TVA 64 (05-9-65)

UNITED STATES GOVERNMENT

Memorandum

GNS 810410 050

TENNESSEE VALLEY AUTHORITY

TO : M. N. Sprouse, Manager of Engineering Design, W11A9 C-K
H. H. Mull, Manager of Construction, E7B24 C-K

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

DATE : April 9, 1981

SUBJECT: WATTS BAR NUCLEAR PLANT - NSRS REPORT R-80-21-WBN

The NSRS Report on the review of the WBN construction project program governing the installation and inspection of safety-related piping and supports conducted December 10-12, 1980, is attached for your information and actions on recommendations. Contained in the report are three recommendations that require action from your organizations to resolve. Other problem areas identified in the report are being addressed by OEDC in the Phase I and Phase II hanger and piping inspection programs and in response to NSRS as a result of reviews relating to IE Bulletin 79-14.

This report and its findings were reviewed with Messrs. Pierce, Cantrell, Killian, and Wilkins of your staffs in a meeting on March 11, 1981.

Please provide us a schedule for implementation of these recommendations by April 24, 1981. The NSRS contact for this report is T. G. Tyler, extension 6590.

H. N Culver
H. N. Culver

MVS

TGT:LML
Attachment
cc: MEDS, E4B37 C-K (Attachment)

READING FILE

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

REVIEW

NSRS Report No. R-80-21-WBN

Subject: Review of the WBN CONST Project Program Governing the Installation and Inspection of Safety-Related Piping and Supports

Date of Review: December 10-12, 1980

Reviewer: Terry G. Tyler — 4/8/81
T. G. Tyler — Date

Reviewer: Bruce F. Siefken — 4/8/81
B. F. Siefken — Date

Approved by: M V Sinkule — 4/8/81
Date

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I. Introduction

During the period December 10-12, 1980, two members of the Nuclear Safety Review Staff (NSRS) conducted a review at Watts Bar Nuclear Plant (WBN). The objectives of the review were:

1. To develop an understanding of the way that the WBN Construction Project (CP) conceptulized, developed, implemented and revised the program that has controlled the installation of safety-related piping and supports. Inherent in this objective was the need to not only understand how the WBN CONST Project fulfilled their responsibilities in this program, but also to understand how the quality assurance (QA) organizations (WBN CONST Site, CONST QA Knoxville, and OEDC QA) and the EN DES branches and design project interacted with this program. The goal of the reviewers was to develop a detailed understanding of the program from identification of the design requirements by the CONST project to documentation of the final "as-constructed" system configuration including problems any person(s)/or groups of persons were having or had with understanding/implementing the program.

2. To determine whether or not the WBN program governing the installation of safety-related piping and hangers contained problems of a similar nature to those found to exist at Sequoyah during the NSRS review of the TVA Program to meet the requirements of IEB 79-14.

3. To make recommendations on ways to resolve significant nuclear safety-related problems that were found during the review and/or to make recommendations on ways to improve the methods and practices that currently govern the installation and inspection of safety-related structures, systems, and components at all of TVA's nuclear plants.

The review was accomplished by conducting individual discussions with managers from the WBN CP organization at the project manager, construction engineer, assistant construction engineer, section supervisor, quality assurance, craft superintendent, and assistant craft superintendent levels and with employees in welding, quality control and records, and crafts. This allowed NSRS to obtain a perspective of the safety-related piping and supports installation and inspection program at each level of the WBN CP organization. It also allowed the WBN CP Staff the opportunity to express their viewpoint of the program including its weaknesses, adequacy, problems encountered with implementation, and suggestions for improvement.

Prior to the actual review the NSRS representative had reviewed the NRC, EN DES, OEDC QA, CONST QA, and WBN CP generated documentation listed in Section V of this report. In addition, conversations had transpired between NSRS and Civil Engineering Branch (CEB) personnel and Nuclear Engineering Branch (NEB) personnel on various aspects of General Construction Specification G-43, "Support and Installation of Piping Systems in Category I Structures," This provided an

understanding of the written requirements for such a program including commitments and a basis for determining the adequacy of the written program. From this review specific questions were generated which served as the basis for the onsite review interviews. The NSRS report on the "TVA Program to Meet IE Bulletin 79-14" (GNS 800814 001) and the two OEDC responses to that report--(1) EDC 800827 020 outlining a two-step program to ensure consistency between the "as-designed" and "as-constructed" safety-related piping and hanger configuration and (2) EDC 801212 013, in draft form at that time, outlining management control improvements for the quality aspects of the piping and supports program--were reviewed for problem similarities and committed programmatic revision applicability at WBN.

This report is for the most part founded on information that was revealed in the discussions that took place during the onsite review. As such, report readers should recognize that changes to current WBN CONST project practices and philosophies are in the process of implementation. NSRS views these changes as positive steps toward resolving some of the problems identified during this review. The overall effectiveness of these changes cannot be determined until after the passage of time.

II. SUMMARY

Assessment of Safety-Related Piping and Supports Program

The majority of the safety-related piping is installed in both units at WBN, while the majority of permanent hangers and supports remain to be installed. Considerable difficulty is being experienced with installing permanent hangers and supports per design requirements. The hanger and support installation problems appear to be due to piping being mislocated as a result of inadequate control by the WBN CP over the preparation of field fabrication sketches by the craft; to interferences with field routed components and structures; and to the use of different reference points by EN DES to locate piping, hangers, and supports.

The quality control program that governs the installation and inspection of safety-related piping, supports, and hangers appears to suffer from problems similar to those identified in NSRS's report on the "TVA Program to Meet IE Bulletin 79-14" for SQN. The Phase I and Phase II program to meet NRC IE Bulletin 79-14 (EDC 800827 020) and the management changes described in EDC 801212 013 should help to correct these problems. However, the passage of some amount of time is required before the effectiveness of these changes can be assessed.

The review revealed that the requirements contained in the quality control program constitute the best effort of the WBN CP to identify and interpret program requirements from the morass of documents containing design requirements, program requirements and

licensing commitments that were provided to them by OEDC. The program as it presently stands could be interpreted as satisfying Criterion V, Instructions, Procedures, and Drawings of 10 CFR 50, Appendix B. However, the program is lacking in that it does not formally implement all of the requirements necessary to ensure that "as-constructed" satisfies "as-designed." This problem with the program may be attributable to one or more of the following:

a. Lack of an assessment of the adequacy of the scope and detailed requirements contained in the quality control program by an organization or organizations independent from the WBN CP or WBN CONST-QA organizations,

b. A misinterpretation of specific program requirements,

c. Design requirements not clearly and concisely defined

d. The seemingly continuous changing of requirements by EN DES or by NRC.

These problems were also found in the SQN program. Actions have been taken to correct these problems as stated in the preceeding paragraph. The adequacy of these actions cannot be determined at this time.

Failure to formally document closure sections in piping and to control preload in piping and nozzles as required in General Construction G-43 is an example of a significant design requirement that was not formally implemented in the quality control program for the installation and inspection of safety-related piping and supports. As a result, varying amounts of preload are suspected to exist in the piping and nozzles; because the 1/16-inch maximum misalignment of closure weld joints was not adhered to. External force of varying degrees was utilized to accomplish piping to piping and piping to nozzle alignments. Existence of this preload has the potential to invalidate the seismic analysis.

The WBN CP quality control, field change request, and nonconformance report programs as they presently stand have been identifying mislocated piping and hanger or support installation problems. These problems are resolved by way of QA audit reports, field change requests, and nonconformance reports. The preload problem had not been identified by the existing programs, and potentially could have gone undetected.

NSRS believes that the existing quality-control, nonconformance, and field change request programs coupled with the Phase I and Phase II programs to meet NRC IE Bulletin 79-14 (EDC 800827 020) could provide assurance that the final "as-constructed" configuration that evolves from the program satisfies the seismic design and analysis requirements. This NSRS belief is contingent upon:

- adequate training of inspectors in Phase I and II

- adequate definition of criteria to assess the results to the Phase II inspections and accept or reject the Phase I results

- adequate implemetation of both programs.

III. RECOMMENDATIONS

A. R-80-21-WBN-01 Preloading Problem

CEB with the assistance of the WBN CP should assess the magnitude and frequency of the existence of preload in the piping. Based on these findings CEB should specify corrective actions, if any, to eliminate the problem and to ensure the validity of the seismic analysis. In addition changes to the existing program that are acceptable to both organizations should be identified and implemented to prevent recurrence of the problem in piping being installed now and in the future.

B. R-80-21-WBN-02 Different Reference Points for Locating Piping and Supports

EN DES organizations that specify locations for piping and supports at WBN should review their current practices to determine the extent of this problem. Based on the findings of the review, these organizations should specify corrective actions to eliminate this problem on present and future drawings under their control and on drawings issued to the WBN CP or already implemented by the WBN CP.

C. R-80-21-WBN-03 Control of the Field Fabrication Sketch Program

The WBN CP should establish and maintain control of the field fabrication sketches for piping yet to be installed in WBN-1 or WBN-2. The goal of such an effort should be to not release a sketch for implementation until after it has been determined by CP engineering personnel to be consistent with the requirements on the design drawings.

IV. DETAILS

Two members of NSRS were onsite during the period December 10-12 to conduct a review of the WBN construction project (CP) program that governs the installation of safety-related piping and hangers. The NSRS personnel could not have asked for better cooperation from those persons contacted during the course of the review. Also, the positive attitude towards achieving a high degree of quality during construction activities expressed by all persons contacted is commendable.

WBN Safety-Related Piping and Supports Program

The review entailed interviewing persons at all levels of the WBN construction project organization to obtain viewpoints on the

program that governs the installation and inspection of safety-related piping and supports. Specific areas covered in the interviews were the conceptualization, development, implementation, adjustment (refinement), problems encountered, suggestions for improvement of the program, and present status of program implementation.

The review revealed that the perspective of the program is consistent at all levels of construction project organization. The situation that currently exists with the installation of safety-related piping and supports may be described as follows:

- Ninety-two and seventy-seven percent of the safety-related piping has been installed in WBN-1 and WBN-2 respectively.

- Permanent hanger installation has progressed to thirty-eight and five percent complete respectively for units 1 and 2.

- Approximately twenty-five percent of the installed piping is determined to not be located per design when initial pipe location inspections are performed.

- This mislocated piping coupled with congestion caused by the installation of field routed (located) structures, systems, and components has made hanger installation per the original design difficult, if not impossible, to achieve.

- Varying amounts of residual stress due to the use of external force to effect weld joint alignment exists in the piping and equipment nozzles.

- The mislocated piping, the inability to install permanent hangers and supports per design, and the existence of residual stress in piping and nozzles taken individually or as combinations have the potential to invalidate the original seismic analysis for the plant safety-related piping systems.

The causal factors for this situation cannot be attributed to one single factor or organization. NSRS's understanding of the evolution of this situation obtained from the interview discussions is as follows:

- The WBN CP quality control program was formulated during the early stages of the construction project (approximately 1973). The program consisted of quality control instructions (QCI's) and quality control procedures (QCP's) which governed the installation and inspection of safety-related structures, systems, and components. The program formulation and background about the program are as follows:

  a. OEDC philosophy and practice in 1973 was for each CP to develop their own quality control program. In addition the OEDC philosophy on quality assurance was that the

Manager of EN DES and the Manager of CONST were responsible for organizing and directing their respective division's QA programs to attain quality objectives. The OEDC QA Manager was responsible for establishing basic QA Program policies and requirements, providing guidance, and overseeing the division's programs. Consequently, this resulted in each division developing their own QA Program with little regard for the other division's program except that provided by OEDC QA.

b. OEDC QA established policy and requirements for the QA programs in each division including a listing of all licensing commitments.

c. Both EN DES and CONST developed their respective QA programs which would govern the work within their organizations. For EN DES the Engineering Procedures (EP's) along with a number of implementing documents were developed. For CONST quality assurance and control program requirements were identified in the CONST QA Program manual.

d. EN DES produced over a time period drawings, specifications, General Construction Specifications, procurement specifications, memorandums, etc. all of which contained requirements or references to requirements that were to govern installation and inspection activities for safety-related structures, systems, or components.

e. All of this information was transmitted to the WBN CP for their use in constructing WBN.

f. Upon receipt of this information, the WBN CP sorted through all of this documentation to identify what activities required preparation of a QCI or QCP, and to identify all of the requirements that governed particular installation and inspection activities. From this the WBN CP developed the QC program (QCP's and QCI's) that would govern all safety-related activities. These QCI's and QCP's were reviewed and approved onsite by WBN CP personnel and WBN CONST-QA personnel. There was no requirement for any organization other than WBN CP and WBN CONST-QA to review the scope and detailed requirements for adequacy and consistency with licensing, program, and design requirements and/or commitments.

g. The WBN CONST-QA group at some point during the formulation of the QC program developed their QA audit program and audit schedule. The purpose of the WBN CONST-QA audit program was to provide some level of confidence that the QC program requirements were being adhered to during installation and QC inspection. The level of confidence to be provided was not formally quantified in any QA program requirements.

The program described in steps a. through g. was an acceptable way of developing and implementing the QC program. In fact, the program was believed to satisfy Criterion V, Instructions, Procedures, and Drawings of 10CFR50, Appendix B. However, two key steps, 1) clear and concise definition of licensing commitments, program requirements, and especially design requirements for use by the CP in developing their program, and 2) review of the QC program to assess the adequacy and consistency of scope and detailed requirements by some organization within OEDC other than the WBN CP, did not transpire. Consequently, the ability of the resultant QC and QA program requirements to ensure at some confidence level that the "as-constructed" configuration satisfied the assumptions and restrictions of the "as-designed" configuration was questionable.

Along these same lines, the WBN CONST-QA program and audit schedule was developed in conjunction with the QC program. This audit program was designed to provide confidence that the QC program requirements were being adhered to during the installation and inspection of safety-related structures, systems, and components. However, during this review it was learned that the audit scopes and frequencies were not statistically based. This needs to be resolved within the framework of the effective functioning of the QA/QC program.

- At WBN as at other TVA CP's, fitters prepare fabrication sketches from the design piping drawings. These fabrication sketches were and are required by the ASME code. The fitters at all TVA CP's have insisted that preparation of these sketches is their responsibility. Both the fitters and WBN CP personnel agree that the requirements on the design piping drawings have to be converted into field fabrication sketches for them to be conducive for field use.

The original plan at WBN called for the fitters to prepare the sketches and for the engineering unit to review and approve each of these fabrication sketches prior to their release for actual work to begin. The review of the first series of fabrication sketches revealed that sketches contained errors in dimensions and component locations. However, due to a shortage of engineering unit personnel to review these sketches, the volume of sketches being produced by the fitters, and the large number of fitters whose work depended on the issuance of the sketches, the decision was made to waive the requirement for engineering unit review and approval of the sketches. Instead, the verification of piping location/ routing was to be made at the time the engineering unit oversaw the installation of permanent hangers and supports. Verification of location/routing of piping was not waived in this instance, rather it was only postponed to a later time frame in the life of the CP.

Later on in the life of the CP, the decision was made to set up a special hanger group separate from the piping group to oversee the installation of hangers and supports. Consequently, the piping location confirmation kept getting put off. The errors revealed in the early engineering unit review of fabrication sketches should have provided an early warning of potential problems in this area. This problem area needs resolution. (See recommendation C, Section III.)

- A problem was encountered in obtaining the permanent hangers and supports in a time frame consistent with the installation of the safety-related piping. Consequently, the permanent piping was installed on temporary hangers. These factors in and of themselves did not create a problem. However, installation of field routed (located) structures, systems, and components too close to the piping to allow installation of the permanent piping and supports transpired. This has and is causing difficulties with the installation of permanent supports.

- Use of different reference points to locate piping and to locate hangers resulted in further complications. Pipe location is referenced off of column centerlines while supports and hangers are referenced off of nominal wall, ceiling or floor faces. The result in some instances is a support or hanger that totally misses the piping that it is supposed to support, even though both pipe and support are installed per their respective design requirements. This problem with the reference points needs to be resolved. (See recommendation B, Section III.)

- The requirements contained in General Construction Specification G-43, "Support and Installation of Piping Systems in Category I Structures," are being formally implemented to varying degrees in the quality control program for the installation of safety-related piping, hangers, and supports. The reasons for this are:

  a. The purpose of G-43 is to establish minimum requirements for the support and installation of piping systems in Category I structures to assure that the piping is installed in such manner as to validate the analyses of the piping systems and insure conformance with the intended design of the system support scheme. This implies that the minimum requirements are all contained in this document. A closer examination reveals that G-43 itself requires knowledge of G-32, appropriate plant piping and support documents, manufacturers' recommended installation procedures, etc., to have a complete understanding of the requirements for the support and installation of piping systems. The situation just described emphasizes the point that the requirements governing the installation and inspection of safety-related piping and supports are not clearly and concisely defined in one or a relatively small number of documents.

b. The applicability of G-43 is to all piping, and piping supports installed in Category I structures with certain exceptions for embedded piping and piping provided as an integral part of prepackaged equipment provided by a vendor. G-43 continues by stating that procurement, materials, fabrication methods and details, inspection, and test requirements of the piping are not within the scope of G-43. This statement appears to conflict with the purpose of G-43 as stated in a. above; because if G-43 is to be the governing document to assure that the piping is installed in such a manner as to validate the analyses of the piping systems, then such areas as procurement, materials, fabrication methods and details, and especially inspection which have the potential to invalidate analyses must be controlled by some document. In light of this applicability philosophy, it is not surprising that G-43 reads as a general requirements document and does not contain a listing of the critical attributes with specific acceptance criteria that should govern the installation and inspection of safety-related piping and supports.

c. G-43 does contain some specific requirements such as the 1/16-inch maximum allowable misalignment of pipe joints while swinging free on supports for closure connection final assembly. This specific requirement is in G-43 to prevent significant preload on the final assembly connection which is a general assumption used in the seismic analysis. The concern for minimizing preload is justified per the seismic analysis; however, this requirement does not reflect conditions achievable from a constructability standpoint or criteria which are consistent with the ASME code on weld joint misalignment and on "cold-springing." Since the ASME code is utilized more extensively than G-43 and since the weld joint fit-up inspections are performed by welding unit personnel, the requirements of G-43 in this area tend to not be vigorously followed. Consequently, a documented program to minimize pre-load due to use of external force to effect closure connection alignment was never implemented at WBN. Piping fit-up was generally checked against the requirements of the ASME code. As a result, instances of preload outside the limits allowed by G-43 and subsequently the seismic analysis may exist in nozzles and piping at WBN. The impact of this situation on the seismic analysis cannot be determined until after the extent and magnitudes of the preload are determined. The most important point is that a situation exists in the field which may indeed invalidate the seismic analysis. Consequently, this situation must be evaluated and resolved. (See recommednation A, Section III.)

The varying degree of formal implementation of the requirements in G-43 has contributed to the problems being experienced with piping, hanger, and support installation. Also, the failure to have a formal program to minimize residual stress in piping and nozzles for closure welds can be attributable to the reasons stated above.

- The WBN CP as other TVA organizations has experienced difficulties with recruiting qualified or qualifiable personnel due to the competition (salary and benefits) for this type of person in the marketplace. This is evident in the results of recent WBN recruiting trips. In addition, the same competition makes it difficult for TVA to retain these people once they are hired. The problem of recruiting and retaining personnel when coupled with pressure to meet the construction schedule forced first line supervisors to adopt crisis management techniques. Consequently, supervisors did not have time to effectively train, utilize, and supervise the activities of their subordinates. This resulted in schedule and cost impacts due to the rework required to correct first time mistakes.

All of these circumstances when combined produce the situation that exists today with the installation of safety-related piping and supports and the seismic analyses for these piping and supports. Refer to figures 1.0 and 2.0 for cause and effect relationships as perceived by NSRS.

NSRS recognizes that the existing quality, field change request, "as-constructed," and nonconformance programs at WBN have been and would continue to formally identify mislocated piping and support installation problems for resolution by EN DES. These programs coupled with the Phase I and Phase II programs implemented after the IEB 79-14 review at SQN would provide assurance that the "as-constructed" configuration satisfied "as-designed" requirements. With regard to the preload or residual stress problem, the WBN CP had already recognized that the sequence of piping installation requirements in G-43 had not been implemented. Per reference 11 a meeting had been held with EN DES in an attempt to resolve this problem. The discussion in the meeting centered around documentation of closure sections and the cost to rework the piping systems to satisfy this requirement. Neither the significance and ramifications of preload nor the requirements to minimize preload in piping and nozzles, although discussed in the section of G-43 on piping installation sequencing, was sufficiently understood by the personnel interviewed until this review by NSRS. This understanding coupled with the knowledge of piping installation practices led to the conclusion that preload as discussed in G-43 does exist to varying degrees in safety-related piping and nozzles at WBN. Consequently, NSRS as stated in Section III of this report recommends that CEB determine the extent and significance of this situation and based on their findings propose actions to resolve this problem.

Comparison with NSRS Review Findings on SQN

NSRS in August 1980 issued a report entitled, "NSRS Assessment Sequoyah Nuclear Plant Compliance with NRC-OIE Bulletin 79-14 "Seismic Analysis for As-Built Safety Related Piping Systems"" (GNS 800814 001). This report and the two OEDC responses to the report were utilized by the reviewers in preparing for the review at WBN. The findings in this review at WBN support NSRS's supposition in the SQN report that " ... similar problems may exist or have the potential to exist with the adequacy of the seismic qualification of the "as-built" safety-related piping and hangers at all of TVA's nuclear plants."

V. Persons Contacted
WBN CP

T. B. Bucy - Supervisor, Hanger Engineering Unit
C. O. Christopher - Assistant Construction Engineer
F. H. Denton - Welding Inspector
J. Evers - Authorized Nuclear Inspector
M. A. Harper - Training Officer
L. J. Johnson - Supervisor, Mechanical Engineering Unit
B. S. Johnson, Jr. - Assistant Construction Engineer
J. M. Lamb - Supervisor, Mechanical Engineering Unit
A.L.B. Mayes - Steamfitter Superintendent
F. M. McGraw - Authorized Nuclear Inspector
R. W. Olson - Construction Engineer
A. S. Perry - Welding Inspector
A. W. Rogers - Supervisor, Quality Assurance Unit
F. Smith, Jr. - Supervisor, Office, Materials, and Civil Engineering Unit
J. B. Tubb - Assistant Electrical Superintendent
J. E. Wilkins - Construction Project Manager
S. Wolfe - Welding Engineer

VI. References

1. Division of Construction QA Manual

2. OEDC QA Program Requirements Manual

3. OEDC QA Manual for ASME Section III Nuclear Power Plant Components

4. TVA Design Specifications

   a. WBNP-DS-1935-2473-1

   b. WBNP-DS-1935-2618

   c. WBNP-DS-1935-2619

5. TVA General Construction Specification G-43, "Support and Installation of Piping Systems in Category I Structures"

6. WBN Construction Specifications

   a. N3G-881, "Identification of Structures, Systems, and Components Covered by the Watts Bar Nuclear Plant Quality Assurance Program"

   b. N3M-868, "Field Fabrication, Assembly, Examination, and Tests for Piping System for Watts Bar Nuclear Plant"

7. Division of Construction Quality Control Instructions

   a. WBNP-QCI 1.8

   b. WBNP-QCI 1.10

   c. WBNP-QCI 1.11

   d. WBNP-QCI 1.17

   e. WBNP-QCI 1.21

   f. WBNP-QCI 1.22

   g. WBNP-QIC 1.28

   h. WBNP-QCI 1.38

   i. WBNP-QCI 4.2

8. Division of Construction Quality Control Procedures

   a. WBNP-QCP 1.7

   b. WBNP-QCP 1.16

   c. WBNP-QCP 3.11

   d. WBNP-QCP 4.7

   e. WBNP-QCP 4.8

   f. WBNP-QCP 4.1[illegible]

   g. WBNP-QCP 4.24

   h. WBNP-QCP 4.28

   i. WBNP-QCP 4.30

9. Watts Bar Nuclear Plant Field Instructions

   a. WBFI-G-7

   b. WBFI-M35

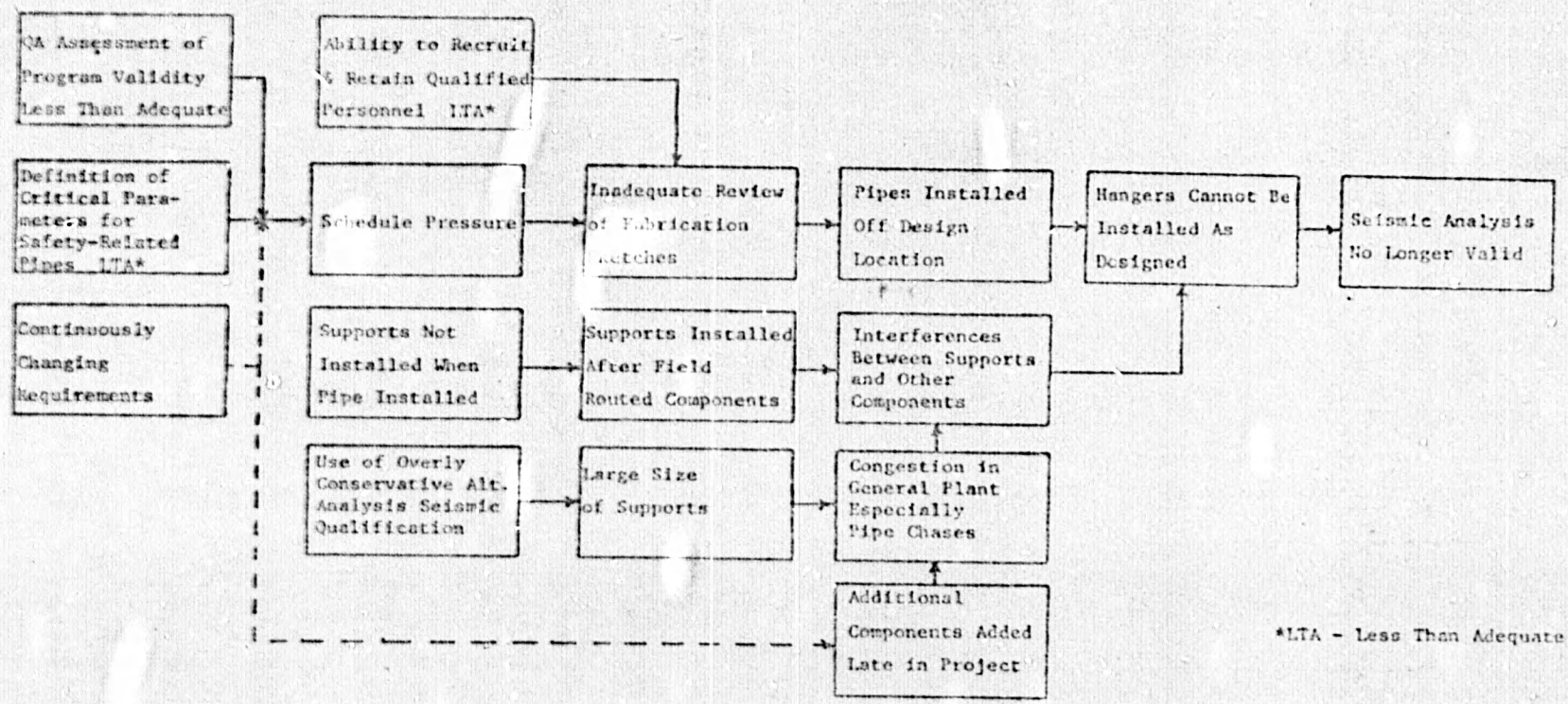
10. Memorandum from R. M. Pierce to J. E. Wilkins dated May 19, 1980 (MEB 800519 019)

11. Memorandum from J. E. Wilkins to R. W. Cantrell dated December 10, 1980 (WBN 801210 003)

12. 10CFR50, Appendix B

FIGURE 2.0

CAUSE/EFFECT CHART FOR
PIPE HANGER INSTALLATION
PROBLEMS

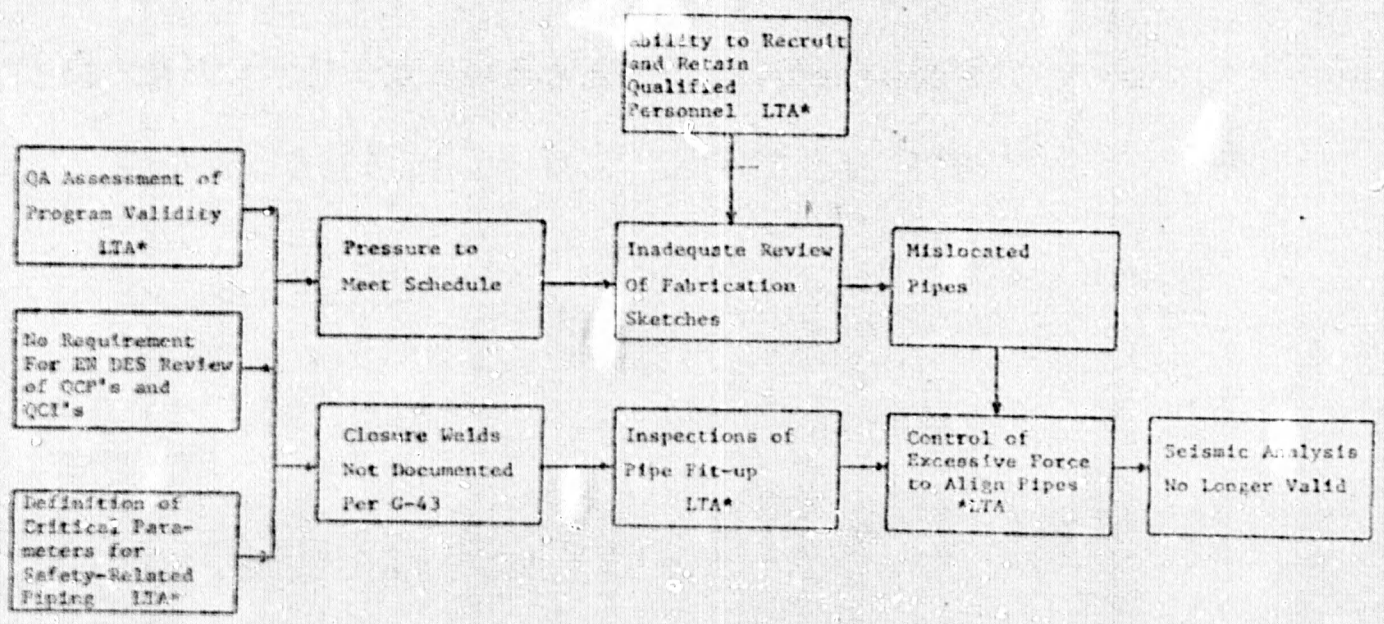


*LTA - Less Than Adequate

FIGURE 1.0

CAUSE/EFFECT CHART FOR

RESIDUAL PIPE STRESS



*LTA - Less Than Adequate

TVA 64 (09-9-68)

UNITED STATES GOVERNMENT

Memorandum

GNS 81 0327 001

TENNESSEE VALLEY AUTHORITY

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

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

DATE : March 26, 1981

SUBJECT: SEQUOYAH NUCLEAR PLANT UNIT 1 - SPECIAL REVIEW OF THE INADVERTENT INITIATION OF THE RESIDUAL HEAT REMOVAL CONTAINMENT SPRAY SYSTEM ON FEBRUARY 11, 1981 - NSRS REVIEW REPORT NO. R-81-05-SQN

Attached is the NSRS report of a special review of the event, activities, and commitments concerning the inadvertent initiation of the residual heat removal containment spray system on February 11, 1981.

Our recommendations are stated in section III with supporting details given in section V. Please provide your reply to our recommendations and your implementation schedule by April 23, 1981.

If you have any questions regarding this matter, please contact M. S. Kidd, extension 4813.

H N Culver

H. N. Culver

~~JOW:KRW~~

Attachment

CC (Attachment):

MEDS, E4B37 C-K

F. A. Szczepanski, 417 UBB-C

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

SPECIAL REVIEW

NSRS REPORT NO. R-81-05-SQN

SUBJECT: TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT - UNIT 1
SPECIAL REVIEW OF THE INADVERTENT INITIATION OF
THE RESIDUAL HEAT REMOVAL CONTAINMENT SPRAY SYSTEM ON
FEBRUARY 11, 1971

DATE OF
ONSITE REVIEW: FEBRUARY 17, 1981

REVIEWERS: Jonnie O Vantrease — 3-25-81
J. O. VANTREASE — DATE

M S Kidd — 3/25/81
M. S. KIDD — DATE

APPROVED BY: M S Kidd for — 3/25/81
K. W. WHITT — DATE

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I. Scope

This special review of the inadvertent initiation of the residual heat removal (RHR) containment spray on February 11, 1981, was conducted to determine the details leading up to the incident, the actions taken during the event, the effectiveness of those actions, and the actions to be implemented to prevent recurrence of this type of incident. To fulfill our overview role to the General Manager and TVA Board, this special review was also conducted to monitor the activities taken by the line organization relating to the event, including their preparation of an investigation report.

The review involved three phases:

1. Telephone conversations with the Division of Nuclear Power (NUC PR) personnel during and following the event,

2. An onsite review by two NSRS reviewers on February 17, 1981, which involved approximately 12 man-hours, and

3. Review of NUC PR's report on the incident (see reference VII.1).

II. Background

On February 11, 1981, while unit 1 was in cold shutdown to comply with Technical Specification requirements regarding ice weight, a unit operator (UO) told an assistant unit operator (AUO) to open RHR valves 1-HCV-74-37 and 1-HCV-74-531 and to verify closed valve 1-FCV-72-40. On the previous shift 1-FCV-72-40 had been stroked to comply with surveillance requirements, but no one had visually inspected the valve to assure closure. Later during the shift, the AUO called the UO about the desired position of the RHR valves. The UO told him to open the valves so that the "B" train of the RHR system could be placed in service. No valve numbers were mentioned during the telephone conversation. The AUO proceeded to open 1-HCV-74-37 and 1-HCV-74-531. Again, the AUO attempted to call the UO to confirm the position of 1-FCV-72-40; however, the telephone was inoperable. He proceeded to open 1-FCV-72-40 which initiated the "A" train of the RHR containment spray.

Almost immediately the UO noticed a rapid decrease in pressurizer level and pressure. The UO notified the shift engineer (SE) and tripped the two running reactor coolant pumps. The situation was diagnosed as a loss of coolant accident (LOCA) and Emergency Operating Instructions (EOI) 0 and 1 were consulted. Containment

evacuation was begun. Containment purge was terminated. Health Physics and Public Safety were notified. Charging flow was increased from the refueling water storage tank (RWST); letdown was continued. "B" train of the RHR was started. RHR suction from the RWST was opened. The suction path from the hot leg of the reactor coolant system (RCS) was not isolated. Pressurizer level was restored but started to decrease. A site emergency was initiated. The evacuation alarm was sounded; all personnel, except three construction workers, were quickly accounted for. Plant access was controlled. A second charging pump was started, and one safety injection pump was started. The AUO entered the main control room and discussed opening 1-FCV-72-40 with another UO who immediately closed the valve terminating the spray. RCS conditions were stabilized. The Nuclear Regulatory Commission (NRC) was notified after several attempts.

Statements were taken from two SE's, one ASE, and one UO (see reference VII.B). The SQN Compliance Staff coordinated and wrote the event report. The NRC sent an investigation team and a confirmation of action letter (see reference VII.G). EN DES was requested to help analyze the event on February 20, 1981. NUC PR representatives met with the NRC in Atlanta on March 6, 1981, to discuss their response to the NRC's confirmation of action letter and implementation of the required actions. NUC PR's incident report was issued on March 5, 1981 (see reference VII.A).

III. Recommendations

On February 17, 1981, NSRS reviewers conducted an exit interview with those persons indicated in section VI of this report. At that time NSRS offered no conclusions or recommendations because the plant personnel were still formulating actions to be taken to prevent recurrence of this type of event and because event information that was available at that time was incomplete (i.e., references VII.B, C, D, E, and K). At the exit meeting, the NSRS reviewers also indicated that it was premature for NSRS to formulate recommendations and conclusions since NUC PR's report on the RHR containment spray event was needed to complete our overview of the event. As a result of the NSRS review of NUC PR's activities during the event and the NUC PR investigations report, we concur with the actions taken or recommended by NUC PR as indicated in their report; however, we believe that the following actions also need to be taken.

A. RHR and Letdown Isolation

The EOI's should be updated to address LOCA's while on RHR cooling. In particular, the isolation of letdown and RHR hot leg suction from the RCS should be accomplished to prevent additional draining of the RCS and possible cavitation of the RHR pumps. Since the operators failed to recognize the need to do this, we also recommend additional operator training on LOCA's while on RHR cooling.

B. Personnel Statements and Logs

NSRS recommends training for the Operations personnel and shift technical advisors (STA's) on preparation of detailed logs and statements, especially those involving an accident or incident. Also, we recommend that someone with operations knowledge and authority read each statement and ask for more detail where required before the personnel leave the plant site after an event. Moreover, NUC PR should consider assigning someone the responsibility of maintaining a log when an event occurs.

C. Data Availability

NSRS recommends that NUC PR investigate other data acquisition methods that are superior to strip chart recorders for fast moving transients. Also, NUC PR should ensure that all two-pen strip chart recorders which record two different parameters be equipped with two different colors of ink to enable reading of the chart.

D. Personnel Evacuation

NUC PR's report indicates that three construction workers did not go to their assembly area. Consequently, NSRS recommends additional training to ensure that everyone understands what they are to do when an evacuation alarm is sounded.

IV. Status of Open Items

A. R-81-05-SQN-01, RHR Isolation

This item remains open pending action by NUC PR on recommendation III.A (see section V.A for details).

B. R-81-05-SQN-02, Personnel Statements and Logs

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

C. R-81-05-SQN-03, Data Availability

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

D. R-81-05-SQN-04, Personnel Evacuation

This item remains open pending action by NUC PR on recommendation III.D (see section V.D for details).

V. Details

The NSRS evaluation of the inadvertent initiation of the RHR containment spray system is based on information gathered during our onsite review of February 17, 1981, conversations with personnel listed in section VI, the NUC PR event report, logs, and statements by the major operations personnel who participated in terminating the event. NSRS agrees with the lessons learned and actions taken by NUC PR as outlined in their report. However, we feel additional actions, as recommended above, are required. These are discussed in detail below.

A. RHR Isolation

The event scenario as outlined by NUC PR in their report and as discussed in the logs and personal statements indicates that the operator never isolated letdown or the RHR suction from the hot leg. Moreover, the suction path from the refueling water storage tank (RWST) was not aligned for the RHR pumps until 10 minutes into the event. NSRS does not understand the operator's logic on this point since available information states that he felt that a LOCA existed. His training should have emphasized the need to inject water into the RCS and the need to prevent cavitation of the RHR pumps. Based on references VII.H, I, and J, the occurrence of a LOCA (i.e., a pipe rupture) while on RHR cooling is a low probability event. However, the event is discussed and operator actions are detailed. Since no plant operating procedures address a LOCA while on RHR cooling, NSRS feels there is great need to include the required operator actions in an EOI with emphasis on isolation of letdown and the RHR hot leg suction path. Also, the operators should receive additional training on this subject.

B. Personnel Statements and Logs

Upon reviewing the statements made by the SE's, ASE, and UO involved in the event, NSRS reviewers determined that some more detail should have been included, especially in the UO's statement. The UO's statement fails to discuss any of his actions other than the original orders to the AUO and the subsequent telephone conversation. One of the SE's statements indicates the UO performed other actions to mitigate the event.

NSRS understands that all of the statements were taken immediately after the unit was brought to a stable condition. NSRS feels this is an excellent policy. However, to ensure adequate detail in future statements, NSRS recommends that someone with operations knowledge, authority, and responsibility review the statements immediately after they are written and before the

participants leave the site to ensure that the statements are complete. It is important to get as much information as possible immediately after termination of the event because people tend to forget as time elapses and because as the event is discussed people analyze their actions in relation to the event and its termination. Detailed statements are essential to the reviewers effort to minimize plant staff time and involvement during the event review period.

When NSRS reviewed the STA's log, the UO's log, and the SE's/ASE's log, very little detail about the event was found in the UO's log or the SE's/ASE's log. The STA's log was a more detailed summary of the events as related to him in conversations and through the other logs. NSRS feels that there is a need for more detailed logs and for someone to attempt to keep a log of the events as they unfold to aid in analysis later. For this reason, NSRS is requesting NUC PR to consider assigning someone the responsibility of maintaining a log when an event occurs. However, we are not suggesting that this person interfere in the shutdown operation of the plant to obtain information for the log.

C. Data Availability

The strip chart recorder data (see reference VII.K) was not very useful in event analysis because:

1. The chart speed was too slow to catch fast transient information.

2. The width of the ink trace masked fluctuations in the data.

3. Two red pens were used on one strip chart to record the temperature for two of the hot legs. The temperature traces crossed making it difficult to read the chart.

Also, the computer printout of some of the parameters was in one-minute intervals which also prevented documentation of quick variations in the data.

Consequently, NUC PR should investigate other methods of recording data that would be more informative after an event and more useful to the event analysis. Also, NUC PR should take measures to prevent a recorder from containing pens of the same color when different parameters are being monitored on that recorder.

D. Personnel Evacuation

The log of Public Safety activities, included as appendix B in NUC PR's report, indicates that the evacuation went smoothly

with one exception. Three construction workers were not in their assembly area(s) as required. Consequently, NSRS feels that additional training is needed so that everyone will know what to do when an evacuation alarm is sounded.

VI. Personnel Contacted

A. Jere M. Ballentine, Plant Superintendent

B. William T. Cottle, Assistant Plant Superintendent, Operations

*C. Michael R. Harding, Supervisor, Plant Compliance

*D. James M. McGriff, Jr., Assistant Plant Superintendent, Health and Safety Services

E. Robert J. Prince, Health Physicist

*Present at exit meeting

VII. Documents Reviewed (References)

A. Memorandum from H. G. Parris to W. F. Willis dated March 5, 1981, entitled, "Response to Chairman Freeman's Request for Information on Inadvertent Spray Actuation at Sequoyah Nuclear Plant Unit 1 (Memorandum dated February 12, 1981, from Craven Crowell to W. F. Willis)(GNS 810309 102)

B. Statements Regarding Incident By:

1. Clyde T. Benton, Shift Engineer

2. Hubert L. Ledford, Unit Operator

3. William O. Lovelace, Jr., Assistant Shift Engineer

4. William R. Ramsey, Shift Engineer

C. Shift Technical Advisor's (STA's) Log for February 11-12, 1981, on SQN-1

D. Unit Operator's Log for February 11, 1981, on SQN-1

E. Shift Engineer/Assistant Shift Engineer's Log for February 11, 1981, on SQN-1

F. Letter from G. F. Stone to Colonel E. P. Tanner dated February 18, 1981 (GNS 810219 001)

G. Confirmation of Action Letter from J. P. O'Reilly to H. G. Parris dated February 23, 1981

H. SQN Final Safety Analysis Report Question 6.53, January 6, 1978

I. SQN Final Safety Analysis Report Question 6.53A, September 29, 1978, amended on December 22, 1978

J. SQN Final Safety Analysis Report Question 6.53B, May 25, 1979

K. Recorder Traces For:

1. Pressurizer Level

2. Pressurizer Pressure

3. Shield Building Radiation Monitor

4. Hot Leg Temperatures for Loops 1, 2, 3, and 4

L. Emergency Operating Instruction EOI 0, "Immediate Actions and Diagnostics, Unit 1 or 2," Revision 6, November 25, 1980

M. Emergency Operating Instruction EOI 1, "Loss of Reactor Coolant, Unit 1 or 2, " Revision 15, September 25, 1980

TVA 64105-9-651

UNITED STATES GOVERNMENT

Memorandum

GNS 81 0505 052

TENNESSEE VALLEY AUTHORITY

TO : H. J. Green, Director of Nuclear Power, 1750 CST2-C
M. N. Sprouse, Manager of Engineering Design, W11A9 C-K

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

DATE : May 5, 1981

SUBJECT: SEQUOYAH NUCLEAR PLANT - NUCLEAR SAFETY REVIEW STAFF REVIEW REPORT NO. R-81-07-SQN

Attached is the NSRS report of a routine review of SQN. The primary purpose of the review was to assess the operational readiness of unit 2. However, while performing this review several concerns were identified which also affected unit 1, necessitating the expansion of the scope into the area of the Technical Specification surveillance requirements and portions of the operator training program. Our review indicated that improvement is needed in the areas of document review, revision, and approval. The report contains several open items (i.e., preoperational tests, commitments to the NRC, etc.) requiring completion prior to unit 2 fuel load. EN DES, CONST, and NUC PR are tracking their items and are working to resolve/complete them.

Our recommendations, as stated in section IV of this report, show seven open items; six require resolution by NUC PR while one item requires action by EN DES. You are requested to inform NSRS of your plans and schedule for implementation of the recommendations by June 1, 1981.

If you have any questions regarding this report or transmittal memorandum, contact K. W. Whitt at extension 6620.

H. N. Culver

H. N. Culver

JOV:LML
Attachment
cc: MEDS, E4B37 C-K (Attachment)

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

REVIEW

NSRS REPORT NO. R-81-07-SQN

SUBJECT: TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT - UNITS 1 and 2

ROUTINE REVIEW

DATE OF ONSITE REVIEW: APRIL 6-9, 1981

REVIEWERS: Janice O. Vantrease — 5-1-81
JANICE O. VANTREASE — DATE

Paul B. Border — 5/1/81
PAUL B. BORDER — DATE

APPROVED BY: K. W. Hatt — 5-1-81
KERMIT W. HATT — DATE

TABLE OF CONTENTS

I. Scope

This was a routine review of Sequoyah Nuclear Plant. The primary purpose of this review was to determine if unit 2 was in a condition of readiness to be licensed for operation. This included a review of the plant organization, training, quality assurance program, procurement program, selected plant procedures required for normal operation, the management controls, the status of the preoperational tests that must be completed prior to fuel load, the status of those items on the Outstanding Work Item List requiring completion prior to fuel load, and the status of the Division of Nuclear Power (NUC PR) and the Division of Engineering Design (EN DES) commitments to the Nuclear Regulatory Commission (NRC) that must be resolved prior to fuel load. The unit 2 startup test program was not covered during this review but will be reviewed later.

In addition to the above, a comparison was made between the unit 1 Technical Specification surveillance requirements and the corresponding plant surveillance instructions.

II. Status of Previously Identified Open Items

Not reviewed.

III. Conclusions

A. Within their scope of responsibility, EN DES is tracking and trying to resolve approximately 51 open items that must be resolved with the NRC prior to unit 2 fuel load.

B. Within their scope of responsibility, the Division of Construction (CONST) is tracking and trying to complete approximately 155 items on the Outstanding Work Item List (OWIL) that must be completed prior to unit 2 fuel load. EN DES has responsibility for an additional six items on the OWIL, and NUC PR must complete another 54 items on the OWIL prior to fuel load.

C. Approximately 43 preoperational tests remain to be completed by NUC PR prior to unit 2 fuel load.

EN DES must review and approve the completed preoperational test data packages which are required prior to unit 2 fuel load.

D. Work on approximately 11 significant/reportable nonconformance reports remains to be completed by NUC PR.

E. Work on approximately 21 NRC open items is to be completed by NUC PR. This includes commitments to IE Bulletins, NUREG's, NRC questions, and inspection reports.

F. Management control of the surveillance program needs to be strengthened.

G. EN DES has not completed implementation of NSRS recommendations on the ERCW pumping station.

H. Management control of the SQN operator training program needs to be strengthened.

I. Approximately 100 temporary alteration control forms on unit 2 safety-related systems had not been reviewed by the Plant Operations Review Committee (PORC) as required by DPM N73011.

IV. Recommendations

A. R-81-07-SQN-1, Employee Concern No. 79-12-01 - Required Material Not in Sequoyah FSAR - Safety Concern on ERCW Pumping Station (See section V.A.5 for details.)

EN DES should amend the SQN FSAR as previously requested in reference NN and as committed to by EN DES in reference OO. Since the barge collision analysis has been completed and since the other recommendations have been addressed in draft FSAR amendments, NSRS does not feel that completion of this item should impact unit 2 fuel load; however, implementation should be completed in a timely manner.

B. Item R-81-07-SQN-2, Lack of Maintenance Instructions (See section V.B.2 for details.)

An instruction or group of instructions should be written by NUC PR for repair and/or replacement of the incore and excore flux monitoring detectors.

C. Item R-81-07-SQN-3, Lack of Management Control of Surveillance Program (See section V.B.3 for details.)

1. NUC PR should review SQA 41 and correct it to include all Technical Specification surveillance requirements.

2. NUC PR should assign responsibility for maintaining SQA 41 as a current document in a written program.

3. NUC PR should address the NSRS's concerns listed in section V.B.3.

4. NUC PR should reconsider the appropriateness of using SQA 41, a document not reviewed by PORC, as the primary basis for scheduling surveillances.

D. Item R-81-07-SQN-4, Inaccurate Organization Representation (See section V.B.4 for details.)

1. The SQN FSAR, N-OQAM, DPM No. N74A20, and the SQN-1 Technical Specifications should be revised by NUC PR to be consistent and to depict the current plant organization.

2. NUC PR should delete table 13.1-1 of the SQN FSAR, if possible, or change it to list those individuals, and their qualifications, who presently hold positions as key staff specialists.

3. Section 13.1.3.1 of the SQN FSAR and N-OQAM, part III, section 6.1, should be revised by NUC PR to require 10 years of responsible power plant experience for the Assistant Plant Superintendent.

E. Item R-81-07-SQN-5, Lack of Management Control in the Area of Nuclear Operator Training Program (See section V.B.5 for details.)

1. NUC PR should revise the N-OQAM and DPM No. N78A13 immediately to detail the operator training program.

2. SQN and Power Operations Training Center (POTC) should revise their procedures to comply with the revised N-OQAM and DPM No. N78A13.

F. Item R-81-07-SQN-6, Errors and Inconsistencies in Sequoyah Nuclear Plant Instructions (See section V.B.6 for details.)

1. SQN procedures and instructions should be reviewed in depth as time permits to assure that up-to-date and accurate guidance is provided to plant personnel in a timely manner.

2. The comments in section V.B.6 should be evaluated by NUC PR and incorporated, as determined to be appropriate, into the applicable instructions in a timely manner.

G. Item R-81-07-SQN-7, Unreviewed Temporary Alteration Control Forms (See section V.B.7 for details)

1. AI-9 should be revised to comply with the requirements of DPM N73011.

2. The status of SQN-2 outstanding temporary alterations on CSSC equipment should be reviewed prior to fuel load.

V. Details

A. Operational Preparedness of SQN-2

1. Basis for NSRS Review

NSRS has the responsibility to evaluate the operational readiness of TVA nuclear plants before they receive licenses. Therefore, NSRS has developed a program for the evaluation of plant programs, procedures, and organizations which implement NRC and TVA commitments (see references A through S, U through MM, and DDD). The NSRS operational preparedness program is divided into five major areas with a checklist for each area. The following is a summary of the major points NSRS evaluated for SQN unit 2:

a. Plant organization

b. Staffing requirements

c. Procurement practices

d. Personnel selection and training

e. Implementation of the Technical Specifications

f. Plant procedures and instructions

g. Hold orders

h. Temporary alterations

i. Drawing requirements

j. Quality assurance program

k. Design control

l. Document control

m. Control of purchased material, equipment, and services

n. Control of special processes

o. Inspection

p. Test control

q. Control of measuring and test equipment

r. Handling, shipping, and storage

s. Inspection, test, and operating status

t. Nonconforming materials, parts, and components

u. Corrective actions

v. Quality assurance audits, inspections, and surveys

w. Quality assurance records

In addition to the above, NSRS also performed a brief review of NUC PR, EN DES, and CONST open items including commitments to NRC to ensure that those items required for fuel load are being tracked and efforts are being made to resolve problems and close out the items.

NSRS also reviewed the status of preoperational tests which must be completed by NUC PR and the results approved by EN DES prior to unit 2 fuel load.

Finally, NSRS evaluated the open items remaining from previous NSRS review reports and employee concerns.

2. Status of the Preoperational Tests

In reference XX, EN DES listed 88 TVA and Westinghouse preoperational tests that must be completed prior to unit 2 fuel load. In addition, EN DES stated that 11 noncritical systems (NCS) preoperational tests must be completed prior to fuel load. Presently a total of 43 preoperational tests remain to be completed as detailed in reference S.

3. Status of Items on the Outstanding Work Item List

Reference YY lists 155 items, which are primarily work plans, that CONST must complete prior to fuel load of unit 2.

EN DES has the responsibility to complete an additional six outstanding items.

NUC PR must complete another 54 outstanding work items; these include preoperational tests, instrument calibrations, functional checkout of equipment, maintenance requests, and work plans.

4. TVA Commitments to NRC

Both EN DES and NUC PR maintain NRC commitment lists. Reference T lists 51 open items that EN DES is tracking and trying to resolve with the NRC prior to fuel load. Basically, these items concern meeting NUREG-0588 requirements for environmental qualification of electrical equipment, resolving significant/reportable nonconformance reports, and resolving the NRC's questions.

NUC PR is tracking 21 open items. These include NUREG commitments, IE bulletins, and NRC inspection items which must be completed prior to unit 2 fuel load. Finally, NUC PR is trying to complete work on approximately 11 significant/reportable nonconformance reports.

5. Item R-81-07-SQN-1, Employee Concern No. 79-12-01 - Required Material not in Sequoyah FSAR - Safety Concern on ERCW Pumping Station

On December 1, 1979, NSRS received an employee concern regarding the design of the new ERCW pumping station. NSRS investigated the concern and issued a report (see reference NN). The following recommendations were made and required to be completed prior to unit 2 fuel load:

a. The FSAR should be amended to address the foundation exploration and improvement for the ERCW pumping station. A level of detail equivalent to that incorporated in the FSAR for other category I structures should be provided. (See section VI.A.)

b. The FSAR should be amended to address the potential hazards to the ERCW pumping station. The amendment

should be worked on a schedule to support unit 2 fuel loading and should include the following as a minimum:

(1) A clear distinction between the description of the ERCW and CCW pumping stations.

(2) Updated figures to properly correspond to the FSAR text. Specifically, figures 2.1-4 and 2.2-2 do not appear to be in complete agreement with the text.

(3) A description of the methodology utilized in addressing the potential hazards resulting from collisions during nonflood conditions, including the possible collision of a barge travelling in the upstream direction.

In reference OO, EN DES established a schedule for meeting the recommendations.

Shortly after NSRS completed its investigation and just before unit 1 received its license, the NRC became concerned about the design of the ERCW pumping station (see references PP through TT and CCC). The NRC transmitted their concern in the form of an FSAR question (No. 2.47).

The NSRS reviewer discussed the status of implementation of our recommendations with EN DES. The reviewer learned that only one recommendation (i.e., V.A.5.b(2), above) had been implemented. Current plans are to implement the remaining recommendations in SQN FSAR amendments 68 and 69. NSRS has been informed of what will be incorporated in the FSAR based on previous analysis and formal correspondence (i.e., refernces PP through TT and CCC). NSRS agrees with what has been done and with draft copies of FSAR amendments 68 and 69. Consequently, implementation of our recommendation is still required, but unit 2 fuel load is no longer dependent upon formal completion of this item.

B. Procedure Review

1. Basis

The Technical Specifications and appendix A of Regulatory Guide (RG) 1.33 require specific procedures and instructions to cover the following general categories: administrative

instructions; general plant operating instructions; procedures for startup, operation, and shutdown of safety-related systems; procedures for control of radioactivity; procedures for abnormal, off-normal, and alarm conditions; procedures for combating emergencies; procedures for control of measuring and test equipment and for surveillance tests, procedures, and calibrations; maintenance instructions; chemical and radiochemical control procedures; and site radiological emergency plan. NSRS used this as a basis to determine if all procedures and instructions have been written and approved for unit 2. Selected procedures were given a cursury review. The results of this review are presented below.

2. Item R-81-07-SQN-2, Lack of Maintenance Instructions

A maintenance instruction for repair and/or replacement of the incore and excore flux monitoring of neutron detectors is required by RG 1.33. NSRS reviewers were unable to identify an instruction to comply with this requirement for either unit 1 or unit 2. Managers in both the Instrument Maintenance and Electrical Maintenance Sections were contacted by NSRS reviewers in an effort to locate and identify the applicable plant instruction. No instruction could be found. Consequently, NSRS recommends that an instruction or a group of instructions be written for repair and/or replacement of the incore and excore flux monitoring detectors.

3. Item R-818-SQN-3, Lack of Management Control of Surveillance Program

As stated earlier, the NSRS reviewers were checking to determine if all surveillance requirements have been addressed by instructions. Because a copy of the proposed Technical Specification for unit 2 was not available, our initial review was performed by comparing the unit 1 Technical Specification surveillance requirements to the existing plant surveillance instructions. Standard Practice, SQA 41, is a cross-index of the surveillance requirements versus the surveillance instruction number. Surveillance Instruction, SI-1, references SQA 41 as the basis for scheduling of the plant surveillance instructions. Therefore, NSRS reviewers used this document in the comparison. It must also be noted that SI-1 is a Plant Operations Review Committee (PORC) reviewed document while SQA 41 is not PORC reviewed. The NSRS review

revealed that the following surveillance requirements had been inadvertently omitted from SQA 41:

4.4.1.3.1

4.4.3.2.1

4.4.3.2.2

4.4.3.2.3

4.8.3.1.a.1.c

This omission was discussed with the supervisors of QA, management services, compliance staff, and operations. NSRS reviewers learned that no one group feels that they have the responsibility to ensure that all Technical Specification requirements are addressed by a specific surveillance instruction in SQA 41. Management services stated that their responsiblity was to update SQA 41 as changes were given to them by other groups at SQN and to schedule the surveillances based on SQA 41. QA stated that after issuance of the operating license and the initial Technical Specifications, they did check to make sure all Technical Specification surveillance requirements were listed in SQA 41; however, as the Technical Specifications have been added or changed, QA had not assumed the responsibility to ensure that SQA 41 is current. After reviewing the FSAR, applicable administrative instructions, and the N-OQAM, NSRS reviewers found no statement regarding this area of responsibility; however, the N-OQAM, part II, section 4.5, did state that "the initial test schedule shall be reviewed by appropriate plant sections to ensure it lists all Technical Specification surveillance requirements."
Therefore, NSRS has the following concerns:

a. Is there a surveillance instruction for each omitted surveillance requirement listed above?

b. If so, has the surveillance been conducted in the appropriate time frame?

c. If a required surveillance was not conducted, has appropriate action been taken?

d. What controls exist, or will be established, to prevent future omission of surveillance requirements from SQA 41?

e. Should PORC review SQA 41 as added assurance that all surveillance requirements will be satisfied since it is the primary basis for scheduling surveillances?

NSRS requests that NUC PR address the above listed concerns. NSRS recommends that NUC PR take action to update SQA 41 and ensure that all surveillance requirements have been, or will be, met. NSRS also recommends that the responsibility for ensuring SQA 41 is current be delineated.

4. Item R-81-07-SQN-4, Inaccurate Organization Representation

The NSRS review of the plant organization included a review of chapter 13 of the FSAR; DPM N74A20; N-OQAM, part I, section 2.1; figure 6.2-2 of the SQN-1 Technical Specifications; and reference WW, which details a proposed change to figure 6.2-2 of the SQN-1 Technical Specifications. None of the documents depicted the same plant organization for SQN. In addition, table 13.1-1 of the FSAR names individuals and their qualifications who hold positions as key staff specialists; this table contains 1974 vintage information. Consequently, this table is extremely out of date.

Also, this review involved a brief look at the experience requirements of individuals in plant positions. ANSI N18.1-1971 requires the Assistant Power Plant Superintendent to have a minimum of 10 years of responsible power plant experience. Section 13.1.3.1 of the FSAR and N-OQAM, part III, section 6.1, state that only eight years is needed. Therefore, a discrepancy exists between plant documents and NRC commitments which must be corrected.

All of the above NSRS concerns on SQN have also been noted as applicable to Watts Bar Nuclear Plant (WBN) in reference UU. In response to our concerns on WBN, reference VV states that WBN will revise their FSAR and N-OQAM to reflect the current organization and the ANSI experience requirements. In addition, DPM N74A20 will be revised to indicate the current plant organization. NSRS agrees with the response to our concerns on WBN; consequently, we recommend that SQN take similar corrective actions. Also, table 13.1-1 of the FSAR should be deleted,

if possible, or changed to list those individuals and their experience who presently hold positions as key staff specialists.

5. Item R-81-07-SQN-5 - Lack of Management Control in the Area of Nuclear Operator Training Program

In reviewing the SQN N-OQAM, AI-14, and chapter 13 of the FSAR and by comparing them to DPM No. N78A13, the POTC hot license program, and the requalification program, several inconsistencies were identified. In comparing these TVA documents to TVA's response to H. R. Denton's letter dated March 28, 1980 (reference ZZ), there was further confusion.

The Sequoyah N-OQAM references DPM No. N78A13 as the controlling document for TVA NUC PR operator training programs.

The SQN AI-14 references the SQN N-OQAM, part III, section 6.1, which in turn, as stated above, references the DPM N78A13.

There is evidence in the referenced POTC programs that NUC PR is in fact meeting some of the commitments made in TVA's response to H. R. Denton, but NSRS was unable to find (as an example) in any of the referenced documents the required (by Denton's March 28, 1980, letter) minimum qualifications Shift Engineers and Assistant Shift Engineers must meet prior to fuel loading. The SQN N-OQAM in paragraphs 1.4.3.1.2 and 1.4.3.1.3 references DPM No. N78A13, but it does not provide this information. In reviewing the AI-14, we found it referenced only 10CFR55. (This was items A.1.a and .b in reference ZZ.)

In items A.2.a and A.2.b of reference ZZ (three months on shift requirement), we again could find no evidence of these requirements in AI-14 or POTC (Hot License Program) training programs.

The TVA NUC PR training program as outlined in the plant N-OQAM, FSAR, and DPM No. N78A13 should implement the NRC requirements, and the program description should be consistent with the implementing documents.

SQN requalification and replacement training should meet the requirements of H. R. Denton's letter, sections A and C of enclosure 1, as specified in SQN Technical Specification 6.4.1. The SQN AI-14 description of requalification

and replacement training does not indicate these requirements are being met. The administrative instruction needs revising.

The SQN FSAR chapter 13.2, which describes the operator training program, does not indicate that Denton's letter requirements are being met. This chapter needs revising.

The findings of NSRS were indicative of a programmatic problem in management control of the very critical area of nuclear safety, licensed operator training.

The NSRS conclusion is that presently the SQN N-OQAM and DPM No. N78A13 are not providing complete management control of the SQN operator training program. It is the NSRS recommendation that NUC PR update their management control procedures (N-OQAM and DPM No. N78A13) promptly and that the plant and POTC then bring their instructions and program descriptions into compliance with the controlling documents.

6. Item R-81-07-SQN-6, Errors and Inconsistencies in Sequoyah Nuclear Plant Instructions

NSRS performed a cursory review of 25 SQN instructions. Detailed below are specific comments for each instruction. It is important to note that any of the following comments taken individually are relatively minor; however, as a whole they are significant. When a reviewer can briefly review a few instructions and find several errors, then questions are raised about the adequacy and completeness of all the instructions and the system for review and approval. Several LER's for SQN-1 resulted from previous problems with the instructions. Therefore, to minimize and eventually prevent such problems, NSRS recommends that all SQN procedures and instructions be reviewed in depth in a timely manner.

a. SI-14, Verification of Containment Integrity, Revision 8

(1) In steps 3, 13, 14, 15, 16, and 18 of SI-14, pipe plugs were checked. However, these pipe plugs were not listed on the SI data sheet.

(2) In step 28 of SI-14, PCV 1-77-838 was listed. However, the applicable data sheet listed PCV 1-77-828.

(3) The data sheet for SI-14 indicated there was a step 7.a in the instruction. However, the body of the instruction did not contain a step 7.a.

b. SI-17, Containment Shield Building Emergency Gas Treatment System Flow, Revision 8, February 17, 1981

(1) In the body of the instruction, steps were included to place the system in service. However, these steps were not included on the data sheet.

(2) There were no instructions to return the system to the standby readiness condition when the SI was completed.

c. SI-166, Summary of Valve Tests for ASME Section XI, Revision 2, January 23, 1981

The cover page stated that this SI is "for Unit 1 only," but it has been assigned to the unit 2 control room.

d. SI-166.1, Full Stroking of Category "A" and "B" Valves During Operation, Revision 3, February 3, 1981

The cover page stated that this SI is "for Unit 1 only," but it has been assigned to the unit 2 control room.

e. SOI-35.1, Generator Hydrogen Cooling System, Revision 10, January 30, 1981

There was no precaution regarding the Technical Specification requirement of maintaining a tank level of at least 50 percent at a pressure of greater than 270 psig in the CO_2 storage tank for fire protection when the generator is being purged. Addition of a precaution statement on this requirement could prevent a licensee event report (LER) in the future.

f. SOI-55, Annunciator Response

(1) SOI-55 was written for unit 1 equipment and panels, but the document has also been assigned to the unit 2 control room. There may be

differences between the equipment and panels in units 1 and 2.

(2) SOI-55 contained punch list items dating back to 1977 which should have been reviewed and resolved since the system on unit 1 is supposedly operational.

g. SOI-67.3, Essential Raw Cooling Water System, Revision 1, October 14, 1980

The cover page stated that this SOI is "for Unit 1 only," but it has been assigned to the unit 2 control room.

h. SOI-68.2, Reactor Coolant Pumps, Revision 14, March 16, 1981

Section V should not only reference an inspection checksheet for reactor coolant pump No. 1 but also should reference inspection checksheets for reactor coolant pumps Nos. 2, 3, and 4.

i. SOI-70.1, Component Cooling Water, Revision 14, February 3, 1981

(1) Instructions for shutdown of the component cooling water pumps have not been provided.

(2) The cover page and the valve checklist stated that this SOI is "for Unit 1 only," but this instruction has been assigned to the unit 2 control room.

j. PHY SI-25, Key Control, Revision 1, February 6, 1980

(1) PHY SI-25 stated that security padlocks for vital areas shall be rotated quarterly. However, the Physical Security Plan and the Public Safety Section Instruction Letters only required annual rotation.

(2) Paragraph 25.3.1 of PHY SI-25 stated that the Shift Engineer (SE) has control of the key cabinets and only he can authorize admittance to the areas controlled by those keys. Since the SE has been required to maintain his office in the main control room, this responsibility

for key control has fallen to the Assistant Shift Engineer (ASE) and clerk who now occupy the SE's office outside the main control room.

k. HCI-G6, Clearance Procedure Requirements, April 6, 1979

This instruction on clearance appears to be redundant with AI-3. This redundancy could cause some confusion as to which is the controlling document.

l. AI-3, Clearance Procedures, Revision 8, January 29, 1981

See item V.B.6.k above.

m. SQM 1, Sequoyah Nuclear Plant Maintenance Program, December 13, 1979

Section III, paragraph 2, stated that maintenance employees must coordinate all requests for clearance with the SE. Also, the SE must handle the need for additional surveillance or radiation monitoring. It appeared to the NSRS reviewer that the ASE was handling these duties since the SE had been relieved of many administrative duties when he was required to maintain his office in the control room.

The following procedures were also briefly reviewed but no comments are offered:

a. PHY SI-2, Access Control of Personnel, Revision 12, August 15, 1980

b. GOI-1, Plant Startup from Cold Shutdown to Hot Standby, Revision 17, January 13, 1981

c. GOI-2, Plant Startup for Hot Standby to Minimum Load, Revision 15

d. FHI-6, Preparation for Refueling, Revision 6, July 24, 1980

e. FHI-7, Refueling Operation - Initial Core Loading, Revision 6, March 24, 1981

f. RCI-2, Radiological Hygiene Training, Revision 6, July 3, 1980

g. RCI-3, Personnel Monitoring, Revision 7, July 3, 1980

h. HCI-E3, Safety Grounds, July 18, 1978

i. HCI-M18, Identification of Piping Systems, July 18, 1978

j. AI-2, Authorities and Responsibilities for Safe Operation and Shutdown, Revision 13, March 2, 1981

k. SMI-0-82-1, Replacement of Diesel Generator Turbocharger, Revision 1

l. SMI-0-79-3, Removal of Nicks and Scratches from Fuel Elements, Revision 1

7. Item R-81-07-SQN-7, Unreviewed Temporary Alteration Control Forms

The SQN temporary alteration forms (TVA 6266) on unit 2 were reviewed in reference to readiness for fuel loading. At the time of review there were over 150 temporary alterations in effect on unit 2. Some on CSSC equipment had been properly controlled by PORC review and Plant Superintendent approval. There were, however, approximately 100 temporary alterations on CSSC equipment which had not been PORC reviewed and approved by the Plant Superintendent. Division procedure N73011, dated November 5, 1980, states that "These requirements become applicable at the time of tentative transfer of a system, structure, or component to NUC PR." It also states that a CSSC alteration shall not be considered temporary if it is to remain in effect over 60 days without issuance of a design change request (DCR). SQN Administrative Instruction AI-9 did not contain either requirement.

DPM N73011 and AI-9 state that quarterly review of the temporary alteration log is to be done by PORC to ensure that they are not being misused and are being handled according to procedure. Many of the SQN-2 CSSC temporary alterations have been in effect since May, June, and July of 1980.

AI-9 also states that the Shift Engineer will control all temporary alterations. At the time of review the TVA 6266 forms in effect were being maintained in an office now being occupied by an Assistant Shift Engineer (ASE).

If the ASE is to handle the temporary alterations, AI-9 and DPM N73011 should be changed to reflect this.

AI-9 did not reflect all of the requirements of DPM N73011, and the quarterly review of temporary alterations did not identify this discrepancy.

VI. Personnel Contacted

William T. Cottle, Assistant Power Plant Superintendent, Operations

Edward A. Craigge, Supervisor, Fire and Safety

James R. Crisp, Supervisor, Administrative Services

James T. Crittenden, Lieutenant, Public Safety

James W. Doty, Supervisor, Mechanical Maintenance

Preston E. Fairfax, Jr., Assistant Shift Engineer

Ronald W. Fortenberry, Supervisor, Nuclear Engineering Group

Michael E. Frye, Instrument Engineer

James W. Gaines, Supervisor, Power Stores

Albert H. Gelston, Electrical Engineer

Samuel E. Griffen, Public Safety Officer

W. Michael Halley, Supervisor, Preoperational Testing

*Robert L. Hamilton, Quality Assurance Engineer

**Michael R. Harding, Supervisor, Compliance Staff

Thomas L. Howard, Jr., Quality Assurance Engineer

Jack R. Hunt, Senior Instrument Mechanical Foreman

Zia M. Kabiri, Supervisor, Plant Services

Robert S. Kaplan, Supervisor, Public Safety

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H. Baxter Norman, Engineering Associate

Boyd M. Patterson, Supervisor, Power Plant Maintenance

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David P. Roberts, Nuclear Engineer

Edward Saputa, Jr., Preoperational Test Engineer

Virgil T. Smith, Electrical Engineer

Paul D. Tallent, Industrial Engineer

Stuart A. Thickman, EN DES, Senior Engineer

James R. Walker, Training Officer

Walter A. Watson, Supervisor, Electrical Maintenance

Archie M. Wilkey, Instrument Engineer

Stephen C. Willard, Nuclear Engineer

Donald L. Williams, EN DES, Nuclear Engineering

VII. Documents Reviewed (References)

A. SQN Final Safety Analysis Report

B. SQN Operational Quality Assurance Manual

C. SQN-1 Technical Specifications

D. SQN-2 Technical Specifications (Proposed)

E. SQN Standard Practices

F. SQN Administrative Instructions

G. SQN System Operating Instructions

H. SQN General Operating Instructions

I. SQN Physical Security Instructions

J. SQN Public Safety Section Instruction Letters

K. SQN Quality Assurance Section Instruction Letters

L. SQN Surveillance Instructions

M. SQN Power Stores Section Instruction Letters

N. SQN Special Maintenance Instructions

O. SQN Hazard Control Instructions

P. SQN Radiological Control Instructions

Q. SQN Fuel Handling Instructions

R. SQN Surveillance Instruction Schedule, 4/81

S. SQN Preoperational Test Schedule, 4/2/81

T. NRC Open Items (EN DES Responsibility)

U. US NRC Regulatory Guide 1.8, "Personnel Qualification and Training"

V. US NRC Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)"

W. US NRC Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operations)"

X. US NRC Regulatory Guide 1.64, "Quality Assurance Requirements for the Design of Nuclear Power Plants"

Y. US NRC Regulatory Guide 1.123, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants"

Z. US NRC Regulatory Guide 1.38, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of items for Water-Cooled Nuclear Power Plants"

AA. US NRC Regulatory Guide 1.58, "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel"

BB. American National Standard ANSI N45.2-1971, "Quality Assurance Program Requirements (Design and Construction)"

CC. American National Standard ANSI N45.2.2-1972, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water Cooled Nuclear Power Plants"

DD. American National Standard ANSI N45.2.6-1973, "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel"

EE. American National Standard ANSI N45.2.13-1975, "Quality Assurance Requirements for Control of Procurement of Items and Servcies for Nuclear Power Plants"

FF. American National Standard ANSI N45.2.23, "Qualification of Quality Assurance Program Audit Personnel for Nuclear Facilities," Draft 3, Revision 0

GG. American National Standard ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel"

HH. American National Standard ANS-3.2/N18.7-1976, "Quality Assurance for the Operational Phase of Nuclear Power Plants"

II. 10CFR50, Appendix B

JJ. TVA Topical Report, TVA-TR75-1

KK. Office of Power Quality Assurance Program

LL. Division Procedure Manual

MM. Letter from H. R. Denton to TVA dated March 28, 1980 (A0 28004 02003)

NN. Memorandum from H. N. Culver to G. H. Kimmons dated April 23, 1980, Employee Concern Case No. 79-12-01 - Safety Concern on Sequoyah Nuclear Plant ERCW Pumping Station (DES 800429 013)

OO. Memorandum from G. F. Dilworth to H. N. Culver dated June 3, 1980, Sequoyah Nuclear Plant - Employee Concern - Case No. 79-12-01 - Safety Concern on ERCW Pumping Station (NEB 800603 250)

PP. Letter from L. M. Mills to A. Schwencer dated December 31, 1980 (A27 801231 004)

QQ. Letter from L. M. Mills to A. Schwencer dated July 28, 1980 (A27 800728 002)

RR. Letter from L. M. Mills to A. Schwencer dated August 5, 1980 (A27 800805 024)

SS. Letter from L. M. Mills to A. Schwencer dated August 11, 1980 (A27 800811 029)

TT. Letter from L. M. Mills to H. R. Denton dated March 13, 1981 (A27 810313 019)

UU. Memorandum from H. N. Culver to H.J. Green dated March 5, 1981, entitled "Watts Bar Nuclear Plant Unit 1 - Nuclear Safety Review Staff Review Report No. R-81-03-WBN" (GNS 810305 001)

VV. Memorandum from C. C. Mason to J. G. Dewease dated April 1, 1981, entitled "Watts Bar Nuclear Plant - Nuclear Safety Review Staff Review Report No. R-81-03-WBN" (L54 810401 954)

WW. Letter from L. M. Mills to H. R. Denton dated March 25, 1981, TVA-SNP-TS-12 (A27 810325 009)

XX. Memorandum from G. F. Dilworth to J. G. Dewease dated May 20, 1980, entitled "Sequoyah Nuclear Plant Unit 2 and Watts Bar Nuclear Plant Unit 1 - Milestones for Completion of Preoperational Tests" (NEB 800512 276)

YY. Outstanding Work Items List dated April 9, 1981

ZZ. Letter from L. M. Mills to H. R. Denton dated November 10, 1980 (A27 801112 005)

AAA. SQN Hot License Program (POTC)

BBB. SQN Requalification Program (POTC)

CCC. Letter from L. M. Mills to A. Schwencer dated April 6, 1981 (NEB 810408 682)

DDD. NUC PR Open Item List, 4/15/81