the "A" main steam valve vault is not protected from tornado missiles. The BLN FSAR states in section 3.5.2, reference T, that if tornado missiles damage main steam lines or MSIVs in the "A" steam valve vault, that the plant is designed to mitigate this event as described in FSAR section 15.1.5 with single failure capability. This section refers to the possibility of an MSIV failing to close. The NRC specifically questioned TVA about such tornado missile-induced breaks in question 410.11. The response to this question does not mention the possibility of a single failure. If one postulates that tornado-generated missiles disable an MSIV and break a steam line in the unprotected "A" steam valve vault, then the failure of an MSIV in the "B" steam valve vault would result in a two-steam-generator blowdown which the closure of the turbine stop valves would not terminate.

VI. REFERENCES

A. NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director NRR to NRR Staff," November 1976

B. NRC's Standard Review Plan, Section 10.3, "Main Steam Supply System," R3, April 1984

C. NRC's Standard Review Plan, Section 15.1.5, "Steam System Piping Failures Inside and Outside of Containment (PWR)," R2, July 1981

D. BLN FSAR Question 430.67 and TVA's response submitted in FSAR Amendment 21, July 1, 1981

E. YCN PSAR Question 022.1 and TVA's response submitted in PSAR Amendment 1, August 20, 1976

F. YCN PSAR Question 022.27 and TVA's response submitted in PSAR Amendment 2, November 1976

G. YCN PSAR Question 022.46 and TVA's response submitted in PSAR Amendment 11, March 31, 1978

H. BLN Design Criteria Diagram Main and Reheat Steam System, 3BW0600-SM, R13

I. BLN Mechanical Composite Piping Drawing 3BW0303-00

J. BLN Equipment Plans - Roof Drawing 3BW0200-00

K. TVA Detailed Design Criteria for Main Steam (Including Secondary Side of Steam Generator), N4-SM-D740

L. B&W's System Description Specification for Secondary System, 15-4044000002

M. Title 10, Code of Federal Regulations, Part 50, Appendix A

N. BLN FSAR Section 3.6, "Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping"

O. BLN FSAR, Section 6.2.1.4, "Containment Analysis for Postulated Secondary Pipe Ruptures"

P. BLN FSAR, Section 10.3, "Main Steam Systems"

Q. BLN FSAR, Section 15.0.2, "Single Failure Philosophy"

R. BLN FSAR, Section 15.1.5, "Steam Line Break"

S. Civil Engineering Branch Report CEB 77-10, "Bellefonte Nuclear Plants Units 1 and 2, Evaluation of the Effects of Postulated Pipe Ruptures Outside Containment"

T. BLN FSAR, Section 3.5, "Systems to Be Protected" (from missiles).

U. Civil Engineering Branch Report CEB 76-13, "Break Exclusion Position for Complying with APCSB 3-1 and MEB 3-1, Bellefonte Nuclear Plant Units 1 and 2," July 8, 1976, Revision 2

GNS '841109 051

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS REPORT NO. I-84-31-BFN

SUBJECT: INVESTIGATION OF BROWNS FERRY NUCLEAR PLANT UNIT 3 STARTUP FROM COLD CONDITION WITH RHR IN SHUTDOWN COOLING MODE ON OCTOBER 22, 1984

DATES OF REVIEW: OCTOBER 22 - NOVEMBER 2, 1984

REVIEWER: K. W. WHITT — DATE 11/9/84

APPROVED BY: M. S. KIDD — DATE 11/9/84

MEDS, W5D83 C-K

TABLE OF CONTENTS

I. INTRODUCTION

On October 22, 1984, unit 3 of Browns Ferry Nuclear Plant (BFN) was brought critical for the stated purpose of performing a full power shutdown margin test in accordance with Refueling Test Instruction RTI-4. The rods were withdrawn and the reactor made critical with one Residual Heat Removal (RHR) loop in service in the shutdown cooling mode. One loop of the recirculation system was in service and one loop was not in service. Rod withdrawal with the reactor in the stated configuration was authorized by plant management. Almost all plant personnel with which this condition was discussed stated that they understood that rods were withdrawn and criticality achieved for the purpose of performing a test and not for the purpose of plant startup.

Preparations for startup had been performed in accordance with Standard Practice (SP) BF-12.18, "Unit Prestartup Review." The purpose of completing the SP was to assure that unit 3 was safe to return to service following a refueling outage. The SP contained the responsibilities and designated the constituents of the prestartup review committee. The prestartup review committee was responsible for assuring that all requirements were met for startup of unit 3. It also established or provided methodology for establishing subcommittees; meeting frequency; schedule for formulation of the committee and commencement of duties; and maintenance, distribution, and content of minutes.

All section supervisors or a substitute had signed off on the SP indicating that their areas of responsibility had no outstanding items that would adversely affect a safe unit startup. The Plant Operations Review Committee had reviewed the SP and the Plant Manager had approved it indicating that all structures, systems and components necessary for safe startup of unit 3 were sufficiently serviceable to perform their intended function.

Since the reactor had been shutdown for an extended period, there was not a sufficient amount of decay heat to raise the moderator temperature to the 180°F which was required by Technical Specifications prior to effecting criticality. For this reason one of the decay heat loops was placed in service with one pump operating and no coolant being provided to the decay heat exchangers. In this mode, heat generated by the RHR pump was added to the reactor moderator. This was the method used to heat the moderator to the required 180°F. Since the RHR system takes suction from the recirculation system and discharges to the recirculation system, one of the recirculation loops was out of service and water was entering the reactor vessel in the reverse direction from normal flow during recirculation system operation.

The coordinating document being used was the Master Refueling Test Instruction (MRTI). This document specified that the reactor be taken critical in the sequence specified by RTI-4, per General Operating Instruction GOI-100-1, sections II.B and C. RTI-4 was the refueling

test instruction that provided instructions for performing the shutdown margin test. GOI-100-1 was the general operating instruction that provided instructions for integrated plant operations including reactor startup. The reactor was made critical and the shutdown margin test was performed. The total time the reactor remained critical was approximately 16 minutes. It was estimated by Operations personnel and a nuclear engineer that the maximum power attained was 0.5 percent of rated power. When the test was completed and the necessary data obtained, the reactor control rods were inserted and the reactor was placed in cold shutdown. If all this work had been performed in full compliance with regulatory requirements, the reactor would, at this point, have been ready for a normal or routine startup and power operation. However, a number of questions regarding possible noncompliance developed as a result of NRC personnel interest and further plant personnel evaluation. These questions included:

1. Were the prestartup preparations performed in accordance with SP BF-12.18 adequate to assure safe plant startup? This question arose because of a number of equipment deficiencies identified by plant personnel and the NRC during and/or following the performance of the shutdown margin test.

2. Were adequate instructions provided for the performance of the shutdown margin test and was the test conducted in accordance with the regulatory requirements?

3. Were adequate program and management controls in effect during the conduct of the test and related reactor manipulations? This question arose because of the potential implications of questions 1 and 2.

NSRS was informed of the event on October 25 and started an investigation the same day. The Office of Nuclear Power (NUC PR) appointed an internal review team on October 25 to review the event. This team arrived on site and started its review on October 26, 1984.

II. SCOPE

This investigation was conducted to determine the circumstances relating to the October 22, 1984 startup with RHR in the shutdown mode. The startup was made for the purpose of performing a full core shutdown margin test at the critical condition. The review included an assessment of the causes of the event and the adequacy of corrective action.

III. CONCLUSIONS AND RECOMMENDATIONS

A. The internal review team assigned by the Manager of NUC PR was a well qualified and diversified team and did a thorough and objective review of the October 22, 1984 event (see section IV.E.1 for details).

Recommendation

I-84-31-BFN-01 - NUC PR should establish an ongoing program to provide internal investigations of potential significant events at TVA nuclear plants.

B. A number of Technical Specifications were violated during the event. Some of these Technical Specifications were perceived by the line organization to be of the "gray area" type that could have been interpreted differently with some justification. Under the present regulatory environment at BFN the degree of conservatism being practiced by the plant staff in the interpretation of regulatory requirements is not sufficient to achieve and maintain a satisfactory regulatory performance record (see sections IV.B.1, .2, and .4 and IV.E.2 for details).

Recommendations

I-84-31-BFN-02 - NUC PR management should identify those areas of regulatory requirements, including Technical Specifications, that are considered to represent intrepretation problems, communicate these identified problems to the NRC for assistance in intrepretation, and then provide clarification in the control documents for all operating plants. Operator training should be modified to include specific attention to the changes resulting from this effort.

I-84-31-BFN-03 - NUC PR should establish a policy and procedures for the interpretation of regulations for plant operation under temporary Technical Specification modifications, waivers, or exceptions.

C. During the event, there were a number of procedure violations. Plant personnel attempted to justify and defend the violations. Little effort was made to show the reviewer that procedural controls were of paramount importance. This indicates that NUC PR has not made a total commitment to procedural controls and procedural adherence (see sections IV.A, IV.B.3, and IV.E.2 for details).

Recommendations

I-84-31-BFN-04 - NUC PR management should establish a written policy to ensure that the necessary commitment is made by all personnel to procedural controls over nuclear activity and to assure that the procedures are strictly adhered to. As a part of management controls a method should be established to assure the policy is implemented and its effectiveness is measured.

D. Procedures that existed at the time of the event were marginally adequate to assure a safe startup if they had been strictly followed.

E. The NUC PR evaluation team report contains several good specific conclusions and recommendations relative to corrective action and prevention control. NSRS concurs with them, and it is not considered necessary to repeat them in this report.

IV. DETAILS

A. Prestartup Preparations

Two areas of prestartup preparations were evaluated. One area was adequacy of preparations performed in accordance with SP BF-12.18. The methodology used in SP BF-12.18 provides documented evidence that the physical plant is ready for a safe startup following a refueling and maintenance outage. Upon completion of SP BF-12.18, plant management should have a high level of confidence that all systems and components necessary to conduct a safe startup are ready for service.

The second area examined was the existence of equipment deficiencies identified subsequent to completion of SP BF-12.18. In this case (unit 3 startup preparation), a number of equipment-type potential deficiencies were identified by plant and NRC personnel during and immediately following the startup to criticality and shutdown margin test of October 22, 1984. If the prestartup preparations in accordance with SP BF-12.18 were sufficient, then serious equipment or material problems should not have been found that would have a significant adverse impact on the safety of reactor startup and operation.

A discussion of the evaluation of each of these areas is provided in the following paragraphs:

1. SP BF-12.18 Activities

This standard practice provides a structured method for tracking and expediting a variety of functional area activities to approach a consolidated endpoint assurance of startup readiness. It establishes a committee representing all the organizational sections of the plant and specifies subcommittee formation and activities.

The prestartup review committee activities performed in preparation for startup of unit 3 were evaluated through the review of meeting minutes, review of the procedure (12.18) administration, and discussions with participating personnel. It was determined that a good effort was made by plant management and all participating employees in getting the variety of elements pulled together into an integrated state of preparedness. The effort resulted in a high degree of resolution of outstanding items and a plant well prepared for safe startup. It appeared that outage personnel had done a particularly good job in getting work packages in an

acceptable condition to support startup. The official and unofficial tracking and expediting methods used by Operations also seemed to be effective in getting plant systems in the desired operational status.

While the actual work performed by the prestartup committee was admirable, procedural practices appeared to be less than desirable. There appeared to be a lack of attention to keeping SP BF-12.18 updated and current. Some committee members indicated that they were not always altogether sure who the official chairman was. They were sure that the performing chairman was not always the Compliance representative as specified by the governing standard practice. The Planning and Scheduling Section had been designated to chair the committee in July 1984, but the procedure had not been changed to reflect this designation. A liberal policy for designating substitutes for signoff indicating satisfactory completion of activities on Appendix B to SP BF-12.18 was also used.

2. Potential Equipment Deficiencies

A total list of 14 potential deficiencies identified primarily by the NRC resident inspector was compiled by the reviewer.

a. The drywell equipment hatch trolley cranks were not padlocked when observed by an NRC resident inspector on October 22, 1984. However, GOI-100-1 had been signed off indicating that the trolley cranks had been locked.

b. Incorrect readings on HPCI/RCIC flow recorder (FR-71-36) on backup control panel. This was observed by a resident inspector on October 22, 1984.

c. Jet pumps not proven operable prior to startup on October 22, 1984. This was identified as a potential Technical Specification violation by a resident inspector.

d. RHR not operable in the low pressure coolant injection (LPCI) mode during startup. This was identified as a potential Technical Specification violation by a resident inspector.

e. Drywell sump level instrument switch (LIS-77-1A) was in the "off" position. This was identified by a resident inspector on October 22, 1984.

f. Temporary gauge installed on condensate storage supply line to HPCI suction. No temporary alteration control form (TACF) currently exists for the gauge. This was identified by a resident inspector on October 22, 1984.

g. Drywell H_2O_2 analyzer has no flow but pressure reads 5 pounds. This was identified by a resident inspector on October 22, 1984.

h. Signoffs not made in GOI-100-1 for two valve checklists. This was identified by a resident inspector on October 22, 1984.

i. MRTI was not followed. No operational approval. No project manager approval. This was identified by a resident inspector on October 22, 1984.

j. The nuclear engineer appears to be in charge of taking the reactor critical. (Reference to MRTI step 30.) This was identified by a resident inspector on October 22, 1984.

k. A resident inspector indicated on October 27, 1984 that he was concerned about the technical analysis involving T-52 (Jet Pump Failure Detection).

l. RCIC steam flow instrument in control room (FI-71-1A and -1B) has four outstanding MRs. This was identifed by a resident inspector on October 27, 1984.

m. Because of recent changes to the pull sheets there is some concern about how rod coupling integrity checks are performed and documented.

n. The issue of gamma exposure from N^{16} with the reactor critical and RHR in shutdown cooling mode was identified as a concern by a resident inspector on October 22, 1984.

Six of the above 14 items were considered to relate directly to possible physical equipment deficiencies. Five of these six were evaluated as a part of this investigation and a brief assessment of each is presented in the following paragraphs:

Item a, Drywell Equipment Hatch Trolley Cranks

This was a correct assessment. The person that signed off the prerequisite in GOI-100-1 failed to verify all items listed in the prerequisite. This item was required to be performed in accordance with MMI-95. However, MMI-95 did not cover padlocking of the trolley hatch cranks. The responsible individual was required to go back and verify all information contained in the prerequisite. Additional disciplinary action was also administered.

Item b, Incorrect Reading on HPCI/RCIC Flow Recorder (FR-71-3G)

The recorder was malfunctioning. A maintenance request (MR) was prepared and the instrument was repaired on October 23, 1984. The primary cause of the problem was a failed capacitor. The recorder is required to be calibrated each refueling outage. The scheduled calibration was performed in September 1984. It was again calibrated on October 23, 1984 following repair.

Item e, Drywell Sump Level Instrument Switch (LIS-77-1A) in "Off" Position

This switch was in the "off" position according to the information obtained by the reviewer. This instrument automatically initiates the drywell floor drain sump pump at a preset sump level. There are two instruments associated with the sump level control system (LIS-77-1A and LIS-77-1B). Each time the sump level rises to the high level setpoint, one of the pumps is automatically started by one of the instruments. When the sump level is lowered to a preset level, the pump automatically stops and an auxiliary relay aligns to allow starting the other pump the next time the sump level reaches the high level setpoint. With LIS-77-1A switch in the "off" position, "A" pump would not start when the sump level reached the high level setpoint. When the sump level increased an additional six inches to the hi-hi level setpoint, "B" pump would automatically start and pump down the sump. Upon initiation of "B" pump on hi-hi sump level an alarm would sound in the control room indicating an abnormal condition.

Item f, Temporary Gauge Installed on Condensate Supply Line to HPCI Suction

The gauge was installed on the condensate supply line as stated. The gauge was probably installed in 1979 during a hydrostatic test of the condensate supply line following repair of the line. It had apparently been left and forgotten about. The gauge and connecting flange was of a higher quality than the condensate line and created no safety problem. At the time the gauge was installed, TACFs were not used for nonelectrical temporary alterations. Therefore, there was no formal system to track mechanical temporary alterations.

Item i, RCIC Steam Flow Instrument in Control Room (FI-71-1A and -1B) Has Four Outstanding MRs

The indicated instruments measure the RCIC steam flow during periods of injection. When there is no flow one would expect the flow indication to be zero. This is the case

when the reactor is in operation with steam in the RCIC steam supply line. However, when the reactor is shutdown and the RCIC steam supply line contains no steam and is at atmospheric pressure, the instruments may not show zero flow. The indication of the instruments has no defined meaning. This is because of the design of the flow mechanism. A 90° vertically installed elbow is used for flow measurement. One tap is taken from the top of the elbow and one from the bottom. When there is no steam pressure in the steam line, the water drains from the upper tap creating an unequalized condition between the two taps and an invalid reading on the instruments. Each time a surveillance test is run on the instruments and the flow indications are outside specified criteria, an MR is required to be written. When the reactor is shutdown and depressurized, the flow indication will usually be out of tolerance. This is the reason for the MRs. The SI should probably be revised so that MRs are not generated during outages. No safety problem exists with respect to the flow instruments since the only time the perceived problem exists is when the RCIC is not required or needed to be operable. The instruments work satisfactorily when required.

As can be seen from the above evaluations, items a, b, e, and f represent actual deviations from the normal and expected condition. Item l does not appear to be a problem of any type. The condition identified for item l has existed throughout the operating life of the plant and is well understood. Some change to the SI is warranted to eliminate the problem identified by the NRC inspector. Items a, b, and e could have had some impact on equipment operation. However, taken individually or collectively, no significant safety problem was considered to have existed. The deficiencies were not considered to be surprising or unexpected following an extended outage of the type unit 3 had gone through. Item f had no impact upon equipment operation but did represent an example of past practices where temporary alterations had not been controlled. Actions being taken as a part of RPIP should identify and correct these situations as the systems are walked down. The above identified five items do not appear to reflect any inadequacy in the preparation made for startup in accordance with SP BF-12.18.

An additional 6 of the list of 14 items which do not relate directly to equipment problems are addressed in other sections of this report. Three items were not evaluated by NSRS except in a very general way (g, h, and n).

B. Reactor Manipulations

On October 22, 1984, unit 3 of BFN was brought critical following an extended refueling and maintenance outage. The reactor was started up for the purpose of performing a shutdown margin test

in accordance with RT1-4, "Full Core Shutdown Margin." The startup was conducted in accordance with GOI-100-1, "Integrated Plant Operations," as specified by the MRTI. The test was performed to provide assurance that the reactor would be subcritical with the most reactive rod fully withdrawn and all other rods fully inserted. Performance of this test required that the most reactive rod be withdrawn for criticality. In this case, the most reactive rod was one of those designated as "B" rods. The reactor is normally brought critical with the "A" rods. The shutdown margin test is performed at criticality and then startup is normally continued on into power operations. However, since the most reactive rod was among the "B" rods, criticality had to be obtained for the first time at BFN on the "B" rod configuration. It was still preferred to startup and go to power on the "A" rod configuration. Therefore, the decision was made to go critical on the "B" rods, perform the shutdown margin test at criticality, shutdown, and then startup again for power operation with the "A" rods.

Due to the long outage, there was insufficient core heat to raise the moderator temperature up to the 180°F required by Technical Specifications for criticality. For this reason, one loop of the RHR system was in service in the shutdown cooling mode with no secondary coolant to the RHR heat exchangers. In this way heat from the RHR pump was used to heat up the moderator temperature to the desired level. The RHR loop was left in the shutdown cooling mode while the reactor was brought critical. This RHR configuration also required one loop of the recirculation system to be out of service. Reactor startup in the configuration described above appeared to be in violation of at least one and possibly two Technical Specifications. A number of procedural violations also seemed apparent. An additional potential violation of Technical Specifications which was unrelated to the systems lineup dealt with the apparent failure to verify the correctness of the rod withdrawal sequence. A discussion of the apparent Technical Specification and procedural deficiencies is provided below.

1. <u>RHR System Aligned in Shutdown Cooling Mode</u>

   The startup of the reactor with the RHR system lined up in the shutdown cooling mode is in violation of Technical Specification 3.5.B.1(1) which requires the LPCI mode of RHR to be operable prior to a reactor startup from a cold condition. The LPCI would not have been available without operator action. Therefore, the LPCI mode of RHR was considered to be inoperable.

2. <u>Jet Pump Operability</u>

   Technical Specification 3.6.E.1 states that "Whenever the reactor is in the startup or run mode, all jet pumps shall be operable." There was some doubt as to whether or not the

operability of all jet pumps was proven prior to the reactor being placed in the startup mode. SI 4.6.E-1, "Jet Pump Operability Test," which was run on October 21, 1984, apparently failed to show conclusively that all unit 3 jet pumps were operable. A note on page 7 states that "step 13 did not pass." In parentheses under the statement were the words "due to low flow conditions." From discussions with nuclear engineers, it seems that failure of step 13 at low flow conditions is not unusual. However, another step on the same page states "SI required prior to startup." It does not appear that SI 4.6.E-1 was performed again prior to startup. On October 22, 1984, the flow transmitter for one of the jet pumps that had demonstrated low flow was found with the equalizer valve open. It is recognized that the jep pumps are passive and the only requirement necessary to prove operability is to show that they are intact. An inoperable flow transmitter does not mean the pumps were inoperable. However, it is not clear that the SI proved that the jet pumps were intact. Thus, the operability of the jet pumps was not demonstrated prior to startup.

3. rocedural Adherence

A number of procedural deficiencies resulted from activities associated with the reactor startup on October 22, 1984, as follows.

a. Master Refueling Test Instruction

Three steps in the instruction were not signed off. They were:

Step 27 - Authorization obtained from Plant Superintendent to continue testing in the heatup to 55 percent power plateau.

This step was to be verified by the shift engineer. The required authorization was to permit closed vessel testing following completion of open vessel testing. The test to be performed would require the reactor to be started up to the critical condition. The shift engineer thought he had the required authorization to pull rods to critical. The Plant Manager (specified as Plant Superintendent in the procedure) was in the control room at the time. The step wasn't signed off apparently because the reactor was being taken critical to perform one test at criticality and not progressing any further but shutting back down as soon as the test was completed.

Step 28 - Preparation for approach to critical complete per GOI-100-1, sections I.A and II.A.

Responsibility for verifying this step had been assigned to the Operations supervisor. The reason given for failure to sign off this step was that the reactor was going critical for a test and not for an actual startup. The intent was to sign off this step before the reactor was brought up on the "A" rods for total startup and power generation.

Step 29 - Authorization obtained from the Plant Superintendent to take the reactor critical.

The required verification signature in this case was that of the shift engineer. Again the shift engineer was of the impression that he had the required authorization to go critical, but didn't need to sign off the step until the shutdown margin test was completed and the second startup with the "A" rods was ready to begin.

In addition to these steps that weren't signed off, the NRC had a concern about step 30 which reads as follows:

> Take the reactor critical in the sequence specified by RTI-4, per GOI-100-1, Sections II.B and C. Perform SRM functional checks and record SRM readings per RTI-4, Sections 8.1 and 8.3. During initial period, take SRM readings and measure reactor K_{eff} for comparison to GE calculated K_{eff} per RTI-4, Section 8.3. Also during initial critical, verify IRM/SRM overlap per RTI-4, Section 8.3.7.

The nuclear engineer was required to verify this step.

Since the first statement directs the reactor to be taken critical and the nuclear engineer was to verify accomplishment, the NRC was concerned that the nuclear engineer was in charge of taking the reactor critical. The instructions could be more clearly stated if the first statement was separated from the remainder of the step and assigned to the shift engineer or other appropriate person for verification. However, there was no intent by the nuclear engineer to be in control of the reactor startup and no intent on the part of Operations personnel to allow such an action.

Failure to sign off steps 27, 28, and 29 represent procedural violations, but no evidence was obtained that indicated an intent to take the reactor critical without proper authorization.

b. GOI-100-1

Step 11.A.9 states "Secure shutdown cooling in accordance with OI-74." Step 11.A.10 states "Start recirculation pumps per OI-68." These two steps were not performed because the shutdown cooling mode of RHR was in service and one loop of the recirculation system was out of service to support startup and the shutdown margin test. This configuration was considered to be necessary for moderator temperature control, and the procedure step requiring change of that configuration was determined not to be applicable in this case. However, no procedure change was made to accommodate this decision. Therefore, the procedure was not followed and Technical Specification 6.3.A.1 and Criterion V of Appendix B to 10CFR50 were violated. These regulations require procedures to be prepared and followed when performing nuclear activities.

4. Rod Withdrawal Sequence

The rod withdrawal sequence was developed by the Reactor Engineering Unit. There was no procedure that provided specific instructions on how the rod withdrawal sequence should be programmed into the rod worth minimizer (RWM) for RTI-4 and verified. The nuclear engineers felt that this should be considered to be within the skill of the engineers. The verification performed consisted of putting the RTI-4 rod withdrawal sequence into the computer and then verifying by computer printout that the same sequence was successfully stored in the RWM computer memory.

Technical Specification 3.3.B.3.c states that "whenever the reactor is in the startup or run modes below 20 percent reactor power, the rod worth minimizer shall be operable." Technical Specification surveillance requirement 4.3.B.3.c specifies, among other things, that the correctness of the control rod withdrawal sequence input to the RWM computer shall be verified to determine the capability of the RWM before reactor startup or shutdown. An error was made in the control rod withdrawal sequence for the unit 3 startup on "B" rods for shutdown margin testing. The sequence was developed by a nuclear engineer. No verification by a second engineer was made. The NSRS reviewer believes that the intent of the Technical Specification discussed above is to require a second party verification. Under this interpretation, the Technical Specification was violated.

The NRC indicated a concern about the way rod coupling integrity checks are performed. This concern was apparently brought about because of recent changes to the pull sheet

format. A review of the pull sheets indicates that the rod coupling integrity checks were performed and documented on October 22, 1984. There are now two different data sheets for rod coupling integrity checks. Several rods are verified coupled by one set of initials for the first 50 percent of rods. The final 50 percent are verified individually. However, the present method of documentation appears to be adequate and satisfactory from a procedural and regulatory position.

C. Testing

The full core shutdown margin test conducted in accordance with RTI-4 appeared to have been performed in a satisfactory manner. While problems were encountered in the development of the rod withdrawal sequence determination after the reactor was critical, there did not appear to be any problem in getting the necessary data and calculating the shutdown margin.

D. Procedure Evaluation

The procedures that existed on October 22, 1984, were marginally adequate for the performance of the proposed activities. The plant systems were in good condition for support of startup as indicated by the effective implementation of SP BF-12.18. The MRTI provided sufficient coordination to allow reactor startup in the proper manner. GOI-100-1 as it existed on October 22, 1984, had been used many times to take the reactor critical. It could have been done again. RTI-4 provides adequate instructions for performance of the shutdown margin test. It was recognized that the startup was somewhat different from those performed in the recent past on any of the three BFN units. The core was cold and a source of heat was necessary to raise the moderator temperature to the necessary 180°F. This was the first time the reactor had been taken critical on the "B" rod configuration, which required the double startup prior to power generation. However, with careful planning and strict procedure adherence, the entire process could have been executed with available procedural systems.

GOI-100-1 had been revised and improved subsequent to the October 22 activities, but it still contained the same disclaimer statement that indicates to the users they don't have to follow it in some cases. The new revision restricts the amount of liberty that can be taken in the way of deviations. A lot of hard work and thought has gone into the development and revision of GOI-100-1. Additional work has been expended in checking GOI-100-1 out by walking it through to get it to the place that everyone concerned is satisfied that it will do its intended function. An equal amount of effort to assure that it will be followed should culminate in a well controlled, integrated reactor operations policy in practice. Management should ensure that this equal or better effort is applied.

E. Overall Evaluation and Root Cause Assessment

1. Office of Nuclear Power Review

The Office of Nuclear Power put together an impressive review team to evaluate the October 22, 1984 event. There was a wide diversity of expertise within the team and each member was well qualified to evaluate the event from his particular professional area. The team charter as outlined by the Manager of NUC PR was sufficiently broad and provided the authority to assure a thorough evaluation. NSRS observation indicated that team members took their responsibility seriously and performed their work well both as individuals and as a team effort. This type of internal review by NUC PR of potentially serious events on a routine basis would definitely be considered a positive action by TVA organizations and regulatory agencies. If this quality of review was performed as a matter of policy, NSRS would not need to conduct investigations of events on a frequent basis. NSRS would, however, monitor the effectiveness of the NUC PR investigation program in a manner similar to existing review of overall programs.

2. NSRS Investigation

It should be recognized that honest, knowledgeable people may hold different views and approach regulatory requirements with different degrees of conservatism. The regulator will tend to be more conservative than the regulated in interpretating regulatory requirements. Herein lies one of the root causes of the October 22 event and some of the events that have occurred in the past and will likely occur in the future unless positive action is taken.

There are numerous gray areas in the BFN Technical Specifications. There also gray areas in the Standardized Technical Specifications. The gray areas are where most of the enforcement problems occur, because they are most susceptible to interpretation. In order to improve regulatory performance significantly at BFN, the gray areas will need to be minimized or BFN management will need to make a total commitment to the most conservative practical interpretation. Such a commitment is not easy for people trying to efficiently generate power at a three-unit BWR. However, anything less will probably fall short of the regulatory record desired by top TVA management and necessary to satisfy the NRC of TVA's ability to operate BFN within regulatory requirements.

Another aspect of Technical Specification adherence that should be considered is the method for obtaining waivers or

exceptions to Technical Specifications. There are occasional situations when plant personnel find themselves under restricted conditions that will require shutdown to prevent Technical Specification violations or they may find that the plant is already being operated in violation of Technical Specifications, but circumstances indicate that shutdown is not the preferred action. In these cases, a waiver or some type of exception from Technical Specifications would be desirable and the practice to be used by TVA to obtain such exceptions should be developed so that all concerned personnel understand their roles and how they are to perform. The methodology to be established should include a listing of the people in TVA to be informed and what action each is responsible for taking. It should also define what TVA management level will contact NRC and to whom or to what position level(s) within NRC the requests for exceptions will be addressed.

Another root cause that needs to be fully recognized and resolved is the differences in concepts of adequate procedural controls between TVA and the NRC. BFN and NUC PR in general have not made the total commitment to procedural control and strict procedural adherence that is necessary to achieve and maintain a good, consistent regulatory record. No one at BFN violates a procedure or a regulation because they want to. There is always a reason the violations occur. Sometimes the procedures are not as clearly written as they should be, but this explanation is considerably overused. Other valid reasons for failing to follow instructions include personnel error, lack of planning, lack of attention to detail, or any number of things that appear to be reasonable justification for doing something a certain way at a given time. Many times the reasoning is sound and justifies doing work in a different way than that specified by a controlling procedure. However, it seldom justifies violation of the procedures. In such cases the procedure should be changed to support the better way. At no time during the investigation did anyone admit that those actions which resulted in violation of procedures were wrong, that a mistake had been made, or that greater commitment to procedure adherence was necessary. Most responses were defensive. The type of statements most commonly heard were: "The procedure was vague." "The procedure provided latitude and allowed deviations." "It is just a matter of interpretation." "We were in a test condition, not a startup." "That procedure was never intended to provide detailed instructions." "We did it that way for years and nobody considered it a violation." "We don't need a procedure to do that because it is within the skill of people performing the function." "The interpretation you are suggesting implies an entirely new way of doing business."

NUC PR management should become involved to the degree necessary to assure a total commitment to procedural controls is made and carried out. This should include development of procedures for all work that could be conservatively considered to require procedure by regulatory requirements and development and implementation of a policy to require that the procedures be followed.

The Regulatory Performance Improvement Plan (RPIP) has helped to substantially improve the situation at BFN during the past nine months. This is particularly evident in the Outage and Operations Sections. The attitudes have improved which is an important indicator. The people want to do a good job. However, they have not been totally convinced that strict adherence to procedures is always in the best interest of doing a good job.

V. PERSONNEL CONTACTED

A. L. Burnett, Assistant Head, Operations
J. Cain, Assistant Shift Engineer
E. D. Cornelius, Supervisor, Mechanical Maintenance
J. D. Glover, Jr., Shift Engineer
D. A. Housley, Nuclear Engineer
B. J. Irby, Supervisor, Instrument Maintenance
G. T. Jones, Plant Manager
L. W. Jones, Chief, Quality Engineering Staff
S. R. Maehr, Head, Planning and Scheduling
A. W. McCaleb, Supervisor, Instrument Maintenance Craft
B. C. Morris, Supervisor, Plant Compliance
J. R. Pittman, Maintenance Superintendent
L. Smith, U-3 Instrument Mechanic Foreman
D. L. South, Shift Engineer
J. E. Swindell, Operations and Maintenance Superintendent
J. D. Thompson, Instrument Mechanic
B. T. Williamson, II, Reactor Engineer
J. D. Wolcott, Supervisor, Reactor Engineering Unit

VI. LIST OF DOCUMENTS REVIEWED

A. GOI-100-1, "Integrated Plant Operations"

B. Master Refueling Test Instruction (MRTI)

C. RTI-4, "Full Core Shutdown Margin Test"

D. SP BF-12.18, "Standard Practice for Prestartup Review"

E. Memorandum from J. A. Coffey to L. M. Mills dated October 24, 1984, "Confirmation of Commitments Regarding Plant Startup Procedures"

F. Confirmation of Action letter from J. P. O'Rielly, NRC, Region II, to H. G. Parris, undated

G. Unit 3 Weekend Notes to Operations personnel dated October 19, 1984

H. Various daily journals and shift turnover checklists

I. Prestartup Committee Minutes for unit 3 startup preparations

J. Plant Operations Review Committee meeting minutes for October 21 and 22, 1984

K. MMI-95, Section 8.0, "Instructions for Opening and Closing of Primary Containment Hatches"

L. Various operating charts, data sheets, and maintenance requests

M. Sections 3, 4, and 6 of the BFN Technical Specifications

N. SI 4.3.B.1.a, "Control Rod Coupling Integrity Check"

O. SI 4.3.B.3.a, "Rod Worth Minimizer and Rod Squence Control System Functional Test"

P. SI 4.6 E.1, "Jet Pump Operability Test"

TVA 74 (08-9-83)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

GNS 850116 050

TO : J. P. Darling, Manager of Nuclear Power, 1750 CST2-C
R. W. Cantrell, Manager of Engineering, W11A9 C-K
C. Bonine, Jr., Manager of Construction, 12-108 SB-K

FROM : K. W. Whitt, Director of Nuclear Safety Review Staff, 249A HBB-K

DATE : January 16, 1985

SUBJECT: NUCLEAR SAFETY REVIEW STAFF (NSRS) REPORT R-84-32-NPS - FOLLOW-UP REPORT OF PREVIOUSLY IDENTIFIED ITEMS

NSRS performed a follow-up review of 22 previously identified items from four earlier reviews for which corrective action had not been verified. The attached report closes four of the items as satisfactorily corrected and four others for NSRS record purposes since they were formally identified and tracked by the Division of Quality Assurance. Three items were left open and require additional response from the responsible organizations. These items and organizations were:

R-81-14-OEDC(BLN)-32 and -41, Office of Engineering
R-82-02-WBN-26, Office of Construction

Eleven items associated with the inadequacy of the TVA QA program were consolidated into a single finding addressing the need for a comprehensive integrated quality program. These eleven items were closed for record purposes. The new item, R-84-32-NPS-01, requires response from the Office of Nuclear Power.

It is requested that responsible organizations submit responses including target dates for completion of proposed action to NSRS by February 15, 1985.

If you have any questions concerning this report, please contact M. A. Harrison at extension 4816 in Knoxville.

K. W. Whitt

K. W. Whitt

MAH:BJN
Attachment
cc (Attachment):
C. W. Crawford, 670 CST2-C
H. G. Parris, 500A CST2-C
MEDS, W5B63 C-K

NSRS FILE

GNS 850116 051

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS REPORT R-84-32-NPS

SUBJECT: FOLLOW-UP REPORT OF PREVIOUSLY IDENTIFIED ITEMS

DATES OF REVIEW: DECEMBER 10 - DECEMBER 20, 1984
JANUARY 7 - JANUARY 11, 1985

REVIEWER: M. A. HARRISON — DATE 1/16/85

APPROVED: M. S. KIDD — DATE 1-16-85

TABLE OF CONTENTS

I. SCOPE

Late in 1982, after the formation of the Office of Quality Assurance (OQA), the NSRS formally transferred a number of concerns with recommendations to OQA for verification of corrective action and closure. Actions by responsible organizations enabled OQA to close many of the transferred items, but 18 of them had not been closed as of September 27, 1984, when responsibility for verification of actions taken was formally returned to NSRS by OQA. NSRS accepted the responsibility for the items and performed follow-up reviews to determine the status of each of these items as well as others remaining open.

The 22 specific items reviewed are identified in section IV, "Details." The items were among those remaining open from the following NSRS reports:

A. R-81-14-OEDC(BLN), Major Management Review of OEDC/Bellefonte

B. R-81-31-NPS, Special Review of Division of Nuclear Power Operator Training

C. R-82-02-WBN, Major Management Review of Watts Bar

D. R-81-11-WBN, Special Review - WBNP

In view of the reorganization and realignment of responsibilities in the Office of Power and Engineering (OPE), NSRS focused on closing as many of the items as possible, and consolidation and updating of items where corrective actions were determined still essential. The status of some items remaining open from the four reports is not included here but will be addressed separately in other reports.

II. CONCLUSIONS AND RECOMMENDATIONS

A. R-84-32-NPS-01, <u>Inadequate TVA Quality Program</u>

<u>Conclusion</u>

Action was incomplete on developing and implementing an integrated quality program for TVA nuclear facilities that includes the following attributes consolidated from previous NSRS findings: (Refer to section IV.D and as indicated below.)

1. Identification of activities affecting quality. (Refer to Details, IV.C.1.)

2. Identification of components, systems, structures to which the quality program is applied. (Refer to Details, IV.A.5, IV.C.2, and IV.C.3.)

3. Identification and definition of industry standards and guidance for control of activities outside the scope of regulatory requirement. (Refer to Details, IV.A.1.)

4. Improvement of the Office of Engineering (OE) quality program and procedures and acceptance of the OE quality program by the Office of Nuclear Power. (Refer to Details, IV.A.3 and IV.A.4.)

5. Incorporation of vendor data into a configuration management control system. (Refer to Details, IV.A.2 and IV.A.7.)

Recommendation

The Office of Nuclear Power as owner-operator should assure that the integrated quality program, currently being planned and developed, will account for the identification and control of the attributes listed above.

III. STATUS OF SELECTED PREVIOUSLY IDENTIFIED OPEN ITEMS

NSRS verified that an adequate, ongoing corrective action program had been implemented to resolve four of the findings and these were closed. Other findings reviewed revealed that actions were in progress or plans were being developed for resolution, but additional management involvement was necessary to achieve resolution. Eleven of these findings were determined by NSRS to be directly related to development and implementation of an adequate integrated quality program for TVA nuclear facilities and were consolidated into a single current conclusion and recommendation, "Inadequate TVA Quality Program." Three items requiring additional corrective action remained open under their previous report designations. The status of actions to correct the concerns is addressed in sections IV.A and IV.C. The items remaining open are:

- R-81-14-OEDC(BLN)-32, Inadequate Storage of Audit Support Records
- R-81-14-OEDC(BLN)-41, QAB (QMS) Auditor Training
- R-82-02-WBN-26, Lack of Approved Procedures for Certain Computer Programs

Four other items were determined still not corrected but are closed for record purposes since the Division of Quality Assurance had included the essence of these findings in an audit, CH-8400-07, requiring response and resolution, and the duplication of effort to track and verify the items is unnecessary.

IV. DETAILS

A. R-81-14-OEDC(BLN), Major Management Review of OEDC/Bellefonte

Nine items remained open from this report at the initiation of the follow-up review. The status of each is addressed below, identified by the original finding designation.

1. R-81-14-OEDC(BLN)-03, Regulatory Guides/Standards

Summary of Original Finding

OEDC was not providing a compilation of the regulatory guides and industry standards to which TVA had committed, other than those endorsing the ANSI N45.2 standards.

Current Status

Resolution of this item had not been achieved and, as a result of the reorganization of OPE, responsibility for identification of industry standards which will be used by TVA to establish requirements for control of their respective functions was in need of clarification. Refer also to sections IV.A.3 and IV.D for additional comment. Item R-81-14-OEDC(BLN)-03 is closed for record purposes. For verification of completion of corrective action R-81-14-OEDC (BLN)-03 is incorporated into R-84-32-NPS-01.

2. R-81-14-OEDC(BLN)-14, Drawing Information System (DIS) Implementation Concerns

Summary of Original Finding

OEDC failed to adequately implement the DIS for Bellefonte configuration control.

Current Status

In an interview with the Chief, Information Management Branch, NSRS learned that the DIS had been incorporated into the Drawing Management System (DMS) and that TVA intended to implement DMS as the overall control system (for drawings) for NUC PR, OE and OC. NSRS was also aware of the existence and efforts of a TVA task force on configuration management through a review performed on outage controls in October 1984. Within the scope of this task force were subtask groups' missions to gain control of information provided to TVA via vendor manuals and drawings as part of the planned configuration management system. NSRS believed the efforts to gain control of this situation and the degree of management attention afforded to be fully adequate. No additional recommendations are offered at this time. This item R-81-14-OEDC(BLN)-14 and related item R-81-14-OEDC(BLN)-31 are closed for record purposes. For purposes of verification of corrective action completion, these findings are consolidated into R-84-32-NPS-01.

3. R-81-14-OEDC(BLN)-17, Program and Implementation Inadequacies of Engineering Procedures (EPs)

Summary of Original Finding

EN DES EPs were inadequate in that they contained conflicting information, provided insufficient assurance of design quality, and were not consistently implemented.

Current Status

In an interview with NSRS, the Manager of the Quality Management Staff (QMS) of OE discussed OE's intention to redesign and streamline Engineering Procedures as stipulated by the Manager of OE. This effort was considered by NSRS as a necessary part of the overall effort to be coordinated and performed by the Division of Quality Assurance (DQA) to establish a QA program for TVA. Engineering or quality assurance programs developed by OE must be evaluated and approved by DQA for the Office of Nuclear Power (ONP) as required by ANSI N45.2.13 and a more cohesive OE program should be the result. NSRS closed R-81-14-OEDC(BLN)-17 for record purposes. For the purpose of verifying corrective action taken, the essence of this item--establishment and implementation of a comprehensive program capable of achieving and assurring quality in design--is being incorporated into R-84-32-NPS-01.

4. R-81-14-OEDC(BLN)-18, Failure to Establish Detailed QA Policy

Summary of Original Finding

EN DES failed to establish a comprehensive detailed QA policy, especially regarding procedural compliance and review of EPs for adequacy.

Current Status

The Manager of OE had issued a memorandum of quality policy to be included in OE Administrative Instructions. This overall quality policy statement was determined fully adequate, but the pending redesign of OE procedures requires that further verification be performed to assure the policy is incorporated into the new OE program. Therefore, for record purposes R-81-14-OEDC(BLN)-18 is closed, but for verification purposes is incorporated into R-84-32-NPS-01.

5. R-81-14-OEDC(BLN)-20, Lack of Control of Safety-Related Systems List

Summary of Original Finding

EN DES failed to develop a single controlled comprehensive listing of safety-related systems and components for BLN. This item was similar to R-82-02-WBN-07 and -09 for Watts Bar.

Current Status

In a memorandum to the Manager of the Office of Construction (OC) dated December 3, 1984, the Manager of OE stated that the baseline Q-list for BLN would be issued by February 1, 1985. As with Q-lists for other plants, it will be necessary to reach agreement among ONP, OE, AND OC on the content of this list. This effort is considered by NSRS to be part of the overall effort to develop and implement a comprehensive TVA QA program and as such is incorporated in R-84-32-NPS-01. R-81-14-OEDC(BLN)-20 is closed for record purposes.

6. R-81-14-OEDC(BLN)-23, Documentation of System Design Bases

Summary of Original Finding

EN DES was not maintaining accurate, permanent, and controlled design bases (or system descriptions) for safety systems. This item was similar to R-82-02-WBN-10 and -11 for Watts Bar.

Current Status

In response to this item and to items R-82-02-WBN-10 and -11 EN DES issued EN DES EP 3.38, "System Description Documents - Preparation, Review, and Approval," Revision 0 dated February 16, 1983, and EP 3.01, "Design Criteria Documents - Preparation, Review, and Approval," Revision 5 dated December 13, 1982. These documents specify the controls and requirements applied to the generation and maintenance of System Descriptions and Design Criteria Documents, including appropriate criteria for deactivation, and a requirement for review by NUC PR.

According to a System Description status report of December 6, 1984 for Watts Bar, over 60 systems had been designated for provision of controlled system descriptions, and a schedule for the completion of each was given. This project was estimated at approximately 50 percent complete, with all descriptions to be completed by August 1, 1985.

For BLN, the BLN design project designated personnel responsible for preparation of over 80 System Descriptions to be

generated in accordance with the procedures identified above. A schedule for the generation of the descriptions was loaded into OE's PC III computer program for tracking and reporting status and changes. Although some System Descriptions were scheduled for issue coincident with system preoperational testing, this program appeared to be adequate and in progress. This item is closed. R-82-02-WBN-10 is closed. R-82-02-WBN-11 is closed.

7. R-81-14-OEDC(BLN)-31, Failure to Input Complete Vendor Information Into the DIS

Summary of Original Finding

EN DES EPs failed to require review of vendor manuals for drawings which should be inputted to DIS, as was required by ID-QAP 6.1.

Current Status

This item is closed for record purposes and is incorporated into R-84-32-NPS-01. Refer to R-81-14-OEDC(BLN)-14, section IV.A.2.

8. R-81-14-OEDC(BLN)-32, Inadequate Storage of Audit Support Records

Summary of Original Finding

OEDC QA groups inadequately stored audit support records, e.g., documents other than the actual report and responses, such as the checklist used and the audit plan.

Current Status

The QMS Manager in an interview with NSRS agreed to enter into MEDS the issued audit report with support records such as the audit plan and checklist. The QMS Manager also stated that this control would be procedurally required under the OE procedure system. Duplicate microfilm storage meets the intent of ANSI N45.2.9 for storage of QA records. This item remains open pending approval of the procedure for QMS QA record storage.

9. R-81-14-OEDC(BLN)-41, QAB Auditor Training

Summary of Original Finding

EN DES failed to establish a written, approved program for training QA auditors.

Current Status

In an interview with NSRS the QMS Manager stated intentions to develop and approve a QMS procedure controlling the training and certification of auditors and lead auditors in accordance with ANSI N45.2.23, which addresses requirements for auditor qualifications. A draft copy of the procedure dated October 29, 1984, was provided to the reviewer and appeared adequate. This item remains open pending approval of the OE or QMS procedure for auditor training and certification.

B. R-81-31-NPS, Special Review of NUC PR Operator Training

Four items remaining open from the operator training review were reviewed during this follow-up. The four deficiencies were identified as concerns regarding training programs administered by the Power Operations Training Center. The Division of Quality Assurance completed a comprehensive audit of operator training and requalification, Audit CH-8400-07, issued July 26, 1984. This audit identified similar general and specific concerns and, in follow-up to the four NSRS findings, determined that corrective action was not yet sufficient to justify closing them. The audit was performed in accordance with TVA commitments to ANSI N45.2.12 and as such requires written response and QA verification of corrective actions taken. Therefore, NSRS has determined that the following four findings from NSRS report R-81-31-NPS are closed:

R-81-31-NPS(POTC)-01
R-81-31-NPS(POTC)-02
R-81-31-NPS(POTC)-03
R-81-31-NPS(POTC)-04

C. R-82-02-WBN, Major Management Review of Watts Bar

Seven items remaining open from the major management review of Watts Bar were followed up to verify corrective action status. Each item is identified below by its original finding designation.

1. R-82-02-WBN-03, Activities Affecting Quality

Summary of Original Finding

This finding identified inconsistencies and omissions in the overall TVA QA program, especially in the areas of identification and control of activities affecting quality and involving interdivisional/interoffice interfaces.

Current Status

The status of this item is addressed in section IV.D. For record and tracking purposes this item is closed. For verification this item is included in R-84-32-NPS-01.

2. R-82-02-WBN-07, Inaccuracies in Identifying the Scope of Work Under QA Control

Summary of Original Finding

OEDC did not provide a comprehensive, consistent, controlled listing of structures, systems, and components to which the QA program for Watts Bar was to have been applied. This item was later determined by OEDC QA to be generic to plants in the construction phase (Bellefonte).

Current Status

The Office of Quality Assurance closed this item for Watts Bar, but left it outstanding for the BLN program. As such, it is similar to item R-81-14-OEDC(BLN)-20, included in R-84-32-NPS-01. Action to achieve agreement among ONP, OC, and OE on the Q-list and its implementation for Watts Bar was scheduled to be taken prior to receipt of an operating license for Watts Bar in accordance with a commitment made to the NRC on November 28, 1984. Item R-82-02-WBN-07 is closed for record and tracking purposes. For verification purposes, this item is incorporated into R-84-32-NPS-01.

3. R-82-02-WBN-09, Lack of Control of Safety-Related Structures, Systems and Components List

Summary of Original Finding

EN DES failed to provide positive control of the identification/designation of safety-related structures, systems, and components to the extent that other organizations, notably CONST, were generating and using "safety-related lists" in the absence of clear design guidance.

Current Status

This item is closed for record and tracking purposes and is included in R-84-32-NPS-01. Refer to items R-81-14-OEDC (BLN)-20, section IV.A.5 and R-82-02-WBN-07, section IV.C.2.

4. R-82-02-WBN-10, Inadequate Documentation of Systems Design Base

Summary of Original Finding

EN DES was maintaining incomplete and inconsistent Watts Bar Design Criteria and FSAR System Descriptions. Design Criteria were not provided for some systems. This item was similar to R-82-02-WBN-11 and R-81-14-OEDC(BLN)-23.

Current Status

See section IV.A.6, R-81-14-OEDC(BLN)-23. This item is closed.

5. R-82-02-WBN-11, Improper Inactivation of Some Watts Bar Design Criteria

Summary of Original Finding

EN DES deactivated some Watts Bar Design Criteria by an uncontrolled practice. This item was similar to R-82-02-WBN-10 and R-81-14-OEDC(BLN)-23.

Current Status

See section IV.A.6, R-81-14-OEDC(BLN)-23. This item is closed.

6. R-82-02-WBN-25, Control of Protective Coating Processes

Summary of Original Finding

EN DES failed to define in a controlled design document the areas, structures, systems, and components to be protected by Service Level I protective coatings.

Current Status

The Watts Bar Q-list issued as controlled drawing 91QL series addressed and identified the areas and structures requiring Service Level I protective coating. Bellefonte controlled drawing series OGP-0025 R-0 provided similar information for that project. R-82-02-WBN-25 is closed.

7. R-82-02-WBN-26, Lack of Approved Procedures for Certain Computer Programs

Summary of Original Finding

CONST failed to provide procedural controls on the development, verification and application of computer programs used to support quality in construction such as the Universal computer program.

Current Status

This item was closed for Watts Bar but remained open for BLN as determined by OQA. An interview with the BLN Compliance supervisor revealed that essentially no progress had been made toward adequate resolution of this item. This item remains open.

D. R-81-11-WBN, Watts Bar Unit 1 - Special Review

Two programmatic findings, R-81-11-WBN-01 and -02, remained unresolved and were followed up during this review.

Summary of Original Findings

These findings reported that the TVA QA program for Watts Bar was inadequate to control or assure compliance with requirements and commitments. The report recommended that a thorough review of commitments and implementing procedures be performed by QA and that the results of that review be used to upgrade the program. These findings were considered by NSRS to be similar in nature and scope to R-82-02-WBN-03, -07, and -09, and to R-81-14-OEDC (BLN)-03, -17, -18, and -20, in that all of these findings dealt generally with examples of failures to define, prescribe, and implement the controls necessary to achieve and assure quality during the design and construction phases of nuclear plants.

The Office of Quality Assurance had planned to develop and assure implementation of an "integrated" QA program for TVA in order to resolve these deficiencies. This project, development of the Management Policies and Requirements Manual, was terminated during the TVA reorganization and formation of OPE.

Current Status

As a result of the July 1984 reorganization of the TVA Power organization, ONP, i.e., the owner-operator, was formed with broad authority and responsibility for establishing and executing an integrated QA program for the design, construction, and operation of TVA nuclear facilities. The ONP Division of Quality Assurance (DQA) was delegated the responsibility of developing and maintaining an overall nuclear QA program. DQA specified that the primary goal in the overall program development would be the definition of QA program policies and requirements applicable to the design, construction, services, and operation of nuclear facilities; and secondarily to define and develop requirements for control of key activities that had previously been inadequately controlled. DQA had issued a formal plan for development of a single QA policy- and requirement-oriented program manual, the "Nuclear Quality Assurance Manual," and scheduled its initial issue for December 31, 1984. An intended function of this manual is to replace the former upper tier program manuals, such as the Interdivisional Quality Assurance Manual, Program Requirements Manual, and Office of Power Quality Assurance Manual, which had inadequately defined and controlled requirements and interfaces.

The Office of Nuclear Power had also identified the need to restructure and upgrade ONP procedures that affect and control the quality, consistency, and safety, among other necessary attributes, of performance of ONP activities in support of ONP goals. In a memorandum dated October 29, 1984, the Manager of

ONP planned the development of a Nuclear Policies and Requirements Manual (NPRM) to establish procedural controls for areas outside the scope of the regulatory compliance-oriented nuclear QA program.

NSRS believes that unless otherwise planned, this development effort should also include the management requirements and acceptance standards for work performed for or in support of ONP by organizations such as OE and OC so that adequate performance requirements are communicated, acknowledged, and achieved throughout TVA. These items, R-81-11-WBN-01 and -02 are closed for record and tracking purposes and are included in R-84-32-NPS-01.

VI. REFERENCES

A. Nuclear Safety Review Staff Reports/Working Files

1. R-81-11-WBN, Watts Bar Special Review dated July 1, 1981 (GNS 810701 051)

2. R-81-14-OEDC(BLN), Major Management Review of the Office of Engineering Design and Construction dated September 29, 1981 (GNS 810930 054)

3. R-81-31-NPS, Special Review of Division of Nuclear Power Operator Training dated March 30, 1982 (GNS 820330 050)

4. R-82-02-WBN, Major Management Review of Watts Bar Nuclear Plant dated June 3, 1982 (GNS 820603 051)

5. R-82-14-OEDC(BLN), Routine Followup Review of R-81-14-OEDC(BLN) dated November 3, 1982 (GNS 821104 052)

6. R-82-24-WBN, Routine Followup Review of R-80-11-WBN, R-81-11-WBN, R-81-28-WBN, and R-82-02-WBN dated November 4, 1982 (GNS 821104 050)

B. Correspondence (Other than included in NSRS Report Working Files)

1. Memorandum from H. N. Culver to J. W. Anderson dated December 29, 1982, "Transfer of Responsibility for Followup and Action on Nuclear Safety Review Staff Review Report Findings (GNS 821229 151)

2. Memorandum from J. W. Anderson to H. N. Culver dated September 27, 1984, "Reassignment of NSRS Items" (OQA 840927 002)

3. Memorandum from H. N. Culver to J. W. Anderson dated December 5, 1984, "Reassignment of NSRS Items" (GNS 841205 050)

4. Memorandum from J. W. Anderson to H. N. Culver dated June 13, 1984, "Updated Responses to NSRS Open Items (OQA 840613 002)

5. Memorandum from J. W. Anderson to H. N. Culver dated August 8, 1984, "Updated Responses to NSRS Open Items (OQA 840808 001)

6. Memorandum from R. M. Hodges to Those listed dated April 11, 1983, "Bellefonte Nuclear Plant - System Descriptions Assignments List" (BLP 830411 011)

7. Memorandum from J. L. Wright to Bellefonte Design Project Files dated April 7, 1983, "System Description Assignments List Coordination Meeting - Notes" (BLP 830405 024)

8. Memorandum from J. W. Anderson to G. H. Kimmons dated December 17, 1983, "NSRS Report R-81-14-OEDC(BLN) - Program Implementation Inadequacies of Engineering Procedures - Item 17" (OQA 831219 003)

9. Letter from J. W. Hufham to J. P. O'Reilly, NRC, dated November 28, 1984, " Watts Bar Nuclear Plant Units 1 and 2 - NRC-OIE Region II Inspection Report 50-390/82-09, 50-391/82-07 - Fourth Revised Response to Item 2" (L44 841128 809)

10. Memorandum from J. C. Standifer to Those listed dated December 6, 1984, "Watts Bar Nuclear Plant System Description - Schedule Update" (WBN 841206 006)

11. Memorandum from R. W. Cantrell to J. W. Anderson dated February 27, 1984, "NSRS Report R-81-14-OEDC(BLN) - Program and Implementation Inadequacies of Engineering Procedures - Item 17" (EEB 840227 001)

12. Memorandum from W. D. Poling to Those listed dated December 3, 1984, "QA Requirements for Regulated Programs" (L00 841203 921) and attachments

13. Memorandum from R. J. Mullin to Those listed dated December 10, 1984, "Quality Assurance Activities Transferred from the Office of Quality Assurance to Other Power and Engineering Organizations" (L16 841207 941)

14. Memorandum from J. P. Darling to Those listed dated October 29, 1984, "Restructuring of Office of Nuclear Power (NUC PR) Procedures" (L00 841025 873)

15. Draft memorandum from J. E. Law to H. N. Culver, "NSRS Deviation R-82-02-WBN-03"

16. Memorandum from J. E. Law to W. R. Brown and R. M. Pierce dated December 4, 1984, "Nuclear Quality Assurance Manual (OQAM) - Owner/Operator Responsibility and Authority" (L16 841204 938)

17. Memorandum from J. P. Darling to J. W. Anderson and R. J. Mullin dated August 9, 1984, "Establishment of Division of Quality Assurance (DQA) - Office of Nuclear Power (NUC PR) - (L20 840809 801)

18. Memorandum from H. N. Culver to J. W. Anderson dated April 27, 1984, "Nuclear Safety Review Staff (NSRS) Open Items" (GNS 840427 050)

19. Memorandum from J. P. Darling to Those listed dated November 11, 1984, "Overall Nuclear Quality Assurance Program - Policy and Status" (L16 841023 885)

20. Memorandum from J. P. Darling to Those listed dated October 1, 1984, "TVA Interdivisional Quality Assurance Procedures Manual (IPM)" (L16 840917 840)

21. Memorandum from J. Killian to J. P. Darling dated July 26, 1984, "OQA Audit Report No. CH-8400-07, Operator Training and Retraining" (OQA 840726 702)

22. Memorandum from R. W. Cantrell to J. W. Anderson dated March 23, 1984, "Watts Bar Nuclear Plant - NSRS Report No. R-82-02-WBN, ITEM NOS. -09, -10, -11, AND -25" (ESB 840323 004)

C. OQA Working Files, NSRS Open Items

D. <u>Procedures</u>

1. EN DES EP-3.37 R0 dated February 26, 1983, "System Description Documents - Preparation, Review, and Approval"

2. EN DES EP-3.01 R5 dated December 13, 1982, "Design Criteria Documents - Preparation, Review, and Approval"

3. 45D from J. S. Colley to QAC Employees dated December 12, 1984, transmitting pen and ink change to EN DES EP-1.29, "EN DES Quality Audit Program"

4. Draft QMS-EP Certification of Audit Personnel - received from J. S. Colley December 12, 1984

TVA 64 (08-0-81)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

001 '85 0607 050

TO : R. W. Cantrell, Manager of Engineering, W12A12 C-K

FROM : K. W. Whitt, Director of Nuclear Safety Review Staff, E12B31 C-K

DATE : June 7, 1985

SUBJECT: BROWNS FERRY NUCLEAR PLANT (BFN) - NUCLEAR SAFETY REVIEW STAFF (NSRS) REPORT NO. I-84-33-BFN - INVESTIGATION OF BFN PIPING AND SUPPORT DESIGN

Attached is the approved subject report. It is essentially identical to the final draft transmitted to you by my informal memorandum of May 1, 1985.

The investigation was initiated due to employee concerns. As noted in the report, one of those concerns, involving lack of documentation of baseline information for analysis and design was substantiated. Additionally, problems were identified involving training and qualification of BFN piping analysis personnel; deviations from design criteria, control of the IE Bulletin 79-14 effort; inadequate or nonexistent procedures to perform assigned work; potential interface problems between the analysis section, support design sections, and site modification group; lack of generic evaluation of prior piping analysis problems; and lack of an effective audit program in this area.

In accordance with the agreement reached during the exit meeting and subsequent telephone conversation between M. S. Kidd and D. B. Bowen on March 29, 1985, NSRS anticipated receipt of your responses to this report by May 31, 1985. That would have allowed concurrent issuance of your response along with the final report. This agreement was reiterated in the referenced May 1, 1985 informal memorandum.

Based on recent discussions with members of your staff and review of draft Office of Engineering (OE) comments on the report dated June 3, 1985, it is obvious that meaningful OE responses have not been generated. Please provide me with written responses to the negative findings (open items) identified in the report by June 21, 1985. Your responses should address these areas:

- Whether your agree or disagree with the conclusions drawn, as supplemented by information in the details section of the report.

  If you disagree with a conclusion, provide the basis for disagreement.

- Corrective actions which have been taken, and results achieved.

NSRS FILE

- Corrective and preventive steps which will be taken, along with scheduled completion dates.

- Results of generic evaluation of the specific examples of problem areas noted in the report. This should encompass similar work activities/processes for Browns Ferry as well as other nuclear plants.

Should you have questions regarding report content or these requests, please contact M. S. Kidd at extension 2289.

K. W. Whitt

K. W. Whitt

MSK:WCS
Attachment
cc (Attachment):
RIMS, SL26 C-K
B. M. Cadotte, E3C00 C-K (w/out attachment)
C. W. Crawford, 670 CST2-C
R. J. Mullin, 1350 CUBB-C
H. G. Parris, 500A CST2-C
J. A. Coffey, Browns Ferry NUC PR

TENNESSEE VALLEY AUTHORITY
NUCLEAR SAFETY REVIEW STAFF
NSRS REPORT NO. I-84-33-BFN

SUBJECT: INVESTIGATION OF BROWNS FERRY NUCLEAR PLANT PIPING AND SUPPORT DESIGN

DATES OF REVIEW: JANUARY 14 - MARCH 29, 1985

TEAM LEADER: Jack T. Muecke — J. T. MUECKE — 6/7/85 DATE

TEAM MEMBER: P R Washer — P. R. WASHER — 6/7/85 DATE

APPROVED BY: M S Kidd — M. S. KIDD — 6-7-85 DATE

RIMS, SL26 C-K

TABLE OF CONTENTS

I. BACKGROUND

This investigation was initiated due to an employee's concern relating to the practices of the Browns Ferry piping analysis section. The practices in question involved the documentation of final analysis results, the random manipulating of field-generated inspection data (piping configuration), and the intimidation of employees to justify overstressed (and potentially unsafe) conditions on existing piping configuration, thereby reducing the number of field modifications required.

Due to the technical nature of the concerns, two investigators with structural analysis and design experience were assigned to perform this investigation.

II. SCOPE

The original objectives of this investigation were to verify the validity of the employee concerns and to evaluate the effect of any substantiated concerns on the overall safety in the plant.

To accomplish these objectives the following actions were taken:

- Interviews were conducted with BFN piping analysis section personnel January 15-18, 1985.
- Analysis problems (computer printouts, calculation summary books, drawings) were reviewed and discussed with selected BFN personnel January 28-February 1, 1985.

The original objectives of the investigation were expanded after completing the interviews and problem review. During that process, specific issues were identified involving the training and qualification of BFN piping analysis personnel, deviations from design criteria, control of the "79-14" effort (see section IV.F.1 for background information on IE Bulletin 79-14), inadequate or nonexistent procedures to perform assigned work, and potential interface problems between the analysis section, support design sections, and site modification group.

The following actions resulted from these additional identified issues:

- Interviews were conducted with BFN support designers from two civil structure sections February 4-8, 1985.
- Support design calculations were reviewed and discussed with BFN personnel February 11-15, 1985.
- Visual inspection of modified supports at BFNP was performed February 19-21, 1985.
- The BFN civil project engineer, 79-14 coordinator, and BFN modifications project supervisor (mechanical section) were interviewed February 15, 19-21, 1985.

These actions were taken in an attempt to trace work associated with the 79-14 effort from the original "as built" inspections performed by NUC PR to the completed modification.

III. CONCLUSIONS/RECOMMENDATIONS

The following conclusions summarize the results of investigating the BFN piping analysis/support design effort. Sections A and B address employee and NSRS concerns, respectively.

A. Employee Concerns

1. I-84-33-BFN-01, Documentation of Baseline for Analysis and Design

Conclusion

Of the analysis problems reviewed, no isometric and support load drawings which involved the emergency equipment cooling water (EECW) and reactor drain and vents system were being issued in conjunction with the corresponding analysis problem, as required by design criteria BFN-50-D707. These drawings represent the baseline used for the analysis and support design. In some cases, support drawings had been issued and field modifications completed without the isometric or load tables being issued (see sections IV.C.2.a, IV.D.3, and IV.E for details).

Recommendation

Prepare and check all isometric and support load drawings and issue them in conjunction with completed analysis summary book. Perform an evaluation of problem N1-110-1R to verify the correct value of the design load for support H-21.

2. Manipulation of Field-Generated 79-14 Inspection Data

Conclusion

Information was provided verbally concerning some dimensional additions to inspection data, although no objective evidence was located to indicate a safety concern at this time (see section IV.D.1 for details).

3. Intimidation of Employees to Reduce Number of 79-14 Field Modifications

Conclusion

Major emphasis had been placed by management on retaining the "as-built" configuration with no modifications when performing analysis and support design. This "emphasis" has

been construed by some employees to be intimidation. No evidence of employees signing inaccurate analyses or support design calculations because of pressure was identified (see section IV.D.1 for details).

B. NSRS Identified Concerns

1. I-84-33-BFN-02, Acceptance of Pipe Stresses and/or Support Stresses in Excess of Defined Allowable Stresses

Conclusion

There have been instances where stresses in excess of the code-defined allowables have been accepted possibly due to the importance management places on accepting the "as-built" configuration of the installed piping. There was no documentation provided in analysis/support design calculations to justify the use of stresses in excess of the allowables (see section IV.D.3 for details).

Recommendation

Although emphasis can be placed on using the "as-built" configuration where possible, everyone should be made aware that the allowable stresses, from the ASME code for piping and AISC code for supports, as defined in the design criteria cannot be exceeded. Also, specific examples of excess stresses provided in the Details section of this report should be corrected and a review performed to ensure others do not exist.

2. I-84-33-BFN-03, Written Guidance and Training for Performing Piping Analysis

Conclusion

The design criteria BFN-50-D707, written to define the general technical requirements of the 79-14 effort, does not provide a complete guide for performing analysis. Analysts were given no additional instructions, training or general orientation to BFN analysis and procedures when they entered the section nor on a periodic basis to verify consistency of approach when information relating to a new analysis method became available. Analysts interviewed provided inconsistent answers concerning questions which involved stress intensification factors, insulation data, flange qualifications, normalized versus unnormalized loads, and capabilities of the TPIPE post processor (see section IV.C.1 for details).

Recommendations

I-84-33-BFN-03A

Technical guidance in the form of a controlled handbook or similar mechanism should be issued and made available to the section.

I-84-33-BFN-03B
General orientation and training specific to BFN practices should be made available on a periodic basis within the section.

3. I-84-33-BFN-04, Calculations Checked by Non-independent, Inexperienced Personnel

Conclusion

Inexperienced personnel were assigned to check the work of the engineer who was training them. Requirements for independent verification, as defined by EP 3.03, were not being met (see section IV.C.2.b for details).

Recommendation

Utilize qualified personnel to maintain independent verification for all problems. Do not assign personnel areas of responsibility until they have received adequate training to qualify for the responsibility.

4. I-84-33-BFN-05, Failure to Implement Discrepancy Status Tracking System as Defined in Special Engineering Procedure (SEP) 81-02 and to Control Progress of 79-14 Through Accurate Scheduling

Conclusion

The tracking system (Y-40) defined in SEP 81-02, "Implementation of NRC-OIE Bulletin 79-14 for Browns Ferry Nuclear Plant," was never fully implemented and was abandoned due to the time involved in maintaining it. No controlled system is presently being utilized to track and document the status of 79-14 discrepancies. The lack of control is further evidenced in past OE scheduling which projected all 79-14 related work for all units to be complete by May 1985. As of September 1984, work was only 31 percent complete. A controlled tracking system is necessary in order to identify the completion of the total scope of work. An accurate schedule which represents real progress is also necessary to complete the work by the revised completion date of 1990 (see sections IV.F.2a. and 2.b for details).

Recommendation

I-84-33-BFN-05A Identify and assimilate the current status of all discrepancies. Establish and implement a system which tracks 79-14 discrepancy resolution from analysis

through completed modification.

I-84-33-BFN-05B
Identify current status of all analysis and support design work which remains for the 79-14 effort. Track actual progress of work performed, document, and reflect in accurate schedules.

5. I-84-33-BFN-06, Failure to Revise SEP 81-02 to Adequately Define 79-14 Program

Conclusion

SEP 81-02 (R0) detailed the initial handling of discrepancies but failed to consider the coordination required to verify completion of field modifications for all discrepancies and the preparation of the final closure documentation for the 79-14 NRC report. Project management defined this completion as "Phase II" of 79-14, which could include a second inspection walkdown similar to the one conducted initially for 79-14. No draft revision to SEP 81-02 which documented this verbal description of Phase II was available for review (see section IV.F.3).

Recommendation

Revise SEP 81-02 to define activities required to complete and document this completion of the 79-14 program, to include a final comparison of as-constructed drawings to as-analyzed drawings.

6. I-84-33-BFN-07, Failure of OE Organization to Perform Generic Evaluation of Problems Identified in "Evaluation of WBN Piping Analysis Review Team(s) Report," (July 1982)

Conclusion

The OE organization failed to review all other piping analysis sections after a 1982 WBN Independent Review Team identified basic root causes for nonconforming conditions in WBN piping. Some of these basic root causes and conclusions were identified again in the BFN analysis section during the investigation (see section IV.G for details).

Recommendation

Review the problems, root causes, and corrective action in the WBN report and the conclusions of this report for applicability to all other projects.

7. I-84-33-BFN-08, Failure of QA to Perform Audits and Surveillance of Identified Problem Areas Within Piping Analysis Sections

Conclusion

No audits or surveillances have been conducted or are scheduled to be performed by QA organizations of the BFN piping analysis section. The Quality Management Staff performed a cursory review of the 1982 WBN root causes and documented their non-existence in 1984 based on the reorganization and special procedures written to eliminate these root causes. No audits/surveillances were performed to verify the implementation and effectiveness of these WBN procedures, or to determine the existence of these root causes in other projects (see section IV.G for details).

Recommendation

Perform periodic audits (including technical audits) to ensure that procedures and design criteria are being followed and to evaluate the effectiveness of the programs being implemented.

IV. DETAILS

A. Background Information

The piping analysis group performs one piece of work in the overall effort of verifying that the as-built configuration for seismic Category I piping systems meet design requirements. The design criteria used to perform the analysis are BFN-50-D707, "Detailed Design Criteria for Analysis of As-Built Piping Systems," and BFN-50-D717, "Detailed Design Criteria for Analysis of Torus Attached Piping (Long-Term Torus Integrity Program)." An EN DES-SEP 81-02, "Implementation of NRC-OIE Bulletin 79-14 for Browns Ferry Nuclear Plant," defines the coordination effort and responsibilities of the various sections involved in the 79-14 effort. An overview of the flow of work for the 79-14 effort and responsibilities of NUC PR, analysis support design, and modification are briefly stated as follows:

1. NUC PR site inspection personnel perform a "walkdown" inspection of the as-built piping configuration utilizing OE design drawings. OE piping drawings and support drawings are marked to show all discrepancies between as-built and design configurations.

2. Inspection data is transmitted to the OE BFN/civil structure section for review. Study packages are compiled by the responsible civil structure section. These packages contain a variety of information which is needed by the piping analyst such as walkdown information, equipment weights, and valve data. Study drawings are also made at this time from the marked-up inspection sheets and included in the study package.

3. Study packages are sent to the BFN piping analysis section who review and evaluate the need to perform a reanalysis of the piping system. (Note: All piping systems inspected to date have required seismic reevaluation and reanalysis.) Piping systems require reanalysis usually as a result of either no documented original analysis, mislocated piping and supports, or inadequate existing support configurations.

   The analysis section must interface with the civil structure section if all required information is not in the study package. Basically, the first three steps go in reverse as the civil structure section then requests the desired information from the NUC PR inspection group.

4. Initial reanalysis is completed and support loads are sent (usually as Preliminary) to the civil structure section. Existing supports are evaluated for new loads and modified if needed. This step requires significant coordination between the piping analyst and support designer to ensure a minimum of modifications while maintaining an overall safe operating condition of the piping system. The analyst completes analysis based upon agreed support conditions and then finalizes support loads.

5. Revised support drawings are issued by the civil structures group in conjunction with the analysis section's issuance of a load table, isometric drawing (drawing which illustrates the as-analyzed piping configuration), and summary book.

6. Required modifications are scheduled into specific outages by NUC PR. Complications in performing a modification are identified by NUC PR in the form of a Field Change Request (FCR). Depending on the complexity of the request, support designers and/or analysts may become involved in the approval. Reanalysis of system may be warranted.

7. Modifications are completed and site QC inspectors inspect and approve physical work. No record of the final approval is transmitted to OE.

B. Method of Investigation

In order to effectively evaluate the employee concerns, a combination of interviews, record review, and site verification was required.

1. Conduct of Interviews

   Interviews involving approximately 60 percent of the BFN piping analysis staff were performed. Similar questions were asked of each interviewee concerning that individual's experience and knowledge of piping analysis, awareness of applicable EPs and design criteria, methods used in perform-

ing an analysis and documenting/transmitting results, and understanding of the responsibilities delegated to the originator and checker of an analysis.

The relationship between the individual and the technical supervisor (C-4), supervisor (M-5), and support design group were also discussed due to the importance of interface in performing the work.

Interviews were conducted with personnel from the support design sections. Each person was asked questions concerning their background and experience, the preparation of study packages, interface practices between support design and analysis, method of interface between support designers and site inspection group, and the documenting of support design calculations and issuance of support drawings.

Interviews were also conducted with the civil project coordinator and civil project engineer to gain an understanding of the current status of all 79-14 work and the schedule/scope of the work remaining.

2. Review of Documents

The piping analysis section is performing an analysis on approximately 26 safety-related piping systems for each of the units at BFN. Selected analyses from each of the three units were reviewed at random. The review traced the initial study package information through the computer input and final output. Justifications of overstressed conditions were also evaluated. Preparers and/or checkers of the analysis were asked questions concerning the work and their understanding of the final results. Associated drawings (isometric and load table) were also reviewed to verify that they had been issued and that the information on them was consistent with the information presented in the analysis summary book.

Twelve analysis problems were selected at random for the initial portion of the document review. Twenty support calculation packages related to the above analysis problems were reviewed. Calculations were reviewed for consistency of information from analysis to support design (see section VI, "Docments Reviewed," for specific problems).

3. Site Verification

Documentation such as an "as-constructed" drawing of the final modification of the piping system was not available in OE. Therefore, to completely evaluate the transfer of information from initial walkdown to completed design, site verification was required. After reviewing randomly selected analyses and support calculations, it was determined

that modifications had been completed only on a limited number of systems. Selected supports from the EECW (unit 1), RHR (unit 3), and reactor drain and vents (units 1 and 2) were chosen for site verification. The verification process involved visual inspection and some measurements of support configurations.

C. BFN Piping Analysis Section

Personnel in the analysis section were extremely cooperative and assisted the investigators in the review of analysis summary books and related correspondence. Most seemed conscientious in the performance of their job and displayed a willingness to discuss programatic problems in order to improve the output of the section. The interviews and document review within the piping analysis section resulted in four adverse findings in the general areas of training, utilization of procedures/design criteria, and technical/administrative supervision.

1. Training of Personnel

The BFN analysis section contains a mixture of engineers and engineering aides at various grade levels. Approximately 50 percent of those interviewed had no prior experience in performing piping analysis before being placed in the BFN section, although 75 percent had previously attended a class in TPIPE (computer program for piping analysis) utilization.

These employees had received limited or no indoctrination to BFN piping analysis when entering the section. The supervisor and civil project engineer expressed the opinion that only qualified people would be hired and placed in the section and therefore should not require additional training. NSRS made no attempt to evaluate how qualified the employees were. It was observed, however, that many inconsistencies existed within the section concerning the understanding of stress intensification factors, insulation data, flange qualifications, normalized versus unnormalized loads, and capabilities of the TPIPE post processor.

The consistent conclusion of those interviewed was that the section contained a distinct core of experienced analysts who usually were aware of the current information on performing analysis. The remaining employees were left to assimilate information concerning the mechanics of performing analysis on their own. The situation was described as an inner group that interfaced frequently with the supervisor, and an outer group that somehow managed to gain information from the other group.

Employees were also attempting to utilize a Yellow Creek Nuclear Plant (YCN) Rigorous Analysis Handbook for guidance in analysis. Project management was aware of these hand-

books and stated that a cookbook solution does not always exist in analysis and that people should not depend on a handbook.

In summary, it was identified during the investigation that employees have not consistently received adequate training in performing BFN piping analysis, information transfer within the group is marginal, and YCN handbooks are being utilized to assist in performing analysis.

NSRS does not consider a rigorous analysis handbook to be a cookbook. In fact, a controlled handbook relating to BFN analysis could be a means of assuring a consistent, updated approach to analysis in a section that had limited information transfer, high employee turnover, and personnel with limited experience.

2. Utilization of Procedures/Design Criteria

All personnel interviewed were asked questions related to the procedures/design criteria utilized in the performance of their job. Most could identify the design criteria numbers of BFN-50-D707 (Analysis of As-Built Piping Systems) and BFN-50-D711 (Analysis of Torus Attached Piping - Long Term Torus Integrity Program) but could not answer questions dealing with specific requirements identified in the criteria. The technical supervisor (C-4) was also unaware of the specific requirements. Although it is emphasized that the analysis section lost two C-4 technical supervisors during 1984 and the present technical supervisor has been in the group for approximately three months performing basically administrative functions.

a. Deviations from Design Criteria BFN-50-D707

During the review of selected analysis problems and corresponding drawings associated with the EECW and the reactor drain and vent systems (N1-167-5RA, N1-D6B-A1, N1-167-3RA, N1-167-3RB, N1-110-1R, N1-210-1R), it was observed that the drawings had not been issued. This nonissuance is contrary to the requirement stated in section 8.2.1.g of the design criteria. Briefly stated, when an analysis is complete, the isometric and support load table drawings are to be issued and the analysis summary books microfilmed in MEDS. The analysis books had been microfilmed. It is emphasized that completed analysis problems are revised periodically. These revisions can affect the isometric drawing, which must reflect the configuration represented by a specific revision as well as the support load table drawings. Therefore, if the isometric and support load tables are not issued, a baseline does not exist for the analysis or support design. Three other systems

(RHR, CRDSD, torus penetration X-231) were reviewed to verify the issuance of drawings with the completed analysis summary book. The work related to the EECW and the reactor drain and vent systems were the only groups identified in which the drawings were not being issued. The engineer responsible for the EECW system explained that all EECW drawings would be issued together for each unit. There is no basis for waiting to issue drawings. There was no explanation given for not issuing the reactor drain and vent drawings. The nonissuance is considered a problem in document, design, and configuration control which could result in inadequate support design.

Drawings must be issued to represent each complete analysis with no exception as required by the design criteria.

b. Deviations From Engineering Procedures (EPs)

EP 3.03, "Design Calculations," requires an independent checker be used in the verification process for all calculations. This independence was not occurring in all problems reviewed. Personnel with no prior experience or training in piping analysis had been assigned to the lead engineer (for EECW) for training and became the "checker" of that engineer's work. Although the concept of using engineers to train and assist inexperienced people is good, independent verification is not achieved when the inexperienced person serves as a checker.

Analysis problem N1-DGB-A1 (R0) was found to be an example of a "trainee" checking the "trainer." The checker was questioned concerning specific methods used in this analysis and readily admitted to a limited understanding of analysis at the time the check was performed. Knowledge was confined to what the "trainer"/preparer had told the "trainee"/checker.

Therefore, the checker could not explain the rationale of methods used.

3. Inadequate Technical/Administrative Supervision

The staff of the BFN analysis section consisted of 25 individuals (excluding the M-5 supervisor) during the time of the investigation. The grades of the engineers and aides ranged from C-1 through C-4 to E-4 through E-6. Of those people interviewed, 100 percent stated that the newly acquired C-4 did not provide any technical guidance to the staff. Everyone was familiar with his background and chose to ask technical questions of other more experienced individuals

within the section. The C-4 had been assigned administrative tasks and limited himself to an administrative role, not being comfortable in providing technical assistance.

The role of a technical supervisor is important in this section. Considering the number of people with limited experience and the significant effect "engineering judgement" could have on the analysis results, technical expertise must be readily available when needed. The staff considered the supervisor to be technically competent but difficult to approach. The availability of a "Design Handbook" or a similar controlled document would reduce the load on the technical supervisor by providing a consistent approach to the work.

D. Support Design Sections

There were four sections within the BFN project which could generate support design calculations. Two of these sections were located at the plantsite, with one primarily responsible for field change request (FCR) approval. The major responsibility for original 79-14 design resulting from analysis remained in Knoxville after the October 1984 reorganization. Therefore, more review emphasis was placed on the Knoxville located sections and their interface with the piping analysis section. Personnel from both of these sections were interviewed to establish their work responsibilities, the interface conditions with the piping analysis section, the status of 79-14 support design and modification work, and the feasibility of obtaining accurate additional field inspection data when warranted.

Several problems were reviewed in the support design groups. The supports reviewed were limited to the RHR, EECW, and reactor drains and vents systems. These systems were the only ones which contained modifications that had been completed at the plantsite and provided a basis for review of the 79-14 effort from analysis through completed modification.

1. Interface wtih Piping Analysis Section

The support and analysis sections communicated informally throughout the analysis process. Initially the piping analyst depended on the support designer to provide a complete study package for each piping system in order to create an accurate model for computer analysis. The quality of inspection data and subsequent study packages had varied significantly, which resulted in frequent communication between analysis and support design. Typically, some valve information would be incomplete or inspection dimensions failed to add up between column lines. Some support designers admitted to addiug dimensions to the original inspection data when questioned by analysts. These dimensional additions were limited to inches and not considered signifi-

cant by the support designer. All support designers interviewed stated that the NUC PR site inspection group would be contacted for a reinspection of an area if the initial field generated sketches were not adequate. A log of all reinspection requests was reviewed to verify that coordination did exist between the site and support sections.

The NSRS investigators found no evidence of undocumented dimensions in the problems reviewed, other than the insignificant ones mentioned above. It is emphasized that although the process for receiving required information was tedius and frustrating, the majority of analysis employees interviewed displayed a conscientious attitude in trying to establish accurate models for their analyses.

Additional interface occurred after analysis was performed when overstressed conditions were identified. The analyst and designer discussed potential solutions required to reduce stresses in order to minimize the number of modifications required per system and to establish the feasibility of additional supports at specific locations. All analysts and support designers interviewed discussed an unwritten management policy of maintaining the present, as-built support configuration with no modifications for each piping system. The "policy" was seen by the majority of personnel as an emphasis to "sharpen pencils" and perform a more detailed evaluation of each support. Some employees identified a new management policy of sending Knoxville based personnel to BFN to ensure supports could be modified as planned before the support design drawings were issued. This was perceived to be a method of reducing the number of modifications. However, NSRS could not identify any evidence of employees signing inaccurate analyses or support design calculations due to these unwritten policies.

Although open communication played an essential role in the 79-14 effort, NSRS confirmed a potential problem of unverified support load information being transferred between the sections. Support designers attempted to design/modify supports based on preliminary support loads generated by the analysis section. Xeroxed copies of preliminary computer runs were given to support designers which enabled them to begin work. Support designers stated that the preliminary, uncontrolled information was usually needed as soon as possible in order to meet schedule requirements and support NUC PR outages. The designers explained that they eventually verified the uncontrolled information against the issued support load drawing before support calculations were issued. Supports R290, R291, R303, and R309 (located in EECW system analysis problems N1-167-3RA and N1-167-3RB) were selected by the investigators to verify this "load check" by support designers. There were notes made in the calculations which indicate that the "current" loads from

the unissued support load table drawings were evaluated. The support calculations were issued in June 1984 with no support load table being issued as of February 1985. No immediate safety concern was identified in the selected supports. However, the practice of leaving an open revision (consistently changing the drawing with revised analysis information without issuing the drawing for each revision) for the load table does represent a safety concern. The support calculations do typically reference the analysis problem number (such as N1-167-3RA) and/or the isometric drawing and support load drawing without stating the revision level. The section supervisor stated that support load table drawings did not have to be issued with support calculations because all support information was available through the analysis summary book. It is emphasized though that the analysis summary books do not contain load table information for each support. They do reference a computer printout which provides loads for each joint of the analysis model. Unfortunately, the analysis joint number and support number are not the same and the isometric drawing is necessary to locate the joint number which corresponds to a specific support. The isometric drawing had not been issued for the EECW system. Therefore, although the analysis summary book and related drawings are referenced in support calculations, the baseline for support calculations is not adequately documented without isometric and support load table drawings being issued. The nonissuance of isometric and load table drawings is contrary to the requirements in design criteria BFN-50-D707.

Four supports on the RHR system were also reviewed to establish the interface between analysis and support design. The associated analysis isometric and load table drawings had been issued for all four of the supports. Three of the four supports reviewed had support design loads which were greater than the load table loads. However, for RHR, unit 3, support H-34, the design load used was less than half of the load found in the current load tables. Support design calculations and analysis load tables need to be consistent and kept current to ensure the final support design is correct. In general, the interface information was being baselined for the RHR system, as required by design criteria BFN-50-D707.

Two supports on the reactor drains and vents system were reviewed to establish the interface between analysis and support design. Support H-21 for unit 1 (located in analysis problem N1-110-1R) and support H-21 for unit 2 (located in analysis problem N1-210-1R) were used to verify the "load check" by support designers. For problem N1-110-1R, the analysis summary book was issued but no checklist (as required by BFN-50-D707) was included. There was no reference in the summary book to an analysis isometric drawing or a

load table drawing. NSRS could not find an issued isometric or load table drawing for this problem. Also, for problem N1-210-1R, there was no issued analysis isometric drawing or load table drawing. These problems further illustrate the identified concern of using interface information from the analysis group by the support design group, which had not been issued and therefore was unbaselined.

2. Interface with NUC PR Field Modification Section

From interviews with the support group and field modification section, it appeared that the exchange of interface information was being carried out properly. The scope of 79-14 required information has changed over the years, which has resulted in additional reinspections. The requests for additional information have been handled promptly. In the areas reviewed, the supplied study package information was based on field inspections. In some packages the original walkdown information was not always complete. However, additional updates and resinspections by the NUC PR Modifications Section have made the study packages accurate and complete.

3. Review of 79-14 Related Support Calculations

The support design review showed that in most cases the criteria were followed and the acceptance criteria were met. However, there were examples of existing supports being accepted even though complete analytical calculations to qualify the support to revised analysis results were not available.

Also, some support calculations were identified which had incomplete stress calculations. Even though the loads were small, on support R-142, RHR, unit 3, there was no calculation of actual stresses. For support R-148, RHR, unit 3, the actual stress was not compared with the allowable nor was the weld stress calculated. The calculations are not complete without a comparison of actual stresses and allowable stresses. For support R-10, RHR, unit 3, the existing lug weld was qualified by dividing the allowable stress by 1.8 since the weld was a "partial" penetration weld. According to the design engineer, the justification for this factor on the "partial" penetration weld is given in a CEB memorandum (CEB 820830 010), although this memorandum is not mentioned in the calculations. The factor in the CEB memorandum was based on proposed ASME Code Case N-318. However, to use this value equations 10 and 11 of the proposed code case N-318 must also be satisfied. These equations were not included in the calculations for support R-10. These calculations do not provide complete comparisons of results to allowables and use an unqualified allowable stress for welds.

For support H-21, reactor drains and vents, unit 1, there was an inconsistency in the calculations of the Analyses and Pipe Support Design Sections and the modifications done by NUC PR. NSRS identified the following activities related to this support but could find no basis for them in the documentation provided.

- The computer output for analysis problem N1-110-1R showed a load of 652 pounds on the spring support H-21. This load exceeded the 550 pound capacity of the existing installed spring, a Power Piping size 705.

- In order to qualify this condition on the spring support, the analyst performed a hand calculation utilizing a revised load of 577 pounds on the spring. The 577-pound value was provided by the support group. There was no justification by the checker as to why the new number was used, although the final conclusion reached was that the spring capacity had not been exceeded with this load. The checker indicated that he was using the judgement and direction of the technical supervisor when accepting the value from the support designer.

- The computer output also had a value of 715 pounds handwritten next to the output value of 652 pounds. There was no explanation for this value which, coincidentally, was very close to the 718 pounds calculated for the H-21 support in unit 2 in analysis problem N1-210-1R. Three load values are listed in the analysis summary book without justification for any of them. Baselined analysis isometric and load table drawings, if available, would eliminate questions of which number was correct. (An evaluation of the analysis needs to be made to determine the correct value of load on the spring.) Due to the confusion of these actions, support H-21 (units 1 and 2) was selected for the site verification portion of the investigation. It was found that the H-21 spring for unit 1 had been replaced with one of adequate capacity if the output value of 652 pounds was correct (see section IV.E for details).

4. <u>"As-Built" Drawing Effort for 79-14 Modifications</u>

As of the date of this investigation, the "as-built" drawing effort on 79-14 modifications had not resulted in any drawings being sent back to the analysts for verification. The design people are relying on QC inspections to ensure that modification is according to the drawings. After the QC inspection, an "as-constructed" drawing will be produced by NUC PR. No one knew if these would be sent back for verification that the "as-constructed" configuration is the same as the "as-analyzed" configuration. This is necessary to

complete TVA's obligation under IE Bulletin 79-14.

E. Verification of Completed Modifications

As a part of the overall 79-14 program, modifications are made to the support system when overstressed conditions exist in the piping or the supports. NSRS attempted to verify that such modifications were made according to the drawings by performing a spot check at the BFN site of supports that had gone through the modification stage.

The first support reviewed in the field was on the unit 3 Residual Heat Removal System. The analysis for this support was done in problem N1-374-2R. The support was R148, which required a change from a 1/2-inch-diameter rod hanger to a Bergen-Patterson support. The modifications group provided a copy to NSRS of Workplan No. 13067, which showed that the support had been changed. Inspection of the support showed that it had been installed according to OE drawing 47A452-809, R1.

The next support reviewed was H-34 on unit 3 RHR system. The analysis, which was problem N1-374-3R, and support design results required this support to be replaced by a Power Piping size 607 with a spring setting greater than 551 pounds. Inspection in the field revealed that a Power Piping size 607 was installed with a spring setting of 625 pounds.

Two other supports that were selected for verification in the field were the Reactor Drain and Vent System, units 1 and 2, H-21. The analysis for unit 1 was N1-110-1R and that for unit 2 was N1-210-1R.

Access to the supports was not possible. Therefore, NUC PR workplans were used as a means of verification. The change has been made on unit 1 by workplan 10335. According to the workplan and "as-constructed" print, the hanger was changed to a Grinnell type A, size 7, with a cold load setting of 672 pounds, which is greater than the load of 652 pounds shown in the computer output. The bottomed-out spring (spring that has its maximum capacity exceeded) had been replaced with one that has adequate capacity according to analysis results. The problem with this particular support was in the baselining of documentation. There are no baselined analysis isometric drawings or load tables. Also, according to the analysis summary book, the support was qualified without needing modification. However, support modification drawings have been issued and the modifications completed without the analysis isometric or load tables, which is contrary to the requirements of the design criteria.

According to the modifications group, the H-21 support had not been modified on unit 2 nor was it scheduled for future outages. The Power Piping size 705 spring that was in place has a capacity of 550 pounds, which was much less than the 718 pounds shown in

the analysis. The spring can "bottom out" and had not been changed as of the date of this investigation. Drawing 47A465-124 R1 calls for the spring to be changed to a Grinnel Type A, size 8 with a capacity of 725 pounds which is adequate for the analysis load of 718 for this support. Again for unit 2, no isometric analysis drawing or load tables were issued. The configuration for unit 2 was very similar to unit 1. The load from the unit 1 analysis was 652 while the load from the unit 2 analysis was 718 pounds.

The last support modification that was verified in the field was EECW, unit 1, support R-299. This modification required a change from a flatbar support to a 3 x 3 angle support with a 1/2-inch baseplate. The inspection verified that the support was modified according to support drawing 47B451-104 R0. The analysis for this support was N1-167-3RB. There was a problem with the baselining of configuration because there were no issued analysis isometric or load tables for this already modified support.

Overall the field review showed that the required modifications were being made according to the issued drawings. However, support drawings are being issued without baselining analysis through issued isometric drawings or load tables. This is contrary to the design criteria and results in an "as-built" configuration that cannot be traced back to an "as-analyzed" configuration. This must be corrected since that is a fundamental requirement of the IE Bulletin 79-14.

F. NRC Bulletin 79-14

The employee concern which initiated this investigation addressed a limited portion of work involved in the 79-14 effort. During the course of the investigation many interpretations of the intent of 79-14 were presented to NSRS. There were also as many interpretations as to the status of 79-14 related work (analysis, support design, modifications). Therefore, NSRS expanded the scope of the original concern to include a review of the 79-14 program.

The review attempted to establish the history of the NRC bulletin, TVA commitments to the NRC, and management controls in place to meet these commitments. This review was limited to the OE organization.

1. General Overview of the Requirements of IE Bulletin 79-14

This section provides an overview of the intent of NRC IE Bulletin No. 79-14, "Seismic Analyses for As-Built Safety-Related Piping Systems." IE Bulletin No. 79-14 was transmitted from NRC to TVA on July 2, 1979. This version requested licensees and construction permit holders to verify that the seismic analysis applied to the actual configuration of safety-related piping systems.

Each licensee had to identify inspection elements to be used in verifying that the seismic analysis input information agreed with the actual configuration of safety-related systems. For portions of systems which were normally accessible, licensees had to inspect one system in each set of redundant systems and all nonredundant systems for conformance to the seismic analysis input information set forth in design documents. The inspection was to include pipe run geometry; support and restraint design, location, function; embedments; pipe attachments; and valve and valve operator locations and weights. The results of the inspection were to be submitted within 60 days of the date of issuance of the bulletin. The licensee was to inspect all other normally accessible safety-related systems and all normally inaccessible safety-related systems and report results of the inspections within 120 days.

If nonconformances were identified, licensees were to:

a. Evaluate the effects upon system operability under specified earthquake loadings and comply with applicable action statements in the technical specifications including prompt reporting.

b. Submit an evaluation of identified nonconformances on the validity of piping and support analyses as described in the FSAR or other NRC-approved documents. If reanalysis was necessary, a schedule for completing reanalysis, comparison of results to acceptance criteria, and description of results of reanalysis was to be submitted.

c. In lieu of b above, a schedule for correcting nonconforming systems so they conformed with design documents could be submitted.

d. Revise documents to reflect as-built configuration and describe measures to ensure future modifications would be in design documents.

The requirements of IE Bulletin 79-14 were changed by Revision 1, which was transmitted to TVA on July 18, 1979. This revision required the "as-built configuration" analysis verification for safety-related piping 2-1/2 inches in diameter and greater and to seismic Category I piping, regardless of size which was dynamically analyzed by computer.

Additional guidance and definition of action required were provided by NRC in Supplement 1 to IE Bulletin 79-14, which was issued on August 15, 1979. This supplement required, for systems selected in accordance with item 2 of the bulletin, the licensee to verify that the inspection elements

meet the acceptance criteria. The licensee was required to remove insulation or provide access where inspection elements could not be viewed because of insulation. Where physical inspection was not practicable, like valve weights and materials of construction, conformance was to be verified by inspection of QA records. Nonconformances were to be evaluated in two phases; by engineering judgment within 2 days followed by an analytical engineering evaluation within 30 days. Where either evaluation showed that system operability was in jeopardy, the technical specifications had to be met and the item 2 and 3 inspections had to be done as soon as possible.

The rest of the accessible safety-related piping systems then had to be inspected in the same manner as the selected accessible piping. All nonconformances had to be evaluated and justification for continued operation had to be provided. The nonconformances were to be evaluated using the same analytical technique or less complex ones if shown to be conservative. Licensees were to submit the evaluations and if they had shown that the analysis was nonconservative, licensees had to submit a schedule for reanalysis or correction of nonconformances. The schedule for correcting nonconformances had to be submitted along with improvements in quality assurance procedures to assure that future modifications would be handled efficiently. The schedule for action and reporting as given in the original bulletin was not changed.

Supplement 2 to NRC IE Bulletin No. 79-14 was issued on September 7, 1979. This supplement gave new guidance. On systems where the customer inspected against the latest revision of drawings, marked them as necessary to define the as-built configuration, and returned to the AE for comparison with seismic analysis input; the seismic analyst would be the one identifying nonconformances. Prompt notification of licensee was required on nonconformances in order to ensure that technical specifications were met. This supplement also stated where visual estimates were to be used that they must be within tolerance requirements. The supplement also required a minimum of 10 percent of obstructed supports be examined by removal of insulation. This was to satisfy the initial inspections with a schedule being provided for the rest of the supports. The supplement also required the measurement of clearances between piping and supports, attachments, and penetrations. If the licensee found areas that were considered impractical to inspect even with the reactor shutdown, he had to report these on a case-by-case basis. The schedule for action and reporting was not changed from the schedule in the original issue of the bulletin.

TVA's efforts to respond to this bulletin resulted in the

development of a criteria for the analysis, which is BFN-50-D707, "Design Criteria for Analysis of As-Built Piping Systems." TVA has attempted to implement the intent of the provisions of IE Bulletin 79-14 as revised and supplemented. The initial evaluation of the effects of nonconformances on the piping analysis and continued operation of the plant has been completed. Modifications were made in areas that jeopardized safe operation of the plant. After those were completed, TVA established a schedule for completing all the 79-14 work. Field sketches were made of the as-built piping and transmitted in packages to the analysts for evaluation. Some systems were analyzed because no previous analysis had been performed. Other systems had to be reanalyzed due to discrepancies or nonconformances. Once modifications are identified, the isometrics and new support drawings are to be issued. At the time of the investigation, none of the modifications had gone through a complete cycle from analysis to completed modification and back to the analyst to verify that the "as-built" (modified) configuration was the same as the "as-analyzed" configuration.

The IE 79-14 bulletin requires, for all safety-related piping 2-1/2 inches in diameter and greater and for all seismic Category I piping that was dynamically analyzed by computer, that the licensee verify the "as-built" configuration agrees with the analysis. The complete response to the bulletin will require that the final "as-built" drawings be compared to the analysis. No one was aware of how this was going to be done.

2. Scheduling/Control of 79-14 Work

Realistic, effective scheduling and control of all work related to the 79-14 is essential if TVA is to meet its NRC commitments (as described in section F.1). NSRS attempted to ascertain the status of all discrepancies, analyses, and support design. No one interviewed could provide this information due to the lack of a status tracking system and ineffectiveness of past scheduling. The status tracking and scheduling are discussed further.

a. Status Tracking

The SEP 81-02 defines the "Y-40" system as a "tracking device used to collect and report piping and pipe support documentation for Browns Ferry Nuclear Plant. This system includes the tracking of documentation required by NRC IE Bulletin 79-14."

Specific reports generated due to this system tracking are identified as the Discrepancy Status Report, Analysis Status Report, and Support Status Report. The Discrepancy Status Report was intended to track any

safety-related problems resulting from significant discrepancies and the status of any work required.

The Analysis Status report was intended to keep track of analysis problems on all three units and every system within BFN.

The Support Status Report was intended to document every support on every system on all units. Any redesign due to reanalysis would be identified as well as the verification of installation.

In actuality, the Y-40 system was never fully implemented and no evidence of any of the reports as previously described could be identified. The Discrepancy Status Report was replaced by a "system" of placing discrepancy sheets into a study package. At the end of the 79-14 effort, all study packages will be reviewed to verify that all discrepancies have been resolved. It is noted that these packages are uncontrolled documents that will not be reviewed until approximately 1990. The Analysis Status Report is presently a list of completed problems for each unit. An attempt to identify the status of all analysis problems was begun in December 1984. No documentation comparable to a Support Status Report could be identified.

In summary, the method of tracking all discrepancies and related work was to place this information in uncontrolled packages of information. Although attempts such as duplicating study package had been made to preclude losing the information, the system was inadequate. Verification of all discrepancies being resolved should not be postponed until the completion of 79-14 work nor dependent upon information which has been uncontrolled for years.

b. Scheduling/Control of 79-14

NSRS reviewed the scheduling of the work related to the 79-14 effort to determine the control project management had in place over analyses and support design. It was concluded that scheduling and control of this work has been ineffective and unrealistic.

Previous OE scheduling of 79-14 analysis/support design work can be described as "reactionary" in that a schedule and work priorities would be revised based on NUC PR's schedule of outages and modifications. OE typically responded to support NUC PR outages without the benefit of having any input to this NUC PR scheduling. Although a schedule had been established by OE showing the completion of all 79-14 work to be completed in 1985, the 79-14 program has historically been

given low priority and could not meet this completion date. (NOTE: The torus integrity program has been the main emphasis in analysis for the past four years.) The completion dates for design and analysis for each of the units were identified as follows:

Unit 1 - August 1, 1984
Unit 2 - May 1, 1985
Unit 3 - November 1, 1984

However, these dates were totally eliminated in September 1984 when a 79-14 scope of work document was issued by OE which included a work schedule that corresponded to NUC PR's six-year plan and extended the completion dates to approximately 1990. The scope of work document for 79-14 also stated that "EN DES is presently 31 percent complete." The NSRS investigators found no prior documentation which stated that 79-14 was not on schedule. It is emphasized that unit 1 and unit 3 should have been complete or near completion in September 1984 based on documentation such as a June 1983 summary of open items issued by BFN which stated that 79-14 work was progressing on schedule. The scope of work document also included percentages of completion for analysis and design for each system on each unit. Analyses showed 78 and 47 percent complete for units 1 and 3, respectively. These percentages could not be verified during the investigation. No one could provide information concerning the total number of analysis problems worked and those remaining for each system. In fact, the analysis section had just begun to identify and complete a record of all completed analysis problems in December 1984. As of February 1985, the cover sheets from each completed analysis summary book were still being located and compiled in a notebook. The status of other incomplete analysis problems (such as prepared, but not checked) was unknown. The analysis section had been working on unit 1 analysis until September 1984 and then totally shelved that work to begin on unit 2. No time was alloted to summarize the general status of that unit 1 work. Project management told NSRS that this major switch in unit emphasis was solely an economic problem beyond their control because they were attempting to support a NUC PR outage.

Past scheduling and control of the 79-14 effort has been unrealistic and overly optimistic based on the low priority given to this work. The ability of management to responsibly control the effort is questionable unless more realistic tracking of actual work is performed, documented, and reflected in accurate schedules.

3. Final Submittal to NRC

According to the requirements of IE Bulletin 79-14, TVA will have to provide analysis results of all the safety-related piping systems to NRC. The results will be from an "as-built" analysis for each system. Where the analysis shows there are discrepancies or nonconformances, these systems will have to be modified so that the systems meet the acceptance criteria. After all the modifications are complete, a final report must be issued to NRC to show that the "as-constructed" configuration and the "as-analyzed" configuration are the same. At the time of this investigation, a plan was not in place to provide this final closeout of the bulletin. SEP 81-02 (R0) only detailed the initial handling of discrepancies but failed to consider the coordination required to verify completion of field modifications for all discrepancies and the preparation of the final closure documentation for the 79-14 NRC report. Project management defined this completion of work as "Phase II" of 79-14, which could include a second inspection walkdown similar to the one conducted initially for 79-14. No draft revision to SEP 81-02 which documented this verbal description of Phase II was available for review.

A revision to SEP 81-02 should be developed to include a plan of tracking, coordinating, and verifying necessary modifications to and controlling documentation required for the final NRC 79-14 report.

G. OE/QA Audits/Surveillance of Piping Ayalysis/Support Design

NSRS included a review of the OE/QA organization in order to identify and evaluate the actions (audits/surveillance) QA had performed and had scheduled to perform in the piping analysis/support design area, and the conclusions drawn from these activities. Previous work identified in this area was limited to audits SS-82-2, D83-2 and 83V-26. Of these, SS-82-2 is the only one which briefly addressed piping analysis/support design work. This audit was conducted in various projects and branches and attempted to identify uncontrolled documents, utilized within OE, which defined requirements. Two of the five audit findings involved piping analysis/support design. One finding was identified in CEB and concerned uncontrolled WBN rigorous analysis handbooks. The other finding was identified in the BFN support design area. It stated that uncontrolled manuals for a computer program (SAGS) had been used by a BFN support design section. However, there was no attempt to evaluate the specific program and implementation of such which controlled piping analysis and support design. Other work identified by NSRS included an appraisal by an independent review team (which included one QA employee) of two task force reports on rigorous and alternate piping analysis for WBN. The two task force groups were to review the increasing number of NCRs for WBN and identify basic root causes.

The independent review team then evaluated the two reports and issued a report in July 1982 (QAS 820723 014). This report identified weaknesses in both the WBN alternate and rigorous analysis programs.

Some of the final conclusions presented by the independent review team are similar to those identified by NSRS during the BFN investigation.

The conclusions (with the corresponding similar conclusion identified in section III, "Conclusions/Recommendations" of this report) are stated as follows:

1. Inaccurate and incomplete piping analysis criteria (see I-84-33-BFN-02)

2. Documentation of WBN piping analysis stored in MEDS is incomplete (see I-84-33-BFN-01).

3. Procedures for performing verifying, and documenting WBN piping analyses do not exist (see I-84-33-BFN-02).

4. Training in the use and application of the design criteria, procedures, methods, and reports used to perform piping analysis is inadequate (see I-84-33-BFN-02B).

5. Inadequate schedule versus available manpower resulted in a lack of independent review (I-84-33-BFN-03).

6. Quality assurance auditors not auditing to the requirements of the controlling documents (I-84-33-BFN-07).

The report presents basic root causes and a number of WBN specific recommendations which would correct the situations, but does not perform a generic evaluation of the root causes to include all plants. In addition, no evidence of OE management utilizing this report as a basis for performing a generic evaluation of other plants could be identified during the investigation.

On October 3, 1984, a memorandum was issued by the Quality Management Staff (QMS 841003 202) which stated that all basic root causes identified in the July 1982 report had been resolved. This conclusion was reached after a "cursory" review had been performed. The rationale for this conclusion was essentially based on the issuance of two SEPs which addressed corrective action for the WBN 1982 report, the reorganization, and the performance of audit SS-82-2. There was no attempt made to verify the implementation of the corrective action addressed in the SEPs through audit/ surveillance activities. Also, audit SS-82-2 (previously described in this report as an audit conducted to locate uncontrolled procedures) cannot be construed to be a technical audit of the analysis program and the basis for

resolving the root cause of "auditors not auditing to controlling documents." Therefore, the basis for the overall QMS conclusion is questionable and unsubstantiated.

In summary, the total QMS involvement in the piping analysis/support design has been limited in SQN, WBN, and BLN, and nonexistent for BFN. Some positive actions have resulted from audits SS-82-2, D83-2 and 83V-26 concerning control and issuance of rigorous analysis handbooks and control of analysis information given to vendors who perform work for TVA. It is emphasized though that when very specific problems (such as those presented in the WBN report) were identified, QA made no attempt to pursue the conclusions/root causes and evaluate these for other projects. Also, the QA organization made no attempt to evaluate the implementation and effectiveness of corrective action before concluding that all WBN root causes had been resolved. No audits of BFN piping analysis/support design sections had ever been conducted or were scheduled to be performed in FY 1985.

V. PERSONS CONTACTED

A. Office of Engineering - Knoxville

1. BFN Civil Piping and Support Section (No. 4)

   Supervisor - B. C. Rosberg Draftsman - 2
   Engineering Aides - 7 Engineer - 9

2. BFN Civil Structures (No. 3)

   Supervisor - E. E. Cole Engineering Aide - 2 Engineer - 4

3. BFN Civil Structures (No. 5)

   Supervisor - F. D. Stidham Engineer - 3

4. Other

   E. G. Beasley - Chief, Quality Management Staff
   J. W. Beason - Engineer, BFN Civil Design Project Staff
   D. B. Bowen * - Division of Engineering, Projects Nuclear
   R. W. Cantrell * - Manager, Office of Engineering G. F. Dilworth, Jr. * - Division of Engineering and Technical Services
   P. A. Schrandt - Supervisor, Quality Management Staff
   D. A. Valentine - Engineer, Quality Management Staff

   * Attended exit meeting only

B. Browns Ferry Nuclear Plant

1. BFN Civil Structures (No. 2)

Supervisor - C. N. Simms Engineer - 3

2. Other

N. R. Beasley	Project Manager, BFEP
G. R. Hall	Manager, Design Services
J. M. Marshall, Jr.	Civil Project Manager
J. Traglia	NUC PR Modifications

VI. DOCUMENTS REVIEWED

A. Analysis Summary Books

Problem Identifier Number	Unit, System
N1-264-3R	Unit 2, Torus Purge Line of the HVAC System from Torus Penetration X-231
N1-364-3R	Unit 3, Torus Purge Line of the HVAC System from Torus Penetration X-231
N1-164-3R	Unit 1, Torus Purge Line of the HVAC System from Torus Penetration X-231
N1-DGB-A1	Units 1-3, Emergency Equipment Cooling Water System (EECW)
N1-167-5RA	Unit 1, Emergency Equipment Cooling Water System (EECW) Inside the Diesel Generator Building
N1-285-2R	Unit 2, Control Rod Drive Scram Discharge System (CRDSD)
N1-110-1R	Unit 1, Reactor Drain and Vent System
N1-210-1R	Unit 2, Reactor Drain and Vent System
N1-274-11R	Unit 2, Residual Heat Removal (RHR), Head Spray Piping
N1-167-1RA	Unit 1, Emergency Equipment Cooling Water System (EECW) Inside the Reactor Building
N1-167-3RB	Unit 1, Emergency Equipment Cooling Water System (EECW)
N1-167-1RA	Unit 1, Emergency Equipment Cooling Water System (EECW) Inside the Reactor Building

B. Supports Calculations

Support Number Identifier	Unit, System (Corresponding Analysis)

R-117, -140, -142, -148, -159	Unit 3, RHR (N1-374-2R)
H-272 and -273 (Temporary)	Unit 2, CRD (N1-285-2R)
R-300, -301, -302, -303, -309 -216, -297	Unit 1, EECW (N1-167-3RG)
R-301	Unit 1, EECW (N1-167-3RA)
H-213, R-299	Unit 1, EECW (N1-167-3RB)
H-34, R-10	Unit 3, RHR (N1-374-3R)
H-21	Unit 1, Reactor Drains and Vents (N1-110-1R)
H-21	Unit 2, Reactor Drains and Vents (N1-210-1R)

C. Procedures/Design Criteria

1. BFN-50-D707, "Detailed Design Criteria for Analysis of As-Built Piping Systems," BFN, Revision 2, May 18, 1983

2. BFN-50-D711, "Detailed Design Criteria for Analysis of Torus Attached Piping (Long-Term Torus Integrity Program)," BFN, Revision 1, May 21, 1984

3. EN DES-SEP 81-02, "Implementation of NRC-OIE Bulletin 79-14 for Browns Ferry Nuclear Plant," Revision 0, December 21, 1981

4. CEB-EP 21.12, "Class 2 and 3 Piping Analysis," Revision 3, December 12, 1978

5. CEB-EP 21.42, "Rigorous and Alternate ASME Class 1 and 2 Piping Analysis Verification," Revision 0, February 22, 1984

6. CEB-EP 21.43, "Documentation of ASME Class 2 and Class 3 Rigorous Piping Analysis," Revision 0, February 1, 1984

D. Memorandums and Reports

1. Memorandum from N. R. Beasley to G. R. Hall dated September 14, 1984, "Browns Ferry Nuclear Plant - Scope of Work for Bulletins 79-02 and 79-14" (BFP 840914 006)

2. Memorandum from M. N. Sprouse to H. J. Green dated July 26, 1982, "Browns Ferry Nuclear Plant - Revised Schedule for Completion of NRC-OIE Bulletins 79-02 and 79-14" (CEB 820826 018)

3. Memorandum from P. A. Schrandt to Quality Management Staff Files dated October 3, 1984, "Corrective Actions Addressing Basic Root Causes Contained in Evaluation of Watts Bar

Nuclear Plant (WBN) Piping Analysis Report" (QAS 820723 014) (QMS 841003 202)

4. Report on "OEDC Responsibilities in Resolving Findings and Open Issues Which Have Been Identified by NRC, NSRS, TVA QA, and Others - Overall Assessment for BWP"

5. Memorandum from R. O. Barnett to J. C. Standifer dated August 30, 1982, "Permanent Structured (LUGS) Integral Attachment Welds to Mechanical Piping" (CEB 820830 010)

TVA 64 (03-9-48)

UNITED STATES GOVERNMENT

Memorandum

Q01 85 0315 051

TENNESSEE VALLEY AUTHORITY

TO : C. Bonine, Manager of Construction, 12-108 SB-K
R. W. Cantrell, Manager of Engineering, W11A9 C-K

FROM : K. W. Whitt, Director of Nuclear Safety Review Staff, 249A HBB-K

DATE : March 15, 1985

SUBJECT: INVESTIGATION OF ALLEGED VIOLATION OF QUALITY ASSURANCE PROCEDURES BY THE OFFICE OF CONSTRUCTION AT SEQUOYAH NUCLEAR PLANT - NUCLEAR SAFETY REVIEW STAFF REPORT NO. I-84-34-SQN

The Nuclear Safety Review Staff (NSRS) investigated an employee concern in December 1984 and January 1985 at Sequoyah Nuclear Plant (SQN). The employee's allegation basically involved the failure to follow QA procedures by a certain named individual as well as himself not being allowed to follow QA procedures. He also alleged that alterations were made to QA training records and that the appropriate CMTR for a heat number was not available at the site.

The NSRS investigation of the employee's allegations substantiated the failure to follow QA procedures by the named individual but found no basis for his allegation of not being allowed to follow QA procedures as well as alterations to QA training records. Although the CMTR for the subject heat number was available at the site, NSRS review of the Heat Number Sort program revealed a number of issues that require further action by the Office of Construction (OC).

This report contains three recommendations. Recommendations I-84-34-SQN-01, Failure to Follow QA Procedures by an Individual at SQN, and I-84-34-SQN-03, Availability of the Material Certification and Requirements for Heat Number Sort Printout Entries, require responses from OC while recommendation I-84-34-SQN-02, Completion of TVA Mark Letter Description on Transfer Requisition Documents, requires response from the Office of Engineering (OE). You are requested to provide NSRS with the actions taken or planned to resolve these issues within 30 days of the date of this memorandum.

If you have any questions concerning the report, please contact Mansour Guity at extension 7637 in Knoxville.

K. W. Whitt

K. W. Whitt

MS MG:LM Attachment
cc (Attachment):
RIMS, SL26 C-K
B. M. Cadotte, E3C80 C-K (w/out attachment)
C. W. Crawford, 670 CST2-C
J. A. Nicholls, E5B29 C-K
H. G. Parris, 500A CST2-C
bc (Attachment):
E. G. Beasley, W12C61 C-K

NSRS FILE

QU1 '85 0315 052

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS REPORT NO. I-84-34-SQN

SUBJECT: INVESTIGATION OF ALLEGED VIOLATION OF QUALITY ASSURANCE PROCEDURES BY THE OFFICE OF CONSTRUCTION AT SEQUOYAH NUCLEAR PLANT

DATES OF REVIEW: December 20, 1984
January 15, 1985
January 29, 1985

REVIEWERS: Manson Guity — M. GUITY — 3/14/85 DATE

M. S. Kidd — M. S. KIDD — 3/14/85 DATE

APPROVED BY: M. S. Kidd — M. S. KIDD — 3/14/85 DATE

TABLE OF CONTENTS

I. BACKGROUND

The Nuclear Safety Review Staff (NSRS) evaluated an Office of Construction (OC) employee's concern at Sequoyah Nuclear Plant (SQN) in January 1983. The evaluation was documented in NSRS memorandum report No. I-83-04-SQN. The concerned employee alleged that there was willful and widespread violation of quality assurance (QA) procedures at SQN.

The same concerned employee contacted NSRS in July of 1983 and stated that he had been pressured into accepting a deficient condition. NSRS then conducted an investigation of the new allegation as well as the previous ones on August 10-12, 1984, and reported the conclusions in NSRS Report No. I-83-21-SQN.

Nuclear Regulatory Commission (NRC) Region II evaluated similar concerns, as a result of this employee's allegation, on October 30-31, 1984, and documented their conclusions in NRC report Nos. 50-327/84-33 and 50-328/84-33.

The same concerned employee contacted NSRS again and restated his allegations (indentical and/or similar issues as previously alleged) and expressed a desire for further investigation. The NSRS investigation team met with the concerned employee on December 20, 1984, at SQN. The employee's allegation and/or items of concern were discussed at length with the employee, and at the conclusion of the interview all items of concern were documented by NSRS investigators; these documented concerns were then reviewed and concurred by the concerned employee.

The employee's concerns were:

<u>Concern No. 1</u>

Individual "A" has not been following QA procedures since 1977, and also, individual "B" (the concerned employee) has not been allowed to follow QA procedures during the same time period. The following three examples were provided to support this allegation.

<u>Example A</u> - Cable reels were not meggered upon rereeling for the reels documented on Data Sheet 6 of Inspection Instruction (II)-32, "Inspection of Materials in Storage and Housekeeping Conditions," for report Nos. MIG-828, -827, 710, -755, -756, and -757.

Individual B stated that the "Data Sheet 6s" of II-32 for these MIG reports were not in the Material Inspection Group trailer and he did not know whether they were in the Quality Control Records Unit or not.

<u>Example B</u> - Cable identification information may not have been transferred properly when cables were rereeled. In some cases cable catalog numbers were used on the II-32 Data Sheet 6s instead of cable reel numbers (SNP numbers) as required by the procedures. The concerned employee stated that this violation occurred during the 1981 through July 1982 timeframe.

Example C - A reel of cable with TVA mark No. "WDU" was transferred to SQN from Watts Bar Nuclear Plant (WBN). TVA mark No. WDU is class 1E cable at SQN, whereas it is non-class 1E at WBN. Individual B was concerned that this reel of cable may have been used in a class 1E system. Individual B provided a form TVA 45D dated January 13, 1983, to the investigation team on this subject.

Concern No. 2

Quality assurance training was not always received as documented on the SQN Report of Training or Instruction form. He alleged that these records were altered by the instructors or clerks after the training sessions were completed and forms signed by the alleger. The alleged alteration dealt with addition/expansion of the description of instruction or training given. He provided documents dated July 18, 1983, and June 27, 1983, to illustrate this allegation.

Concern No. 3

Heat numbers for QA material (steel) may be entered into the "log book" without Certified Material Test Reports (CMTRs) being in the record vault. Heat No. 7438383 was provided as an example. He stated that he had not been able to locate CMTRs for three other heat numbers in the past but that he did not remember those numbers.

II. SCOPE

The NSRS objectives were to (1) obtain a precise and complete definition of the employee's concerns, whether it was the restatement of previously evaluated concerns or identification of new concerns, and (2) to evaluate, indepth, all items of concern as identified in (1) above, regardless of whether or not it had been previously evaluated, to determine the validity of the concerns and, if valid, the safety significance of the concern.

III. CONCLUSIONS/RECOMMENDATIONS

A. Concerns 1.A and 1.B

I 84-34-SQN -01, Failure to Follow QA Procedures by an Individual at SQN

Conclusion

The alleged failure to follow QA procedures by individual A for the two examples cited by individual B were substantiated (see section IV.A for details).

Recommendation

The NSRS recommends conduct of a survey of the individual A's activities to determine other areas where he could have potentially failed to follow QA procedures and their safety implications.

B. Concern 1.C

I-84-34-SQN-02, Completion of TVA Mark Letter Description on the Transfer Requisition Documents

Conclusion

The allegation that the individual B was not allowed to follow QA procedures was not substantiated. However, it was concluded for other reasons that additional intrasite controls were needed for the transfer of material between sites (see section IV.B for details).

Recommendation

The NSRS recommends that complete and full description of TVA mark letters for items be placed on the Transfer Requisition documents.

C. Concern 2

Conclusion

The allegation that the QA training was not always received as documented was not substantiated (see section IV.C for details).

D. Concern 3

I-84-34-SQN-03, Availability of the Material Certification and Requirements for Heat Number Sort Printout Entries

Conclusion

The specific allegation was not substantiated; however, there appear to be inconsistencies in the implementation of the compilation of required materials certification as well as ambiguity in the program established to control it (see section IV.D for details).

Recommendation

The NSRS recommends performance of an indepth review of the Heat Number Sort program at SQN in response to the issues detailed in section IV.3 as well as obtaining CMTRs for the subject thermowells.

IV. DETAILS

A. Concerns 1.A and 1.B

Individual B alleged that individual A had not been following QA procedures since 1977, and to substantiate this allegation, individual B cited two separate areas of activities performed by individual A where the alleged failure to follow QA procedures had occurred.

Concern No 1.A

Individual B alleged that individual A did not follow QA procedures during performance of inspection activities related to meggering cable reels upon rereeling of the cable and provided copies of II-32, Data Sheet 1, report Nos. MIG-710, -755, -756, -757, -827, and -828 to NSRS investigators to support this allegation. Individual B alleged that cable reels listed on the above listed MIG reports were not meggered when rereeled.

Concern No. 1.B

Individual B alleged that cable identifications may not have been transferred properly by individual A when cables were rereeled. In some cases, catalog numbers were used rather than SNP numbers on the storage inspection reports for the period of 1981 through July 1982.

Concerns 1.A and 1.B are closely interrelated and therefore are detailed jointly as follows:

SQN II-32, "Inspection of Materials in Storage and Housekeeping Conditions," specifies inspection requirements for all QA storage areas and for performing maintenance inspections on permanent QA material and equipment for housekeeping. This procedure requires the Materials Inspection Group (MIG) to perform housekeeping and storage inspection of all QA items from the time of receipt until the materials/items have left the control of the Warehouse Service Unit (WSU) supervisor. The results of these inspections are documented on the II-32 Data Sheet 1 and forwarded to Quality Control Records Unit (QCRU) for safekeeping. II-32 requires verification of cable storage to ensure it meets the requirements of SNP Construction Procedure No. P-12 (SNP CP P-12), "Storage of Material and Housekeeping Requirements," and that the cable reels are not damaged or deteriorated. SNP CP P-12 describes the methods to be used for the storage of QA items and housekeeping requirements as well as providing a listing of required frequency of inspections for all QA storage areas.

SNP CP P-12 further describes the method by which WSU is to discharge its responsibility of labeling of all insulated electrical cable reels. This responsibility includes assignment of an "SNP" unique reel number to the cable reel and placement/attachment of this "SNP" number, manufacturer's reel number, and TVA mark number on the reel itself. This SNP unique number provides the required traceability of the cable to the point of end-use from the storage areas as well as back to the contract as required by SNP CP P-12 and as such becomes TVA's unique cable reel identification.

The investigators reviewed II-32 Data Sheet 1s for the period of November 19, 1979, through September 28, 1984, which covered II-32 Data Sheet 1 report Nos. MIG-362 through -1043. Out of a total of 681 Data Sheet 1s reviewed, only 41 dealt with rereeling

of cable reels, 18 of which were authored by individual A starting from September 8, 1981, through September 28, 1984. Individual A had consistently identified cable reels with TVA cable identifications other than SNP number identification which is contrary to II-32. Only 6 Data Sheet 1 reports out of the remaining 23, authored by others, had failed to utilize TVA SNP numbers. In all of the 24 reports where SNP numbers were not used (18 by individual A and 6 by others), a TVA identification number known as "catalog number" had been used. The Standard Operating Procedure (SOP) No. 320, "Locating and Cataloging Permanent Material for Engineering Control," was issued in September 1979 to describe the method of identifying the existing material in warehouse storage (at that time only), determine material status as to use, and reordering shortages as necessary. This procedure required assignment of a catalog number to each existing cable reel and attachment of a tag with that catalog number to the reel as well. The catalog number tag on the cable reel was in addition to SNP number tag as required per SNP CP P-12 and as discussed earlier. An IBM Office System 6 (OS6) was utilized to store and maintain the required information per SOP 320.

The investigators determined that the information available through OS6 provided an easily accessible cross reference between any cable catalog number and SNP number. Furthermore, a hard copy of cable catalog information available through OS6 had been made and forwarded to QCRU for safekeeping and future cross references.

The result of interviews with individual A was not conclusive as to the reasons that individual A might have had for not listing SNP numbers, other than the fact that "most probably SNP tags were not easily visible at the time of inspections due to the improper method of placement of cable reels in the warehouse storage facilities." The investigators concluded that had this in fact been the reason, the inspector should have documented the improper placement of the cable reels as part of his inspection activities since he was charged with that responsibility as well. As a part of his inspection activities, his duties included ensuring that materials are placed in such a fashion that the material identification is easily visible and the material itself is readily accessible for removal/withdrawal. Therefore, the investigators concluded that the apparent reason could have very well been the individual A's lack of attention to the details of the procedure and/or not recognizing that the procedure required SNP numbers rather than catalog numbers.

The investigators concluded that the individual A had in fact failed to follow QA procedures for this particular activity as alleged by individual B. Although there was failure to follow a particular aspect of II-32 by individual A, this failure has not and could not have brought about a nuclear safety implication at all, specifically since the data base availability through OS6 provides adequate information for the traceability of cable reels and cross referencing to SNP numbers.

With regard to the nonperformance of meggering activities by individual A, the investigators initially ascertained the authenticity of the six II-32 Data Sheet 1 reports provided by individual B. These six data sheets were documentation of the inspection activities by individual A during the November 9, 1981 through August 9, 1982 timeframe.

Additionally, the investigators randomly selected one cable reel from each of 8 out of 18 II-32 Data Sheet 1s that had been authored by individual A as discussed earlier in this section. Through the OS6, the corresponding SNP numbers were found for the above cable reels as well as all those listed in the six examples that individual B had provided, and subsequently a search of MEDS and records still available in the QCRU was made to locate II-32 Data Sheet 6s for the above cable reels.

II-32, in part, requires documentation of cable rereeling on the Data Sheet 6 to indicate that the cable reel was rereeled and subsequently the cable insulation was checked with a megger for acceptability of insulation. The Data Sheet 6 requires documentation of the megger identification number as well as the calibration due date. Each inspector is required to complete the "daily megger check log" sheet when a megger is checked out to indicate the megger identification number as well as the inspector's signature attesting to having satisfactorily calibrated the megger prior to and after use.

The NSRS investigators could not locate any of the II-32 Data Sheet 6s for those cable reels randomly selected and could only locate Data Sheet 6s for 4 out of 21 cable reels that individual B had alleged were not meggered. These four Data Sheet 6s were not made available through MEDS or QCRU. They were located by individual A in the MIG's unit files. The daily megger check log was then reviewed to determine if a megger had been checked out by the individual A during the time period that the four cable reels were to have been meggered. The daily megger check log did not have any entry by individual A for that time period.

The investigators then reviewed the daily megger check log for the time periods that cable reels were to have been rereeled and the rereeling meggered by individual A. The daily megger check log sheets did not have any entry by individual A for that time period either.

Individual A explained, during the interview, that the reason for absence of entries in the daily megger check log sheet is that he had neglected to make the entries. He also stated that he had conducted the megger check and had prepared the required Data Sheet 6s and since he was not custodian of the QA records, he did not know what might have happended to those Data Sheet 6s.

The investigators concluded that there was no objective evidence to substantiate that the meggering was actually accomplished for

the above-mentioned cables and therefore determined that allegations made by individual B relative to the above subject matter are founded. It must be recognized that the cable insulation resistance test conducted during the storage period is not the final test. Each insulated cable is tested again for insulation resistance acceptability upon completion of cable pulling activities in the plant. Therefore, the investigators concluded that had there been some insulation damages during the rereeling activities that had gone undetected (since the cable reels were not meggered), the final meggering would have detected the insulation damage after the cable pull and prior to the cable being placed in service.

The NSRS recommends that an indepth survey of all activities performed by individual A be conducted to determine (1) if he had failed to follow QA procedures in other activities that he had performed and (2) the safety implication of those failures, if any.

B. Concern No. 1.C

Individual B alleged that he was not allowed to follow QA procedures from 1977 to present, and to illustrate his allegation he stated that a cable reel transferred from WBN to SQN may have been used in class 1E system. This cable reel, a non-class 1E cable at WBN, had an SQN mark letter for class 1E cable. During the subsequent interview process, he alleged that he had been told to accept the nonconformance report dispositioning prematurely.

The Standard Operating Procedure No. 310, "Requsition From and Return of Permanent Material to the Warehouse," defines the manner in which permanent materials are requisitioned from and returned to the warehouse. This procedure states that the responsible engineering unit shall review the Storeroom Requisition (form 575) to verify the availability of the material; and furthermore, the responsible engineer must stamp the form 575 with the appropriate QA stamp. The MIG inspector is responsible for witnessing the issuance of all QA material and determining the status of returned material.

SNP Inspection Instruction No. 30, "Receipt Inspection," describes the manner in which all permanent plant QA material shall be inspected when received onsite. This inspection instruction states that all electrical cable must be certified by the Division of Engineering Design (EN DES), presently Office of Engineering (OE), and listed on the certification sheet as certified. It further requires placement of all cables received without certification in nonconforming status until certification notification has been received from EN DES.

EN DES accomplishes this task through (1) requiring the shipping site to initiate TVA transfer compliance form TX1 and (2) requiring the receiving site to complete the TX1 form and forward

it to EN DES so that the process of certification of the cable for use in class 1E systems can take place. Once EN DES has received the TX1 and completed the certification process, then the cable certification notification will be transmitted to the receiving site. The receiving site will then release the cable from the nonconforming status.

Transfer Requisition No. 833041 was issued by EN DES to transfer one reel of cable 2000 feet, size 12 AWG, 1/c white, type CPJ, TVA mark letter "WDU" from WBN to SQN for Engineering Change Notice (ECN) L-5495. ECN L-5405 was issued in response to DCR 1007 which requested design and construction of an outage office and shop facility known as Field Services Building. This building is a non-Category I structure and does not contain any safety-related equipment nor does its addition impact safety-related systems, structures, or components. The Field Services Building is powered from non-Class 1E electrical system and is not served by any safety-related services.

WBN shipping ticket No. G194764 was issued on January 12, 1983, for shipment of one reel of cable, 2000 feet, size 12 AWG, 1/c, type CPJ, mark WDU, reel N30761, class II to SQN. The shipment arrived at SQN on January 13, 1983, and upon receipt inspection was nonconformed [nonconformance report (NCR) 2821] due to the fact that TVA mark letter "WDU" at SQN was size 12 AWG, 1/c, white, type CPJ, class 1E, and not class II (non-class 1E) as it was at WBN. A form 45D dated January 13, 1983, from design personnel in Knoxville to personnel in the Electrical Engineering Unit (EEU) at SQN acknowledged that TVA mark letter "WDU" identified non-class 1E cable at WBN where as it was class 1E at SQN and Bellefonte Nuclear Plant (BLN). This 45D directed the SQN EEU personnel to nonconform the cable and further stated that another transfer requisition was in the process to transfer the needed cable from BLN to SQN.

The recommended disposition for NCR 2821 was to designate the cable for use at the Field Services Building and return the excess, if any, to the warehouse for further dispositioning. This recommendation was discussed with EN DES personnel in Knoxville and approved on January 24, 1983. Based on this approval, the Quality Control (QC) inspector (individual B) removed the nonconforming status from the cable on January 27, 1983.

A Storeroom Requisition No. 1690 had been submitted on January 12, 1983, for the said cable reel to be used for work on ECN L-5495. This requisition was approved by SQN EEU personnel and the material released to the crafts foreman on January 31, 1983.

The Transfer Requisition No. 833041 was for work on ECN L-5405, Field Service Building. A non-class 1E cable requirement meets the needs for this type installation and there would have been no safety implications if all 2000 feet of this cable had been released and used within this building. NSRS concluded that all

2000 feet of the said cable reel was used for non-class 1E system and therefore the concern was not substantiated.

The potential of non-class 1E cable being installed in a class 1E system appears to be greater when cables are transferred from one site to the other site where the TVA cable mark letters do not define the same cable at both sites. It must be recognized that all TVA purchased cables have TVA cable mark letters printed on the outer cable jacket every few feet without any differentiation as to the plant for which it was purchased. If a cable, with permanent marking, at the use location must meet more stringent requirements than those the cable was purchased for, then additional management controls must be exercised to ensure proper control of end use and whereabouts of the excess cable.

Such controls were already in place and enforced appropriately. Although NSRS was unable to ascertain why class 1E cable was to be used for a nonsafety-related system, the receipt inspection process nonconformed the cable reel as it was discovered to be class II cable. A TX1 form was not processed for the cable as it was determined that the end use would be nonsafety-related application and TX1 process was not required. Had the receipt inspection failed to recognize the class II pedigree, a TX1 form would have been submitted to EN DES and the resultant response would have been acknowledgment by EN DES of lack of certification for use in class 1E system. Although these two major points of control appeared to be sufficient for this type activity, the intersite transfer of cables must receive additional attention as there are a number of other cable mark letters that are class 1E at one site and non-class 1E at other site(s).

NSRS recommends that transfer requisitions (either draft or final) contain full description of the mark letter of the item as defined in that site's Master Bill of Material description in order to preclude further recurrence.

The investigators concluded that the recommended method of the disposition for NCR 2821 was a proper and appropriate resolution and determined that if there were any prompting or pressure, it would have been justified and well within management rights to exercise. It is equally important to note that no indication of such pressure was detected by the investigators. The recommended dispositioning for the said NCR had been approved by the appropriate personnel, although individual B may not have agreed with it. Individual B's assigned responsibilities were to ensure that the disposition was as approved on the NCR and not necessarily concurring with it. As such, he was charged with ensuring that the cable reel was designated for use at the Field Services Building. In order for the cable reel to be removed from the nonconforming status and released for use at the Field Services Building, the NCR disposition had to be implemented and verified. Individual B was charged with performance of these verification activities.

The investigators concluded that this specific example did not substantiate individual B's allegation of having been told to accept the NCR 2821 disposition prematurely and therefore not being allowed to follow QA procedures.

C. Concern No. 2

Individual B alleged that the reports of QA training provided were altered to include materials that were not covered during the training sessions. He further alleged that the alterations occurred after he had signed the documents, and he provided copies of two such documents to the investigators.

The investigators conducted onsite interviews with nine individuals whose names had appeared on the copies of the "SQN Report of Training or Instructions" forms dated June 27 and July 18, 1983, which were provided to NSRS by individual B. Also, telephone interviews were conducted with two individuals who are presently at other TVA sites. These 11 individuals (including the alleger) had participated in the training sessions in the capacity of instructor and/or supervisor and/or trainee. The results of these interviews and review of documents, records, and procedures revealed the following:

All units conducted a safety meeting each Monday morning. The basic thrust of the Monday morning safety meeting was to discuss and disseminate the safety bulletins on safety (industrial) issues. These meetings generally lasted no longer than 15 to 20 minutes and were conducted by the supervisor or his designee within the unit's work area. Unit supervisors passed the safety bulletin around to the attendees, who then placed their signatures on the safety bulletin, and subsequently a copy was made and forwarded to the safety engineer's office for safe keeping. The attendees' signatures on the safety bulletin signified their attendance at the meeting where the said safety bulletin was discussed. Supervisors had been using a standard form memorandum to document the conduct of the safety meeting as well as listing the names of those in attendance. This standard form was a typed memorandum from the respective units to the safety engineer or unit's files with the subject matter being "Sequoyah Nuclear Plant Report of Training or Instruction." The body of the memorandum started with the "description of instruction or training given:" and on to the "employees receiving training or instruction" under which the names and social security numbers of all employees within the unit and a signature column for attendees appeared. This was followed by signature lines for the instructor and the supervisor attesting that the above-mentioned employees have received the subject training.

Some supervisors would have had the description of instruction or training given typed, others would have it handwritten, and at times it was a combination of a typed and handwritten description, but in all cases safety bulletins were listed as part of the content of the training.

The attendance roster, whether it was the safety bulletin or the standard form, was either passed around the room for signature when the training was being conducted or signatures were obtained at the conclusion of the session, at which time the names of those not in attendance were crossed over.

The instructor/supervisor discussed other work-related issues as needed and appropriate during these Monday morning safety meetings, such as revisions to procedures and the content of revised material that might have been beneficial and related to the group's work activities. The degree and depth of discussions around these subject matters was directly related to the need-to-know basis of the trainees and scope of their activities. Normally the supervisor/instructor accentuated the issues of importance to his unit or otherwise provided a copy of the revised procedures for review by the attendees. This part of the Monday morning safety meeting appeared to have been an informal method of dissemination and/or accentuation of relevant information to the unit personnel with a high degree of flexibility provided to the instructor as to the methods and styles. At times there might also have been a brief discussion of problems encountered by the unit personnel, procedural or otherwise.

Although the interviews conducted revealed that at times additions/modifications were made to the description of instruction or training given after the attendees had signed the attendance roster, all the interviewees, except the alleger, expressed confidence that the subject matter added must have been discussed during the Monday morning safety meetings. There are a number of reasons for additions/modifications to the "description of instruction or training given" after the attendees had signed the roster, in particular, for the two examples provided by the alleger. The supervisor of that unit had the "safety bulletin" typed on the form for the description of the instruction or training given as a matter of practicing efficiency and would hand write whatever else he covered during the session at the completion of the training session. The investigators determined that these forms and the process of documentation of the training material covered during the Monday morning safety meetings were not considered to be part of the formal QA training/indoctrination.

SNP CP P-48, "Personnel Quality Assurance Training," is the procedure that defines the QA training requirements for QC employees at SQN. This procedure requires documentation of QA training on attachment A of P-48. Attachment A is a standard form memorandum to the "Quality Control Records Unit" from individual units (unit designation is typed on this form by the units) with the subject line of "Sequoyah Nuclear Plant Report of QA Training or Instruction." The body of the memorandum includes space for "Description of instructions or training given," date of presentation, as well as the duration of presentation. Included in the body of this attachment is the listing of those

employees that received the training, instruction, or indoctrination followed by the signature lines for the instructor and supervisor attesting to the fact that the above listed employees received the training, instruction, or indoctrination described above. It was noted that attachment A of P-48 was a different form than the one used to document conduct of safety meetings. Furthermore, it was noted that this attachment did not require the attendees' signatures as did the form for Monday morning safety meetings.

It is NSRS's conclusions that the documents the alleger provided to substantiate this allegation are not offical QA records documenting QA training at SQN. They are records to document the safety meetings and materials covered during those meetings. Although it is concluded that there were alterations to the description contents of these instructions, NSRS concluded that there was no intent on the part of instructors or supervisors to falsify the records of instruction and that no procedures were violated by making such alterations.

D. Concern No. 3

Individual B alleged that the heat numbers for QA material (steel) may be entered into the SQN Heat Number Sort without CMTRs being in the records vault. Heat No. 7418383 was provided as an example.

SQN on May 21, 1971, received a shipment on contract No. 71057-52827 which included six pieces of structural steel plates with heat No. 7438383 and ASTM A-36 designation as permanent QA material. The TVA receiving report (form 209) indicated that mill certifications were received and were attached to the form 209. A request for the retrieval of the said mill certificate was made to the QCRU which promptly provided a copy of the Certificate Tests dated May 19, 1979. This certificate provides the physical and chemical test results of the subject steel and heat number and is signed by the Chief Metallurgist of the supplier who certified that the material has been tested in accordance with the contract specifications (ASTM A36) and the results conforms to the contract specifications.

NSRS could not substantiate this specific allegation, but in order to have a higher level of confidence, the investigators randomly selected five additional heat numbers from the SQN Heat Number Sort printout to ensure availability of CMTRs onsite. The five heat numbers are 428990, 434221, 19047, 2-TW-67-426, and 1815533, and further details about each heat number follow:

Example 1, Heat No. 428990 - Item 106 (class G material) of TVA contract 73C55-92789-6 was shipped by J. M. Tull Metals Company and received by SQN on December 1, 1972, with a signed metuallurgical report from Carpenter Technology Corporation. This report contains the results of chemical composition analysis and physical properties of the material for the above stated heat number.

The document certified that the listed data for the material are true and correct.

Example 2, Heat No. 434221 - Item 21 of TVA contract 76K-52-86714 -2 was shipped by Sandvik Steel, Incorporated, and received by SQN on September 23, 1975, with a Material Certificate signed by the Quality Assurance Manager. This certificate contains the results of the chemical analysis and a mechanical tests and certification statement attesting to the fact that the material with this heat number has been manufactured and tested in accordance to purchase order and requirements of listed specifications.

Example 3, Heat No. 19047 - Items shipped by Industrial Piping Company to SQN on September 29, 1972, included material with heat No. 19047. Daniel Bolt Company certificate of tests for the material with this heat number provides results of the chemical and physical properties of the said material and a certification statement that the material meets all the stated requirements on the certificate of tests. This certificate is signed by Daniel Bolt Company's Metallurgical Department.

Example 4, Heat No. 2-TW-67-426 - Item 1-2 of TVA contract 77K17-820638 was shipped by Weed Instrument Company and received by SQN on September 21, 1977. SQN received six Test Wells on this shipment as TVA class C, and upon receipt inspection a nonconformance report (NCR 464) was issued since the material's manufacturer's certificate of compliance (COC) was not received.

Weed Instrument Company's notarized certification of September 27, 1977, was supplied to TVA at a later date which became the basis for dispositioning of the NCR and releasing the Test Wells for use.

Review of the detailed requirements of the subject contract revealed specific call for CMTRs for Thermowells as well as the results of all required tests and examinations such as hydrostatic tests.

It appears that the receipt inspection failed to accurately identify the proper certification requirements at the time of receipt. As discussed previously, the contract certification requirement for the subject item was a CMTR and not a COC as stated in the NCR 464, and as a result, the subsequent efforts were expended to satisfy the NCR requirement and not the contract requirement. It should be noted that 2-TW-67-426 is not, in fact, a heat number as listed in the Heat Number Sort, rather TW-67-426 is TVA's instrument unique identifier of which two were purchased.

Example 5, Heat No. 1815533 - Heat No. 1815533 material supplied was for TVA contract 77K53-821308, Item No. 105, which was shipped by Capitol Pipe and Steel Prod and received at SQN on March 28, 1978, with a notarized certification from Lone Star Screw Company. This document provides the chemical analysis data

for the subject heat number and certifies that the data provided is a true copy of the data furnished by the producing mill and that it conforms to the requirements of the listed specifications.

As a result of reviewing the documents for the aforementioned examples, discussions with knowledgeable personnel, and review of other documents, the following was compiled:

History

SQN Heat Number Sort printout presumably includes all material received onsite that has a "good" heat number. A "good" heat number appears to be a material that has some kind of certification, MTR, CMTR, or COC. It appears that this program was established as a result of a task force recommendation some time in the mid-70s, and further history and/or information surrounding this printout is vague and very little could be found in the documents reviewed or interviews conducted.

Program

The present program requires the receipt inspector to ensure that the required certification, as listed or referred to in the contract, accompanies the material delivery. It also requires the receipt inspector to ensure the validity of the heat number or heat code by identifying that the number or code marked or stamped on the material agrees with that on the applicable material certification or if the number or code has been previously validated. He then attaches the certification documents to the form 209 which is eventually forwarded to QCRU. The QCRU personnel ensures that CMTRs or MTRs are available for the heat number prior to encoding the heat number into the printout.

This printout appears to be utilized for verification of heat number for issue and release, as well as the result of fitup and cleanliness inspection activities per SNP II-74 as documented on SNP II-74 data card. This data card, once submitted to QCRU, is reviewed to ensure that the heat number listed on it appears in the Heat Number Sort.

Discussions with plant personnel revealed that there may have been some confusion, at one time, about what might constitute a proper certification prior to entry of the heat number into the Heat Number Sort. It was also acknowledged that the QCRU personnel responsible for accepting the certification for a given heat number may not have had adequate knowledge to differentiate among MTRs, CMTRs, and COCs.

Although the present program requires CMTR or MTR prior to encoding the heat number into the program, it was difficult to determine what the past practices have been in view of the fact that some heat numbers with COC could have been encoded into the program such as the one revealed in example 4.

As a result, the investigators could not confidently determine (1) the purpose(s) for which the Heat Number Sort was generated, (2) the specific administrative controls designed and practiced to maintain the integrity and adequacy of the program, and (3) the complete scope of application and utilization of the printout.

Therefore, it is NSRS's recommendation that a search of documents/files/procedures be made so that a history of the Heat Number Sort can be reconstructed to include the following:

(1) The purpose/reason the Heat Number Sort was generated.

(2) Those management controls designed and implemented to administer the program.

(3) The function the program was designed to serve and the function it has been serving, if different.

(4) The confidence QC management has with the available information in the program (Heat Number Sort printout).

NSRS further recommends that appropriate actions be taken to obtain CMTRs for items listed for example 4.

V. LIST OF PERSONNEL CONTACTED

Name	Attended Entrance Meeting	Conducted During Investigation	Attended Exit Meeting
Richard M. Adams, Jr., Mechanical Engineering and Materials Unit		X	
Lillard R. Baily, Mechnical Engineering and Materials Unit		X	
Douglas A. Bateson, Quality Manager's Staff		X	
*James E. Blackburn, Mechanical Engineering Unit		X	
Claude E. Carson, Quality Control Inspection Unit		X	
Leo S. Cash, Quality Control Inspection Unit		X	
J. J. Chenkus, Jr., Safety Engineer		X	
Lindell D. Delius, Quality Control Inspection Unit		X	
Thomas W. Foley, Quality Control Inspection Unit		X	

	Attended Entrance Meeting	Conducted During Investigation	Attended Exit Meeting
C. E. Greek, Construction Engineer		X	
Ronnie E. Griffith, Mechanical Engineering and Materials Unit		X	
*Michael D. Harris, Mechanical Engineering Unit		X	
Frankey D. Henderson, Mechanical Engineering and Materials Unit		X	
Howard W. Knox, Quality Engineering Branch (OE)		X	
William H. Loomis, Mechanical Engineering and Materials Unit		X	
Sarah B. Miller, Quality Manager's Staff		X	
William Mita, Electrical Engineering Support Branch (OE)		X	
John Nicholls, Project Manager	X		X
Glenn A. Roberts, Mechanical Engineering and Materials Unit		X	
Heyword R. Rogers, Compliance Staff	X		
Melisa C. Shivers, Quality Manager's Staff		X	
*Lorna L. Wakefield, Mechanical Engineering Unit		X	

*This individual is no longer with SQN Office of Construction.

VI. REFERENCES

A. SNP Standard Operating Procedure No. 310, "Requisition from and Return of Permanent Material to the Warehouse," R4

B. SNP Standard Operating Procedure No. 320, "Locating and Cataloging Permanent Material for Engineering Control," R0

C. SNP Standard Operating Procedure No. 601, "Receipt Inspection of Permanent Plant Material," R2

D. SNP Inspection Instruction No. 30, "Receipt Inspection," R7

E. SNP Inspection Instruction No. 32, "Inspection of Materials in Storage and Housekeeping Conditions," R10

F. SNP Inspection Instruction No. 39, "Heat Code Transfer and ASTM Designator Transfer," R2

G. SNP Inspection Instruction No. 74, "Fitup and Cleanliness Inspection," R6

H. SNP Construction Procedure No. P-2, "Handling Nonconformances," R18

I. SNP Construction Procedure No. P-4, "Control and Calibration of Measuring and Test Equipment," R15

J. SNP Construction Procedure No. P-8, "Quality Assurance Records," R16 and Addenda 1 and 2

K. SNP Construction Procedure No. P-12, "Storage of Material and Housekeeping Requirements," R13

L. SNP Construction Procedure No. P-31, "Identification and Marking of Permanent Material," R2

M. SNP Construction Procedure No. P-34, "Heat Number Validation," R1

N. SNP Construction Procedure No. P-48, "Personnel Quality Assurance Training," R1 and R2

O. SNP Heat Number Users Guide, November 1977

P. SNP Records Retrieval Instruction, R0

Q. Account Procedure 1, "Receipt of Material," June 30, 1977

R. Encoding Heat Numbers Into Printout

S. NSRS Report No. I-83-04-SQN, "Sequoyah Nuclear Plant - Concerns Relating to Quality Assurance Activities within the Construction Electrical Engineering Unit"

T. NSRS Report No. I-83-21-SQN, "Employee Concern Regarding Violation of Quality Assurance Procedures by the Division of Construction at Sequoyah Nuclear Plant"

U. NSRS Report No. R-84-21-SQN, "Follow-up to Employee Concern Regarding Violation of Quality Assurance Procedures by the Division of Construction at Sequoyah Nuclear Plant"

V. Nuclear Regulatory Commission, Office of Inspection and Enforcement, Region II, Report Nos. 50-327/84-33 and 50-328/84-33

W. SNP Division of Construction Audit Report No. SN-C-81-07, "Orientation and Training"

X. SNP II-32, Data Sheet 1 reports, MIG-362 through -1043

Y. Daily Megger Check Log entries from 12/27/79 through 1/2/85

Z. SNP II-32, Data Sheet 6 reports

AA. Transfer Requisition No. 833041

AB. SNP Nonconformance Report No. 2821

AC. Shipping Ticket No. G194764

AD. Form TVA 575, Storage Requisition No. 1690

AE. SNP Engineering Change Notice No. L5495

AF. SNP Construction Specification No. N2G-877, "Identification of Structures, Systems, and Components Covered by the Sequoyah Nuclear Plant Quality Assurance Program," R3

AG. SNP Construction Specification No. N2M-865, "Field Fabrication Assembly Examination and Tests for Pipe and Duct System," R3

AH. SNP Design Change Request No. 1007

AI. SNP Report of Training or Instructions

AJ. SNP Report of Quality Assurance Training, Instructions, or Indoctrination

AK. Certification documents for the materials with the following heat Nos.: 7538383, 428990, 434221, 19047, 2-TW-67-426, and 1815533

001 '85 0325 052

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS REPORT NO. R-85-02-SQN/WBN

SUBJECT: SEQUOYAH AND WATTS BAR NUCLEAR PLANTS - SPECIAL REVIEW OF MANUFACTURER-IDENTIFIED POTENTIAL MISAPPLICATION OF SWAGELOK TUBE FITTINGS AT WESTINGHOUSE REACTOR SEAL TABLES

DATES OF REVIEW: JANUARY 22-23, 1985 - SEQUOYAH
JANUARY 24-25, 1985 - WATTS BAR

REVIEWER: G. G. Brantley — G. G. BRANTLEY — DATE 3/22/85

APPROVED BY: M. S. Kidd — M. S. KIDD — DATE 3/25/85

RJMS SL28 C-K

TABLE OF CONTENTS

LIST OF FIGURES

I. BACKGROUND

Recent events at Sequoyah Nuclear Plant (SQN) and at least one other pressurized water reactor (PWR) involving the seal table portion of the bottom-mounted incore instrumentation system (BMIIS) have heightened the interchange of information concerning activities associated with this system. A copy of a Crawford Fitting Company (CFC) distributor information exchange addressing potential misapplication of their SWAGELOK tube fittings for seal table high-pressure mechanical seal applications was provided to the TVA Board of Directors in November 1984 by a local vendor of SWAGELOK tube fittings (references VI.A.1 and 2). The NSRS was requested to perform a review of the concerns expressed in the distributor information exchange. The review was performed in January 1985 and consisted of an evaluation of regulatory, industry, and TVA documents; inspections of seal table equipment; and discussions with industry [Westinghouse (W) and CFC] and Office of Nuclear Power (NUC PR) personnel.

II. SCOPE

This special review was performed to determine the important points of the concerns expressed by CFC about potential misapplication of their SWAGELOK tube fittings on the W designed BMIIS seal tables, if those important points are pertinent at SQN and Watts Bar Nuclear Plant (WBN), and to what extent the pertinent points had been addressed by the line organizations. Actions taken to implement related recommendations made by W were also evaluated during the review.

III. CONCLUSIONS AND RECOMMENDATIONS

The NSRS concluded that some of the concerns expressed by CFC were pertinent at SQN and WBN. There was some evidence that use of the low-pressure seals during refueling operations had caused degradation of five high-pressure mechanical seals on SQN unit 2. Both plants had high-pressure mechanical seals on their seal tables (unit 1 at both plants) that were composed of mixed fittings from different manufacturers.

NUC PR personnel at SQN with assistance from W and CFC had performed recommended inspections and evaluations and had determined that both units were safe to operate with the existing conditions. No related problems had been encountered with the seals during subsequent operation. Work instructions had been revised to include some recommendation made by CFC and W to prevent degradation of seal integrity during maintenance activities.

WBN personnel had made modifications to unit 1 to assure that all the components of the high-pressure mechanical seals were from one manufacturer (CFC) and to minimize potential for seal degradation during maintenance and outage activites. The same modification will be made for unit 2 prior to startup of that unit. The maintenance work instructions for the BMIIS were only in the initial stages of preparation.

NSRS did not identify any conditions that constitute an immediate safety concern related to the operation of SQN or the startup of WBN. While the actions taken by the SQN and WBN staffs appear to be appropriate, some specific conclusions and recommendations are offered to enhance the consistency and clarity of related maintenance instructions, lessen the probability of seal degradation, and increase the awareness of the critical and somewhat delicate nature of these seals and tube fittings in general. Those conclusions and recommendations are as follows:

A. R-85-02-SQN/WBN-01(NUC PR), Office-Wide Awareness Bulletin for Tube Fitting Maintenance Activities

Conclusion

There have been recent and significant industry events involving failures of pressurized tube fittings during maintenance activities. There are some common identified contributors to these failures and to failures of tube fittings in general (see sections IV.B, C, E, F, and G for details).

Recommendation

A NUC PR office-wide awareness bulletin or similar mechanism should be prepared and distributed to the nuclear plants. The bulletin should discuss tube fitting design; assembly, reassembly, and inspection criteria; policy on interchanging components; failure modes (including those identified by the SQN and WBN maintenance craft personnel); hazards involved in working on pressurized fittings; and should specify special precautionary measures when maintenance on pressurized fittings is necessary. The desired bulletin should be incorporated into a permanent instruction at each plant for future awareness of new employees.

B. R-85-02-SQN/WBN-02(SQN/WBN), Maintenance, Operating, and Test Instructions

Conclusion

Instructions at SQN did not contain sufficient clarity, precautions, warnings, and other measures to provide the desired level of confidence that the high-pressure mechanical seals will not be degraded during maintenance activities or to lessen the severity of the consequences of a failed seal. WBN instructions for maintenance activities on the BMIIS had not been prepared at the time of this review (see sections IV.E, F, and G for details).

Recommendation

SQN

Applicable maintenance, operating, and test instructions should be revised as necessary to provide consistent guidance for system assembly, reassembly, and inspection of all SWAGELOK and mixed

fittings; address replacement of ferrule assemblies on previously undisturbed tubing; address lubrication and inspection of fitting threads to minimize or detect wearing, galling, and cross-threading; specify limiting forces while using the low-pressure seal; add cautions and warnings against interchanging fitting components, cross-threading, turning of fitting bodies, excessive forces, working on seals while the primary system is pressurized above atmospheric, and increasing primary system pressure while thimble tubes are disconnected from the overhead path transfer system.

WBN

WBN should incorporate the recommendations discussed above into their applicable maintenance, operating, and test instructions.

IV. DETAILS

The following information is provided to facilitate an understanding of the design and operation of the BMIIS, the nature of the seal fitting manufacturer's and NRC concerns, and the actions taken or to be taken by the SQN and WBN staffs to address the concerns.

A. Incore Instrumentation System

1. System Description

The tube fittings that are the subject of this review make up the high-pressure mechanical seals in the BMIIS (see figures 1 and 3). The BMIIS was designed to measure neutron densities at 58 different locations in the reactor core. The neutron densities are measured by 6 miniature fission chamber detectors that are driven into and withdrawn from the reactor core by electric drive units and cables through 58 small (0.201-inch inside diameter) stainless steel tubes called thimble tubes. Each thimble tube (~ 103-117 feet long) is housed in its respective guide tube which is 1 inch in diameter and is essentially an extension of the reactor pressure vessel. The guide tubes terminate at a common header type device referred to as the "seal table" (see figure 2). The thimble tubes are held in place against normal reactor operating pressure by high-pressure mechanical seals made up of tube fittings consisting of two ferrule assemblies (each assembly has two ferrules), a reducer union, and two nuts (see figure 3). Each seal establishes two reactor pressure boundaries (one on the guide tube and the other on the thimble tube). Improper assembly, excessive forces, or damage to these mechanical seals can cause leakage of reactor coolant and under severe conditions can cause partial or complete ejection of the thimble tube from its guide tube. Leakage from these seals cannot be isolated and it may be necessary to shutdown, cooldown, and depressurize the reactor to allow draining the coolant level in the reactor vessel below the elevation of the seal table to

stop the leakage. Ejection of the thimble tube can result if the reactor is at some pressure (partial ejection of a thimble tube occurred at another nuclear plant with the reactor pressure at 400 psi) and separation occurs between one of the ferrule assemblies and its respective guide or thimble tube. The ejection of the thimble tube can be partial if the overhead path transfer assembly is in place and the tubing is connected up. The entire thimble tube can eject from its guide tube if the reactor pressure is high enough, the separation of the ferrule assembly from its thimble or guide tube is complete, the thimble tubes are disconnected from the overhead path transfer assembly, and the assembly is rolled back from over the seal table.

2. Westinghouse Design of the High-Pressure Mechanical Seals

a. SQN

The reactor seal table and its associated high-pressure mechanical seals were designed and the components supplied to TVA by W. The W drawings supplied to TVA specified that the high-pressure mechanical seals at SQN were to be composed of SWAGELOK tube fittings manufactured by CFC (references VI.A.14 and 15). The systems at SQN were initially assembled using W specifications (reference VI.A.12).

b. WBN

The BMIISs including the seals at WBN had not been completely assembled. The W drawings supplied to TVA specified that the high-pressure mechanical seals could be constructed of SWAGELOK tube fittings manufactured by CFC or equivalent GYROLOK fittings manufactured by Hoke, Inc. (reference VI.A.16). The drawings indicate that the system is to be assembled in accordance with W specifications (reference VI.A.12). These specifications address SWAGELOK tube fittings only.

Following this review NSRS determined that assembly of the WBN unit 1 high-pressure seals was completed as of March 12, 1985, and consists entirely of SWAGELOK components.

3. Retraction and Reinsertion of Thimble Tubes During Refueling Operations

Before the irradiated nuclear fuel can be removed from the reactor vessel during refueling operations, it is necessary to fill the reactor vessel and refueling cavity with water to provide shielding against radiation and cooling for the fuel. The thimble tubes must be retracted from the reactor core region 15-20 feet into their guide tubes before any nuclear fuel is moved to prevent damage to the thimble tubes

or to the nuclear fuel. To accomplish this operation the water level in the reactor is lowered below the elevation of the seal table. The overhead transfer assembly is disconnected from the thimble tubes at the union flare fittings (see figure 3) and the overhead assembly is rolled from over the top of the seal table. The 5/8- or 3/4-inch (1/2-inch for WBN) range nuts that form the lower reactor pressure boundary seals and hold the ferrule assembly in place on the guide tubes are loosened and approximately 15-20 feet of each thimble tube along with the 5/16-inch reactor pressure boundary seal is pulled out of its respective guide tube. A low-pressure seal supplied by W and consisting of a slotted plug (must be slotted to fit around the thimble tube), two half-metal washers, and a rubber castrate ring not supplied by W are then installed around each thimble tube. The slotted plug is threaded into the 5/8-, 3/4-, or 1/2-inch range nut and tightened to form the low-pressure seal (see figure 4). The reactor water level can then be raised to fill the reactor vessel and the refueling cavity and movement of irradiated and new nuclear fuel can begin. The low-pressure seals prevent the water in the refueling cavity (higher than the seal table) from leaking at the seal table. Following refueling operations the low-pressure seals are removed, the thimble tubes are reinserted, and the high-pressure mechanical seals are reassembled for normal operation. Final integrity of the high-pressure seals after reassembly cannot be demonstrated until the primary system is repressurized after the refueling outage.

B. NRC IE Information Notice No. 84-55, "Seal Table Leaks at PWRs"

IE Information Notice No. 84-55 was issued by the NRC on July 6, 1984, to notify licensees of a potentially generic problem involving reactor coolant leaks from the BMIIS seal tables. The notice contained information relating to events involving leaks at SQN and at another nuclear plant. The following is a brief summary of the information contained in the notice relating to these two events.

1. SQN

Workers were in the seal table room brush-cleaning thimble tubes while the reactor was at 30 percent power. The thimble tubes were disconnected from the overhead path transfer assembly such that the high-pressure mechanical seal fittings (referred to as a "SWAGELOK fitting" in the notice) were the only devices restraining the thimble tubes. While one of the thimble tubes was being brushed a fitting broke loose ejecting the thimble tube from its guide tube.

2. Other Nuclear Plant

It was believed that slight bowing of a thimble tube caused the fitting (referred to in the notice as a "SWAGELOK fitting") making up the high-pressure mechanical seal to be

improperly seated, thus causing a leak of reactor coolant. When support devices holding the fitting in place were removed in an attempt to straighten the tubing the fitting "broke loose" at the guide tube causing an unisolatable reactor coolant leak of 18 gallons per minute (gpm). Subsequent examination of the fittings found that the ferrule assemblies on all but seven of the guide tubes had been displaced from their original positions. Review of the procedure for assembly of the high-pressure and low-pressure seals revealed that the low-pressure seal fittings could have displaced the ferrule assemblies toward the end of the guide tube (see figure 4) causing improper reassembly of the high-pressure mechanical seals resulting in the initial leak. Overtorquing the fittings while attempting to stop the leak probably overstressed the ferrule assembly and allowed the seal to break loose when the support devices were removed.

The NRC notice indicated that in both cases maintenance was being conducted on a high-pressure system with what was equivalent to single valve protection. The NRC recommended that licensees review their maintenance procedures to ensure that maintenance of any system under hot, pressurized conditions should be thoroughly evaluated before allowing personnel to perform the work and to ensure that maintenance under those conditions is minimized.

C. CFC Concerns as Expressed in the Distributor Information Exchange and Other Documents

1. CFC Distributor Information Exchange, "Westinghouse Nuclear Plant Seal Tables"

The CFC information exchange was sent to distributors of CFC SWAGELOK tube fittings in October 1984. In the information exchange CFC indicated that the problems that had been occurring at seal tables at several nuclear plants were not the result of SWAGELOK fitting failures, but had resulted from fitting modifications made by the reactor designer (W). The following is a summary of the CFC specified contributing factors to problems at the seal tables:

a. Use of the Low-Pressure Seal During Refueling Operations

Use of the W modified and supplied low-pressure seals had forced ferrule assemblies on the guide tubes out of their sealing position. Over-tightening of the range nut and the slotted plug could cause the range nut to act as a gear puller thus displacing the two-piece ferrule assembly out of its original position and up toward the end of the guide tube increasing the probability of seal failure when the system is pressurized after the refueling operations. The low-pressure seals were not standard SWAGELOK components but were SWAGELOK components modified and supplied by W.

b. Wall Thickness of the Guide Tubes

Heavier than recommended wall thickness of guide tubes could result in inadequate "SWAGEING" action (see figure 5) on the tube resulting in improper sealing and holding functions. When the SWAGELOK fitting is tightened on tubing, the two-piece ferrule assembly deforms the tubing in a manner that causes a slight indention of the tubing and a slight increase in the outside diameter of the tubing above the leading edge of the ferrule assembly. This action forms the seal and secures the ferrule assembly in place on the tubing.

c. Hardness of the Guide Tubes

Machining down the guide tubes from larger outside diameter (od) heavy wall tubing may increase the hardness of the tubing thus prohibiting proper "SWAGEING" as discussed above.

d. Interchanged (Mixed) Fitting Components

Fitting components had been mixed and resulted in other than all SWAGELOK components. According to CFC this violates the basic design of the SWAGELOK fitting as their fitting components are designed and manufactured to exact angles and close tolerances and are not designed to be used with components supplied by other manufacturers as those fittings' angles and tolerances may be different. CFC indicated that their position against mixing or interchanging parts of tube fittings was longstanding and had been clearly defined in their product literature. Additionally they include a "caution" card (see figure 6) stating their position in every box of fittings they ship.

e. Thimble Tube Expansion

The thimble tubes are made from undersized tubing which is expanded in order to use standard sized fittings. The expansion results in work hardening of the tubing which could prohibit proper "SWAGEING" of the tubing.

f. Replacement of Ferrule Assemblies

Ferrule assemblies had been cut from the guide tubes and replaced in exactly the same position where previous ferrule assemblies were when jacked out of position by the low-pressure seals. According to CFC this could prohibit proper "SWAGEING" action of the new ferrule assembly.

g. Misaligned Thimble Tubes and Cross-Threaded Nuts

Thimble tubes had been misaligned and installed with nuts cross-threaded on the smaller end of the reducer union. As many as three threads had been torn off the body.

The CFC summary of their concerns was that their SWAGELOK fittings were not being used as designed.

In a letter to the NRC in November 1984 (reference VI.A.3), CFC indicated that their concerns involving expanded thimble tubes and the hardness of the guide tubes had been resolved during discussions with W and that their SWAGELOK fittings when properly assembled per the recommended instructions in their catalog are compatible with and suitable for use in the W designed seal table. CFC indicated that the fitting that was involved in the thimble tube ejection at SQN had only a SWAGELOK body (union) with nuts and ferrules from another manufacturer. They took strong exception to the use of the term "SWAGELOK fittings" in the NRC Information Notice No. 84-55, in that the fitting involved was not entirely comprised of SWAGELOK components.

2. Nuclear Operations and Maintenance Information Service (NOMIS) Report 3298A, "Instrumentation, Incore Detectors, Experience with Wear and Galling of SWAGELOK Fittings"

CFC provided NSRS with a NOMIS report (reference VI.A.7) in which four nuclear plants had reported problems with wear and galling of threads of the high-pressure fittings at the seal table to the extent that some of the fittings had to be replaced.

CFC informed NSRS that some of their fittings are lubricated with a silver-based lubricant that provides permanent lubrication on the threads for the life of the fitting to minimize wear and galling, but the fittings used for W seal table applications were not supplied with this lubricant. This was confirmed by W during a discussion with NSRS. Both CFC and W recommend that a lubricant (W recommends Neolube) be used on the male threads for initial assembly and each subsequent reassembly to minimize wear and galling of the threads.

D. Westinghouse Technical Bulletin NSID-TB-84-09, "Primary System Leaks at Seal Tables"

A W technical bulletin (reference VI.A.8) was issued in October 1984. It discussed the two events relating to the PWR and seal tables. Additionally, it indicated that subsequent to the thimble tube ejection at SQN it was reported that in many of the seal table tubing fittings, nuts, and ferrules from one fitting manufacturer were being used with fitting bodies (unions) from

another manufacturer. It was pointed out that one plant had operated in that configuration since initial startup (2-3 years) with no leakage and that the fittings had experienced approximately 25 temperature and pressure cycles, including refueling outages and trips at power.

In the bulletin W recommended or indicated the following:

1. Recommended that thimble tubes be cleaned during a scheduled outage with the primary system depressurized.

2. Recommended that fittings always be reassembled per the manufacturer's instructions and cautioned that if the fitting body is turned the sealing surface of the fitting body and/or the ferrule assembly may be damaged, and the pressure retaining capability of the fittings could be degraded.

3. Indicated that the intended design was for the mating components of the seal table pressure retaining fittings to be from the same manufacturer. W indicated that they did not recommend operation with mixed fittings but that operation with mixed fittings was not a safety concern for the following reasons:

   a. One plant had operated with mixed fittings for 2-3 years with no leaks.

   b. No leakage was found during limited hydro testing conducted by W in 1972 on mixed fittings at 1.7 times the normal system pressure.

   c. The fittings experience only pressure loading during normal operation.

   d. Adequate safety systems exist to safely shut a plant down in the event the system malfunctions.

   e. There was no apparent connection between the mixed seal table fittings and the SQN thimble ejection incident.

4. Recommended that the ferrule assemblies on the guide tubes be inspected for signs of displacement at the next outage when the thimbles were retracted as a routine part of the refueling operations.

5. Recommended that if displaced ferrule assemblies were found that they be replaced and that they should be installed on a previously undisturbed surface of the guide tube.

6. Strongly recommended against any tightening or loosening operations on the seal table reducer union fittings while the primary system is pressurized.

E. Pertinent Points Raised by CFC, Westinghouse, and the NRC

The important points raised by CFC, W, and the NRC that are considered pertinent to the integrity of the high-pressure mechanical seals at the reactor seal tables and to safety while performing maintenance on tube fittings in general, along with the plant responses to them, are as follows:

1. Use of the Low-Pressure Seal During Refueling Operations

SQN, in accordance with W recommendations, inspected the ferrule assemblies on the unit 2 guide tubes during the recent refueling outage after the low-pressure seal plugs had been removed. They found some seal degradation (loose or displaced ferrules) on five guide tubes that could have been caused by use of the low-pressure seal cap. These ferrule assemblies were subsequently inspected by a CFC representative and the plant staff and were determined to be satisfactory for service. No problems were encountered with these seals during the startup of unit 2.

A guide tube ferrule assembly inspection step had been added to MI-1.9, "Bottom-Mounted Instrument Thimble Tube Retraction and Reinsertion," that should be effective in preventing reassembly of the high-pressure seals with displaced ferrule assemblies on the guide tube. However, step 5.1.16 of SQN MI-1.9 implies that for a newly installed ferrule assembly on the guide tube the ferrule assembly can be SWAGED with the low-pressure seal cap. It is doubtful that the low-pressure seal cap can properly SWAGE the newly installed ferrule assembly as it is not designed for that purpose. Step 5.1.17 allows snugging the low-pressure seal with a wrench if leakage occurs while raising the water level in the reactor cavity, but no maximum torque value is specified to prevent displacing the ferrule assemblies on the guide tubes.

MI-1.9 should be revised to clarify that the low-pressure seal cap cannot be used to properly SWAGE newly installed guide tube ferrule assemblies. Reference should be made to SMI-0-94-3, "Seal Table High-Pressure Seal Repair," for high-pressure mechanical repair or proper replacement of new ferrule assemblies on the guide tubes. Some maximum value past handtight should be specified for the low-pressure seal and a caution should be added to prevent exceeding the maximum value which could cause displacement of the ferrule assemblies on the guide tubes.

The above recommendation should be incorporated into the appropriate instruction for retraction and reinsertion of the thimble tubes at WBN as their instructions are prepared.

2. Interchanged (Mixed) Fitting Components

a. High-Pressure Mechanical Seal Makeup at the Seal Table

W had supplied the high-pressure mechanical seals for SQN and WBN seal tables. These seals were a mixture of SWAGELOK and GYROLOK, and the W drawings and specifications did not contain any precautions against mixing components.

SQN personnel had inspected the high-pressure seals on units 1 and 2. On unit 1 at least 37 of 58 high-pressure mechanical seals had a mixture of SWAGELOK and GYROLOK parts. Based upon engineering evaluations, SQN and W personnel concluded that operation with mixed fittings on unit 1 is not a safety concern. All the fitting components inspected on unit 2 were SWAGELOK, and there is reasonable assurance that the fitting components not inspected (ferrule assemblies on the thimble tubes) were also SWAGELOK.

WBN personnel had inspected the portion of the high-pressure mechanical seals that had been installed on unit 1 for cold hydro testing (guide tube ferrule assemblies, range nuts, and cold hydro caps). These fittings were a combination of SWAGELOK and GYROLOK components. A modification was accomplished on March 12, 1985 using all SWAGELOK components that will enable the seals to be disassembled and reassembled during and following refueling outages without disturbing the ferrule assemblies on the guide and thimble tubes that form the high-pressure seal boundaries. The fittings on unit 2 had not been installed. It is planned to install all SWAGELOK components with the modification on unit 2.

b. Maintenance Activities on Tube Fittings in General

NSRS interviewed mechanical maintenance engineering and craft personnel at SQN and WBN to determine their thoughts and experiences concerning interchanging (mixing) components of tube fittings. From the interviews NSRS determined that there was no formal policy established addressing mixing components of tube fittings. Maintenance craft personnel (steamfitters, foremen, general foremen, and a maintenance planner) indicated that they were aware that components should not be mixed but they were not the only craft that worked on fittings. They identified some common contributors to tube fitting failures that were encountered in their corrective maintenance activities on tube fittings. These contributors included mixed fitting components, ferrules installed backwards, no ferrules, ferrule assemblies made up of ferrules from different

manufacturers, fittings cross-threaded, and tubing not inserted properly to seat against the fitting body. The craft personnel interviewed recommended that some type of training be provided for all crafts involved in tube fitting maintenance activities. NSRS concurs with the craft personnel and recommends that the training be in the form of a NUC PR awareness bulletin to be shared with all the nuclear plants.

3. Replacement of Ferrule Assemblies

Review of SQN maintenance instructions associated with the BMIIS revealed that where replacement of defective ferrule assemblies was discussed there were no provisions in the instructions to assure that the new ferrule assemblies would be installed on previously undisturbed tubing. Where replacement of defective ferrules is addressed in maintenance instructions it should be specified that new ferrules should be replaced on previously undisturbed tubing or a special evaluation should be required to determine that replacement in the previous position is acceptable. Guidance for the evaluation and acceptance criteria should be given.

4. Wearing, Galling, and Cross-Threading of Seal Table High-Pressure Mechanical Seals

The SQN Maintenance Instruction (MI-1.9) used for retraction and reinsertion of the thimble tubes during refueling operations and which also provides instruction for disassembly and reassembly of portions of the high-pressure seals did not require the application of a lubricant during reassembly of the seals nor did it contain a caution to warn against cross-threading. A suitable lubricant should be selected and specified for use on the high-pressure seal during reassembly applications to minimize wearing and galling and a "caution" should be added to warn against cross-threading. Inspections for wearing, galling, and cross-threading should be required before reassembly of the seals.

5. Maintenance on Thimble Tubes and Seals While Pressurized

SQN had incorporated a prerequisite into their maintenance instruction for cleaning thimble tubes that will assure that cleaning activities are initiated only while shutdown with the primary system depressurized. The portion of MI-1.9 that allowed tightening of the high-pressure seals to stop leakage while the primary system was pressurized had been removed from that instruction.

6. Manufacturer's Reassembly Instructions (Criteria)

See section IV.F of this report.

F. High-Pressure Mechanical Seal Initial Assembly, Reassembly, and Inspection Criteria

1. CFC Criteria

CFC had established in their product literature (reference VI.A.5) initial assembly and reassembly criteria for their SWAGELOK tube fittings for normal applications and additional assembly criteria for their fittings used for high-pressure application/high safety factor systems. They provide a "gap inspection gage" (see figure 7) to be used to assure proper pullup on both applications (normal and high safety factor systems) for initial assembly only. Their criteria and the inspection gage can only be used on fittings made up of all SWAGELOK components and is not applicable for fittings made up with mixed components or fittings made up entirely with components from another manufacturer.

2. Westinghouse Criteria

The criteria specified by W (see references VI.A.12 and 13) for initial installation of the high-pressure mechanical seals was very similar to the CFC criteria for high safety factor systems and made reference to the SWAGELOK catalog. There was no initial assembly or reassembly criteria specified for all GYROLOK fittings or mixed fittings of SWAGELOK/GYROLOK components supplied by W. The criteria for cold hydro caps which initially establishes the seal on the guide tube is not in accordance with the CFC criteria for high safety factor systems even though once installed the initial and permanent seal on the guide tube is established. No criteria was specified for the GYROLOK hydro caps that were supplied to WBN and subsequently installed on unit 1.

The reassembly criteria is inconsistent with the CFC criteria in that a "snugging" step to assure that the nut is tightened past the previous position is omitted.

The W criteria omits the gap inspection gage to insure proper fitup on initial installation.

3. SQN and WBN Criteria

The criteria specified in instructions for a new thimble tube installation and high pressure mechanical seal repair is based upon the W initial assembly specifications and repair procedures which are in some cases inconsistent or have omissions. There are no criteria specified for the mixed component seals. SQN is currently considering replacing the mixed component high-pressure mechanical seals with all SWAGELOK components to allow use of the applicable CFC criteria.

The SQN instructions for maintenance activities on the high-pressure mechanical seals at the seal tables should be reviewed and clarified to establish consistent criteria for initial assembly (high safety factor system), reassembly, replacement of new ferrule assemblies on previously undisturbed tubing (or an alternative evaluation), and gap inspection gage testing for initial assembly and reassembly. It should be made clear that the gap inspection gage has only limited value on reassembly applications and would only identify a grossly loose fitting. Before the next refueling outage reassembly criteria should be specified for the seals with mixed components and if different from the CFC criteria for SWAGELOK tube fittings, the seals with mixed components should be uniquely identified in the instructions. An alternative would be to change out all the mixed seals to seals made up of components from one manufacturer using criteria specified by that manufacturer.

G. Other Precautions, Warnings, and Measures to Prevent High-Pressure Mechanical Seal Degradation

Some cautions and measures had been added to SQN instructions to prevent degrading the seals during maintenance activities. However, sufficient cautions and warnings had not been added to provide a high degree of confidence that the seals would not be degraded during maintenance activities. The following cautions or measures should be incorporated into applicable plant instructions at SQN and appropriate existing or new instructions at WBN to warn against causing damage to the high-pressure mechanical seals and to minimize the consequences of a severe seal failure:

1. The W caution note (from reference VI.A.8) to prevent turning of the fitting body while tightening fitting nuts should be included in SMI-0-94-3 and SMI-1-94-5, "Thimble Tube Installation."

2. A caution note should be added to MI-1.9, MI1.10, "Incore Flux Thimble Cleaning and Lubricating," SMI-0-93-4, and SMI-1-94-5 that indicates that excessive forces on the high-pressure mechanical seals at the seal table can cause seal degradation and subsequent failure at pressure. The nature of the forces should be identified (e.g., bending, torque, etc.).

3. A caution note should be added to the four instructions addressed in item 2 that warns against any maintenance on the seals while the primary system is pressurized above atmospheric pressure. The caution should indicate that any tightening or loosening of the seals with the primary system above atmospheric pressure requires an unique procedure reviewed by PORC and approved by the Plant Manager on a case-by-case basis.

4. MI-1.9 should be revised to require an inspection of the threads of the fittings for galling, wearing, or cross-threading, and application of an approved lubricant to the male threads on the high-pressure seal fittings before reassembly.

5. A precaution should be added to SQN's GOI-1, "Plant Startup from Cold Shutdown to Hot Standby," SI-146, "Reactor Coolant System Leak Test," and SI-250, "Reactor Coolant System Hydrostatic Pressure Test," which warns against increasing reactor pressure above atmospheric with the thimble tubing disconnected from the path transfer unit. If a seal fails during pressurization of the primary system causing ejection of a thimble tube, the consequences should be less severe with the path transfer unit connected to the thimble tube.

V. LIST OF PERSONNEL CONTACTED

A. Industry

1.	L. L. Dietz	Ridge Valve and Fitting Company
2.	B. Flusche	Crawford Fitting Company
3.	W. D. Wilson	Crawford Fitting Company
4.	K. A. Kloes	Westinghouse
5.	R. U. Mathieson	Westinghouse
6.	S. Groth	INPO

B. Office of Nuclear Power

1. Division of Nuclear Services

a.	J. H. Fox	Metallurgical Engineer

2. SQN

a.	H. L. Abercrombie	Site Director
b.	R. E. Alsup	Compliance Section Supervisor
c.	G. S. Boles	Mechanical Maintenance Engineering Supervisor
d.	R. D. Bates	Steamfitter Foreman
e.	M. R. Harding	Engineering Group Supervisor
f.	G. B. Kirk	Acting Compliance Section Supervisor
g.	J. B. Krell	Maintenance Superintendent
h.	K. E. Lewis	Mechanical Maintenance Engineer
i.	D. L. Love	Supervisor Mechanical Maintenance Section
j.	J. Leighton	Steamfitter
k.	M. D. Pickard	Steamfitter
l.	D. C. Queen	Mechanical Maintenance Engineer
m.	P. R. Wallace	Plant Manager

3. WBN

a.	S. M. Anthony	Compliance Section Engineer
b.	E. A. Elam	Mechanical Maintenance Engineer

c.	G. W. Hurley	Mechanical Maintenance General Foreman
d.	S. E. Jenkins	Assistant Power Storeroom Supervisor
e.	P. C. McCulley	Power Storeroom Supervisor
f.	C. D. Nelson	Acting Mechanical Maintenance Supervisor
g.	K. L. Reed	Mechanical Maintenance Planner
h.	R. C. Sauer	Compliance Section Supervisor

VI. REFERENCES (DOCUMENTS REVIEWED)

A. Industry

1. Letter to C. H. Dean, Jr., from Lawrence L. Dietz, Ridge Valve and Fitting Company, "Westinghouse Seal Tables," dated November 14, 1984 (GNS 841130 100)

2. Crawford Fitting Company Distributor Information Exchange, "Westinghouse Nuclear Plant Seal Tables," dated September 1984.

3. Letter to Edward L. Jordan, USNRC, from William D. Wilson, Crawford Fitting Company, dated November 16, 1984

5. Swagelok Tube Fitting Catalog No. C-983, Crawford Fitting Company, 1983

6. Swagelok, "A Report on Initiation and Interchange," Crawford Fitting Company," 1984

7. Nuclear Operations and Maintenance Information Service Report 3298A, "Instrumentation, Incore Detectors, Experience With Wear and Galling of Swagelok Fittings," November 1, 1983

8. Technical Bulletin NSID-TB-84-09, "Primary System Leaks at Seal Tables," Westinghouse Nuclear Service Divison," October 11, 1984

9. Westinghouse letter number MED-PTE-1108(84) to R. Mathieson from G. J. Ohare, "TVA Memo From P. R. Wallace Dated 12/10/84," dated December 14, 1984

10. Westinghouse letter to R. Mathieson from R. Howard, "Seal Table Fittings Intermix - SEQ 1," dated May 2, 1984

11. Westinghouse letter number MED-PTE-1029(84) to R. Mathieson from G. J. O"Hare, "Seal Table Fitting Inspection," dated November 14, 1984

12. Westinghouse Specification 616A230, "Bottom Mounted Instrumentation Assembly Specification," Revision 3, dated January 9, 1971

13. Westinghouse Procedure No. MP 2.3.1, "Seal Table High Pressure Seal Repair," Revision 1, May 19, 1983

14. Westinghouse drawing number 113E516, Sheet 1 of 4, "TVA (Sequoyah Unit No. 1) Instrumentation - Bottom Mounted," TVA Approved February 16, 1972 (As-Designed)

15. Westinghouse drawing number 113E767, Sheet 1 of 4, "TVA (Sequoyah Unit No. 2) Instrumentation - Bottom Mounted," TVA approved August 26, 1980

16. Westinghouse drawing number 1096E91, sheet 1 of 4, "WAT-WBT (Watts Bar Units 1 and 2) Instrumentation Bottom Mounted"

B. Regulatory

1. USNRC IE Information Notice No. 84-55, "Seal Table Leaks at PWRs," July 6, 1984

C. TVA

1. NSRS

a. Memorandum from H. N. Culver to W. F. Willis, "Response to Board Comment - Sequoyah and Watts Bar Nuclear Plants - Manufacturer-Identified Misapplication of Swagelok Tube Fittings at Westinghouse Seal Tables," December 10, 1984 (GNS 841210 050)

b. Memorandum from K. W. Whitt to J. P. Darling, "Sequoyah and Watts Bar Nuclear Plants - Manufacturer-Identified Potential Misapplication of Swagelok Tube Fittings at Westinghouse Reactor Seal Tables - Nuclear Safety Review Staff (NSRS) Report No. R-85-02-SQN/WBN," January 15, 1985 (GNS 850115 050)

2. Power and Engineering

a. Division of Operations Support

(1) Central Laboratories Report No. M86-84-0110A, "Tube Fittings for 1/4-inch O.D. SS Tubing, Sequoyah Nuclear Plant, Unit 1 - N191," dated October 10, 1984

b. Office of Nuclear Power

(1) Division of Nuclear Services

(a) Draft Report, "Sequoyah Nuclear Plant Unit 1, D-12, Traveling Incore Probe Thimble Tube Separation Special Tests," May 17, 1984

(2) Sequoyah Nuclear Plant

(a) MI-1.9, Revision 4, "Bottom Mounted Instrument Thimble Tube Retraction and Reinsertion," December 14, 1984

(b) MI-1.10, Revision 1, "Incore Flux Thimble Cleaning and Lubrication," December 12, 1984

(c) SMI-1-94-5, Revision 1, "Thimble Tube Installation," May 25, 1984

(d) SMI-0-94-3, Revision 0, "Seal Table High Pressure Seal Repair," November 12, 1984

(e) SI-146, Revision 12, "Reactor Coolant System Leak Test," May 1, 1984

(f) SI-250, Revision 2, "Reactor Coolant System Hydrostatic Pressure Test"

(g) GOI-1, Revision 50, "Plant Startup from Cold Shutdown to Hot Standby", January 7, 1985

(3) Watts Bar Nuclear Plant

(a) GOI-1, Revision 38, "Plant Startup from Cold Shutdown to Hot Standby Unit 1 or 2," January 31, 1985

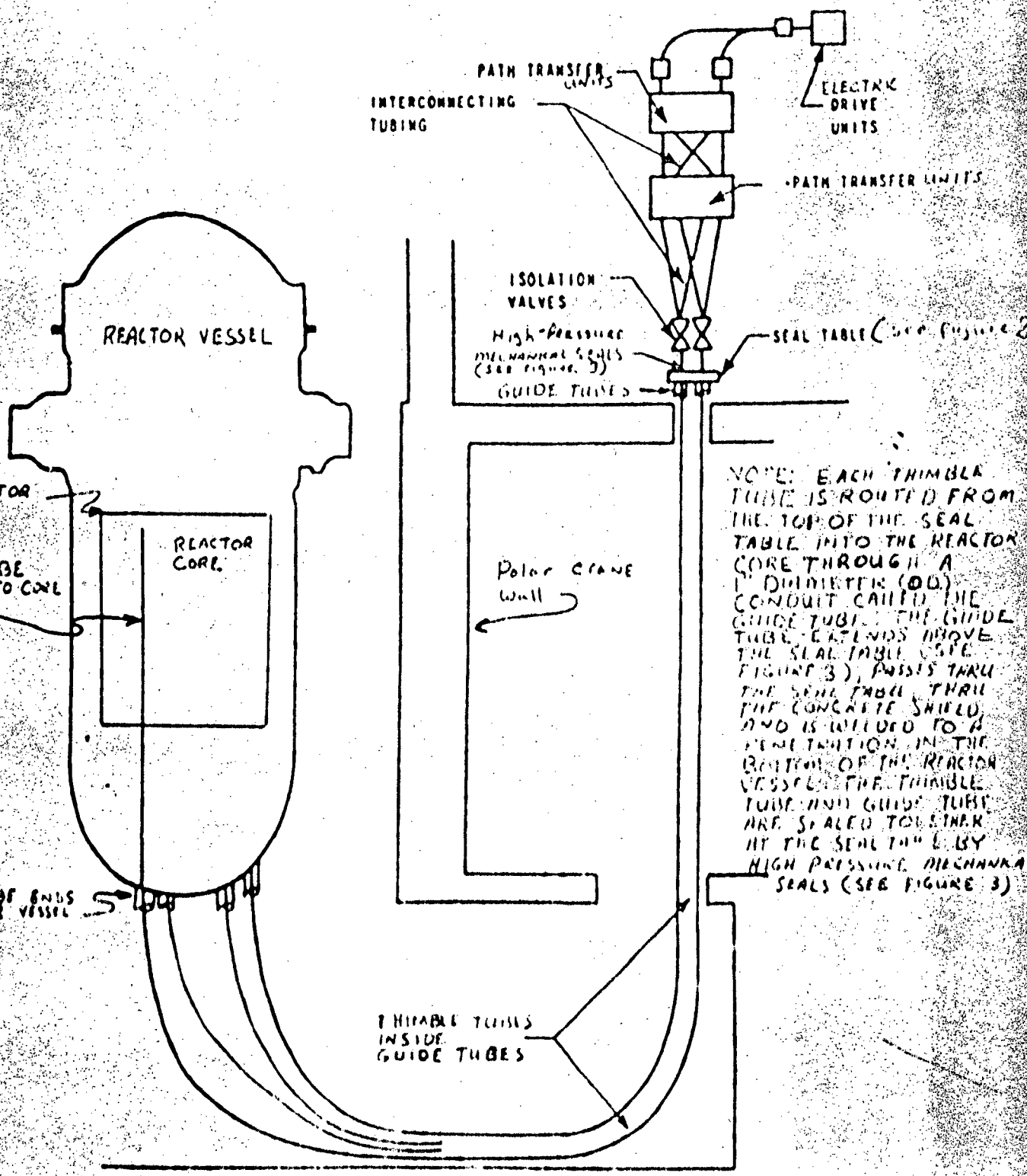


Figure 1. [illegible] (DMTS)

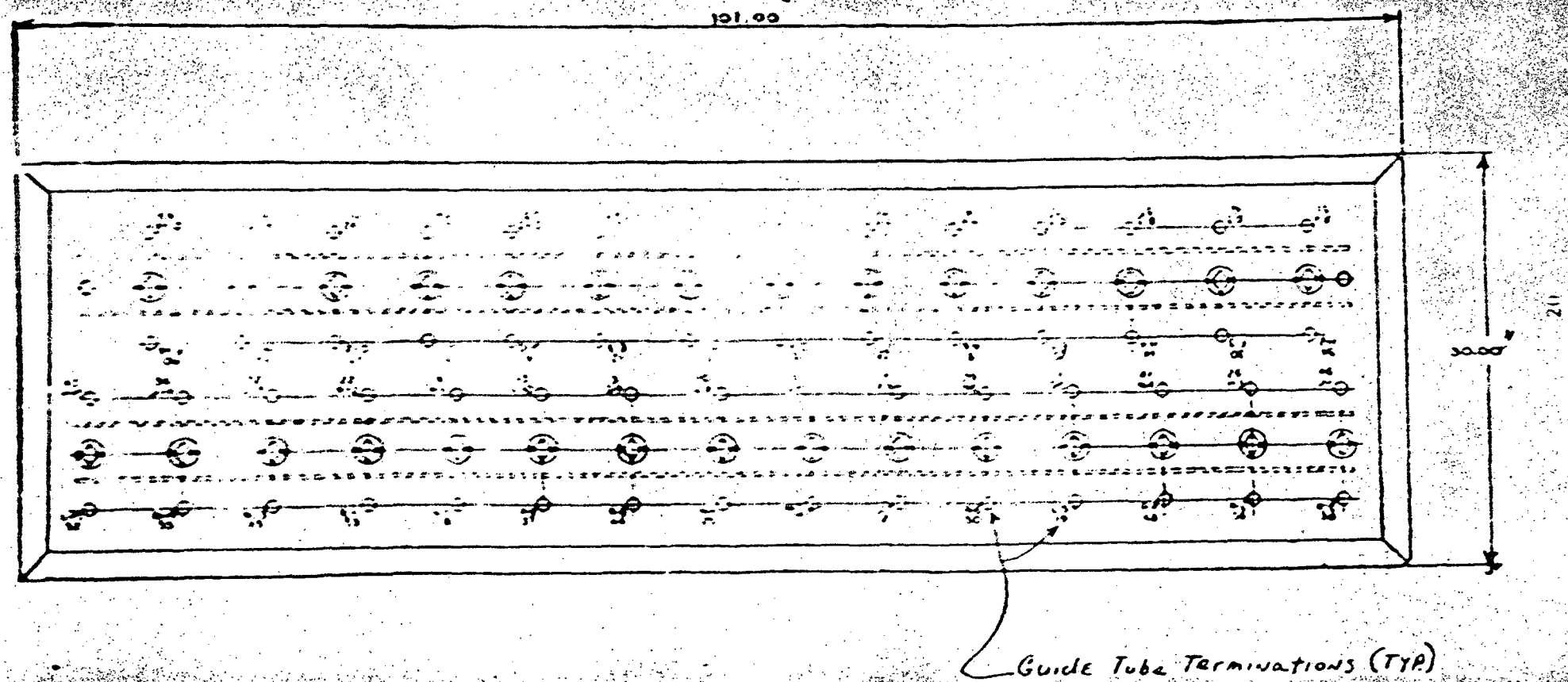
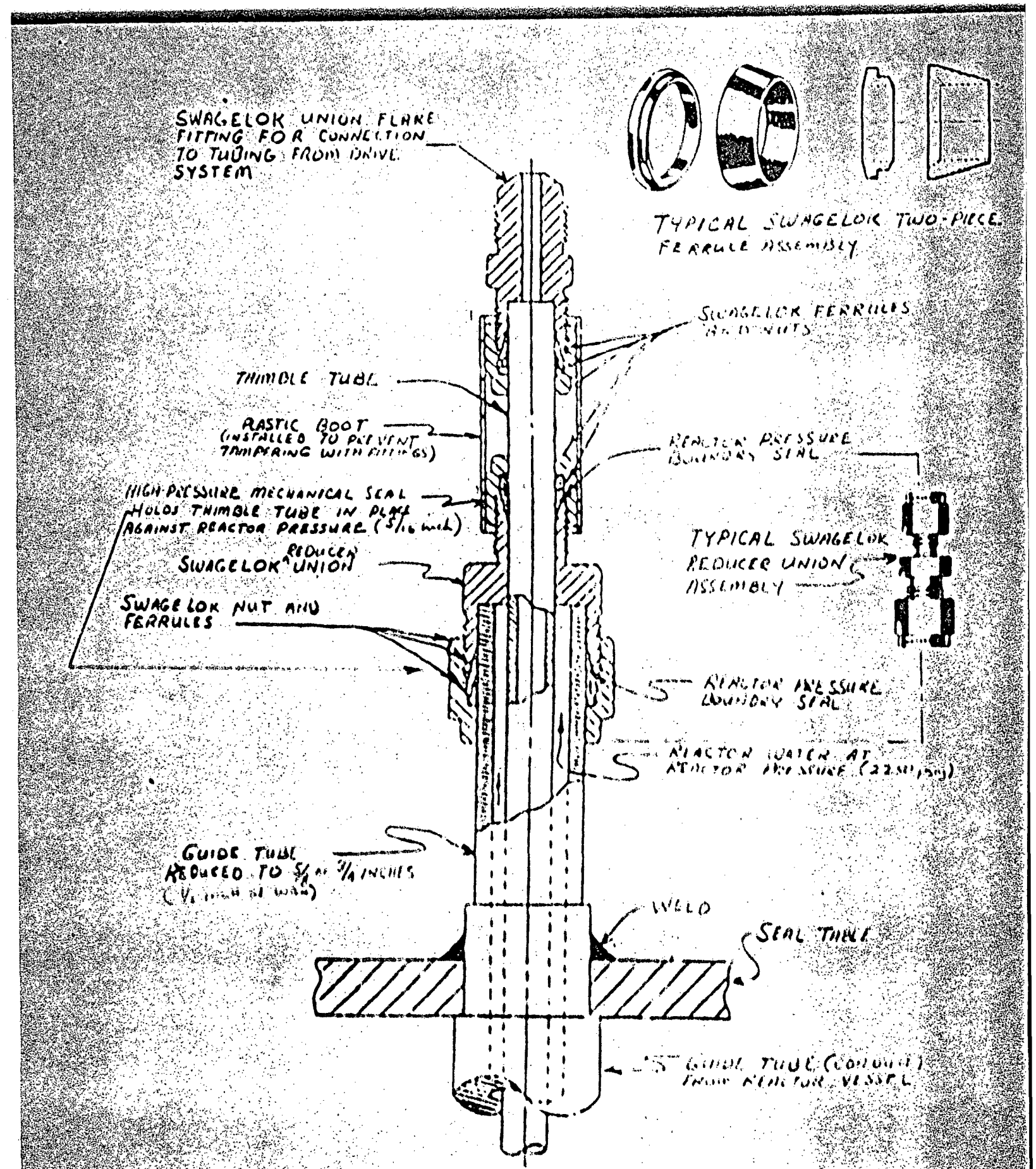
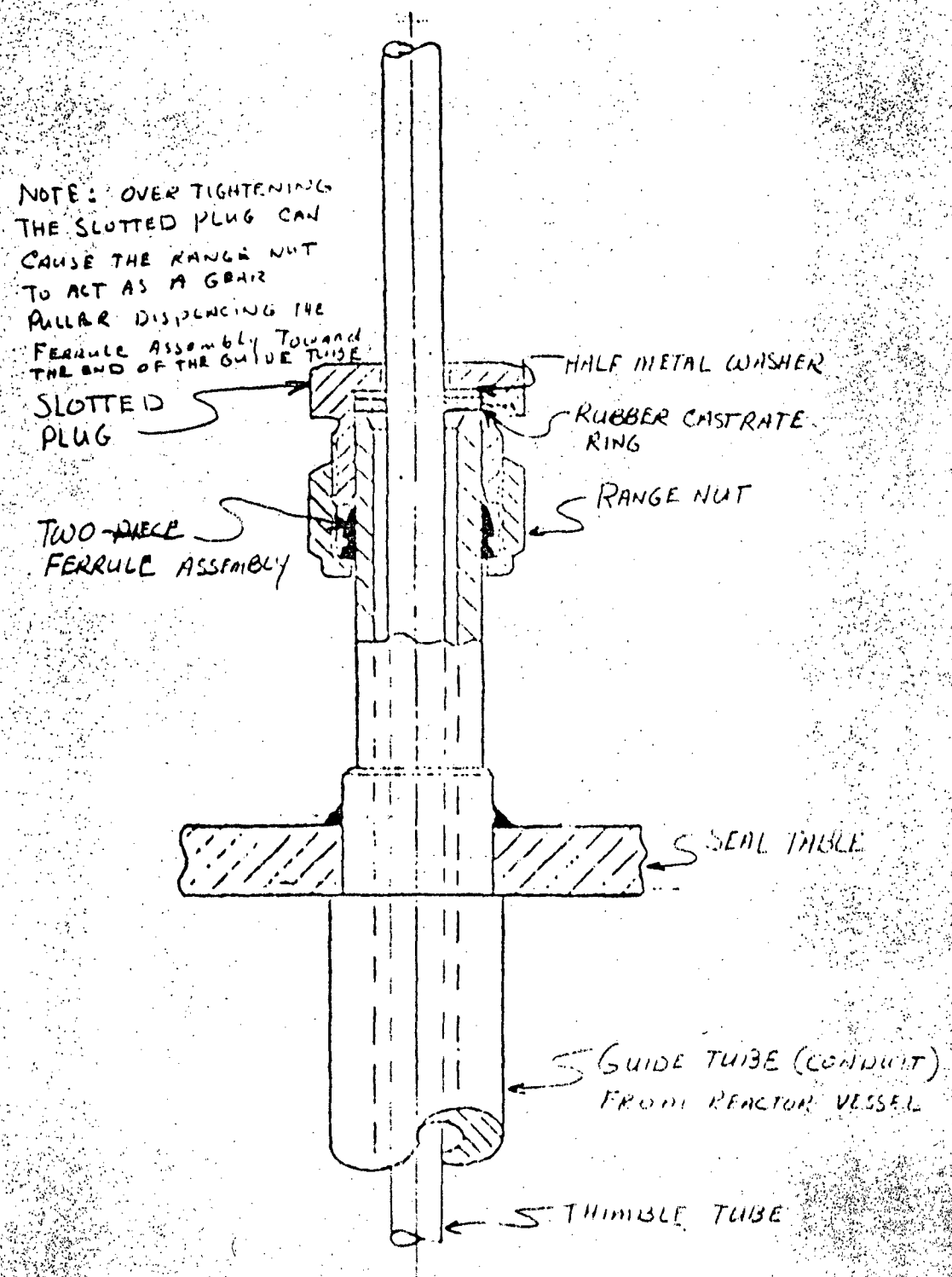


FIGURE 2
REACTOR SEAL TABLE



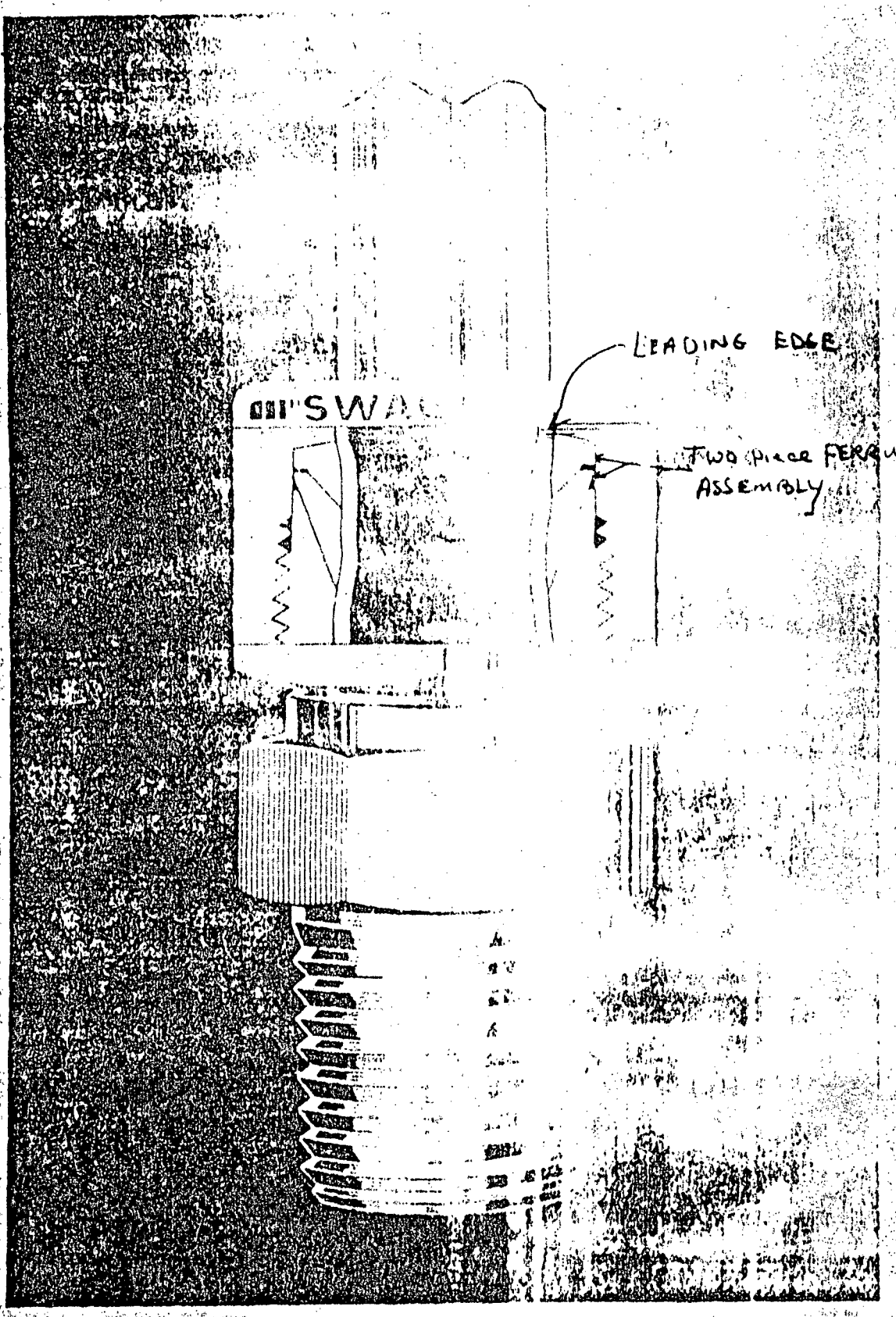
SEAL TABLE HIGH-PRESSURE MECHANICAL SEAL ASSEMBLY (TYPICAL 1 OF 58)

FIGURE 3



LOW-PRESSURE SEAL USED DURING REFUELING OPERATIONS WHEN THIMBLES ARE RETRACTED

FIGURE 4



"SWAGEING" ACTION OF A SWAGELOK FITTING ON TUBING
FIGURE 5

CAUTION

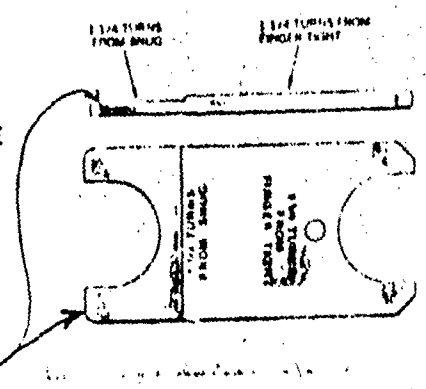
DO NOT MIX OR INTERCHANGE PARTS OF TUBE FITTINGS MADE BY OTHER MANUFACTURERS.

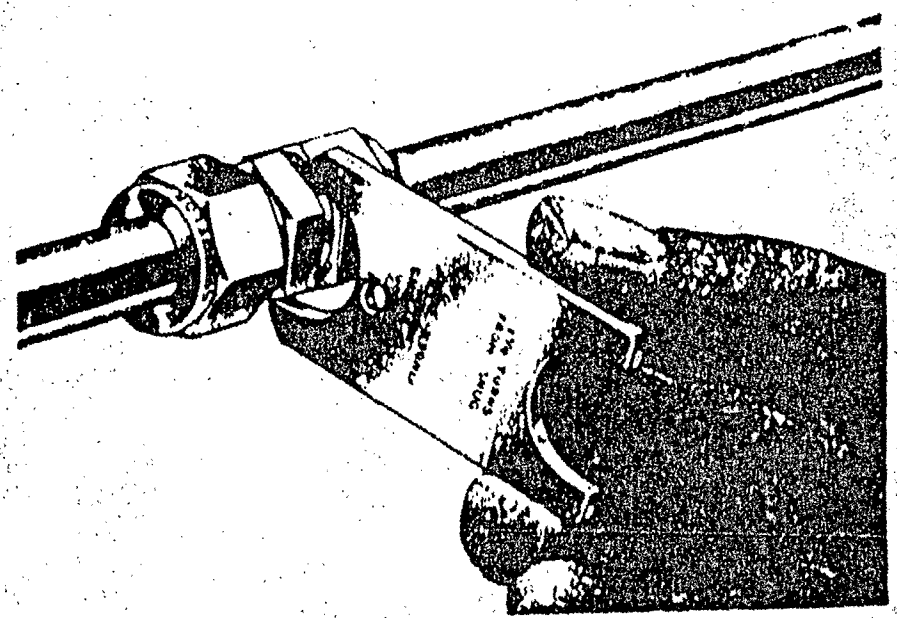
SWAGELOK Tube Fittings are manufactured to exacting tolerances. The critical interaction of precision parts as designed is essential to reliability and safety. Using parts of fittings made by other manufacturers with SWAGELOK Tube Fitting parts will not provide reliable connections. Damage or injuries may result from interchanging or mixing parts of tube fittings made by other manufacturers with SWAGELOK Tube Fitting parts.

SWAGELOK CAUTION CARD

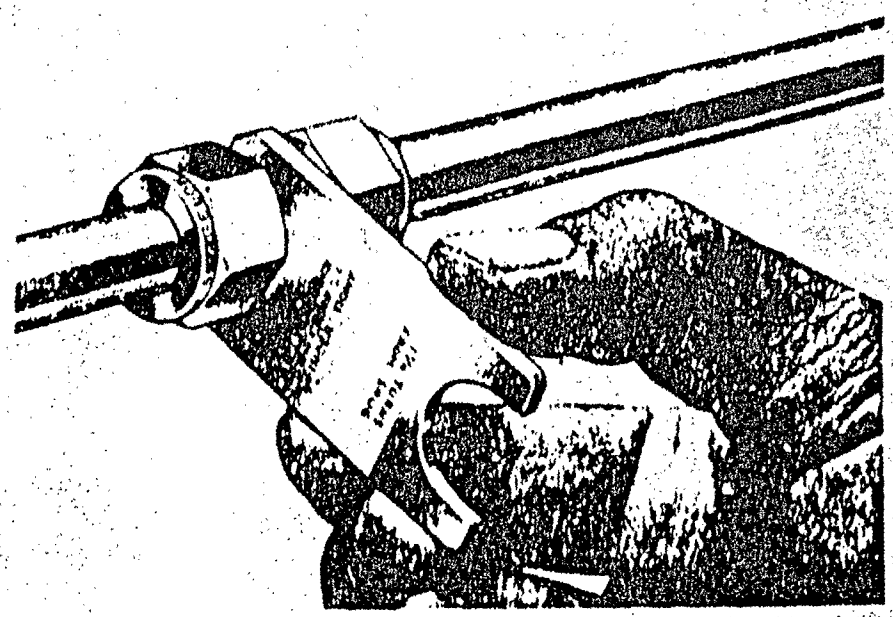
FIGURE 6

NOTE: THIS PART OF THE GAGE IS TO BE USED AFTER INITIAL ASSEMBLY OF SWAGELOK TUBE FITTINGS FOR HIGH PRESSURE APPLICATIONS, HIGH SAFETY FACTOR SYSTEMS





Gap Inspection Gage does not fit between nut and body hex. Fitting is sufficiently tightened.



Gap Inspection Gage fits between nut and body hex. Fitting is *not* sufficiently tightened.

SWAGELOK GAP INSPECTION GAGE

FIGURE 7